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Licensee Event Report (LER) Compilation

For month of June 1988

Oak Ridge National Laboratory

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

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Abstract

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

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

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[1] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 88-004
REACTOR BUILDING HYDROGEN CONCENTRATION INSTRUMENT INOPERABLE DUE TO INADEQUATE POST-MODIFICATION TESTING.
EVENT DATE: 021088 REPORT DATE: 040588 NSSS: BW TYPE: PWR
VENDOR: COMSIP DELPHI INC.

(NSIC 208961) ON 2/10/88, WHILE PERFORMING AN OPERABILITY VERIFICATION TEST ON THE LEAD HYDROGEN ANALYZER, THE REMOTE CONTROL ROOM HYDROGEN CONCENTRATION INDICATION WAS OBSERVED TO READ ZERO WHEN A HYDROGEN TEST GAS WAS APPLIED TO THE ANALYZER. TROUBLESHOOTING BY MAINTENANCE AND ENGINEERING PERSONNEL REVEALED THAT TWO WIRES LOCATED IN THE LEAD HYDROGEN ANALYZER PANEL WERE NOT TERMINATED ON CORRECT TERMINAL BLOCK LOCATIONS. THE WIRING DISCREPANCY WAS CORRECTED AND THE AS-BUILT WIRING CONFIGURATION WAS DETERMINED. THE LEAD HYDROGEN ANALYZER WAS CALIBRATED AND RETURNED TO SERVICE ON 2/14/88. THE REDUNDANT STANDBY HYDROGEN ANALYZER WAS REMOVED FROM SERVICE TO DETERMINE IF A SIMILAR PROBLEM EXISTED. A HYDROGEN TEST GAS WAS APPLIED TO THE STANDBY ANALYZER AND PROPER INDICATIONS WERE OBSERVED. THE STANDBY ANALYZER WAS RETURNED TO SERVICE ON 2/15/88. THE CAUSES WERE (1) A FAILURE TO REFLECT AS-BUILT CONFIGURATION OF THE PANEL WIRING IN THE DESIGN DRAWINGS AFTER INSTALLATION OF THE REMOTE INDICATION TO MEET NUREG-0737 REQUIREMENTS AND (2) A FAILURE TO PROVIDE ADEQUATE GUIDANCE FOR POST-MODIFICATION TESTING AFTER A DECEMBER 1986 DESIGN MODIFICATION. CURRENT DESIGN MODIFICATION PROCEDURES CONTAIN ADEQUATE GUIDANCE FOR POST-MODIFICATION TESTING AND AS-BUILT VERIFICATIONS.

[2] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 88-004
VIBRATION INDUCED CLOSURE OF AIR VOLUME DISTRIBUTION DAMPER IN THE CONTROL ROOM EMERGENCY AIR CONDITIONING SYSTEM RESULTS IN DEGRADED SYSTEM COOLING AND AIR MIXING CAPABILITY.
EVENT DATE: 021188 REPORT DATE: 041188 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 1 (PWR)
VENDOR: AMERICAN AIR FILTER CO., INC.

(NSIC 209033) ON 2/11/88, DURING AN AIR BALANCE OF THE CONTROL ROOM EMERGENCY AIR CONDITIONING SYSTEM, AN OBSERVED DEGRADED SYSTEM FLOW LED TO THE DISCOVERY THAT A VOLUME DAMPER IN THE SYSTEM DUCTWORK TO THE ANO-1 CONTROL ROOM WAS CLOSED. THE DAMPER WAS OPENED AND SYSTEM FLOW RETURNED TO NORMAL. IT IS BELIEVED THAT THE VOLUME DAMPER CLOSED DUE TO VIBRATION. A WINGNUT ATTACHED TO THE DAMPER OPERATING SHAFT AND USED TO SECURE THE DAMPER IN POSITION WAS TIGHTENED TO MINIMIZE INADVERTENT CLOSURE. THE SYSTEM PROVIDES EMERGENCY COOLING AND AIR MIXING FOR THE ANO-1 AND ANO-2 CONTROL ROOMS. THE CLOSED VOLUME DAMPER DEGRADED THE SYSTEM TOTAL AIR FLOW AND THE OCCURRENCE WAS DETERMINED TO BE A CONDITION THAT ALONE COULD HAVE PREVENTED THE FULFILLMENT OF A SAFETY FUNCTION. THE OCCURRENCE WAS ALSO A CONDITION PROHIBITED BY TECH SPEC FOR BOTH UNITS. THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINOR IN THAT OTHER COOLING METHODS WERE AVAILABLE AND COULD HAVE BEEN USED AND AIR MIXING WOULD HAVE OCCURRED DUE TO PRESSURIZATION BY THE CONTROL ROOM EMERGENCY RECIRCULATION AND FILTRATION SYSTEM. FUTURE CORRECTIVE ACTIONS INCLUDE REVIEWS TO EVALUATE IF SAFETY-RELATED VOLUME DAMPERS IN OTHER PLANT VENTILATION SYSTEMS ARE SUSCEPTIBLE TO INADVERTENT CLOSURE.

[3] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 88-003
UNPLANNED AUTOMATIC ACTUATION OF ENGINEERED SAFETY FEATURES ACTUATION SYSTEM DUE TO DEENERGIZING AN ELECTRICAL DISTRIBUTION SYSTEM VITAL POWER PANEL FOR MAINTENANCE.
EVENT DATE: 031088 REPORT DATE: 041188 NSSS: CE TYPE: PWR
VENDOR: COLT INDUSTRIES, INC.
ITE/GOULD

(NSIC 208975) ON 3/10/88 AT APPROXIMATELY 0320 HOURS, AN UNPLANNED AUTOMATIC ACTUATION OF THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) OCCURRED

WHEN A 120 VOLT AC VITAL POWER PANEL (2RS-2) WAS DEENERGIZED FOR MAINTENANCE. THE ACTUATION RESULTED IN GENERATION OF A SAFETY INJECTION ACTUATION SIGNAL (SIAS) AND A CONTAINMENT COOLING ACTUATION SIGNAL (CCAS). THE UNIT WAS DEFUELED WITH ALL FUEL ASSEMBLIES STORED IN THE SPENT FUEL POOL AT THE TIME OF THE EVENT. A SIGNIFICANT AMOUNT OF PLANT EQUIPMENT WAS OUT OF SERVICE FOR MAINTENANCE RELATED ACTIVITIES THEREFORE MINIMUM EQUIPMENT ACTUATIONS OCCURRED AS A RESULT OF THE SIAS AND CCAS. THE "B" TRAIN EMERGENCY DIESEL GENERATOR (EDG) 2K4B, STARTED AS DESIGNED UPON RECEIPT OF THE SIAS. HOWEVER, A FEW SECONDS LATER THE ENGINE SHUTDOWN RELAY ACTUATED CAUSING THE ENGINE TO AUTOMATICALLY STOP. OPERATIONS PERSONNEL RESTORED POWER TO 2RS-2 AND RESET THE ESFAS. EQUIPMENT WHICH HAD ACTUATED AS A RESULT OF THE SIGNALS WAS REALIGNED TO DESIRED CONFIGURATIONS. THE CAUSE OF THE UNPLANNED ESFAS ACTUATION WAS DETERMINED TO BE LOSS OF POWER TO THE "B" CHANNEL ESFAS BY DEENERGIZING 2RS-2 IN CONJUNCTION WITH THE PRESENCE OF A TRIP SIGNAL IN CHANNEL "C." THE CAUSE OF THE EDG AUTOMATIC SHUTDOWN WAS DUE TO DRIFT OF A TIME DELAY (TD) RELAY IN THE ENGINE PROTECTION CIRCUIT.

[4] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 88-005
CONTINUOUS FIRE WATCH PERSONNEL FOUND ASLEEP.
EVENT DATE: 031988 REPORT DATE: 040788 NSSS: CE TYPE: PWR

(NSIC 208872) ON 3/19/88 AT 0440 HOURS, THE SHIFT FIRE WATCH FOREMAN FOUND A FIRE WATCH ON STATION ASLEEP. THE FIRE WATCH WAS POSTED IN THE ANO-2 UPPER NORTH PIPING PENETRATION ROOM. THE FIRE WATCH HAD BEEN AT THE POST FOR APPROXIMATELY 15 MINUTES WHEN FOUND ASLEEP. AT THE TIME OF THE EVENT, ANO-2 WAS IN A REFUELING MODE. THE FIRE WATCH POST HAD BEEN ESTABLISHED DUE TO MODIFICATIONS BEING PERFORMED IN THE AREA WHICH REQUIRED BREACHING A FIRE BARRIER. THE EVENT IS NOT CONSIDERED TO BE SAFETY SIGNIFICANT AS THIS ROOM IS MONITORED BY SMOKE DETECTORS WITH CONTROL ROOM ALARM WHICH WERE OPERABLE AND THE FIRE LOAD IN THE ROOM IS LOW. HOWEVER, TECH SPEC 3.7.11 REQUIRES THAT A CONTINUOUS FIRE WATCH BE ESTABLISHED WHEN A FIRE BARRIER IS NOT FUNCTIONAL. THEREFORE, THE OCCURRENCE OF THE FIRE WATCH SLEEPING WHILE STATIONED ON POST WAS CONSIDERED TO BE A VIOLATION OF THIS TECH SPEC REQUIREMENT. THE INDIVIDUAL WAS IMMEDIATELY RELIEVED OF FIRE WATCH DUTIES AND EMPLOYMENT WAS TERMINATED. FIRE WATCH PERSONNEL ARE CURRENTLY ROTATED TO DIFFERENT POSTS APPROXIMATELY EVERY 30 MINUTES TO MINIMIZE THE POTENTIAL FOR THIS TYPE OF OCCURRENCE. ADDITIONALLY, ROUTINE TOURS BY FIRE WATCH ROVERS AND OTHER PLANT PERSONNEL SHOULD DETECT THIS TYPE OCCURRENCE.

[5] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 88-006
CABLE SPREADING ROOM FIRE WATER SYSTEM REMOVED FROM SERVICE TO PREVENT INADVERTENT ACTUATION DUE TO CONSTRUCTION ACTIVITIES BEING PERFORMED IN AREA.
EVENT DATE: 040288 REPORT DATE: 042188 NSSS: CE TYPE: PWR

(NSIC 209034) ANO-2 IS SHUT DOWN FOR A REFUELING OUTAGE. DURING THE OUTAGE, A MODIFICATION OF THE UNIT'S CABLE SPREADING ROOM TO PROVIDE AN AREA FOR INSTALLATION OF NEW CORE PROTECTION CALCULATOR COMPUTERS AT A FUTURE DATE IS BEING PERFORMED. THE CABLE SPREADING ROOM IS EQUIPPED WITH A FIRE PROTECTION WATER DELUGE SYSTEM WHICH IS AUTOMATICALLY ACTUATED UPON DETECTION OF HEAT AND SMOKE IN THE AREA. MODIFICATION OF THE AREA REQUIRED CONSTRUCTION ACTIVITIES WHICH INCLUDED WELDING AND GRINDING OPERATIONS. ON 3/15/88 AT 1050 HRS, TO PRECLUDE INADVERTENT AUTOMATIC ACTUATION OF THE DELUGE SYSTEM, THE SYSTEM WAS MANUALLY ISOLATED BY CLOSING A VALVE IN THE FIRE WATER SUPPLY LINE. A CONTINUOUS FIRE WATCH WITH BACKUP FIRE SUPPRESSION EQUIPMENT WAS ESTABLISHED FOR THE AREA PRIOR TO CONTINUOUS FIRE WATCH WITH BACKUP FIRE SUPPRESSION EQUIPMENT WAS ESTABLISHED FOR THE AREA PRIOR TO REMOVING THE SYSTEM FROM SERVICE AS REQUIRED BY TECH SPEC 3.7.10.2. TECH SPEC 3.7.10.2 ALSO REQUIRES SUBMISSION OF A SPECIAL REPORT TO THE COMMISSION IF THE SYSTEM IS NOT RESTORED TO AN OPERABLE STATUS WITHIN 14 DAYS. IN ORDER TO COMPLETE THE MODIFICATION OF THE AREA, IT IS NOT POSSIBLE TO RETURN THE SYSTEM TO SERVICE WITHIN THE 14 DAY TIME PERIOD.

MODIFICATIONS TO THE AREA ARE PROCEEDING AND IT IS ANTICIPATED THAT THE SYSTEM WILL BE RESTORED TO AN OPERABLE STATUS.

[6] BEAVER VALLEY 1 DOCKET 50-334 LER 88-001 REV 01
UPDATE ON STEAM GENERATOR TUBE PLUGGING.
EVENT DATE: 020888 REPORT DATE: 040488 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELEC CORP.-NUCLEAR ENERGY SYS

(NSIC 208854) DURING THE SIXTH REFUELING OUTAGE, BEAVER VALLEY CONDUCTED INSPECTIONS TO OBTAIN U-TUBE WALL THICKNESSES ON ALL THREE STEAM GENERATORS (SG). ALL INSERVICE TUBES ON ALL THREE SGS WERE INSPECTED BY MULTI-FREQUENCY EDDY CURRENT TESTING. A TOTAL OF 72 TUBES WERE PLUGGED (37 TUBES ON THE 1A SG, 20 TUBES ON THE 1B SG AND 15 TUBES ON THE 1C SG). FORTY-SEVEN (47) TUBES WERE FOUND TO HAVE DEGRADATION IN EXCESS OF 40%, 19 TUBES IN ROWS 1 AND 2 WERE FOUND WITH DEGRADATION LESS THAN 40%, BUT LOCATED IN THE U-BEND RADIUS PORTION OF THE TUBE AND WERE PLUGGED PREVENTIVELY AND 6 TUBES WERE IDENTIFIED AS SUSCEPTIBLE TO THE NORTH ANNA FAILURE MECHANISM. OF THE PLUGGED TUBES ON THE 1A SG, 29 EXHIBITED DEGRADATION IN THE U-BEND PORTION, 4 ON THE 1ST SUPPORT BAR ON THE COLD LEG SIDE (SBCL), 2 ON THE 2ND SBCL, 1 ON THE 3RD SBCL AND 1 ON THE 6TH SBCL. OF THE PLUGGED TUBES ON THE 1B SG, 10 EXHIBITED DEGRADATION IN THE U-BEND PORTION, 3 ON THE 1ST SBCL, 5 ON THE 1ST SUPPORT BAR, HOT LEG SIDE (SBHL), 1 ON THE 1ST, 2ND, 3RD SBHL AND 1 NEAR THE TUBE SHEET. OF THE PLUGGED TUBES ON THE 1C SG, 9 EXHIBITED DEGRADATION IN THE U-BEND PORTION, 3 ON THE 1ST SBCL, 2 ON THE 1ST SBHL AND 1 ON THE 2ND SBCL. THIS INFORMATION IS BEING REPORTED IN ACCORDANCE WITH TECH SPEC 4.4.5.5.A, FOLLOWING THE INSERVICE INSPECTIONS OF ALL THREE SGS CONDUCTED THIS SIXTH REFUELING OUTAGE.

[7] BEAVER VALLEY 2 DOCKET 50-412 LER 87-032 REV 01
UPDATE ON REACTOR TRIP DUE TO 100% LOAD REJECTION TEST.
EVENT DATE: 102487 REPORT DATE: 112387 NSSS: WE TYPE: PWR

(NSIC 208837) ON 10/24/87, THE 100% LOAD REJECTION TEST (IST 2.04.06) WAS PERFORMED. THE OPERATORS, AS PER PROCEDURE, MANUALLY OPENED THE MAIN OUTPUT BREAKERS TO INITIATE A LOSS OF LOAD TRANSIENT. CONDENSER STEAM DUMPS AUTOMATICALLY OPENED IN RESPONSE TO THE LOSS OF LOAD. THE RESULTANT STEAM FLOW TRANSIENT CAUSED ALL THREE STEAM GENERATOR LEVELS TO DROP RAPIDLY. A REACTOR TRIP OCCURRED ON LO-LO STEAM GENERATOR LEVEL. ALL AUXILIARY FEED PUMPS AUTO-STARTED TO RECOVER STEAM GENERATOR LEVELS. DUE TO TURBINE SPEED FLUCTUATIONS, ALL THREE REACTOR COOLANT PUMPS (RCPS) TRIPPED ON UNDERFREQUENCY. THIRTY SECONDS AFTER THE REACTOR TRIP, STATION LOADS WERE (AS DESIGNED) AUTOMATICALLY TRANSFERRED FROM ON-SITE TO OFF-SITE POWER. HOWEVER, THE "A", "B" AND "AE" 4KV BUSSES DID NOT SUCCESSFULLY TRANSFER DUE TO PHASE DIFFERENTIAL BETWEEN ON-SITE AND OFF-SITE POWER. THE #1 EMERGENCY DIESEL GENERATOR AUTO-STARTED AND REENERGIZED THE "AE" BUS. THE "A" AND "B" BUSSES WERE MANUALLY REALIGNED TO OFF-SITE POWER. THE "A" RCP WAS RESTARTED. OPERATORS STABILIZED THE PLANT USING THE REACTOR TRIP RESPONSE PROCEDURE. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. THIS EVENT WAS BOUNDED BY FSAR SECTION 15.2.6 (REACTOR TRIP FROM 100%, COINCIDENT WITH LOSS OF OFF-SITE POWER).

[8] BEAVER VALLEY 2 DOCKET 50-412 LER 88-007
REACTOR TRIP DUE TO REACTOR COOLANT PUMP TRIP CAUSED BY A LOSS OF 4KV BUS 2A MOTOR LOADS.
EVENT DATE: 040488 REPORT DATE: 050488 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELEC CORP.-NUCLEAR ENERGY SYS

(NSIC 209234) ON 4/4/88 AT 0800 HOURS, WITH THE UNIT IN POWER OPERATION AT 100% REACTOR POWER, A SURVEILLANCE TEST OF THE 4KV AND 480 VAC NORMAL BUS UNDERVOLTAGE (UV) PROTECTION WAS INITIATED. THIS TESTING INVOLVES USING A BLOCKING RELAY IN

THE UV CIRCUITRY WHILE TESTING THE UV PROTECTION RELAYS. DURING TESTING OF THE 2A 4KV BUS, ACTUATION OF THE UNDERVOLTAGE RELAYS OCCURRED CAUSING A LOSS OF 2A 4KV BUS MOTOR LOADS. THIS CAUSED THE "A" REACTOR COOLANT PUMP TO TRIP INITIATING A REACTOR TRIP, AT 0847 HOURS, DUE TO THE BREAKER POSITION OF THE COOLANT PUMP. THE OPERATORS STABILIZED THE PLANT IN HOT STANDBY USING THE EMERGENCY OPERATING PROCEDURES. THE CAUSE FOR THE UV PROTECTION RELAY ACTUATIONS WAS DETERMINED TO BE A MALFUNCTIONING BLOCKING RELAY. THE CONTACT SPACING WAS FOUND TO BE OUT OF ADJUSTMENT CAUSING IMPROPER OPERATION. THIS RELAY WAS ADJUSTED AND CALIBRATED USING A RELAY CALIBRATION PROCEDURE. THE UNDERVOLTAGE PROTECTION SURVEILLANCE TEST WAS PERFORMED SATISFACTORILY AND THE RELAY WAS RETURNED TO SERVICE. THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC AS A RESULT OF THIS INCIDENT. THE UNDERVOLTAGE PROTECTION IS DESIGNED TO SHED THE LOADS ON THE RESPECTIVE BUS BEFORE ANY DAMAGE TO THE LOADS ARE RECEIVED DUE TO THE OPERATION AT A REDUCED VOLTAGE. THIS TYPE OF EVENT IS DISCUSSED IN THE FINAL SAFETY ANALYSIS REPORT, SECTION 8.3.1.1.11.

[9] BRAIDWOOD 1 DOCKET 50-456 LER 87-031 REV 02
UPDATE ON CONTROL ROOM VENTILATION SHIFT TO THE EMERGENCY MAKEUP MODE AS A RESULT OF SPURIOUS ACTUATION OF A RADIATION MONITOR DUE TO DESIGN DEFICIENCY.
EVENT DATE: 061387 REPORT DATE: 033088 NSSS: WE TYPE: PWR

(NSIC 208867) AT 1908 ON JUNE 13, 1987, AT 1126 ON OCTOBER 16, 1987, AND AGAIN AT 0225 ON DECEMBER 28, 1987, IT WAS DISCOVERED THROUGH MAIN CONTROL ROOM ANNUNCIATION THAT TRAIN OA OF THE CONTROL ROOM VENTILATION SYSTEM HAD SHIFTED TO ITS EMERGENCY MAKEUP MODE OF OPERATION. THESE ACTUATIONS WERE ATTRIBUTED TO THE PRESSURE SWITCHES OF MONITORS OPR31J AND OPR32J WHICH SEND AN ELECTRICAL IMPULSE WHICH IS READ BY THE MONITOR AS A RADIATION SPIKE. IN ALL THREE OCCURRENCES ALL MONITOR CHANNEL ACTIVITY READINGS RETURNED TO NORMAL WITHIN 30 MINUTES AND THE LINEUP FOR CONTROL ROOM VENTILATION WAS SUBSEQUENTLY RESTORED. THE FIRST OCCURRENCE WAS CONSIDERED TO BE AN ISOLATED EVENT AND NO ADDITIONAL ACTION WAS TAKEN. AFTER THE THIRD OCCURRENCE, WORK REQUESTS WERE WRITTEN TO INSTALL NOISE SUPPRESSING ELECTROCUBES IN THE MONITOR CIRCUITS. THERE HAVE BEEN NO PREVIOUS OCCURRENCES DUE TO KEYING A RADIO CAUSING AN ENGINEERED SAFETY FEATURE ACTUATION.

[10] BRAIDWOOD 1 DOCKET 50-456 LER 88-006
PARTIAL LOSS OF AUXILIARY BUILDING NON-ACCESSIBLE FILTER PLENUM POSITION INDICATION DUE TO AN ADMINISTRATIVE AND MANAGEMENT DEFICIENCY.
EVENT DATE: 031388 REPORT DATE: 040688 NSSS: WE TYPE: PWR

(NSIC 208977) ON MARCH 12, 1988, A REQUEST WAS MADE TO REMOVE RELAYS FROM THE AUXILIARY BUILDING VENTILATION SYSTEM TO VERIFY THAT THEY WERE ENVIRONMENTALLY QUALIFIED. THE ASSOCIATED OUT-OF-SERVICES WERE PREPARED AND AT 0210 ON MARCH 13, 1988, WERE PLACED. AT 0830 ON MARCH 13, 1988, DURING A TURNOVER WALKDOWN OF THE CONTROL ROOM PANELS, A LACK OF DAMPER POSITION INDICATION FOR THE NON-ACCESSIBLE PLENUMS INLET AND OUTLET DAMPERS WAS NOTED. IT WAS DETERMINED THAT THE OUT-OF-SERVICES REMOVED POWER TO THE INLET (FAIL OPEN) AND OUTLET (FAIL CLOSED) DAMPERS ASSOCIATED WITH FIVE OUT OF THE SIX CHARCOAL BOOSTER FANS. AT 0945 ON MARCH 13, 1988, THE AUXILIARY BUILDING VENTILATION SYSTEM WAS RETURNED TO NORMAL. THE CAUSE WAS AN ADMINISTRATIVE AND MANAGEMENT DEFICIENCY IN THAT THE PERSONNEL ASSIGNED THE TASK OF DETERMINING AND VERIFYING THE ISOLATION POINTS LACKED THE EXPERTISE AND ADEQUATE REFERENCE MATERIAL REQUIRED TO PERFORM THE FUNCTION. THIS EVENT HAS BEEN REVIEWED WITH THE INDIVIDUALS INVOLVED AND WILL BE REVIEWED WITH THE REST OF THE OPERATING SHIFT PERSONNEL. ALSO, A REVIEW OF THE OUT-OF-SERVICE PROGRAM AND ASSOCIATED MATERIAL AVAILABLE TO THE OPERATORS WILL BE PERFORMED. THERE HAVE BEEN NO PREVIOUS OCCURRENCES OF INADEQUATE OUT-OF-SERVICE VERIFICATION ON ANY VENTILATION SYSTEM.

[11]   BRAIDWOOD 1   DOCKET 50-456   LER 88-008
REACTOR COOLANT SYSTEM LEAKAGE DUE TO BROKEN RELIEF VALVE DISC PIN.
EVENT DATE: 032588   REPORT DATE: 042588   NSSS: WE   TYPE: PWR
VENDOR: CROSBY VALVE & GAGE CO.

(NSIC 209187) ON MARCH 25, 1988 AND MARCH 27, 1988, OPERATORS NOTED A DECREASING VOLUME CONTROL TANK LEVEL WHICH CAUSED INCREASED MAKE-UP. REACTOR COOLANT WATER INVENTORY BALANCE SURVEILLANCES CONFIRMED THAT UNIDENTIFIED LEAKAGE WAS IN EXCESS OF 1 GALLON PER MINUTE (GPM). THE SOURCE OF THE MARCH 25, 1988 OCCURRENCE WAS THOUGHT TO BE AN IMPROPERLY LOCKED CLOSED VALVE WHICH WAS INADVERTENTLY BUMPED OFF ITS CLOSED SEAT. THE RESIDUAL HEAT REMOVAL (RHR) PUMP SUCTION RELIEF VALVES MAY HAVE CONTRIBUTED TO THE GENERATING STATION EMERGENCY PLAN UNUSUAL EVENT FOR BOTH OCCURRENCES. LEAKAGE PAST THE SEATS BY MEASURING THE DOWNSTREAM TEMPERATURE INDICATED THE SOURCE OF LEAKAGE. SUBSEQUENT INVESTIGATION OF ONE OF THE RELIEF VALVES INDICATED THAT THE DISC INSERT PIN WAS BROKEN AS A RESULT OF IMPROPER NOZZLE RING SETTING. THE 1A RHR SUCTION RELIEF VALVE HAS BEEN REPAIRED AND REINSTALLED. THE 1B RELIEF VALVE WILL BE TESTED AND REPAIRED AS NECESSARY PRIOR TO RESTART OF THE UNIT. THERE HAVE BEEN NO PREVIOUS OCCURRENCES OF CROSBY RELIEF VALVE FAILURES.

[12]   BRAIDWOOD 2   DOCKET 50-457   LER 88-001
SAFETY INJECTION PUMP DISCHARGE VALVE LOCKED CLOSED DUE TO OEPRATOR MISCOMMUNICATION.
EVENT DATE: 030588   REPORT DATE: 040788   NSSS: WE   TYPE: PWR

(NSIC 208978) ON MARCH 5, 1988, A MECHANICAL MAINTENANCE FOREMAN REQUESTED THAT A LOCK AND CHAIN BE REMOVED FROM THE 2B SAFETY INJECTION (SI) PUMP MANUAL DISCHARGE ISOLATION VALVE, 2SI8921B, TO FACILITATE UNRELATED MAINTENANCE ON A FLANGE. AN OPERATOR WAS DISPATCHED AND THE LOCK AND CHAIR WERE REMOVED, FOLLOWING THE MAINTENANCE ACTIVITY, A SECOND OPERATOR WAS DISPATCHED TO REPLACE THE LOCK AND CHAIN. AT 1650 ON MARCH 13, 1988, IT WAS DISCOVERED THAT VALVE 2SI8921B WAS LOCKED CLOSED. THE VALVE WAS RESTORED TO IT'S NORMAL LOCKED OPEN POSITION. SUBSEQUENT INVESTIGATION REVEALED THAT IT WAS CLOSED AT 1?4? ON MARCH 5, 1988. ROOT CAUSE WAS A MISCOMMUNICATION BETWEEN A LICENSED OPERATOR AND THE SECOND NON-LICENSED OPERATOR. CONTRIBUTING TO THE LENGTH OF TIME THAT THE VALVE WAS MISPOSITIONED WAS A LACK OF COGNIZANCE THAT REMOVING A CHAIN AND LOCK, AS IN THIS CASE, CONSTITUTES A CHANGE IN POSITION. INDIVIDUALS INVOLVED IN THIS EVENT HAVE PARTICIPATED IN A REVIEW OF THIS EVENT WITH SENIOR OPERATING MANAGEMENT. APPROPRIATE DISCIPLINARY ACTION COMMENSURATE WITH THEIR PERFORMANCE HAS BEEN ACCOMPLISHED. A REVIEW OF THIS EVENT STRESSING PROPER COMMUNICATIONS WILL BE PERFORMED. PROPER GUIDANCE ON THE USE OF THE COMPONENT ABNORMAL POSITION LOG WILL BE REVIEWED WITH OPERATING SHIFT PERSONNEL. NO PREVIOUS OCCURRENCES.

[13]   BRAIDWOOD 2   DOCKET 50-457   LER 88-005
COMPONENT COOLING WATER SURGE TANK LEVEL INSTRUMENTATION DISCREPANCIES DUE TO A DESIGN ERROR.
EVENT DATE: 032188   REPORT DATE: 042188   NSSS: WE   TYPE: PWR
OTHER UNITS INVOLVED: BRAIDWOOD 1 (PWR)

(NSIC 209188) INVESTIGATION OF THE UNIT 2 COMPONENT COOLING (CC) SURGE TANK REVEALED THAT THE ASSOCIATED LEVEL INSTRUMENTATION, MAIN CONTROL BOARD (MCB) INDICATORS AND LOW CC SURGE TANK LEVEL PUMP TRIP SWITCHES WERE CROSSWIRED. THIS WAS CAUSED BY AN ERROR IN THE DESIGN ORGANIZATION. THE RESULTANT INSTALLATION CONTRIBUTED TO THE TESTING ERRORS BY THE BRAIDWOOD TESTING ORGANIZATION. THE ARCHITECT ENGINEER (AE) INITIATED CHANGES IN RESPONSE TO WESTINGHOUSE LETTER CAW-4151/CBW-3396. THE AE DESIGN CHANGES DID NOT CONSIDER THE POWER SUPPLIES FOR THE LEVEL INSTRUMENTS, MCB LABELING, HIGH/LOW LEVEL ALARMS, OR SIGNIFICANT EVENT RECORDER ALARM MESSAGE REVISIONS REQUIRED TO PROPERLY CONFIGURE THE CC SURGE TANK AS A RESULT OF A LACK OF UNDERSTANDING OF THE BAFFLE PLATE. TESTING DID REVEAL

DRAIN NUMBER ERRORS, BUT FURTHER INVESTIGATION WAS NOT CONDUCTED SINCE THE TEST STEPS WERE OTHERWISE COMPLETED AS ORIGINALLY WRITTEN. CAUTION CARDS WERE ADDED TO THE MCB LEVEL INDICATION AND LOCAL INDICATORS AND DRAIN VALVES TO ALERT OPERATING PERSONNEL TO THE PROPER INSTRUMENTATION TO MONITOR DURING TANK LEVEL CHANGE EVOLUTIONS. LOW LEVEL PUMP TRIP FUNCTION WAS DISABLED AND MODIFICATIONS HAVE BEEN INITIATED TO CORRECT THESE ERRORS. THE CC SURGE TANK WILL BE PROPERLY LABELED TO PREVENT SIMILAR OCCURRENCES. NO PREVIOUS OCCURRENCES OF DESIGN ORGANIZATION ERRORS DUE TO INADEQUATE DESIGN CHANGES.

[14] BROWNS FERRY 1 DOCKET 50-259 LER 86-024 REV 02
UPDATE ON LOSS OF SECONDARY CONTAINMENT CAUSED BY DESIGN OVERSIGHT.
EVENT DATE: 082286 REPORT DATE: 041988 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 209038) ON AUGUST 22, 1986, IT WAS DETERMINED BY TVA DESIGN ENGINEERS THAT VARIOUS NONSAFETY-RELATED PIPING SYSTEMS PENETRATING THE REACTOR BUILDING-TURBINE BUILDING WALL ARE NOT SEISMICALLY QUALIFIED. THIS CONDITION, IS NOT CONSISTENT WITH THE DESIGN DESCRIPTION IN SECTION 5.3 OF THE FINAL SAFETY ANALYSIS REPORT (FSAR), WHICH STATES THAT SECONDARY CONTAINMENT PENETRATIONS ARE SEISMICALLY CONSTRUCTED. AS A CONSERVATIVE MEASURE, PLANT MANAGEMENT DECLARED THE SECONDARY CONTAINMENT INOPERABLE, AND PLANS PENDING AT THE TIME TO UNLOAD FUEL FROM THE UNIT 3 CORE WERE HALTED. A TVA DESIGN OVERSIGHT PERMITTED DRAWINGS TO BE ISSUED WHICH DID NOT INCLUDE ANY REQUIREMENTS TO SEISMICALLY QUALIFY SECONDARY CONTAINMENT PENETRATIONS, AS DESCRIBED IN THE FSAR. A FUEL HANDLING ACCIDENT IS THE ONLY EVENT THAT COULD CHALLENGE SECONDARY CONTAINMENT UNDER THE PRESENT PLANT CONDITIONS. AN ANALYSIS WAS PERFORMED WHICH FOUND DOSES RESULTING FROM SUCH AN EVENT TO BE WELL WITHIN 10CFR100 GUIDELINES. TVA WILL DEMONSTRATE THAT THE PLANT IS CAPABLE OF MAINTAINING A NEGATIVE 1/4-INCH OF WATER PRESSURE IN SECONDARY CONTAINMENT FOLLOWING A DESIGN BASIS EARTHQUAKE PRIOR TO UNIT 2 STARTUP.

[15] BROWNS FERRY 1 DOCKET 50-259 LER 87-020
PRIMARY CONTAINMENT ISOLATION DUE TO REACTOR WATER CLEANUP TEMPERATURE SWITCH FAILURE.
EVENT DATE: 080287 REPORT DATE: 090187 NSSS: GE TYPE: BWR
VENDOR: FENWALL, INC.

(NSIC 209200) ON AUGUST 2, 1987 AT 0100, WITH THE UNIT DEFUELED, THE TEMPERATURE SWITCH MONITORING THE NONREGENERATIVE HEAT EXCHANGER OUTLET TEMPERATURE INITIATED AN UNEXPECTED ISOLATION OF THE REACTOR WATER CLEANUP SYSTEM THROUGH THE PRIMARY CONTAINMENT ISOLATION SYSTEM. THIS WAS AN ACTUATION OF AN ENGINEERED SAFETY FEATURE. THE TEMPERATURE SWITCH PERFORMS THE NONSAFETY RELATED FUNCTION OF PROTECTING THE FILTER DEMINERALIZER RESINS FROM HIGH TEMPERATURES. THE POWER SUPPLY AMPLIFIER BOARD TO THE TEMPERATURE SWITCH WAS REPLACED, THE TEMPERATURE SWITCH WAS RECALIBRATED, THE ISOLATION WAS RESET AND RWCU WAS RETURNED TO SERVICE APPROXIMATELY 21 1/2 HOURS AFTER THE ISOLATION WAS INITIATED. A REVIEW OF THE MAINTENANCE HISTORY OF THE TEMPERATURE SWITCH HAS NOT INDICATED A PREVIOUS FAILURE OF THIS KIND. NO FURTHER CORRECTIVE ACTIONS ARE PLANNED.

[16] BROWNS FERRY 1 DOCKET 50-259 LER 88-013
RADIATION MONITOR SPIKE INITIATES CONTROL ROOM EMERGENCY VENTILATION.
EVENT DATE: 032888 REPORT DATE: 042688 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 209128) ON MARCH 28, 1988, AT 1430 HOURS, WITH ALL THREE UNITS DEFUELED, BOTH CONTROL ROOM EMERGENCY VENTILATION (CREV) TRAINS INADVERTENTLY STARTED DUE TO A HIGH RADIATION SIGNAL FROM THE CONTROL ROOM AIR INLET RADIATION MONITOR

(RM). A HIGH RADIATION SIGNAL IS A DESIGNED START SIGNAL FOR THE CREV TRAINS. IN PREPARATION FOR INSTALLING INSULATION PLANT CRAFTSMEN WERE VACUUMING NEAR THE UNIT 3 CONTROL BAY RM. A HIGH RADIATION SIGNAL WAS RECEIVED FROM THIS RM WHICH RESULTED IN CONTROL ROOM VENTILATION ISOLATION AND BOTH CREV TRAINS STARTING. THIS IS CONSIDERED AN UNANTICIPATED ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF). THE ASSISTANT SHIFT OPERATIONS SUPERVISOR VERIFIED, BY THE REVIEW OF THE RM CHART RECORDER, THAT A SPIKE ON RM 0-RM-90-259B HAD CAUSED THE ESF ACTUATION. TWELVE MINUTES LATER THE UNIT OPERATOR RESET THE ISOLATION AND RETURNED THE CREV TO STANDBY READINESS. INSTRUMENT AND CONTROLS PERSONNEL WILL INVESTIGATE THE POSSIBILITY OF A RADIO FREQUENCY TRIPPING 0-RM-90-259B USING A RADIO FREQUENCY GENERATOR AND DETERMINE IF THE RM IS ADEQUATELY SHIELDED. THE SHIFT OPERATIONS SUPERVISOR REQUESTED THE MODIFICATION CARPENTERS PLACE A PROTECTIVE PLYWOOD SHELL AROUND 0-RE-90-259B TO ENSURE THAT AFFECTED SAFETY EQUIPMENT IS PROTECTED AS NECESSARY, WHEN MODIFICATIONS ARE BEING PERFORMED.

[17] BROWNS FERRY 1 DOCKET 50-259 LER 88-014
SURVEILLANCE TESTING OF LIQUID RADIOACTIVE WASTE DISCHARGE ISOLATION VALVES INCOMPLETE DUE TO INADEQUATE PROCEDURES.
EVENT DATE: 032988 REPORT DATE: 042688 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 209250) AS A RESULT OF THE PROGRAMMATIC UPGRADE OF THE BROWNS FERRY SURVEILLANCE INSTRUCTIONS (SI) IT WAS REPORTED ON MARCH 29, 1988, THAT THE SI WHICH TESTS THE AUTOMATIC ISOLATION LOGIC OF THE LIQUID RADIOACTIVE WASTE DISCHARGE ISOLATION VALVES DID NOT FULLY TEST ALL OPERATIONAL CONFIGURATIONS AND THAT THE FLOW SWITCH ON THE COOLING TOWER BLOWDOWN LINE WAS NOT IDENTIFIED AS A TECH SPEC REQUIRED INSTRUMENT AND HAD NOT BEEN CALIBRATED ON A REGULAR SCHEDULE. TECH SPEC 4.8.A.3 REQUIRES ANNUAL TESTING OF THE AUTOMATIC ISOLATION VALVES. FAILURE TO TEST THE LOGIC AND CALIBRATE THE FLOW SWITCH WAS A VIOLATION OF THE TECH SPEC SURVEILLANCE REQUIREMENT. NO INSTANCES OF INADVERTENT DISCHARGE DUE TO LOGIC FAILURE HAVE BEEN IDENTIFIED. ALL THREE UNITS WERE DEFUELED AT THE TIME OF DISCOVERY. ADMINISTRATIVE CONTROLS WERE ESTABLISHED TO ENSURE IMPROPER RELEASES WERE NOT PERMITTED UNTIL SUCH TIME AS THE AUTOMATIC ISOLATION LOGIC COULD BE TESTED FOR EACH UNIT RESPECTIVELY. SIS HAVE BEEN PREPARED WHICH FULLY TEST THE PUMP INTERLOCK LOGIC. A CALIBRATION PROCEDURE WILL BE PREPARED FOR THE FLOW SWITCH. THE DISCOVERY OF THIS DEFICIENCY IS CONSIDERED A GOOD INDICATION OF THE QUALITY AND CAPABILITY OF THE UPGRADED PROGRAM, THEREFORE ADDITIONAL RECURRENCE CONTROL ACTIONS ARE NOT CONSIDERED NECESSARY.

[18] BRUNSWICK 2 DOCKET 50-324 LER 88-007
PINHOLE LEAKS AND LINEAR INDICATIONS IN THE INSERT AND WITHDRAW LINES OF CONTROL ROD DRIVES.
EVENT DATE: 031988 REPORT DATE: 041888 NSSS: GE TYPE: BWR
VENDOR: BROWN & ROOT INC.

(NSIC 209009) DURING THE UNIT 2 1988 REFUEL/MAINTENANCE OUTAGE, VISUAL AND LIQUID PENETRANT INSPECTIONS OF THE UNIT CONTROL ROD DRIVE (CRD) INSERT/WITHDRAW LINES REVEALED THE EXISTENCE OF PINHOLE AND LINEAR INDICATIONS IN THE LINES. THESE PROBLEMS WERE INITIALLY REVEALED ON 3/19/88, DURING THE VISUAL LEAK INSPECTION IN ACCORDANCE WITH THE HYDROSTATIC PRESSURE TEST OF THE REACTOR PRESSURE VESSEL, PERIODIC TEST (PT) 80.1. IN EACH CASE, A RUST-COLORED DEPOSIT OF UNKNOWN ORIGIN WAS FOUND IN THE AFFECTED AREAS OF THE LINES. THE INDICATIONS ARE ATTRIBUTED TO CHLORIDES CONTAINED IN THE DEPOSITS, WHICH LED TO TRANSGRANULAR STRESS CORROSION CRACKING OF THE LINE MATERIAL, SCHEDULE 80, TYPE 304, STAINLESS STEEL. THE AFFECTED LINE SECTIONS WERE APPROPRIATELY REPAIRED, REPLACED, OR EVALUATED AS ACCEPTABLE FOR OPERATION AND WERE RETURNED TO SERVICE. DURING THE NEXT UNIT 1 REFUEL/MAINTENANCE OUTAGE, THE UNIT CRD INSERT/WITHDRAW LINES WILL BE INSPECTED TO DETERMINE IF THE SAME CONDITION EXISTS ON THAT UNIT.

[19]   BRUNSWICK 2   DOCKET 50-324   LER 88-008
FULL REACTOR PROTECTION SYSTEM TRIP RESULTING FROM A DOWNSCALE INOPERATIVE TRIP OF AVERAGE POWER RANGE MONITOR D WITH REACTOR PROTECTION SYSTEM SHORTING LINKS REMOVED.
EVENT DATE: 033188   REPORT DATE: 042988   NSSS: GE   TYPE: BWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 209226) DURING THE UNIT 2 1988 REFUEL/MAINTENANCE OUTAGE, A FULL REACTOR PROTECTION SYSTEM (RPS) TRIP OCCURRED AT 0532 HOURS ON 3/31/88, WHEN MODE SWITCH (S1) OF AVERAGE POWER RANGE MONITOR (APRM) D WAS ROTATED BEYOND THE SWITCH MECHANICAL STOP POSITION. THIS ENERGIZED THE MONITOR INOPERATIVE CIRCUITRY. THE RPS SHORTING LINKS WERE REMOVED (WHICH ALLOWS ANY TRIP SIGNAL TO CAUSE A FULL RPS TRIP) FOR THE WITHDRAWAL OF CONTROL ROD DRIVE (CRD) 46-27 AND THE CONTROL OPERATOR (CO) WAS IN THE PROCESS OF VERIFYING OPERABILITY OF THE NEUTRON MONITORING SYSTEM. THERE WAS NO ADVERSE SAFETY SIGNIFICANCE DUE TO THIS EVENT. S1 HAD BEEN REPLACED ON 1/29/88, WITH AN IMPROPERLY CONFIGURED SWITCH RESULTING FROM AN ERROR MADE IN PROCUREMENT OF THE SWITCH FROM THE SUPPLIER. A SIMILAR PROBLEM AND ROOT CAUSE WAS ALSO FOUND INVOLVING THE METER PANEL FUNCTION SWITCH OF APRM F. S1 WAS REPLACED WITH A PROPERLY CONFIGURED SWITCH AND APRM D WAS RETURNED TO SERVICE AT 1320 HOURS ON 4/1/88. THE METER PANEL FUNCTION SWITCH OF APRM F WAS PROPERLY CONFIGURED AND RETURNED TO SERVICE ON 4/20/88. THESE EVENTS WILL BE EVALUATED AND APPROPRIATE ACTION WILL BE TAKEN TO PREVENT FUTURE REOCCURRENCES. A PRIOR SIMILAR EVENT WAS REPORTED IN LER 1-88-005.

[20]   BRUNSWICK 2   DOCKET 50-324   LER 88-009
FULL REACTOR PROTECTION SYSTEM (RPS) TRIP WHILE SELECTING A CONTROL ROD FOR WITHDRAWAL WITH RPS SHORTING LINKS REMOVED DURING REFUELING/MAINTENANCE OUTAGE.
EVENT DATE: 033188   REPORT DATE: 042688   NSSS: GE   TYPE: BWR

(NSIC 209132) DURING THE UNIT 2 1988 REFUEL/MAINTENANCE OUTAGE, WHEN CONTROL ROD DRIVE (CRD) 10.39 WAS SELECTED, A FULL REACTOR PROTECTION SYSTEM (RPS) TRIP OCCURRED AT 1817 HOURS ON 3/31/88, DUE TO AN UPSCALE TRIP OF INTERMEDIATE RANGE MONITOR (IRM) D WITH THE RPS SHORTING LINKS REMOVED. IRM D WAS BYPASSED AND WITHIN TEN MINUTES, THE RPS TRIP SIGNAL WAS RESET. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL AS CONTROL RODS WERE FULLY INSERTED IN THE CORE PRIOR TO THE RPS TRIP. AT 1855 HOURS, THIS EVENT WAS DUPLICATED DURING PLANNED TROUBLESHOOTING ACTIVITIES. CHECKS OF THE SIGNAL CABLE CONNECTORS AT THE MONITOR INSTRUMENTATION DRAWER AND PREAMPLIFIER INPUT AND OUTPUT DID NOT REVEAL PROBLEMS WHICH COULD BE ASSOCIATED WITH THE EVENT. THIS EVENT IS ATTRIBUTABLE TO SPURIOUS ELECTRONIC NOISE ASSOCIATED WITH OPERATION OF THE REACTOR MANUAL CONTROL SYSTEM (RMCS). ELECTRONIC NOISE SUPPRESSORS HAVE BEEN INSTALLED IN THE RMCS CIRCUITRY OF BOTH UNITS AS CORRECTIVE ACTION TO PRIOR SIMILAR EVENTS INVOLVING ELECTRONIC NOISE IN THE CIRCUITRY. IRM D WAS RETURNED TO SERVICE FOLLOWING A DETERMINATION THAT THE MONITOR WAS NOT SUSCEPTIBLE TO SPIKING RESULTING FROM ELECTRONIC NOISE IN THE RMCS CIRCUITRY.

[21]   BYRON 1   DOCKET 50-454   LER 87-021
CONTROL ROOM VENTILATION ACTUATION DUE TO DISTRIBUTION SYSTEM VOLTAGE TRANSIENT WHEN ON OFFSITE LINE TRIPPED.
EVENT DATE: 091787   REPORT DATE: 092587   NSSS: WE   TYPE: PWR
OTHER UNITS INVOLVED: BYRON 2 (PWR)

(NSIC 209197) ON SEPTEMBER 17, 1987, AT 0612, WITH UNIT 1 IN POWER OPERATION (MODE 1) AT 97% REACTOR POWER AND UNIT 2 IN POWER OPERATION (MODE 1) AT 83% REACTOR POWER, PROCESS RADIATION MONITORS OPR31J (MAIN CONTROL ROOM OUTSIDE AIR INTAKE 'A') AND OPR32J (MAIN CONTROL ROOM OUTSIDE AIR INTAKE 'A') SENSED AN UNDERVOLTAGE CONDITION AND TRANSFERRED TO THE INTERLOCK MODE. THE INTERLOCK SIGNAL TRANSFERRED THE MAIN CONTROL ROOM VENTILATION SYSTEM TO ITS ENGINEERED SAFETY FEATURES CONFIGURATION. A TRIP OF AN OFFSITE 345KV TRANSMISSION LINE

CAUSED A VOLTAGE TRANSIENT ON THE COMMONWEALTH EDISON GRID WHICH CAUSED THE MONITORS TO TRANSFER TO THE INTERLOCK MODE. MODIFICATIONS TO LOWER THE UNDERVOLTAGE TRIP SETPOINTS WERE INSTALLED IN 1985 AND HAVE REDUCED THE MONITOR'S SENSITIVITY TO VOLTAGE TRANSIENTS CAUSED BY LARGE PUMP STARTS AND MOST GRID DISTURBANCES. THIS IS CONSIDERED AN ISOLATED OCCURRENCE AND NO FURTHER CORRECTIVE ACTION IS PLANNED AT THIS TIME. AN EVENT SIMILAR TO THIS HAS OCCURRED IN THE PAST (LER 86-026-00).

[22]   BYRON 2   DOCKET 50-455   LER 87-018
REACTOR TRIP ON LOW STEAM GENERATOR LEVEL WHEN THE 2B MAIN FEEDWATER PUMP TRIPPED DUE TO PERSONNEL ERROR.
EVENT DATE: 100187   REPORT DATE: 102387   NSSS: WE   TYPE: PWR

(NSIC 209198) BYRON UNIT 2 WAS OPERATING AT 92% POWER WHEN THE 2B MAIN FEEDWATER PUMP TRIPPED. THE UNIT WAS RAMPED BACK TO 50% POWER. THE REACTOR TRIPPED DUE TO A LOW-2 STEAM GENERATOR LEVEL IN ONE STEAM GENERATOR. ALL SAFETY SYSTEMS RESPONDED AS DESIGNED. THE CAUSE WAS THE INADVERTENT ACTUATION OF THE OVERSPEED TRIP PLUNGER BY A CONTRACTOR WORKING ON THE 2B FEEDWATER PUMP HIGH PRESSURE STOP VALVE. TO PREVENT RECURRENCE THE OPERATING AND MAINTENANCE PERSONNEL WILL BE REQUIRED TO READ THIS REPORT. THE INSTALLATION OF A GUARD ON THE TRIP PLUNGER WILL BE INVESTIGATED.

[23]   BYRON 2   DOCKET 50-455   LER 88-002
COMPONENT COOLING WATER SURGE TANK LEVEL INSTRUMENTATION DISCREPANCIES DUE TO A DESIGN ERROR.
EVENT DATE: 032988   REPORT DATE: 042888   NSSS: WE   TYPE: PWR

(NSIC 209165) ON MARCH 29, 1988, DISCREPANCIES WERE DISCOVERED IN THE UNIT 2 COMPONENT COOLING WATER SYSTEM (CC) SURGE TANK INSTRUMENTATION. CC SURGE TANK LEVEL TRANSMITTER 2LIT-670, INSTALLED ON THE 'B' SIDE OF THE TANK, SUPPLIES THE "2A CC SURGE TANK LEVEL" MAIN CONTROL BOARD (MCB) METER. SIMILARLY, CC SURGE TANK LEVEL TRANSMITTER 2LIT-676, INSTALLED ON THE 'A' SIDE OF THE TANK, SUPPLIES THE "2B CC SURGE TANK LEVEL" MCB METER. ADDITIONALLY, 2LIT-670 IS POWERED FROM THE 'A' TRAIN INSTRUMENT POWER SUPPLY ALTHOUGH IT ACTUALLY MEASURES TRAIN 'B' SURGE TANK LEVEL. A LIKE DISCREPANCY EXISTS ON 2LIT-676. ALSO THE CC SURGE TANK LOW LEVEL SWITCHES WOULD TRIP THE CC PUMP MOTOR CIRCUIT BREAKER ON THE OPPOSITE TRAIN OF AN ACTUAL LOW LEVEL CONDITION. THIS REPORT IS SUBMITTED VOLUNTARILY DUE TO THE POTENTIAL SIGNIFICANCE OF THESE DISCREPANCIES. THE ROOT CAUSE OF THE EVENT WAS AN ERROR BY THE DESIGN ORGANIZATION. THE ARCHITECT/ENGINEER (A/E) DESIGNERS INCOMPLETELY IMPLEMENTED A MODIFICATION OF THE CC SYSTEM, WHICH WAS INTENDED TO ACHIEVE TRAIN UNIFORMITY BETWEEN THE CC SYSTEM AND THE RESIDUAL HEAT REMOVAL SYSTEM. THE A/E DESIGNER FAILED TO UNDERSTAND THE SIGNIFICANCE OF A BAFFLE PLATE IN THE CC SURGE TANK. THIS RESULTED IN THE FAILURE TO MODIFY THE TANK'S INSTRUMENTATION. THERE HAVE BEEN NO PREVIOUS OCCURRENCES OF THIS EVENT.

[24]   BYRON 2   DOCKET 50-455   LER 88-003
TECH SPEC ACTION STATEMENT NOT SATISFIED DURING UNINTENDED 2B DIESEL GENERATOR INOPERABILITY DUE TO INCORRECT DRAWING.
EVENT DATE: 032988   REPORT DATE: 042688   NSSS: WE   TYPE: PWR

(NSIC 209166) ON MARCH 29, 1988, THE 2B DIESEL GENERATOR (DG) LEFT BANK STARTING AIR SYSTEM WAS REMOVED FROM SERVICE TO REPAIR A LEAKING VALVE. UPSTREAM AND DOWNSTREAM ISOLATION POINTS WERE CHOSEN FROM A PIPING AND INSTRUMENTATION DRAWING (P&ID) TO PERMIT THE MAINTENANCE. THE P&ID INCORRECTLY REPRESENTED THE ACTUAL PIPING ARRANGEMENT IN THE PLANT, THEREFORE, THE RIGHT BANK OF THE STARTING AIR SYSTEM WAS ALSO ISOLATED DURING THE INTENDED ISOLATION OF THE LEFT BANK. THIS CONDITION RESULTED IN THE ISOLATION OF ALL STARTING AIR FROM THE 2B DG, THUS, MAKING IT INOPERABLE. THE INOPERABILITY WAS IDENTIFIED ON MARCH 31, 1988, WHEN

AN EQUIPMENT OPERATOR NOTICED THAT THE "UNIT AVAILABLE FOR EMERGENCY" INDICATING LIGHT WAS EXTINGUISHED, AND THE TECH SPEC LIMITING CONDITION FOR OPERATION ACTION REQUIREMENT WAS IMPLEMENTED. THE FOLLOWING CORRECTIVE ACTIONS HAVE BEEN OR ARE BEING TAKEN: 1. CAUTION CARDS AND LABELS EXPLAINING THE PIPING DISCREPANCIES HAVE BEEN HUNG LOCALLY AND IN THE MAIN CONTROL ROOM. 2. THE P&ID WILL BE CORRECTED TO REPRESENT ACTUAL PLANT CONDITIONS. 3. DG AUXILIARY EQUIPMENT LABELS WILL HAVE BYRON PART NUMBERS. 4. THE OPERATING ROUNDS PROCEDURE WILL REQUIRE PERIODIC CHECKS OF THE "UNIT AVAILABLE FOR EMERGENCY" INDICATING LIGHT. 5. TRAINING PROGRAMS WILL ADDRESS THE INDICATING LIGHT. SIMILAR EVENTS HAVE NOT OCCURRED PREVIOUSLY.

[25] CALLAWAY 1 DOCKET 50-483 LER 86-007 REV 02
UPDATE ON MISSING FIRE BARRIER PENETRATION SEALS.
EVENT DATE: 031886 REPORT DATE: 040888 NSSS: WE TYPE: PWR

(NSIC 208833) ON 3/18/86 WHILE IN MODE 6 (REFUELING), 23 FIRE BARRIER PENETRATIONS WERE IDENTIFIED WHICH DID NOT HAVE INTERNAL CONDUIT SEALS INSTALLED. THE FIRE BARRIERS WERE DECLARED INOPERABLE AND FIREWATCH PATROLS VERIFIED TO BE IN PLACE. SUBSEQUENT INSPECTIONS OF THE PLANT'S INTERNAL CONDUIT SEALS WERE COMPLETED ON 8/18/87 AND IDENTIFIED 308 CONDUITS WHICH WERE MISSING AT LEAST ONE SEAL. THESE CONDUITS WERE NOT SEALED DURING CONSTRUCTION. OF THESE 308 CONDUITS, 39 WERE DETERMINED TO CONSTITUTE VIOLATIONS OF THE LIMITING CONDITION FOR OPERATION FOR TECH SPEC 3.7.11. ALTHOUGH IT WAS DETERMINED THAT FIREWATCH PATROLS HAD BEEN ASSIGNED FOR A MAJORITY OF THE AFFECTED AREAS SINCE RECEIPT OF THE OPERATING LICENSE, TOTAL COMPLIANCE WITH THE ANCILLARY T/S'S CANNOT BE DETERMINED DUE TO THE COMPOUNDING ASPECT OF THE T/S ACTION STATEMENTS. THEREFORE, THE CONDITION IS ASSUMED TO BE REPORTABLE. THE SEALS WERE REPAIRED BY 8/24/87. THE APPROPRIATE ADMINISTRATIVE PROCEDURE WAS REVISED TO ASSURE FUTURE CONDUIT SEALS ARE INSTALLED PER DESIGN REQUIREMENTS. UNION ELECTRIC JOINED A GROUP OF UTILITIES TO ANALYZE THE NECESSITY OF SEALS IN ALL CASES. ALTHOUGH THE ANALYSIS SHOWED THAT SEALS ARE NOT NECESSARY IN MANY CONFIGURATIONS, UNION ELECTRIC DETERMINED THAT NO MODIFICATIONS TO THE FIRE PROTECTION PROGRAM ARE NECESSARY.

[26] CALLAWAY 1 DOCKET 50-483 LER 87-001 REV 01
UPDATE ON ACTION STATEMENT IMPROPERLY ENTERED WHEN FIRE BARRIER INSPECTIONS COMPLETED LATE.
EVENT DATE: 021287 REPORT DATE: 050488 NSSS: WE TYPE: PWR

(NSIC 209213) THIS REPORT DOCUMENTS A FAILURE TO ESTABLISH APPROPRIATE FIREWATCHES AS THE DIRECT RESULT OF NOT COMPLETING THE VISUAL INSPECTION OF FIRE BARRIER PENETRATION SEALS AND FIRE RATED ASSEMBLIES WITHIN THE SPECIFIED TIME INTERVAL OF TECHNICAL SPECIFICATION (T/S) SURVEILLANCE 4.7.11.1. PRECAUTIONARY HOURLY FIREWATCH PATROLS WERE IN PLACE FOR AN OUTAGE. IT WAS ASSUMED THESE PATROLS WERE ADEQUATE; HOWEVER, FURTHER REVIEW INDICATED THAT APPROPRIATE FIREWATCHES WERE NOT ESTABLISHED. ADDITIONALLY, FOUR FIRE RATED ASSEMBLIES WERE DISCOVERED DEFICIENT DUE TO CONSTRUCTION ERRORS AND THE T/S FIREWATCH REQUIREMENTS FOR THESE WERE THEREFORE NOT MET SINCE RECEIPT OF THE OPERATING LICENSE. THE CAUSE OF EXCEEDING THE SURVEILLANCE DATE FINISH DATES WAS DUE TO UTILITY NON-LICENSED, MANAGEMENT PERSONNEL ERROR IN THE DETERMINATION THAT THE SEALS/ASSEMBLIES WERE OPERABLE UNTIL PROVEN INOPERABLE BY THE SURVEILLANCE INSPECTION. APPROPRIATE PERSONNEL WERE COUNSELED ON THE NEED TO SATISFY T/S ACTION STATEMENTS. ADMINISTRATIVE PROGRAMS WERE REVIEWED AND ENHANCED TO ENSURE PROPER, TIMELY CORRECTIVE MAINTENANCE OF BREACHED FIRE BARRIERS. A REVIEW WAS CONDUCTED OF OTHER SURVEILLANCES REQUIRING LONG-TERM INSPECTIONS OF A LARGE POPULATION OF PASSIVE EQUIPMENT.

[27] CALVERT CLIFFS 2 DOCKET 50-318 LER 87-009 REV 01
UPDATE ON LOSS OF MAIN GENERATOR PERMANENT MAGNET GENERATOR.
EVENT DATE: 122187 REPORT DATE: 041588 NSSS: CE TYPE: PWR
VENDOR: COPES-VULCAN, INC.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 208904) UNIT 2 TRIPPED ON DECEMBER 21, 1987 FROM 100% POWER DUE TO A LOSS OF LOAD SIGNAL. THE LOSS OF LOAD WAS CAUSED BY A FAILURE OF THE UNIT 2 MAIN GENERATOR PERMANENT MAGNET GENERATOR DUE TO THE STATOR FRAME BEING MISALIGNED. DURING THE POST-TRIP COOLDOWN, AN ATMOSPHERIC DUMP VALVE MALFUNCTIONED (FAILED TO FULLY SHUT) CAUSING A MORE RAPID COOLDOWN THAN NORMAL. THE OPERATORS SECURED THE STEAM LOADS TO MINIMIZE COOLDOWN AND SUPPLIED THE STEAM GENERATORS FROM THE AUXILIARY FEEDWATER SYSTEM. THE COOLDOWN WAS TERMINATED WHEN THE ATMOSPHERIC DUMP VALVE WAS MANUALLY ISOLATED. THE FOLLOWING CORRECTIVE ACTIONS WERE TAKEN. 1. TURBINE GENERATOR MAINTENANCE PROCEDURES WERE SIGNIFICANTLY UPGRADED. 2. THE ATMOSPHERIC DUMP VALVE WAS REPAIRED AND RETURNED TO SERVICE. THE FAILED ROLL PIN WAS TESTED BY THE MATERIALS LABORATORY. THE PIN WAS VERIFIED TO BE MADE OF THE PROPER MATERIAL. SUBSEQUENT MAINTENANCE OF OTHER ATMOSPHERIC DUMP VALVES HAS NOT REVEALED SIMILAR PROBLEMS. ATMOSPHERIC DUMP VALVE MAINTENANCE PROCEDURES HAVE BEEN UPGRADED.

[28] CALVERT CLIFFS 2 DOCKET 50-318 LER 88-002 REV 02
UPDATE ON LOSS OF FEED TRIP DUE TO THE LOSS OF INSTRUMENT BUS 22.
EVENT DATE: 012288 REPORT DATE: 041588 NSSS: CE TYPE: PWR

(NSIC 208918) DURING TROUBLESHOOTING OF THE THREE PHASE UNIT 2 COMPUTER INVERTER, POWER WAS LOST TO THE NON-VITAL 208/120V A.C. INSTRUMENT BUS 22 (2Y10). THE DE-ENERGIZATION OF 2Y10 RESULTED IN REDUCED FEEDWATER FLOW AND CAUSED THE REACTOR TO BE TRIPPED NEAR THE LOW STEAM GENERATOR LEVEL SETPOINT. THE LOSS OF INSTRUMENT BUS 22 (2Y10) WAS CAUSED BY PERSONNEL ERROR WHEN A PLANT ELECTRICIAN'S MISINTERPRETATION OF ELECTRICAL PRINTS AND UNCLEAR COMMUNICATIONS WITH THE INVERTER VENDOR LED TO THE PLACEMENT OF TEMPORARY JUMPERS IN THE INVERTER, CAUSING A DIRECT SHORT CIRCUIT THAT WAS REFLECTED BACK TO THE MAIN POWER FEED CIRCUIT BREAKER CAUSING IT TO TRIP THUS DE-ENERGIZING INSTRUMENT BUS 22 (2Y10). CORRECTIVE ACTIONS ARE: 1. A STUDY OF THE PROPER SIZE FUSING AND FUSE TO CIRCUIT BREAKER COORDINATION IS IN PROGRESS. 2. CLARIFY THE INVERTER MANUFACTURERS ELECTRICAL PRINTS TO SHOW THE ACTUAL PLACEMENT, IN THE CIRCUIT, OF THE POWER FACTOR CORRECTION CAPACITORS. 3. INVESTIGATE THE USE OF "SPECIAL" PROCEDURES IN TROUBLESHOOTING COMPLEX EQUIPMENT. 4. ALL MAINTENANCE ELECTRICIANS WILL BE TRAINED ON THE DETAILS OF THIS EVENT AS PART OF OUR CONTINUING TRAINING PROGRAM.

[29] CALVERT CLIFFS 2 DOCKET 50-318 LER 88-003
FAILURE OF A STEAM GENERATOR ISOLATION CHECK VALVE.
EVENT DATE: 031788 REPORT DATE: 041688 NSSS: CE TYPE: PWR
VENDOR: CHAPMAN DIV OF CRANE CO.

(NSIC 209008) ON MARCH 17, 1988, UNIT 2 WAS IN MODE 5, COLD SHUTDOWN, FOR A MAINTENANCE OUTAGE. THE CHECK VALVE WHICH PROVIDES ISOLATION OF NO. 22 STEAM GENERATOR FROM NO. 21 STEAM GENERATOR IN THE EVENT OF A MAIN STEAM LINE BREAK UPSTREAM OF THE MAIN STEAM ISOLATION VALVES, WAS DISASSEMBLED FOR INSPECTION. UPON DISASSEMBLY, THE DISK WAS FOUND TO BE SEVERELY BENT, MAKING THE VALVE INCAPABLE OF PERFORMING ITS DESIGN FUNCTION. THE ORIGINAL VALVE WAS REPLACED WITH AN ANCHOR/DARLING 6" - 900# TILTING DISK CHECK VALVE. THE VALVE WAS REVERSE FLOW TESTED AND PLACED INTO SERVICE. THE FAILURE OF THE VALVE WAS A RESULT OF NORMAL WEAR OF THE HINGE PINS AND BUSHING AREA, IN CONJUNCTION WITH STEAM EROSION/CUTTING OF THE DISK SEATING SURFACE. LONG TERM PREVENTIVE ACTIONS WILL BE TO PERIODICALLY EITHER REVERSE-FLOW TEST THE VALVES OR TO DISASSEMBLE AND INSPECT THE VALVE INTERNALS. THE FREQUENCY OF THE TEST WOULD BE EVERY REFUELING

OUTAGE, AND THE FREQUENCY OF INSPECTION WOULD BE TO INSPECT ONE VALVE EACH REFUELING OUTAGE. SIMILAR VALVES INSTALLED ON UNIT 1 WILL BE REVERSE FLOW TESTED DURING THE APRIL 1988 UNIT 1 REFUELING OUTAGE. ANY CORRECTIVE ACTIONS FOR THE UNIT 1 VALVES WILL BE BASED ON THE TEST RESULTS.

[30] CATAWBA 1 DOCKET 50-413 LER 87-042 REV 01
UPDATE ON DIESEL GENERATOR AUTO START AND SUBSEQUENT FAILURE OF AN EMERGENCY LOAD GROUP TO ENERGIZE DUE TO EQUIPMENT MALFUNCTIONS.
EVENT DATE: 111787 REPORT DATE: 041288 NSSS: WE TYPE: PWR
VENDOR: CUTLER-HAMMER
WESTINGHOUSE ELECTRIC CORP.

(NSIC 208911) ON 11/17/87, DUKE POWER PERSONNEL WERE TESTING A 50F COOLING FAN CONTROL RELAY IN 6.9KV BREAKER LTD-5 WHEN THE BREAKER TRIPPED AND LOCKED OUT AT 1102 HOURS DUE TO AN OUT OF ADJUSTMENT WIPER THAT SHORTS TWO TERMINALS IN THE RELAY BEING STRAINED DUE TO THE TESTING. THE MALFUNCTIONING RELAY CAUSED A PHASE TO PHASE CURRENT INBALANCE AND ACTIVATED A GROUND FAULT RELAY WHICH TRIPPED LTD-5. AS A RESULT POWER TO 4.16 KV ESSENTIAL BUS LETB WAS LOST AND CAUSED A DIESEL GENERATOR 1B AUTO START. ESSENTIAL 600V LOAD CENTERS 1ELVB AND 1ELXD DID NOT ENERGIZE AS REQUIRED BECAUSE AN 8.5 SECOND UNDERVOLTAGE TIMER HAD DRIFTED TO 9.7 SECONDS AT A PREVIOUS TIME. THEREFORE SINCE THE 1 SECOND LOAD SHED TRIP SIGNAL WHICH FOLLOWS THE TIMING OUT OF THE 8.5 SECONDS TEST TIMER WAS STILL AVAILABLE ON THE INCOMING BREAKERS OF LOAD GROUP 1 WHEN THE D/G SEQUENCER TRIED TO CLOSE FOLLOWING THE EXPIRATION OF ITS 10 SECOND TIMER (WHICH RUNS SIMULTANEOUSLY TO THE 8.5 SECOND UNDERVOLTAGE TIMER) 1ELXB AND 1ELXD FAILED TO ENERGIZE. THE OPERATOR AT THE CONTROLS MANUALLY ENERGIZED 1ELXB AND 1ELXD APPROXIMATELY 20 MINUTES AFTER INITIATION OF THE EVENT. THE WIPER ON THE 50F RELAY WAS READJUSTED AND SATISFACTORILY TESTED. APPROPRIATE DUKE POWER PERSONNEL HAVE BEEN TRAINED TO DISABLE GROUND FAULT RELAYS PRIOR TO TESTING PHASE RELAYS.

[31] CATAWBA 1 DOCKET 50-413 LER 88-014
INOPERABLE FIRE BARRIER IN VIOLATION OF TECH SPECS DUE TO THE INSTALLATION OF TELEPHONE WIRE BECAUSE OF A MANAGEMENT DEFICIENCY.
EVENT DATE: 011388 REPORT DATE: 040188 NSSS: WE TYPE: PWR

(NSIC 208947) ON JANUARY 13, 1988, CATAWBA CONSTRUCTION AND MAINTENANCE DEPARTMENT (CMD) ELECTRICIANS IMPROPERLY INSTALLED A TELEPHONE IN THE AUXILIARY FEEDWATER PUMP TURBINE CONTROL PANEL ROOM. THE METHOD USED FOR THE INSTALLATION DID NOT COMPLY WITH THE ELECTRICAL INSTALLATION SPECIFICATION FOR PENETRATION OF A FIRE BARRIER AND UNKNOWINGLY RENDERED A FIREWALL TECHNICALLY INOPERABLE. THE ERROR WAS DISCOVERED AT 1105 HOURS, ON MARCH 2, 1988, DURING A FIRE PROTECTION AUDIT. THE PENETRATION WAS PROPERLY SEALED AT APPROXIMATELY 1145 HOURS, WHICH ENDED THE INOPERABILITY. THE UNIT HAD OPERATED IN ALL MODES EXCEPT MODE 6, REFUELING, DURING THIS PERIOD OF TIME. THIS INCIDENT HAS BEEN ATTRIBUTED TO A MANAGEMENT DEFICIENCY DUE TO A BREAKDOWN IN THE REVIEW PROCESS. CMD SCHEDULING PERSONNEL HAVE NOT ROUTED CERTAIN TELEPHONE RELATED WORK ITEMS TO CMD TECHNICAL SUPPORT FOR REVIEW OF THEIR QUALITY ASSURANCE CONDITION AS REQUIRED BY STATION DIRECTIVES. FURTHERMORE, THE REQUIRED REVIEW BY CMD TECHNICAL SUPPORT HAS NOT BEEN CONSISTENTLY IMPLEMENTED AS REQUIRED BY STATION DIRECTIVES. CMD RECALLED ALL AFFECTED ITEMS IN PROGRESS WHICH HAD NOT RECEIVED A TECHNICAL SUPPORT REVIEW, AND THE REVIEWS WERE PERFORMED. CMD WILL ROUTE ALL APPROPRIATE ITEMS THROUGH TECHNICAL SUPPORT FOR REVIEW IN THE FUTURE. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS EVENT.

[32] CATAWBA 1 DOCKET 50-413 LER 88-015
DEGRADED PERFORMANCE OF UNIT 1 AUXILIARY FEEDWATER SYSTEM AND REQUIRED HOT SHUTDOWN OF BOTH UNITS DUE TO ASIATIC CLAM INFESTATION IN THE NUCLEAR SERVICE WATER SYSTEM.

EVENT DATE: 030988 REPORT DATE: 040888 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 208948) ON MARCH 9, 1988, AT APPROXIMATELY 1825 HOURS, UNIT 2 TRIPPED FROM APPROXIMATELY 20% FULL POWER (SEE LER 414/88-12). DURING THE TRANSIENT, ALL THREE AUXILIARY FEEDWATER (CA) PUMPS STARTED AUTOMATICALLY AS DESIGNED. HOWEVER, MOTOR DRIVEN CA PUMP (MDCAP) 2A SWAPPED SUCTION AUTOMATICALLY TO THE NUCLEAR SERVICE WATER (RN) SYSTEM WHEN A SUSTAINED LOW SUCTION PRESSURE CONDITION WAS SENSED, AND RAW WATER FROM LAKE WYLIE ENTERED TWO STEAM GENERATORS (S/GS). AFTER THE INITIAL TRIP RECOVERY, IT WAS NOTED THAT CA FLOW TO S/GS 2A AND 2B HAD DEGRADED FOLLOWING THE SUCTION SWAP. TWO WORK REQUESTS WERE WRITTEN TO INSPECT THE INTERNALS OF THE CA PUMP 2A TO S/G 2A AND 2B FLOW CONTROL VALVES. THE INSPECTIONS REVEALED THAT THE CAVITROL CAGES FOR THESE VALVES WERE CLOGGED WITH SHREDDED ASIATIC CLAM SHELLS. FOLLOWING DISCOVERY, ALL CA PUMPS FOR BOTH UNITS WERE DECLARED INOPERABLE. THIS RESULTED IN BOTH UNITS BEING TAKEN TO MODE 4, HOT SHUTDOWN. AT THE TIME THIS INCIDENT OCCURRED, UNIT 1 WAS IN MODE 1, POWER OPERATION, AT 97% POWER AND UNIT 2 WAS IN MODE 3, HOT STANDBY. THIS INCIDENT HAS BEEN ATTRIBUTED TO ASIATIC CLAM LARVAE FROM LAKE WYLIE ENTERING THE RN SYSTEM AND GROWING TO MATURITY IN NORMALLY STAGNANT LINES WHICH PROVIDE ASSURED WATER SUPPLIES TO VARIOUS SAFETY RELATED SYSTEMS (INCLUDING THE CA SYSTEM).

[33] CATAWBA 1 DOCKET 50-413 LER 88-017
INADVERTENT ACTUATION OF A REACTOR TRIP BREAKER DUE TO A PERSONNEL ERROR.
EVENT DATE: 031788 REPORT DATE: 041 88 NSSS: WE TYPE: PWR

(NSIC 209235) ON MARCH 17,1988, AT APPROXIMATELY 1438 HOURS, REACTOR TRIP BREAKER A (RTA) OPENED UNEXPECTEDLY. A TECHNICIAN AND A CONTROL ROOM OPERATOR (CRO) WERE IN THE PROCESS OF PERFORMING A PROCEDURE SECTION FOR RETURNING THE SOLID STATE PROTECTION SYSTEM (SSPS) TRAIN A TO NORMAL FROM TEST WHEN THIS INCIDENT OCCURRED. A REVIEW OF ACTIVITIES PRIOR TO THE INCIDENT REVEALED THAT A STEP IN THE PROCEDURE REQUIRING THE TRAIN A NUCLEAR INSTRUMENTATION POWER RANGE TRIP BLOCK TO BE REINSTALLED HAD BEEN SIGNED OFF WITHOUT THE ACTION BEING TAKEN. THIS RESULTED IN A TRAIN A REACTOR TRIP SIGNAL BEING GENERATED, TRIPPING THE TRAIN A REACTOR TRIP BREAKER. THE REACTOR TRIP BYPASS BREAKER A WAS SHUT AT THE TIME, PREVENTING A REACTOR TRIP. UNIT 1 WAS IN MODE 1, POWER OPERATION, OPERATING AT 98% POWER AT THE TIME OF THIS INCIDENT. THIS INCIDENT HAS BEEN ATTRIBUTED TO A PERSONNEL ERROR. THE TECHNICIAN AND THE CRO FAILED TO FOLLOW A CORRECT PROCEDURE. RTA WAS RECLOSED AND THE PROCEDURE WAS SUCCESSFULLY COMPLETED. THIS INCIDENT HAS BEEN REVIEWED WITH THE PERSONNEL INVOLVED AND WAS ALSO REVIEWED WITH ALL SHIFT SUPERVISORS. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS EVENT.

[34] CATAWBA 2 DOCKET 50-414 LER 88-011
ESSENTIAL SWITCHGEAR INCOMING BREAKER FAILS TO TRIP DURING TESTING DUE TO OPEN SLIDING LINK IN CONTROL CIRCUIT.
EVENT DATE: 122987 REPORT DATE: 040188 NSSS: WE TYPE: PWR

(NSIC 208949) DURING ENGINEERED SAFEGUARDS FEATURES ACTUATION PERIODIC TESTING ON DECEMBER 19, 1987, THE INCOMING BREAKER TO 4160 VOLT ESSENTIAL SWITCHGEAR 2ETB FAILED TO TRIP WHEN EXPECTED. CONCURRENT BLACKOUT AND SAFETY INJECTION (S/I) SIGNALS HAD BEEN SIMULATED WHICH REQUIRED THE ALTERNATE INCOMING BREAKER TO 2ETB TO OPEN. THE ASSOCIATED DIESEL GENERATOR (D/G) STARTED AND THE LOAD SEQUENCER PERFORMED AS EXPECTED BY CONNECTING THE S/I LOADS TO 2ETB. THE FAILURE OF THE INCOMING BREAKER TO TRIP DURING THE SEQUENCER ACTUATED LOAD SHED REQUIRED THE D/G TO BE PARALLELED WITH THE OFFSITE POWER SYSTEM. SUBSEQUENT INVESTIGATION FOUND A SLIDING LINK OPEN IN THE ALTERNATE INCOMING BREAKER'S CONTROL CIRCUIT WHICH PREVENTED THE BREAKER FROM TRIPPING ON AN S/I SIGNAL. THE SLIDING LINK WAS CLOSED AND TESTING WAS SUCCESSFULLY COMPLETED. THE D/G PARALLELED PROPERLY DURING THIS INCIDENT AND OFFSITE POWER VOLTAGES REMAINED NORMAL. THE UNIT WAS IN MODE 5, COLD SHUTDOWN, AT THE TIME OF THIS INCIDENT. THIS EVENT WAS DETERMINED TO BE

REPORTABLE ON MARCH 3, 1988. THE ACTIVITY WHICH OPENED AND DID NOT RECLOSE THE SLIDING LINK COULD NOT BE IDENTIFIED. AN ALTERNATE TRAIN OF ESSENTIAL POWER AND OFFSITE POWER WAS MAINTAINED OPERABLE AS REQUIRED BY TECHNICAL SPECIFICATIONS. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS EVENT.

[35] CATAWBA 2 DOCKET 50-414 LER 88-012
REACTOR TRIP FOLLOWED BY AN AUXILIARY FEEDWATER SUCTION SWAP TO THE NUCLEAR SERVICE WATER SYSTEM DUE TO EQUIPMENT FAILURES.
EVENT DATE: 030988 REPORT DATE: 040888 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 208950) ON MARCH 9, 1988, AT APPROXIMATELY 1825 HOURS, CONTROL ROOM OPERATORS (CROS) WERE SWAPPING FROM THE STEAM GENERATOR (S/G) MAIN FEEDWATER (CF) BYPASS VALVE TO THE S/G CF CONTROL VALVE FOR S/G 2B, DURING UNIT STARTUP FOLLOWING END-OF-CYCLE 1 REFUELING OUTAGE. WHEN THE CRO PLACED S/G CF CONTROL VALVE IN AUTO, THE VALVE OPENED UNEXPECTEDLY. THE CROS IMMEDIATELY TOOK MANUAL CONTROL OF THE VALVE AND THE CF PUMP TURBINE (PT) TO STABILIZE FEEDFLOW TO S/G 2B. LEVELS IN S/GS C AND D CONTINUED TO RISE. AT 1825:36:317 HOURS, S/G 2D HI HI LEVEL OCCURRED TRIPPING THE TURBINE AND CFPT, AND INITIATING FEEDWATER ISOLATION. LEVEL IN S/G 2A WAS DECREASING AND AT 1825:44:179 HOURS, REACTOR TRIP OCCURRED FROM 15% POWER DUE TO LOW LOW LEVEL IN S/G 2A. AFTER THE INITIAL TRIP RECOVERY, THE CROS DISCOVERED THAT MOTOR DRIVEN AUXILIARY FEEDWATER (CA) PUMP TRAIN A HAD SWAPPED SUCTION TO THE NUCLEAR SERVICE WATER (RN) SYSTEM. THE UNIT WAS RETURNED TO MODE 2, STARTUP, ON MARCH 18, 1988, AT 1501 HOURS, AND MODE 1, POWER OPERATION, AT 2035 HOURS. THIS INCIDENT HAS BEEN ATTRIBUTED TO AN EQUIPMENT FAILURE. THE UNEXPECTED OPENING OF THE S/G CF CONTROL VALVE WAS ATTRIBUTED TO A DEFECTIVE PRINTED CIRCUIT CARD AND A DEFECTIVE CONTROLLER DRIVER CARD. DUKE POWER PERSONNEL REPLACED THE DEFECTIVE CARDS.

[36] CATAWBA 2 DOCKET 50-414 LER 88-013
FEEDWATER ISOLATION CAUSED BY STEAM GENERATOR HIGH LEVEL DUE TO A VALVE FAILURE AND A PERSONNEL ERROR.
EVENT DATE: 031488 REPORT DATE: 041388 NSSS: WE TYPE: PWR
VENDOR: BORG-WARNER CORP.

(NSIC 209023) ON MARCH 14, 1988, AT 1158 HOURS, WHILE A FLUSH OF AUXILIARY FEEDWATER (CA) SYSTEM PIPING WAS IN PROGRESS, STEAM GENERATOR (S/G) 2A NARROW RANGE LEVEL REACHED THE HIGH HIGH LEVEL SETPOINT AND CAUSED A FEEDWATER ISOLATION. THE UNIT WAS IN MODE 4, HOT SHUTDOWN, AT THE TIME. THIS INCIDENT HAS BEEN ATTRIBUTED TO AN EQUIPMENT FAILURE. 2CA62A, CA PUMP 2A TO S/G 2A ISOLATION VALVE, DID NOT FULLY CLOSE AND CAUSED THE S/G TO BE OVERFILLED. NPRDS REPORTABILITY OF THIS FAILURE IS UNDER EVALUATION. THIS INCIDENT HAS ALSO BEEN ATTRIBUTED TO A PERSONNEL ERROR. WHILE PERFORMING A VALVE ALIGNMENT TO SUPPORT A CHEMISTRY ACTIVITY, A CONTROL ROOM SUPERVISOR MISTAKENLY CLOSED THE INCORRECT VALVE AND CAUSED THE ISOLATION OF ALL BLOWDOWN FROM THE S/GS. THIS ISOLATION HINDERED EFFORTS TO MAINTAIN S/G LEVEL DURING THE SUBSEQUENT SWELL OF THE S/G 2A INVENTORY. THE S/G BLOWDOWN SYSTEM WAS RETURNED TO SERVICE AND THE S/GS WERE RETURNED TO NORMAL LEVEL. THE FEEDWATER ISOLATION SIGNAL WAS RESET AND ALL VALVES AND PUMPS RETURNED TO THEIR PREVIOUS ALIGNMENT. A PROBLEM INVESTIGATION REPORT HAS BEEN INITIATED TO FURTHER EVALUATE THE CAUSE OF 2CA62A FAILURE TO CLOSE. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS EVENT.

[37] CLINTON 1 DOCKET 50-461 LER 88-011
UNKNOWN PLUGGED INLET LINES CAUSE DRYWELL AIR COOLERS' CONDENSATE FLOW MONITORING SYSTEM TO BE INOPERABLE RESULTING IN MISSED DRYWELL ATMOSPHERE GRAB SAMPLES.
EVENT DATE: 031188 REPORT DATE: 050688 NSSS: GE TYPE: BWR
VENDOR: ITT-BARTON

(NSIC 209243) ON APRIL 7, 1988, WITH THE PLANT IN MODE 4 (COLD SHUTDOWN), TECHNICIANS IDENTIFIED THAT TWO DRYWELL AIR COOLERS' CONDENSATE FLOW RATE TURBINE METERS WERE INOPERABLE BECAUSE THEIR INLET LINES WERE CLOGGED WITH DEBRIS. PRIOR TO THIS IDENTIFICATION ON APRIL 7, THESE METERS HAD BEEN INOPERABLE FOR AN INDETERMINATE PERIOD OF TIME. ON MARCH 11, 1988, WITH THE PLANT IN MODE 1 (POWER OPERATION), THE DRYWELL ATMOSPHERE GASEOUS RADIOACTIVITY MONITORING SYSTEM WAS REMOVED FROM SERVICE DUE TO THE GAS CHANNEL READING LOW DURING CALIBRATION. TECHNICAL SPECIFICATION 3.4.3.1 REQUIRES GRAB SAMPLES OF DRYWELL ATMOSPHERE TO BE TAKEN AND ANALYZED WHEN BOTH THE DRYWELL AIR COOLERS' CONDENSATE FLOW RATE AND DRYWELL ATMOSPHERE GASEOUS RADIOACTIVITY MONITORING SYSTEMS ARE INOPERABLE. SINCE THE TURBINE METERS WERE NOT KNOWN TO BE CLOGGED AND INOPERABLE UNTIL APRIL 7, GRAB SAMPLING WAS NOT INITIATED ON MARCH 11. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO FOREIGN MATERIAL ENTERING THE SYSTEM DURING PLANT CONSTRUCTION, OPERATION OR MAINTENANCE AND CAUSING THE TURBINE METERS TO FAIL. CORRECTIVE ACTION INVOLVED CLEANING THE INLET LINES AND THE TURBINE METERS AND REMOVING OBSTRUCTIONS. THE DRYWELL AIR COOLERS' CONDENSATE DRAIN LINE FLOW DETECTION SYSTEM IS BEING EVALUATED TO DETERMINE WHAT ENHANCEMENTS COULD BE MADE TO PREVENT RECURRENCE OF THIS EVENT.

[38] CLINTON 1 DOCKET 50-461 LER 88-007
FAILURE TO UNDERSTAND DIESEL GENERATOR (DG) VENTILATION OFF-NORMAL OPERATION RESULTS IN IMPAIRED FIRE PROTECTION AND TEMPERATURE CONTROL IN THE DG ROOMS.
EVENT DATE: 031488 REPORT DATE: 041388 NSSS: GE TYPE: BWR

(NSIC 208952) ON MARCH 14, 1988, SURVEILLANCE TESTING OF THE CARBON DIOXIDE (CO2) FIRE SUPPRESSION SYSTEM FOR THE EMERGENCY DIESEL GENERATOR (DG) ROOMS IDENTIFIED THAT THE VENTILATION FAN FOR THE DIVISION II DG FAILED TO TRIP DURING CO2 INITIATION INTO THE ROOM. REVIEW OF THE LOGIC DRAWINGS FOR THE FANS INDICATED THAT THE VENT FANS WILL NOT TRIP AND DAMPERS WILL NOT ISOLATE WHEN CO2 IS MANUALLY INITIATED, THE ROOM TEMPERATURE IS BELOW 70 DEGREES FAHRENHEIT (F), THE VENT FAN IS STARTED MANUALLY, AND THE DG IS NOT RUNNING. UNDER SUCH CONDITIONS, CO2 INITIATED INTO THE ROOM WOULD BE REMOVED BY THE VENTILATION SYSTEM SINCE THE FAN DOES NOT TRIP. ON MARCH 24, FURTHER REVIEW OF THE PROBLEM IDENTIFIED THAT THE DG VENT FAN WILL NOT TRIP AT THE 65 DEGREE FLOW ROOM TEMPERATURE LIMIT IF THE VENT FAN IS STARTED MANUALLY. AS A RESULT, THE ROOM MAY NOT BE MAINTAINED BETWEEN 65 DEGREES F AND 104 DEGREES F AS REQUIRED BY THE FINAL SAFETY ANALYSIS REPORT. THE CAUSE OF THIS EVENT IS ENGINEERING AND OPERATIONS PERSONNEL BEING UNAWARE THAT THESE CONDITIONS OCCUR UNDER OFF-NORMAL OPERATION AND AN INCORRECT ALARM SETPOINT. THE DG VENTILATION/CO SYSTEM WILL BE MODIFIED TO TRIP THE VENT FAN UNDER ANY CO2 INITIATION. THE DG LOW ROOM TEMPERATURE ALARMS WILL BE RAISED TO 65 DEGREES F.

[39] CLINTON 1 DOCKET 50-461 LER 88-009
IMPROPER USE OF AN IMPACT MATRIX RESULTS IN INSTRUMENT AIR ISOLATION DURING LOAD DRIVER CIRCUIT CARD TESTING DUE TO LIFTING WRONG LEAD WIRE TO PRECLUDE ISOLATION.
EVENT DATE: 040188 REPORT DATE: 050288 NSSS: GE TYPE: BWR

(NSIC 209242) ON APRIL 1, 1988, WITH THE PLANT IN MODE 4 (COLD SHUTDOWN), INSTRUMENT AIR SYSTEM CONTAINMENT ISOLATION VALVES 1IA006/7 AUTOMATICALLY CLOSED. THE EVENT OCCURRED WHEN CONTROL AND INSTRUMENTATION TECHNICIANS RE-INSTALLED A LOAD DRIVER CIRCUIT CARD DURING PERFORMANCE OF NUCLEAR SYSTEM PROTECTION SYSTEM UNTESTED ISLAND (UTI) LOAD DRIVER TESTING FOR THE DIVISION II RESIDUAL HEAT REMOVAL SYSTEM. THE PROCEDURE USED BY THE TECHNICIANS WAS A GENERIC TEST PROCEDURE FOR 91 LOAD DRIVER CIRCUIT CARDS AND DID NOT CONTAIN EITHER A MATRIX IDENTIFYING POSSIBLE PLANT IMPACT OF THE TESTING OR INSTRUCTIONS FOR LIFTING LEAD WIRES TO PRECLUDE AN ACTUATION. TECHNICIANS INCORRECTLY USED AN IMPACT MATRIX AS A WORK DOCUMENT TO LIFT A LEAD WIRE. THE IMPACT MATRIX DEVELOPED BY TECHNICIANS IDENTIFIED THE WRONG WIRE TO BE LIFTED TO PREVENT AN ACTUATION. PERSONNEL WILL BE TRAINED ON PROPER USE OF AN IMPACT MATRIX. ADDITIONAL CORRECTIVE ACTIONS

INCLUDE INCORPORATION OF ALL NON-TECHNICAL SPECIFICATION UTI REQUIREMENTS INTO THE PREVENTIVE MAINTENANCE PROGRAM AND DEVELOPMENT OF SPECIFIC MAINTENANCE TASKS FOR EACH UTI REQUIREMENT. THE TASKS WILL INCLUDE AN IMPACT MATRIX AND LEAD WIRE LIFTING INSTRUCTIONS.

[40] CLINTON 1 DOCKET 50-461 LER 88-010
OVERSIGHT BY UTILITY-LICENSED OPERATORS RESULTS IN FAILURE TO COMPLETE SHIFT CONTROL ROOM OPERATOR SURVEILLANCE LOG.
EVENT DATE: 040188 REPORT DATE: 042188 NSSS: GE TYPE: BWR

(NSIC 209027) ON APRIL 1, 1988, AT 1615 HOURS, WITH THE PLANT IN MODE 4 (COLD SHUTDOWN), CONTROL ROOM OPERATORS DISCOVERED THAT THE SHIFT CONTROL ROOM OPERATOR SURVEILLANCE LOG - MODE 4, 5 (REFUELING) DATA SHEET HAD NOT BEEN COMPLETED ON THE PREVIOUS SHIFT. TECHNICAL SPECIFICATIONS REQUIRE THIS SURVEILLANCE TO BE PERFORMED AT LEAST ONCE PER TWELVE HOURS WITH A PERMITTED OVERRUN OF TWENTY-FIVE PERCENT OF THE SURVEILLANCE INTERVAL. THE SURVEILLANCE LOG WAS LAST COMPLETED AT 0025 HOURS ON APRIL 1. THE OPERATORS IMMEDIATELY PERFORMED THE SURVEILLANCE LOG. THE SURVEILLANCE WAS SATISFACTORILY COMPLETED SEVENTY-FIVE MINUTES AFTER THE FIFTEEN HOUR LIMIT. THE CAUSE OF THE EVENT IS ATTRIBUTED TO AN OVERSIGHT BY THE CONTROL ROOM OPERATOR (CRO) AND THE LINE ASSISTANT SHIFT SUPERVISOR (LASS) WITH THE LEVEL OF CONTROL ROOM ACTIVITIES CONTRIBUTING TO THE CAUSE. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED COUNSELLING OF THE CRO AND LASS, A NIGHT ORDER ENTRY REINFORCING THE AUTHORITY OF THE CROS TO LIMIT CONTROL ROOM ACTIVITY, USE OF THE MAIN CONTROL ROOM TRACKING BOARD TO REMIND OPERATORS TO COMPLETE THIS SHIFT SURVEILLANCE, AND ASSIGNING THE RESPONSIBILITY FOR ENSURING COMPLETION OF THIS SURVEILLANCE TO THE "B" CONTROL ROOM OPERATOR.

[41] CONNECTICUT YANKEE DOCKET 50-213 LER 88-007
RESIDUAL HEAT REMOVAL SYSTEM NOT OPERATING IN MODE 5.
EVENT DATE: 031088 REPORT DATE: 040688 NSSS: WE TYPE: PWR

(NSIC 208841) ON 3/10/88 AT 1659 WHILE PERFORMING A PLANT HEATUP FROM MODE 5 (RCS PRESSURE = 309 PSIG, RCS TEMP = 108 DEGREES F) IN PREPARATION FOR A REACTOR COOLANT SYSTEM (RCS) HYDROSTATIC PRESSURE TEST, A CONTROL OPERATOR COMPLETING AN OPERATING PROCEDURE CHECKLIST DISCOVERED THAT THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM HAD BEEN SHUTDOWN FOR GREATER THAN ONE HOUR (1 HOUR, 22 MINUTES) RESULTING IN A VIOLATION OF THE PLANT TECH SPEC. FOLLOWING THE DISCOVERY OF THIS EVENT, A REVIEW OF THE OPERATIONS LOG REVEALED THAT A SIMILAR VIOLATION OCCURRED ON 3/5/88, DURING A PLANT HEATUP WHEN THE OPERATING RHR PUMP WAS SHUTDOWN FOR 1 HOUR, 2 MINUTES WHILE STILL IN MODE 5. AT THE TIME OF THE DISCOVERY, THE PLANT HAD ALREADY ENTERED MODE 4 WHERE OPERATION OF THE RHR SYSTEM IS NOT REQUIRED. THEREFORE, NO IMMEDIATE CORRECTIVE ACTION TO PLACE THE RHR SYSTEM BACK IN SERVICE WAS NECESSARY. THESE EVENTS ARE THE RESULT OF A FAILURE TO OBSERVE A SPECIFIC PRECAUTION IN THE PLANT OPERATING PROCEDURE FOR PLANT HEATUP. ALL LICENSED OPERATORS HAVE BEEN MADE AWARE OF THE VIOLATION AND CHANGES HAVE BEEN MADE TO THE APPROPRIATE OPERATING PROCEDURES TO PREVENT RECURRENCE. THIS EVENT IS BEING REPORTED UNDER 10CFR50.73(A)(2)(I)(B) SINCE IT INVOLVES A CONDITION PROHIBITED BY THE PLANT'S TECH SPEC.

[42] CONNECTICUT YANKEE DOCKET 50-213 LER 88-008
ZERO POWER REACTOR TRIP DUE TO SPURIOUS HIGH STARTUP RATE.
EVENT DATE: 031988 REPORT DATE: 041588 NSSS: WE TYPE: PWR

(NSIC 208990) AT APPROXIMATELY 0530 ON MARCH 19, 1988, WITH THE PLANT IN MODE 2 (REACTOR POWER AT 10E-11 AMPS), AN AUTOMATIC REACTOR TRIP OCCURRED DURING STARTUP PHYSICS TESTING. THE TRIP WAS DUE TO A SPURIOUS, HIGH STARTUP RATE TRIP SIGNAL CAUSED BY ELECTRICAL NOISE FROM A NEARBY ANNUNCIATOR CIRCUIT. ALL SYSTEMS RESPONDED AS EXPECTED. CORRECTIVE ACTION WAS TO PLACE A SUPPRESSION DIODE ACROSS

THE ANNUNCIATOR RELAY COIL. LONG TERM CORRECTIVE ACTION IS TO UPGRADE THE NUCLEAR INSTRUMENTATION SYSTEM DURING THE NEXT REFUELING OUTAGE. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(IV) SINCE IT INVOLVED AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM.

[43] CONNECTICUT YANKEE DOCKET 50-213 LER 88-009
ZERO POWER REACTOR TRIP DUE TO SPURIOUS HIGH STARTUP RATE.
EVENT DATE: 032288 REPORT DATE: 041588 NSSS: WE TYPE: PWR

(NSIC 208991) AT APPROXIMATELY 0625 ON MARCH 22, 1988, WITH THE PLANT IN MODE 2 (REACTOR POWER AT 10E-9 AMPS), AN AUTOMATIC REACTOR TRIP OCCURRED DURING STARTUP PHYSICS TESTING. THE TRIP WAS DUE TO A SPURIOUS, HIGH STARTUP RATE TRIP SIGNAL CAUSED BY ELECTRICAL NOISE. ALL SYSTEMS RESPONDED AS EXPECTED. THE SOURCE OF THE NOISE COULD NOT BE DETERMINED. CORRECTIVE ACTION WAS TO REPLACE A DETERIORATED PORTION OF NUCLEAR INSTRUMENTATION DETECTOR CABLE, THE DETECTOR AND A SOURCE RANGE RELAY. LONG TERM CORRECTIVE ACTION IS TO UPGRADE THE NUCLEAR INSTRUMENTATION SYSTEM DURING THE NEXT REFUELING OUTAGE IN 1989. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(IV) SINCE IT INVOLVED AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM.

[44] CONNECTICUT YANKEE DOCKET 50-213 LER 88-010
ERROR FOUND IN LARGE BREAK LOCA ANALYSIS.
EVENT DATE: 032488 REPORT DATE: 042188 NSSS: WE TYPE: PWR

(NSIC 208992) DURING LARGE BREAK LOCA (LB-LOCA) MODEL DEVELOPMENT, IT WAS DISCOVERED THAT A DISPARITY EXISTED BETWEEN THE LIMITING CONDITION LOW PRESSURE SAFETY INJECTION (LPSI) FLOW RATE IN THE ORIGINAL 1971 CALCULATION AND THAT CALCULATED BY CONNECTICUT YANKEE ATOMIC POWER COMPANY (CYAPCO). THE NUCLEAR REGULATORY COMMISSION (NRC) WAS NOTIFIED OF THIS CONDITION AT 1724 ON MARCH 24, 1988 IN ACCORDANCE WITH 10CFR50.72(B)(1)(II)(B). THE PLANT WAS IN MODE 2 AT THIS TIME. THE CAUSE OF THIS EVENT WAS AN APPARENT CALCULATIONAL ERROR. LB-LOCA REANALYSIS IS CURRENTLY SCHEDULED FOR NRC SUBMITTAL IN JUNE 1989 AS PART OF THE DESIGN BASIS UPGRADE PROGRAM. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(II)(B) SINCE A CONDITION EXISTED WHICH WAS OUTSIDE THE DESIGN BASIS OF THE PLANT.

[45] CONNECTICUT YANKEE DOCKET 50-213 LER 88-011
SURVEILLANCE FREQUENCY EXCEEDED FOR OVERPOWER TRIP TESTS.
EVENT DATE: 040888 REPORT DATE: 050688 NSSS: WE TYPE: PWR

(NSIC 209210) ON APRIL 8, 1988 AT 1745, WITH THE PLANT OPERATING IN MODE 1 (80% POWER), IT WAS DETERMINED THAT THE REQUIRED TEST INTERVAL FOR SURVEILLANCE PROCEDURE SUR 5.2-9, "NIS OVERPOWER SETPOINT CHECK" HAD BEEN EXCEEDED BY APPROXIMATELY ONE WEEK. THE HADDAM NECK TECHNICAL SPECIFICATIONS REQUIRE THAT THE NUCLEAR POWER TRIP FUNCTION BE CHECKED AT A FREQUENCY OF TWICE PER MONTH. THE ROOT CAUSE OF THIS EVENT WAS ATTRIBUTED TO PERSONNEL ERROR. THE TEST WAS SUBSEQUENTLY PERFORMED ON APRIL 8, 1988 AT 1800 WITH SATISFACTORY RESULTS. A DEPARTMENT PROCEDURE WILL BE ISSUED BY JUNE 30, 1988 THAT REINFORCES INDIVIDUAL RESPONSIBILITIES ASSOCIATED WITH THE PERFORMANCE OF SURVEILLANCES. THIS EVENT IS BEING REPORTED UNDER 10CFR50.73(A)(2)(I)(B) SINCE IT INVOLVES A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATION.

[46] COOK 1 DOCKET 50-315 LER 88-002
ICE BUILDUP IN ICE CONDENSER FLOW PASSAGES DUE TO SUBLIMATION.
EVENT DATE: 032688 REPORT DATE: 042588 NSSS: WE TYPE: PWR

(NSIC 209120) ON MARCH 26, 1988, WITH UNIT 1 IN MODE 3 (HOT STANDBY), FLOW

PASSAGE INSPECTIONS OF THE ICE CONDENSER REVEALED FROST AND ICE BUILDUP ON THE LATTICE FRAMES OF GREATER THAN 3/8 INCH IN A TOTAL OF TWELVE FLOW PASSAGES IN FOUR OF THE TWENTY-FOUR ICE CONDENSER BAYS. THE FLOW PASSAGES WERE DECLARED INOPERABLE AT 1530 HOURS. TECHNICAL SPECIFICATION (T/S) 4.6.5.1.B.3 LIMITS FROST OR ICE BUILDUP IN FLOW PASSAGES TO A NOMINAL THICKNESS OF 3/8 INCH. ACCORDING TO THIS T/S, BUILDUP EXCEEDING THIS LIMIT IN TWO OR MORE FLOW PASSAGES PER BAY IS EVIDENCE OF ABNORMAL DEGRADATION. THOUGH OUR EVALUATION HAS CONCLUDED THAT THE DEGRADATION IS NOT SERIOUS, WE BELIEVE ISSUANCE OF THIS VOLUNTARY LER IS APPROPRIATE SINCE SOME DEGRADATION HAS BEEN IDENTIFIED. ACTIONS TAKEN TO CORRECT THE ABNORMAL DEGRADATION INCLUDED MANUAL CLEANING OF THE FLOW PASSAGES AND AN INTERNAL INVESTIGATION OF THE EVENT. THE FLOW PASSAGES WERE DECLARED OPERABLE ON MARCH 27, 1988, AT 0600 HOURS. THE RESULTS OF T/S SURVEILLANCES REGARDING FROST AND ICE THAT FORMS IN THE FLOW PASSAGES IS BEING MONITORED TO ENSURE THAT ANY ADVERSE TRENDS IN THE AMOUNT OF ICE AND FROST BUILDUP BETWEEN SURVEILLANCES WILL BE IDENTIFIED. THE IMPACT OF FROST AND ICE BUILDUP IN THE FLOW PASSAGES IS ALSO BEING STUDIED IN CONJUNCTION WITH THE OTHER UTILITIES WITH ICE CONDENSER CONTAINMENTS.

[47] COOK 2 DOCKET 50-316 LER 88-003
REPETITIVE VIOLATION OF ESF INSTRUMENTATION LIMITING CONDITIONS FOR OPERATION TOLERANCES DUE TO HIGHLY RESTRICTIVE ALLOWABLE VALUES.
EVENT DATE: 031188 REPORT DATE: 041188 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 1 (PWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 209007) ON MARCH 11, 1988 AN EQUIPMENT TREND INVESTIGATION WAS BEING PERFORMED ON 4KV BUS LOSS OF VOLTAGE RELAYS AND THE 4KV BUS DEGRADED VOLTAGE RELAYS (EIIS/EK-27). THE 'AS FOUND' CONDITION OF THESE RELAYS DURING PAST CALIBRATION CHECKS HAS GENERALLY BEEN FOUND TO BE BEYOND THE TECHNICAL SPECIFICATION (T.S.) ALLOWABLE VALUES. EACH RELAY WAS ADJUSTED TO WITHIN ALLOWABLE VALUES AT THE TIME IT WAS DISCOVERED OUT OF SPECIFICATION. ALL RELAYS WERE FUNCTIONAL AND WOULD HAVE PERFORMED THE ESF FUNCTION, ALTHOUGH AT A SLIGHTLY DIFFERENT VOLTAGE THAN SPECIFIED IN T.S. A SURVEY OF OTHER PLANTS AND RELAY MANUFACTURERS INDICATE THAT THE RELAYS ARE FUNCTIONING CONSISTENT WITH THE MANUFACTURER'S SPECIFICATIONS AND THAT THE T.S. ALLOWABLE VALUES ARE TOO RESTRICTIVE FOR THIS TYPE OF FUNCTION AND THE RELAY BEING USED. THE LICENSEE HAS NOT BEEN ABLE TO LOCATE REPLACEMENT RELAYS WITH THE REQUIRED SETPOINT SENSITIVITY IN THE INDUSTRY. AN ENGINEERING REVIEW IS IN PROGRESS TO EVALUATE THE MAXIMUM ALLOWABLE VOLTAGE TOLERANCES FOR THE TWO APPLICATIONS. A TECHNICAL SPECIFICATION CHANGE REQUEST WILL BE SUBMITTED AS SOON AS THIS EVALUATION HAS BEEN COMPLETED. IN THE INTERIM, WE WILL INCREASE THE CALIBRATION FREQUENCY WILL BE INCREASED FROM EVERY EIGHTEEN MONTHS TO MONTHLY.

[48] COOPER DOCKET 50-298 LER 87-020
APPARENT NONCOMPLIANCE WITH SURVEILLANCE TESTING FREQUENCY REQUIREMENTS.
EVENT DATE: 082787 REPORT DATE: 092587 NSSS: GE TYPE: BWR
VENDOR: COOPER ENERGY SERVICES

(NSIC 209202) DURING A RECENT NRC INSPECTION, AN APPARENT NONCOMPLIANCE WITH THE DIESEL GENERATOR SURVEILLANCE TESTING FREQUENCY ASSOCIATED WITH THE ANNUAL INSPECTIONS PERFORMED IN 1984 WAS IDENTIFIED. SPECIFICALLY, THE INSPECTIONS CONDUCTED IN 1984 WERE NOT ACCOMPLISHED WITHIN 15 MONTHS (ANNUAL PLUS AN ALLOWABLE EXTENSION OF 25 PERCENT) OF THEIR PERFORMANCE IN LATE MAY - EARLY JUNE 1983. IN 1984, THE PLANT WAS IN OPERATION DURING THE MAY - JUNE TIME FRAME, AND EXCEPT FOR A TWO DAY SHUTDOWN IN AUGUST, OPERATION CONTINUED UNTIL SEPTEMBER 15, THE DATE FOR THE START OF THE REFUELING/PIPE REPLACEMENT OUTAGE. THE CAUSE OF THIS APPARENT NONCOMPLIANCE WITH TECH SPEC SURVEILLANCE INTERVAL FOR THE DIESEL GENERATOR ANNUAL INSPECTIONS (PARAGRAPH 4.9.A.2.F, OF THE CNS TECH SPECS) WAS DUE TO THE UNDERSTANDING OF THE TERM "ANNUAL" WHICH EXISTED AT THAT TIME. "ANNUAL"

WAS UNDERSTOOD TO MEAN ONCE PER YEAR, THEREFORE, THE REQUIRED INSPECTIONS WERE TO BE ACCOMPLISHED AT SOME TIME BETWEEN JANUARY 1 AND DECEMBER 31 OF EACH YEAR. AS A RESULT OF DISCUSSIONS CONDUCTED SUBSEQUENT TO THIS EVENT REGARDING THE DEFINITION OF SURVEILLANCE TESTING INTERVALS, PROCEDURAL CHANGES WERE MADE INCORPORATING STANDARD TECH SPEC DEFINITIONS. IN ADDITION, A CHANGE TO THE TECH SPEC WAS INITIATED TO LENGTHEN THE REQUIRED DIESEL GENERATOR INSPECTION INTERVAL TO 18 MONTHS.

[49]   COOPER   DOCKET 50-298   LER 88-004
ACTUATION OF ESF GROUP ISOLATIONS SUBSEQUENT TO A PLANNED MANUAL SCRAM DUE TO MOMENTARY LOW REACTOR VESSEL WATER LEVEL.
EVENT DATE: 030588   REPORT DATE: 040488   NSSS: GE   TYPE: BWR

(NSIC 208933) ON MARCH 5, 1938, AT 4:31 A.M. AN AUTOMATIC ACTUATION OF GROUP ISOLATIONS 2, 3, AND 6 (PRIMARY CONTAINMENT, REACTOR WATER CLEANUP AND SECONDARY CONTAINMENT INCLUDING INITIATION OF THE STANDBY GAS TREATMENT SYSTEM) OCCURRED DUE TO A MOMENTARY "SHRINK" IN REACTOR VESSEL WATER LEVEL FOLLOWING A PLANNED MANUAL SCRAM. A PLANT SHUTDOWN, IN PREPARATION FOR THE 1988 REFUELING OUTAGE, WAS IN PROGRESS. JUST PRIOR TO THE SCRAM, REACTOR THERMAL POWER WAS APPROXIMATELY 20 PERCENT. THE WATER LEVEL TRANSIENT WHICH OCCURRED IS ONE WHICH CAN TYPICALLY BE EXPECTED UPON A SCRAM (EITHER AUTO OR MANUAL) FROM POWER. PRIOR TO MANUALLY SCRAMMING THE REACTOR, WATER LEVEL HAD BEEN RAISED 10 INCHES ABOVE THE NORMAL LEVEL TO 45 INCHES. THIS ACTION WAS TAKEN, IN ANTICIPATION OF THE EXPECTED MOMENTARY "SHRINK", SO AS TO AVOID THESE UNNECESSARY ACTUATIONS. HOWEVER, THE ENSUING WATER LEVEL "SHRINK" FROM THE SCRAM WAS OF SUFFICIENT MAGNITUDE (33 1/2 INCHES) THAT THE LOW LEVEL REACTOR WATER LEVEL SENSORS WERE TRIPPED. REACTOR VESSEL WATER LEVEL WAS IMMEDIATELY RECOVERED THROUGH OPERATION OF THE NORMAL FEEDWATER SYSTEM AND PLANT SYSTEMS WERE RESTORED TO THEIR NORMAL CONDITION.

[50]   COOPER   DOCKET 50-298   LER 88-005
INADVERTENT ISOLATION OF THE REACTOR WATER CLEANUP SYSTEM IN REDUCED PRESSURE DURING A PLANT COOLDOWN DUE TO LOW PUMP NET POSITIVE SUCTION HEAD.
EVENT DATE: 030588   REPORT DATE: 040488   NSSS: GE   TYPE: BWR

(NSIC 209012) ON 3/5/88, DURING A PLANT COOLDOWN, THE REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATED (A GROUP 3 ISOLATION) WHEN THE RWCU PUMP LOST ITS REQUIRED NET POSITIVE SUCTION HEAD (NPSH). THE POTENTIAL FOR LOSS OF PUMP SUCTION DUE TO CAVITATION DURING A PLANT COOLDOWIN IS KNOWN AND ADDRESSED IN APPROPRIATE PLANT PROCEDURES. DURING THIS EVENT, PRESSURE/TEMPERATURE CONDITIONS REACHED THE POINT WHERE MARGINAL NPSH CONDITIONS EXISTED AT THE RWCU PUMP SUCTION. HOWEVER, BEFORE ACTION COULD BE IMPLEMENTED TO MANUALLY REMOVE THE RWCU SYSTEM FROM SERVICE, PUMP CAVITATION OCCURRED AND RESULTED IN A GROUP 3 ISOLATION. AT THE TIME OF THIS EVENT, THE PLANT WAS SHUT DOWN. A PLANT COOLDOWN WAS IN PROGRESS IN PREPRATION FOR THE 1988 REFUELING OUTAGE WITH REACTOR PRESSURE AT APPROX. 225 PSIG. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO SYSTEM DESIGN IN THAT A VERY SMALL MARGIN EXISTS FOR RWCU PUMP NPSH DURING MODES OF OPERATION WHERE FEEDWATER IS NOT AVAILABLE TO ENHANCE RECIRCULATION SUCTION LINE SUBCOOLING. AS A RESULT OF PRIOR SIMILAR EVENTS, PROCEDURAL GUIDANCE REGARDING PUMP OPERATION DURING COOLDOWN HAD BEEN IMPLEMENTED. FURTHER CORRECTIVE ACTION TO BE TAKEN INCLUDES A REVIEW AND, IF APPROPRIATE, REVISION OF THE PERTINENT OPERATING PROCEDURES TO PROVIDE FURTHER GUIDIANCE OR CLARIFICATION REGARDING SYSTEM OPERATION UNDER PLANT CONDITIONS.

[51]   COOPER   DOCKET 50-298   LER 88-006
UNPLANNED AUTOMATIC ACTUATION OF DIESEL GENERATOR STARTING LOGIC DUE TO 4160V AC BREAKER ACTUATION DURING RELAY COVER REPLACEMENT.
EVENT DATE: 031688   REPORT DATE: 041288   NSSS: GE   TYPE: BWR

(NSIC 208925) ON MARCH 16, 1988, AT 5:05 P.M., AN UNPLANNED AUTOMATIC ACTUATION OF DIESEL GENERATOR NO. 1 STARTING LOGIC OCCURRED WHEN THE NORMAL POWER SUPPLY BREAKER (1AF) TO 4160V CRITICAL BUS F UNEXPECTEDLY TRIPPED OPEN. AT THE TIME OF THE EVENT, A COVER FOR AN OVERCURRENT RELAY FOR BREAKER 1AF WAS BEING REPLACED. THIS RELAY HAD JUST BEEN INSTALLED AFTER BEING REMOVED FROM THE BREAKER CUBICLE FOR CALIBRATION AND TESTING. IT IS SUSPECTED THAT DURING COVER INSTALLATION, THE TARGET RESET LEVER WHICH IS ATTACHED TO THE COVER, INADVERTENTLY CONTACTED THE OVERCURRENT RELAY SEAL-IN CONTACTS, CAUSING THE RELAY TO ACTUATE. THIS RESULTED IN A TRIP OF BREAKER 1AF WHICH, IN TURN, INITIATED CLOSURE OF BREAKER 1AS, THE EMERGENCY TRANSFORMER FEEDER, AND AUTOMATIC STARTUP OF DIESEL GENERATOR NO. 1. AT THE TIME OF THIS EVENT, THE PLANT WAS SHUTDOWN FOR THE 1988 REFUELING OUTAGE, WITH REACTOR COOLANT TEMPERATURE AT 75F AND THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM IN OPERATION IN THE SHUTDOWN COOLING MODE. THIS SITUATION WAS IMMEDIATELY INVESTIGATED BY ELECTRICAL MAINTENANCE SUPERVISION. THE RELAY WAS REMOVED FROM THE CUBICLE, INSPECTED, SATISFACTORILY RETESTED, RE-INSTALLED AND ITS COVER REPLACED. BREAKER 1AF WAS THEN CLOSED, RE-ENERGIZING 4160V CRITICAL BUS F FROM ITS NORMAL SOURCE.

[52] COOPER DOCKET 50-298 LER 88-007
UNPLANNED ACTUATIONS OF THE REACTOR PROTECTION SYSTEM AND GROUP ISOLATIONS 2 AND 6 DURING DESIGN CHANGE ACTIVITIES DUE TO INCORRECT DESIGN CHANGE INSTRUCTIONS.
EVENT DATE: 032588 REPORT DATE: 042588 NSSS: GE TYPE: BWR

(NSIC 209131) ON 3/25/88, WHILE SHUTDOWN FOR THE 1988 REFUELING OUTAGE, TWO UNPLANNED ACTUATIONS OF THE REACTOR PROTECTION SYSTEM (RPS) OCCURRED. THE FIRST OCCURRED AT 5:58 P.M., WHEN AN INCORRECT LEAD WAS LIFTED DURING DESIGN CHANGE ACTIVITIES. THE SECOND OCCURRED AT 6:46 P.M. WHEN POWER FOR THE B RPS WAS BEING TRANSFERRED FROM ITS NORMAL TO THE ALTERNATE SOURCE TO SUPPORT ANOTHER DESIGN CHANGE ACTIVITY. DURING THE TRANSFER, THE B RPS WAS MOMENTARILY DE-ENERGIZED. DUE TO THE FACT THAT THE SCRAM DISCHARGE VOLUME (SDV) HIGH LEVEL TRIP SWITCHES FROM THE FIRST ACTUATION WERE STILL TRIPPED, THE SECOND TRIP OCCURRED. THE FIRST RPS ACTUATION WAS ACCOMPANIED WITH ACTUATIONS OF GROUPS 2 AND 6 ISOLATIONS (PRIMARY AND SECONDARY CONTAINMENT, INCLUDING INITIATION OF THE STANDBY GAS TREATMENT (SGT) SYSTEM). AS ANTICIPATED PRIOR TO TRANSFERRING THE B RPS POWER SUPPLY, GROUPS 3 AND 7 ISOLATIONS (REACTOR WATER CLEANUP SYSTEM AND REACTOR WATER SAMPLING) OCCURRED COINCIDENT WITH THE MOMENTARY RPS DE-ENERGIZATION. THE CAUSE OF THESE EVENTS WAS DUE TO USE OF INCORRECT FIELD DRAWINGS DURING DEVELOPMENT OF WORK INSTRUCTIONS FOR A DESIGN CHANGE ACTIVITY. THESE UNPLANNED RPS AND ENGINEERED SAFETY FEATURE (ESF) GROUP ISOLATION ACTUATIONS POSED NO SAFETY SIGNIFICANCE TO ON-GOING PLANT ACTIVITIES. THE SYSTEMS, WHEN ACTUATED, PERFORMED AS DESIGNED.

[53] COOPER DOCKET 50-298 LER 88-008
FAILURE OF RHR INBOARD INJECTION VALVES TO CLOSE DURING SURVEILLANCE TESTING.
EVENT DATE: 040688 REPORT DATE: 050688 NSSS: GE TYPE: BWR

(NSIC 209356) ON APRIL 6, 1988, WHILE CONDUCTING SURVEILLANCE TESTING, THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM PCIS ISOLATION FUNCTIONAL TEST WAS NOT SUCCESSFULLY COMPLETED. LOGIC RELAYS 10A-K63A AND B WHICH INITIATE CLOSURE OF RHR INJECTION VALVES RHR-MOV-MO25A AND B DID NOT ACTUATE AT THE APPROPRIATE STEPS IN THE SURVEILLANCE TEST PROCEDURE. WHEN THIS APPARENT DEFICIENCY WAS IDENTIFIED, THE 1988 REFUELING OUTAGE WAS IN PROGRESS WITH THE REACTOR VESSEL HEAD REMOVED, THE REFUELING CAVITY FLOODED, RHR IN OPERATION IN THE SHUTDOWN COOLING MODE, AND PRIMARY CONTAINMENT INTEGRITY NOT REQUIRED. AS PROVIDED FOR IN PARAGRAPH 3.2.A OF THE CNS TECH SPECS, WHEN PRIMARY CONTAINMENT INTEGRITY IS NOT REQUIRED, INSTRUMENTATION DESIGNED TO INITIATE PRIMARY CONTAINMENT ISOLATION SYSTEM FUNCTIONS IS NOT REQUIRED TO MEET THE SPECIFIED LIMITING CONDITIONS FOR OPERATION (LCO). DUE TO PLANT CONDITIONS, INVESTIGATION OF THIS TEST DEVIATION HAS NOT YET BEEN COMPLETED. THIS ACTION, HOWEVER, WILL BE COMPLETED PRIOR TO

ESTABLISHING PLANT CONDITIONS WHERE THE REQUIREMENTS OF PARAGRAPH 3.2.A. OF THE CNS TECH SPECS MUST BE MET, AND APPROPRIATE CORRECTIVE ACTION TAKEN TO CORRECT THE CAUSE. THE CAUSE OF THE EVENT AND CORRECTIVE ACTION TAKEN WILL BE PROVIDED IN A SUPPLEMENTARY REPORT.

[54] COOPER DOCKET 50-298 LER 88-009
SETPOINT VARIANCE AND OPERABILITY CONCERNS ASSOCIATED WITH SAFETY RELIEF VALVES AND SAFETY VALVES DISCOVERED DURING SURVEILLANCE TESTING.
EVENT DATE: 040788 REPORT DATE: 050688 NSSS: GE TYPE: BWR
VENDOR: DRESSER INDUSTRIES, INC.
TARGET ROCK CORP.

(NSIC 209292) DURING PERFORMANCE OF SAFETY RELIEF VALVE (SRV) AND SAFETY VALVE (SV) TESTING REQUIRED BY THE CNS TECH SPECS, FOUR (4) OF THE EIGHT (8) INSTALLED SRVS AND TWO (2) OF THE THREE (3) INSTALLED SVS WERE SENT TO WYLE LABORATORIES IN HUNTSVILLE, ALABAMA, TO BE BENCHED CHECKED. THREE PROBLEMS WERE DISCOVERED AS FOLLOWS: 1) ONE SRV WHICH SHOULD HAVE LIFTED AT 1100 PSIG +/= 11 PSI ACTUALLY LIFTED AT 1220 PSIG, 2) ONE SRV SET TO ACTUATE AT 1080 PSIG +/- 11 PSI COULD NOT BE TESTED AS RECEIVED DUE TO A PILOT ASSEMBLY TO MAIN BODY FLANGE LEAK. THE PILOT ASSUMBLY WAS REMOVED AND SEPARATELY TESTED WITH SATISFACTORY RESULTS, AND 3) ONE SV WHICH SHOULD HAVE LIFTED AT 1240 PSIG +/- 13 PSI ACTUALLY LIFTED AT 1268 PSIG. BENCH CHECKS CONDUCTED ON THE OTHER THREE VALVES WERE SATISFACTORY. THE VALVES WERE INSPECTED AND REFURBISHED IN ACCORDANCE WITH STANDARD PROCEDURES EMPLOYED BY WYLE LABORATORIES AND THE RESPECTIVE VALVE MANUFACTURERS AND SUBSEQUENTLY TESTED SATISFACTORILY. THE INSPECTIONS CONDUCTED PROVED TO BE INCONCLUSIVE FROM THE PERSPECTIVE OF IDENTIFYING THE CAUSE OF THE OBSERVED SETPOINT VARIANCE. THESE DISCREPANCIES WERE EVALUATED BY GENERAL ELECTRIC AND DETERMINED TO HAVE NO SAFETY SIGNIFICANCE.

[55] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-013 REV 01
UPDATE ON VOLUNTARY ENTRY INTO TECH SPEC 3.0.3 FOR TROUBLESHOOTING AND REPAIR OF EF I&C SYSTEM.
EVENT DATE: 071687 REPORT DATE: 033188 NSSS: BW TYPE: PWR
VENDOR: VITRO LABORATORIES

(NSIC 208981) ON JULY 12, 1987, CRYSTAL RIVER UNIT 3 WAS OPERATING AT 85% RATED THERMAL POWER, GENERATING 530 MWE. STEAM LINE AND FEEDWATER ISOLATION FUNCTIONAL TESTING WAS IN PROGRESS. DURING THE SURVEILLANCE, INDICATION OF A PARTIAL TRIP IN THE EMERGENCY FEEDWATER INITIATION AND CONTROL (EFIC) SYSTEM WAS RECEIVED. ON THREE SEPARATE OCCASIONS (JULY 16, 17, AND 18, 1987) THE OUTPUT BREAKERS OF BOTH "A" AND "B" EFIC CHANNELS WERE OPENED, DISABLING THE AUTOMATIC FUNCTION OF THE SYSTEM. THIS WAS DONE TO PRECLUDE THE POSSIBILITY OF SPURIOUS ACTUATIONS WHILE REPAIRS WERE BEING MADE. OPERATION WITH BOTH EFIC AUTOMATIC ACTUATION CHANNELS DISABLED IS A CONDITION PROHIBITED BY THE CR-3 TECH SPECS. THEREFORE, THESE ACTIONS AMOUNTED TO VOLUNTARY ENTRY INTO THE REQUIREMENTS OF SPECIFICATION 3.0.3. THE PARTIAL TRIP WAS CAUSED BY A DEGRADED LIGHT EMITTING DIODE (LED) IN THE OPTICAL TRANSMITTER OF THE CHANNEL C EMERGENCY FEEDWATER INITIATE CIRCUIT. THE LED WAS REPLACED.

[56] CRYSTAL RIVER 3 DOCKET 50-302 LER 88-008
UNKNOWN CAUSE OF INSTRUMENT DRIFT LEADS TO LOSS OF REQUIRED SAFETY FUNCTION.
EVENT DATE: 010488 REPORT DATE: 041488 NSSS: BW TYPE: PWR
VENDOR: ROSEMOUNT, INC.

(NSIC 208962) ON JANUARY 4, 1988 CRYSTAL RIVER UNIT 3 WAS MAINTAINING MODE 5 (COLD SHUTDOWN) CONDITIONS SUBSEQUENT TO A REFUELING OUTAGE. A SURVEILLANCE PROCEDURE FOR CALIBRATION OF LEVEL TRANSMITTERS USED FOR EMERGENCY FEEDWATER INITIATION AND CONTROL (EFIC) ON THE ONCE THROUGH STEAM GENERATORS (OTSG) HAD

BEEN PERFORMED. FROM THE SURVEILLANCE IT WAS DETERMINED THAT THREE OUT OF FOUR OF THE LOW LEVEL TRANSMITTERS ON THE 'B' OTSG WERE OUT OF TOLERANCE SUFFICIENTLY TO HAVE PREVENTED THE ACTUATION OF EMERGENCY FEEDWATER TO THE 'B' OTSG. AT ITS REQUIRED LOW LEVEL SETPOINT AND IN FACT WOULD NOT HAVE ACTUATED EMERGENCY FEEDWATER TO THE 'B' OTSG. ALL EFIC LEVEL INSTRUMENTS WERE RECALIBRATED TO WITHIN SPECIFIED LIMITS PRIOR TO THE PLANT ASCENDING INTO MODE 3 (HOT STANDBY) IN WHICH EFIC OPERABILITY IS REQUIRED. THE CAUSE OF THE INSTRUMENT ERROR IS UNKNOWN. WHILE THREE OF THE FOUR TRANSMITTERS ON THE 'B' OTSG WERE OUT OF TOLERANCE, ALL FOUR OF THE TRANSMITTERS ON THE 'A' OTSG WERE WITHIN THE ALLOWED INSTRUMENT TOLERANCES. THEREFORE, THE 'A' OTSG WAS AVAILABLE FOR DECAY HEAT REMOVAL IN THE EVENT OF LOSS OF LEVEL IN THE 'A' OTSG. HIGH PRESSURE INJECTION/POWER OPERATED RELIEF VALVE COOLING WAS AVAILABLE FOR DECAY HEAT REMOVAL IF NEEDED. NUCLEAR ENGINEERING WILL EVALUATE THE INSTRUMENT ERROR TO DETERMINE THE CAUSE. CORRECTIVE ACTION WILL BE CONSIDERED BASED ON THE EVALUATION.

[57] CRYSTAL RIVER 3 DOCKET 50-302 LER 88-007
UNKNOWN CAUSE RESULTS IN REACTOR BUILDING SPRAY PUMP OPERATING BELOW ITS DESIGN FLOW.
EVENT DATE: 012788 REPORT DATE: 040688 NSSS: BW TYPE: PWR

(NSIC 208870) AS A RESULT OF THE REEVALUATION OF THE EMERGENCY DIESEL GENERATOR LOADING ISSUE, ADDITIONAL PUMP FLOW TESTING WAS CONDUCTED TO DETERMINE THE ACTUAL KW LOAD OF MAJOR ENGINEERED SAFEGUARDS PUMPS. IN REVIEWING THE ABOVE TEST DATA A DISCREPANCY WAS NOTED BETWEEN THE MANUFACTURER'S PUMP CURVE (HEAD-FLOW) AND THE TEST DATA. A NONCONFORMING OPERATIONS REPORT (NCOR 88-28) WAS GENERATED TO DOCUMENT THE APPARENT LOW DISCHARGE HEAD OF THE 'A' TRAIN BUILDING SPRAY PUMP (BSP-1A). A SUBSEQUENT LETTER TO THE NRC STATED THE APPLICABLE SURVEILLANCE PROCEDURE HAD BEEN RE-RUN AND THE RESULTS CONFIRMED THE PREVIOUS TEST RESULTS (THE DISCREPANCY REMAINED). ENGINEERING CONFIRMED BSP-1A IS CAPABLE OF OPERATING AT A REDUCED CAPACITY OF 1460 GPM WHICH IS BELOW ITS ORIGINAL DESIGN REQUIREMENT OF 1500 GPM; HOWEVER, RECENT CALCULATIONS HAVE SHOWN A MINIMUM FLOW OF 1200 GPM IS ADEQUATE. THE CAUSE OF THE CONDITION IS NOT KNOWN AT THIS TIME. AN INVESTIGATION AND INSPECTION TO DETERMINE THE CAUSE WILL BE PERFORMED DURING THE NEXT REFUELING OUTAGE AND ADDITIONAL CORRECTIVE ACTION WILL BE TAKEN AS WARRANTED.

[58] CRYSTAL RIVER 3 DOCKET 50-302 LER 88-009
VIOLATION OF APPENDIX R III.O DUE TO INSUFFICIENT RESERVE VOLUME IN RC PUMP LUBE OIL COLLECTION SYSTEM CAUSED BY LACK OF AWARENESS BEHIND DESIGN BASIS.
EVENT DATE: 030388 REPORT DATE: 041488 NSSS: BW TYPE: PWR

(NSIC 208959) ON JANUARY 10, 1988, CRYSTAL RIVER UNIT 3 WAS IN MODE 1 (POWER OPERATION) AND ESCALATING IN POWER FOLLOWING A REFUELING OUTAGE. THE LEVEL IN THE LUBE OIL COLLECTION TANK FOR THE REACTOR COOLANT PUMPS WAS SLOWLY TRENDING UP WITH NO APPARENT LOSS OF OIL. THIS WAS INDICATIVE OF A SMALL COOLING WATER LEAK (LESS THAN 0.01 GALLONS PER MINUTE). THE LEVEL TRENDED UP UNTIL, ON JANUARY 13, 1988, IT REACHED THE PROCEDURAL LIMIT OF 20 PERCENT. THE LEVEL WAS THEN PERIODICALLY PUMPED DOWN TO MAINTAIN COMPLIANCE WITH APPENDIX R III.O. ON MARCH 3, 1988, A REFINED CALCULATION BETTER DEFINING THE INDICATED LEVEL LIMIT WAS PUBLISHED. THE NEW LIMIT WAS 15.5 PERCENT. DURING THE PERIOD BETWEEN JANUARY 10 AND MARCH 2, 1988, THE PLANT WAS PERIODICALLY OPERATING OUTSIDE THE DESIGN BASIS AS COMMITTED FOR COMPLIANCE TO 10CFR50 APPENDIX R. THE CAUSE OF THE VIOLATION WAS A PROCEDURAL DEFICIENCY. IT WAS LATER DISCOVERED ADDITIONAL VIOLATIONS OCCURRED PRIOR TO DECEMBER 1987, CAUSED BY LACK OF AWARENESS BY PLANT PERSONNEL OF THE DESIGN BASIS BEHIND THE LUBE OIL COLLECTION TANK RESERVE VOLUME. THE PROPER LEVEL IS BEING MAINTAINED BY DRAINING THE TANK PRIOR TO REACHING THE VOLUME LIMIT.

[59] CRYSTAL RIVER 3 DOCKET 50-302 LER 88-010
RELEASE MONITOR TRIP SETPOINT ABOVE TECH SPEC LIMIT DUE TO INADEQUATE PROCEDURE.
EVENT DATE: 032588 REPORT DATE: 042588 NSSS: BW TYPE: PWR

(NSIC 209253) ON MARCH 25, 1988 WHILE OPERATING AT 100% RTP AND PRODUCING 834 MWE, THE AUXILIARY BUILDING AND FUEL HANDLING AREA EXHAUST DUCT MONITOR, RM-A2, WAS DISCOVERED SET TO TRIP AT A VALUE THAT COULD HAVE ALLOWED THE DOSE RATE AT THE SITE BOUNDARY TO EXCEED THE TECH SPEC LIMIT. THE CAUSE OF THE EVENT WAS A PROCEDURAL DEFICIENCY. THE OFFSITE DOSE CALCULATION MANUAL (ODCM) AND THE PROCEDURE FOR CALCULATING THE TRIP SETPOINTS FOR THE RELEASE MONITORS DID NOT PROVIDE A SUFFICIENT METHODOLOGY FOR CASES WHERE THE NET COUNT RATE (OBSERVED COUNT RATE MINUS BACKGROUND) WAS BELOW THE LOWER LIMIT OF DETECTABILITY FOR THE MONITOR. THE RELEASE MONITOR WAS DECLARED INOPERABLE, AND SAMPLING OF THE EFFLUENT RELEASE PATHWAY EVERY TWELVE HOURS WAS INITIATED, AS REQUIRED BY TECH SPECS. THE ODCM AND THE PROCEDURES FOR DETERMINING THE RELEASE MONITOR TRIP SETPOINTS WERE REVISED. THE GASEOUS AND RADIOACTIVE MATERIALS RELEASE HISTORY WAS RESEARCHED, IT WAS DETERMINED THAT NO LIMITS HAD BEEN EXCEEDED.

[60] DAVIS-BESSE 1 DOCKET 50-346 LER 88-004 REV 01
UPDATE ON INTENTIONALLY MISSED HOURLY FIRE WATCH DUE TO RADIOLOGICAL CONTROL AREA EVACUATION.
EVENT DATE: 012088 REPORT DATE: 041488 NSSS: BW TYPE: PWR

(NSIC 209014) ON JANUARY 20, 1988 AT 1900 HOURS, THE HOURLY FIRE PATROL WAS NOT PERFORMED AS REQUIRED BY THE ACTION STATEMENTS OF TECHNICAL SPECIFICATIONS 3.3.3.8, AND 3.7.10. THE HOURLY FIRE PATROL WAS MISSED BECAUSE OF AN INADVERTENT GASEOUS RELEASE INSIDE THE AUXILIARY BUILDING. AS A RESULT OF THE RELEASE, THE RADIOLOGICAL CONTROL AREA (RCA) WAS EVACUATED AS A PRECAUTIONARY MEASURE FOR ALARA CONCERNS. THE GASEOUS RELEASE RESULTED IN RADIATION DOSE 100 TIMES LESS THAN THE TECHNICAL SPECIFICATION REPORTING LIMITS. AT 100 HOURS, THE HOURLY FIRE PATROLS RESUMED.

[61] DAVIS-BESSE 1 DOCKET 50-346 LER 88-005 REV 01
UPDATE ON INOPERABLE FIRE BARRIER WITH INOPERABLE FIRE DETECTION.
EVENT DATE: 012188 REPORT DATE: 041588 NSSS: BW TYPE: PWR

(NSIC 208971) ON JANUARY 21, 1988, AT 2000 HOURS, THE SHIFT SUPERVISOR WAS NOTIFIED THAT AN INOPERABLE FIRE BARRIER, 426-N/427-S, DID NOT HAVE OPERABLE FIRE DETECTION ON EITHER SIDE. THIS CONDITION HAD EXISTED FOR APPROXIMATELY 45 HOURS. THIS IS A VIOLATION OF TECH SPEC 3.7.10 BECAUSE A CONTINUOUS FIRE WATCH HAD NOT BEEN ESTABLISHED WITHIN 1 HOUR. THE SHIFT SUPERVISOR RESET THE ALARMS, WHICH RETURNED THE FIRE DETECTION ZONES TO AN OPERABLE STATUS. OPERATIONS PERSONNEL DID NOT RECOGNIZE THAT FIRE DETECTORS WERE INOPERABLE ON BOTH SIDES OF AN INOPERABLE FIRE BARRIER. A LIST OF FIRE DETECTION ZONES AND THEIR CORRESPONDING FIRE BAOWHICH WAS BARRIERS WILL BE DEVELOPED FOR OPERATIONS PERSONNEL. UNTIL THE LIST IS DEVELOPED, A CONTINUOUS FIRE WATER WILL BE ESTABLISHED WHEN A FIRE DETECTION ALARM CAN NOT BE CLEARED WITHIN 1 HOOUR. DURING DEVELOPMENT OF THIS LIST IT WAS DISCOVERED THAT ONE INOPERABLE FIRE BARRIER WHICH WAS BELIEVED TO HAVE FIRE DETECTION ON ONE SIDE, DID NOT. THIS WAS A RESULT OF AN INCORRECT DRAWING FOLLOWING IMPLEMENTATION OF FACILITY CHANGE REQUEST (FCR) 81-0100. THIS MODIFICATION WILL BE REVIEWED TO ENSURE ALL DESIGN DRAWINGS AFFECTED HAVE BEEN UPDATED APPROPRIATELY. SINCE THE COMPLETION OF FCR 81-0100 THE MODIFICATION PROCESS HAS BEEN REVISED TO INCLUDE A MORE THOROUGH REVIEW. THIS SHOULD IDENTIFY THIS TYPE OF OVERSIGHT PRIOR TO COMPLETION OF DESIGN PACKAGES.

[62]   DAVIS-BESSE 1   DOCKET 50-346   LER 88-008
NUCLEAR SAFETY RELATED EQUIPMENT POTENTIALLY IMPACTED BY NON-SEISMIC EQUIPMENT.
EVENT DATE: 031288   REPORT DATE: 040888   NSSS: BW   TYPE: PWR

(NSIC 208972) ON MARCH 12, 1988 AT 0930 HOURS, NUCLEAR SAFETY RELATED (NSR) EQUIPMENT WAS DISCOVERED TO BE LOCATED WITHIN THE FALLING ARC DISTANCE OF NON-SEISMIC ELECTRICAL PANEL C4601. SUBSEQUENT INVESTIGATION REVEALED THREE MORE NON-SEISMIC PANELS SIMILAR TO C4601. A WALKDOWN OF THESE PANELS IDENTIFIED ONE OTHER PANEL (C5751) WITH NSR EQUIPMENT LOCATED WITHIN ITS FALLING ARC DISTANCE. ORIGINAL INSTALLATION OF C4601 AND C5751 WAS CORRECT. DURING THE PREPARATION OF SUBSEQUENT MODIFICATIONS THE POTENTIAL FOR PANELS C4601 AND C5751 TO FALL AND IMPACT NSR EQUIPMENT WAS OVERLOOKED. NSR EQUIPMENT NEAR PANEL C5751 WILL BE RELOCATED AND PANEL C4601 WILL BE SEISMICALLY RESTRAINED DURING THE FIFTH REFUELING OUTAGE. A SAMPLE OF NON-SEISMIC FREE STANDING ELECTRICAL PANELS IS TO BE CONDUCTED TO IDENTIFY ANY SIMILAR CONDITIONS AND ANY ADDITION CORRECTIVE ACTIONS. OUTSIDE ORGANIZATIONS INVOLVED IN THE HAZARDS ANALYSIS REVIEW WILL ALSO BE TRAINED. THIS FINDING IS BEING REPORTED UNDER 10CFR50.73(A)(2)(II)(B) AS A CONDITION OUTSIDE THE DESIGN BASIS.

[63]   DIABLO CANYON 2   DOCKET 50-323   LER 87-023
ACCUMULATOR NOZZLE CRACKING DUE TO INTERGRANULAR STRESS CORROSION.
EVENT DATE: 100987   REPORT DATE: 042288   NSSS: WE   TYPE: PWR
VENDOR: DELTA SOUTHERN CO.

(NSIC 209126) THIS VOLUNTARY LER IS BEING SUBMITTED FOR INFORMATION PURPOSES ONLY AS DESCRIBED IN ITEM 19 OF SUPPLEMENT NUMBER 1 TO NUREG-1022. SINCE OCTOBER 1985, UNIT 2 HAD EXPERIENCED SEVERAL VERY SMALL LEAKS ON THE EMERGENCY CORE COOLING SYSTEM ACCUMULATOR TANK SAMPLE AND FILL LINE NOZZLES. THESE LEAKS HAVE ALL BEEN MINOR RESULTING IN ONLY SMALL DEPOSITS OF BORIC ACID PRECIPITATE. THE LEAKS RESULTED FROM INTERGRANULAR STRESS CORROSION CRACKING (IGSCC) WITH THE EXCEPTION OF ONE LEAK WHICH RESULTED FROM A WELD DEFECT INTRODUCED DURING MANUFACTURING. ALL LEAKING NOZZLES WERE IDENTIFIED AND SUCCESSFULLY REPAIRED OR REPLACED, AND SURVEILLANCE INSPECTION FREQUENCY HAS BEEN INCREASED. ALTHOUGH SUPPLIED FROM THE SAME MANUFACTURER, UNIT 1 ACCUMULATORS HAVE NOT EXPERIENCED LEAKS AS IN UNIT 2.

[64]   DRESDEN 2   DOCKET 50-237   LER 88-001
DIESEL GENERATOR AIR START PIPING OUTSIDE FSAR STRESS ALLOWABLES DUE TO APPARENT ORIGINAL DESIGN DEFICIENCY.
EVENT DATE: 032988   REPORT DATE: 042588   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 209168) ON MARCH 29, 1988 WITH UNIT 2 OPERATING AT 97% RATED THERMAL POWER AND UNIT 3 SHUTDOWN FOR REFUEL, THE TECHNICAL STAFF SUPERVISOR WAS NOTIFIED BY SARGENT & LUNDY ARCHITECT ENGINEERS THAT THE UNIT 2 AND UNIT 3 DIESEL GENERATOR AIR START PIPING EXCEEDED THE CODE STRESS ANALYSIS ALLOWABLES SPECIFIED IN THE FINAL SAFETY ANALYSIS REPORT (FSAR). THE AIR START PIPING WAS DETERMINED TO EXCEED THE CONSERVATIVE FSAR DESIGN CRITERIA UPON COMPLETION OF ENGINEERING ANALYSIS OF INSPECTION DATA. THE PIPING CONCERNS WERE RAISED IN DECEMBER OF 1987 DURING AN UNRELATED WALKDOWN PERFORMED BY SARGENT & LUNDY AND COMMONWEALTH EDISON BOILING WATER REACTOR ENGINEERING DEPARTMENT PERSONNEL. THE IDENTIFIED DISCREPANCIES CONSISTED OF AIR START FILTERS AND AIR START PIPING FOR BOTH DIESEL GENERATOR SYSTEMS BEING ATTACHED TO HANDRAILS. THE AIR START PIPING FOR THE UNIT 2 AND UNIT 3 DIESEL GENERATORS FAILED TO MEET FSAR PIPING STRESS REQUIREMENTS DUE TO AN ORIGINAL DESIGN DEFICIENCY. THE SAFETY SIGNIFICANCE WAS CONSIDERED MINIMAL SINCE THE PIPING STRESSES WERE CALCULATED TO BE WITHIN THE OPERABILITY LIMITS. THE AIR START PIPING ON THE UNIT 2 AND UNIT 3 DIESEL GENERATORS WILL BE MODIFIED TO ENSURE THAT THE PIPING STRESSES COMPLY WITH THE FSAR REQUIREMENTS. THIS IS THE

FIRST OCCURRENCE OF PIPING STRESS DEFICIENCIES ON THE DIESEL GENERATOR SYSTEMS AT DRESDEN STATION.

[65] DRESDEN 3 DOCKET 50-249 LER 88-003
FLUED HEAD ANCHOR SUPPORTS IN EXCESS OF FSAR DESIGN CRITERIA DUE TO DESIGN AND CONSTRUCTION DEFICIENCIES.
EVENT DATE: 032388 REPORT DATE: 041888 NSSS: GE TYPE: BWR

(NSIC 209044) ON MARCH 23, 1988 AT 1115 HOURS DURING NORMAL UNIT 3 OPERATION AT 2426 MW THERMAL POWER (96%) DRESDEN STATION MANAGEMENT WAS NOTIFIED BY THE BWR ENGINEERING DEPARTMENT THAT THREE PRIMARY CONTAINMENT PIPE PENETRATION FLUED HEAD ANCHORS (FHA'S) DID NOT MEET FINAL SAFETY ANALYSES REPORT (FSAR) DESIGN REQUIREMENTS. THE DEFICIENCIES WERE DISCOVERED AS A RESULT OF ANALYSIS OF FHA DATA FROM FHA INSPECTIONS PERFORMED IN JANUARY OF 1988. TWO OF THE FHA'S WERE DETERMINED TO NOT MEET FSAR PIPE RUPTURE DESIGN REQUIREMENTS DUE TO A DESIGN DEFICIENCY WHEN THE ANCHORS WERE REDESIGNED DURING THE 1986 UNIT 3 RECIRCULATION PIPING REPLACEMENT REFUEL OUTAGE. THE THIRD FHA WAS DETERMINED TO BE DEFICIENT DUE TO A BRACE THAT WAS IDENTIFIED AS MISSING DUE TO AN ORIGINAL CONSTRUCTION DEFICIENCY. A PRELIMINARY ASSESSMENT OF THE THREE FHA'S DETERMINED THAT THE ANCHORS ARE WITHIN OPERABILITY LIMITS AND WILL PERFORM THEIR INTENDED FUNCTIONS UNDER ALL DESIGN BASIS EVENTS. NO TECH SPEC REQUIREMENTS WERE VIOLATED. FOR THESE REASONS, THIS EVENT WAS CONSIDERED TO BE OF MINIMAL SAFETY SIGNIFICANCE. REPAIRS TO THE THREE FHA'S WILL BE PERFORMED UNDER MODIFICATION M12-3-88-20 WHICH IS SCHEDULED FOR COMPLETION PRIOR TO START UP FOLLOWING THE UNIT 3 1988 REFUEL OUTAGE. NO PREVIOUS OCCURRENCES OF FHA DEFICIENCIES WERE DISCOVERED.

[66] DRESDEN 3 DOCKET 50-249 LER 88-005
HPCI SYSTEM INTENTIONALLY MADE INOPERABLE TO FACILITATE PRE-PLANNED PREVENTIVE MAINTENANCE TESTING.
EVENT DATE: 032688 REPORT DATE: 041888 NSSS: GE TYPE: BWR

(NSIC 209045) ON MARCH 26, 1988 AT 1830 HOURS DURING A NORMAL UNIT 3 SHUTDOWN IN PREPARATION FOR A SCHEDULED REFUELING OUTAGE, THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM WAS INTENTIONALLY MADE INOPERABLE TO FACILITATE PREVENTIVE MAINTENANCE HPCI TURBINE OVERSPEED TRIP TESTING IN ACCORDANCE WITH DRESDEN OPERATING SURVEILLANCE (DOS) 2300-2. DOS 2300-2 REQUIRES THE HPCI TURBINE TO TRIP BETWEEN 5000 AND 5100 RPM. AT 2205 HOURS THE HPCI TURBINE WAS SATISFACTORILY TESTED TO TRIP AT 5013 RPM. THE ROOT CAUSE OF THIS EVENT WAS AN INTENTIONAL STATION MANAGEMENT DECISION TO PERFORM PREVENTIVE MAINTENANCE TESTING OF THE HPCI TURBINE OVERSPEED TRIP SYSTEM IN A CONTROLLED AND PRE-PLANNED MANNER. OVERSPEED TRIP SYSTEM TESTING COULD NOT HAVE BEEN PERFORMED WITHOUT MAKING HPCI INOPERABLE. THE HPCI TURBINE SATISFACTORILY TRIPPED WITHIN ITS ACCEPTABLE TEST RANGE THEREFORE NO CORRECTIVE ACTIONS WERE NECESSARY WITH RESPECT TO HPCI. IN ACCORDANCE WITH TECH SPEC REQUIREMENTS, REACTOR PRESSURE WAS REDUCED TO LESS THAN 90 PSIG 13 HOURS SUBSEQUENT TO DECLARING HPCI INOPERABLE. THE LAST PREVIOUS OCCURRENCE OF HPCI BEING DECLARED INOPERABLE WAS REPORTED BY LICENSEE EVENT REPORT #87-017 ON DOCKET #050249.

[67] DRESDEN 3 DOCKET 50-249 LER 88-006
HPCI AREA TEMPERATURE SWITCHES EXCEEDED TECH SPEC LIMIT DUE TO INSTRUMENT SETPOINT DRIFT.
EVENT DATE: 032988 REPORT DATE: 042688 NSSS: GE TYPE: BWR
VENDOR: UNITED ELECTRIC CONTROLS COMPANY

(NSIC 209175) ON MARCH 29, 1988 AT 0630 HOURS DURING A UNIT 3 REFUEL OUTAGE, HIGH PRESSURE COOLANT INJECTION (HPCI) AREA TEMPERATURE SWITCHES 3-2370D AND 3-2371D TRIPPED AT 215.7F AND 250.0F RESPECTIVELY WHILE PERFORMING THE HPCI TEMPERATURE SWITCH CALIBRATION SURVEILLANCE. THE TEMPERATURE SWITCHES ARE REQUIRED TO TRIP

WITHIN THE TECH SPEC LIMIT OF LESS THAN OR EQUAL TO 200F. WHILE THE INSTRUMENT MAINTENANCE DEPARTMENT CONTINUED TO CALIBRATE THE REMAINING TEMPERATURE SWITCHES, FIVE ADDITIONAL SWITCHES TRIPPED ABOVE THE TECH SPEC LIMIT. THE CAUSE OF THE HPCI TEMPERATURE SWITCHES' FAILURE TO TRIP WITHIN THE REQUIRED LIMIT WAS ATTRIBUTED TO INSTRUMENT SETPOINT DRIFT. THE IMMEDIATE CORRECTIVE ACTION CONSISTED OF RECALIBRATING ALL THE TEMPERATURE SWITCHES TO WITHIN THE STATION LIMIT OF 175F - 185F. TO PREVENT A FAILURE OF THIS TYPE FROM RECURRING, THE TEMPERATURE SWITCHES WILL EVENTUALLY BE REPLACED WITH A MORE RELIABLE TYPE OF SWITCH. ADDITIONALLY, AUGMENTED INSPECTIONS WILL BE PERFORMED UNTIL THE SWITCHES ARE REPLACED. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL SINCE THE REMAINING NINE TEMPERATURE SWITCHES WERE CAPABLE OF ISOLATING THE HPCI SYSTEM UPON DETECTION OF A STEAM LEAK. A PREVIOUS SIMILAR OCCURRENCE WAS REPORTED BY LER 86-28 ON DOCKET #050237.

[68]   FARLEY 1   DOCKET 50-348   LER 88-006 REV 01
UPDATE ON POTENTIAL INOPERABILITY OF THE A TRAIN CHARGING PUMP DUE TO GAS ACCUMULATION.
EVENT DATE: 030188   REPORT DATE: 042588   NSSS: WE   TYPE: PWR
OTHER UNITS INVOLVED: FARLEY 2 (PWR)

(NSIC 209122) AT APPROXIMATELY 1700 ON 3-1-88 IT WAS DETERMINED THAT HYDROGEN WAS ACCUMULATING IN THE SUPPLY LINE FROM THE RESIDUAL HEAT REMOVAL (RHR) PUMP TO THE CHARGING PUMP SUCTION FOR THE A TRAIN ON EACH UNIT. THE AMOUNTS OF HYDROGEN THAT ACCUMULATED COULD POSSIBLY HAVE CAUSED DAMAGE TO THE A TRAIN CHARGING PUMP IF THE RECIRCULATION PHASE OF SAFETY INJECTION (I.E., SUCTION FROM THE CONTAINMENT SUMP VIA RHR PUMP TO THE CHARGING PUMP) HAD BEEN REQUIRED. NORMAL CHARGING PUMP OPERATION AND THE INJECTION PHASE OF SAFETY INJECTION WERE NOT AFFECTED BY THE HYDROGEN ACCUMULATIONS. THE HYDROGEN WAS VENTED FROM THE PIPING AND A PERIOD VENTING PROGRAM WAS INITIATED TO ENSURE THE HYDROGEN ACCUMULATION WAS HELD TO ACCEPTABLE LEVELS. THE NORMALLY OPERATING CHARGING PUMP ON EACH UNIT HAS BEEN ESTABLISHED AS THE B OR C PUMP BECAUSE OF THE IDENTIFIED PROPENSITY FOR GAS ACCUMULATION IN THE SUPPLY LINE FROM THE RESIDUAL HEAT REMOVAL PUMP TO THE CHARGING PUMP SUCTION FOR THE A TRAIN WHEN THE A PUMP IS RUNNING. THIS EVENT WAS CAUSED BY THE PIPING CONFIGURATION IN THAT THE LAYOUT OF THE A TRAIN SUPPLY LINE FROM RHR PIPING ALLOWED HYDROGEN TO ACCUMULATE.

[69]   FARLEY 1   DOCKET 50-348   LER 88-009
FIRE DAMPERS INOPERABLE DUE TO FAILURE TO CLOSE WITH AIR FLOW.
EVENT DATE: 041288   REPORT DATE: 050588   NSSS: WE   TYPE: PWR

(NSIC 209256) ON 7/13/87, FNP SUBMITTED A SPECIAL REPORT (LER 87-011-00) CONCERNING INOPERABLE CONTROL ROOM FIRE DAMPERS. AS A RESULT, A FIRE DAMPER MAINTENANCE AND TESTING PROGRAM IS IN PROGRESS. AS A PART OF THIS PROGRAM, FIRE DAMPERS 1-110-1? (TESTED ON 4-5-88), 1-110-20 (TESTED ON 4-6-88), 1-110-24 (TESTED ON 4-6-88), 1-122-07 (TESTED ON 4-11-88), 1-122-02 (TESTED ON 4-12-88), AND 1-122-03 (TESTED ON 4-12-88) WOULD NOT CLOSE WITH AIR FLOW IN THE SYSTEM. THESE EVENTS WERE CAUSED BY DESIGN DEFICIENCY IN THAT THE FIRE DAMPERS WILL NOT CLOSE FULLY WITH AIR FLOW. DESIGN CHANGES HAVE BEEN INITIATED TO EVALUATE THE OPTIONS AVAILABLE AND PROVIDE THE APPROPRIATE DESIGN TO ENSURE THE PROPER OPERATION OF THE FIRE DAMPERS. THESE DESIGN CHANGES ARE EXPECTED TO BE IMPLEMENTED WITHIN THE NEXT NINE TO TWELVE MONTHS. TECH SPEC 3.7.12 REQUIRES THESE FIRE DAMPERS TO BE RETURNED TO OPERABLE STATUS WITHIN SEVEN DAYS OR A SPECIAL REPORT MUST BE SUBMITTED WITHIN THE FOLLOWING 30 DAYS. THEREFORE THIS SPECIAL REPORT IS BEING SUBMITTED. ALL TECH SPEC ACTION STATEMENT REQUIREMENTS FOR THE FIRE DAMPERS ARE BEING MET.

[70] FARLEY 1 DOCKET 50-348 LER 88-008
SEVENTEEN STEAM GENERATOR TUBES EXCEED PLUGGABLE LIMIT.
EVENT DATE: 042088 REPORT DATE: 050288 NSSS: WE TYPE: PWR

(NSIC 209229) THE FOLLOWING STEAM GENERATOR TUBE PLUGGING REPORT IS BEING SUBMITTED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 4.4.6.5.A. DURING THE CURRENT UNIT 1 CYCLE 8 - 9 REFUELING OUTAGE, SEVENTEEN TUBES EXCEEDED THE PLUGGABLE LIMIT BASED ON EDDY CURRENT TESTING RESULTS. FOUR TUBES WERE PLUGGED IN THE 1A STEAM GENERATOR, ONE TUBE WAS PLUGGED IN THE 1B STEAM GENERATOR, AND TWELVE TUBES WERE PLUGGED IN THE 1C STEAM GENERATOR. THIS TUBE PLUGGING WAS COMPLETED ON 4-20-88. THE COMPLETE RESULTS OF THE STEAM GENERATOR TUBE 1 INSERVICE INSPECTION WILL BE REPORTED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 4.4.6.5.B.

[71] FARLEY 2 DOCKET 50-364 LER 87-004 REV 01
UPDATE ON STEAM GENERATOR TUBE DEGRADATION.
EVENT DATE: 111387 REPORT DATE: 041888 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 209042) THE FOLLOWING REPORT IS BEING SUBMITTED IN ACCORDANCE WITH TECH SPEC 4.4.6.5.A AND C. PRIOR TO THE FIFTH REFUELING OUTAGE OF FARLEY UNIT 2, ALABAMA POWER COMPANY DEVELOPED AN EDDY CURRENT INSPECTION PLAN TO INSPECT ALL NON-PLUGGED TUBES IN ALL THREE STEAM GENERATORS. THIS INSPECTION WAS TO COVER THE FULL LENGTH OF TUBES ABOVE ROW 2. ROW 2 WAS TO BE INSPECTED FROM THE HOT LEG TUBE END TO THE TOP SUPPORT PLATE ON THE COLD LEG. ALL ROW 1 TUBES ARE PLUGGED. DURING THE COURSE OF THE INSPECTION, THE PROGRAM WAS EXPANDED TO INCLUDE THE COLD LEG OF ROW 2 TUBES. ALL OF THESE INSPECTIONS WERE DONE WITH THE BOBBIN COIL PROBE. PROBING PERFORMED IN A SAMPLE NUMBER OF TUBES USING THE ROTATING PANCAKE COIL (RPC) CONFIRMED THAT THE INDICATIONS FOUND AT TUBESHEET AND SUPPORT PLATE LOCATIONS WERE CLASSIFIED CONSISTENT WITH THOSE FOUND AT THE LAST REFUELING OUTAGE (REFERENCE LER 86-004-00). NO TUBES WERE REPAIRED, 109 TUBES WERE PLUGGED AND 88 TUBES WERE DESIGNATED F*. INVESTIGATIONS AND EVALUATIONS PERFORMED IDENTIFIED FOUR AREAS WHERE TUBE DEGRADATION WAS OBSERVED: ANTIVIBRATION BAR (AVB) WEAR, PRIMARY WATER STRESS CORROSION CRACKING (PWSCC) IN THE TUBESHEET AREA, OD STRESS CORROSION CRACKING AT SUPPORT PLANTES, AND RANDOM NONCORRELATABLE DEGRADATION. THESE ARE SIMILAR TO THE MECHANISMS REPORTED IN LER 86-004-00.

[72] FARLEY 2 DOCKET 50-364 LER 87-010 REV 01
UPDATE ON UNIT SHUT DOWN DUE TO PRESSURE BOUNDARY LEAKAGE.
EVENT DATE: 120987 REPORT DATE: 041188 NSSS: WE TYPE: PWR

(NSIC 208957) AT 2255 ON 12-09-87, WITH THE UNIT AT 33% POWER FOLLOWING A REFUELING OUTAGE, IT WAS OBSERVED THAT THE CONTAINMENT COOLER DRAIN POT LEVELS WERE ABNORMALLY HIGH. A REACTOR COOLANT SYSTEM (RCS) LEAKAGE CALCULATION CONFIRMED THAT THE RCS UNIDENTIFIED LEAKAGE HAD INCREASED. A CONTAINMENT ENTRY WAS MADE AND A LEAK WAS IDENTIFIED IN THE VICINITY OF THE B LOOP RESISTANCE TEMPERATURE DETECTOR (RTD) MANIFOLD. A UNIT SHUTDOWN WAS MADE TO REPAIR THE LEAK. AFTER THE SHUTDOWN AND UPON CLOSER EXAMINATION, THE PRESSURE BOUNDARY LEAKAGE WAS IDENTIFIED TO BE FROM THE RCS LOOP B COLD LEG SAFETY INJECTION LINE BETWEEN A CHECK VALVE AND THE RCS LOOP. THE LEAK RESULTED FROM A THROUGH WALL DEFECT IN A WELDED JOINT BETWEEN A LONG RADIUS ELBOW AND A STRAIGHT SECTION OF PIPE. THE SECTION OF PIPING CONTAINING THE DEFECT HAS BEEN REPLACED. RESULTS OF THE METALLURGICAL EVALUATION OF THE FAILED JOINT HAVE IDENTIFIED A FATIGUE MECHANISM AS THE CAUSE FOR CRACK INITIATION AND PROPAGATION. A REVIEW OF CONSTRUCTION RADIOGRAPHS AND NONDESTRUCTIVE EXAMINATIONS (ULTRASONIC TESTS AND RADIOGRAPHIC TESTS) PERFORMED AS A RESULT OF THIS EVENT REVEALED THAT NO PROBLEMS EXIST ON SIMILAR PIPING WELDS IN THE COLD LEG INJECTION LINES OF EITHER UNIT.

[73] FARLEY 2 DOCKET 50-364 LER 88-004
PERSONNEL ERROR RESULTS IN REQUIRED FIRE WATCH PATROL NOT BEING ESTABLISHED.
EVENT DATE: 031588 REPORT DATE: 041488 NSSS: WE TYPE: PWR

(NSIC 208941) AT 0900 ON 3-15-88, IT WAS DISCOVERED THAT A FIRE WATCH PATROL HAD NOT BEEN ESTABLISHED AS REQUIRED. AT 1000 ON 3-8-88, TWO PENETRATIONS WERE RELEASED TO BE BREACHED. HOWEVER, THE SHIFT FOREMAN WHO RELEASED THESE PENETRATIONS DID NOT RECOGNIZE THAT ONE OF THESE PENETRATIONS IS A FIRE BARRIER PENETRATION. CONSEQUENTLY, THE REQUIRED FIRE WATCH PATROL WAS NOT ESTABLISHED. THIS EVENT WAS CAUSED BY PERSONNEL ERROR. THE SHIFT FOREMAN INVOLVED IN THIS EVENT HAS BEEN COUNSELED.

[74] FERMI 2 DOCKET 50-341 LER 87-048 REV 03
UPDATE ON INADEQUACIES IN TECHNICAL SPECIFICATION SURVEILLANCES FOUND DURING SURVEILLANCE REVIEW.
EVENT DATE: 100887 REPORT DATE: 041588 NSSS: GE TYPE: BWR

(NSIC 208905) A REVIEW OF TECHNICAL SPECIFICATION SURVEILLANCE PROCEDURES HAS DISCOVERED SEVERAL VIOLATIONS OF TECHNICAL SPECIFICATIONS. THE REQUIREMENT TO PERFORM A CHANNEL CHECK FOR THE REACTOR PROTECTION SYSTEM DRYWELL HIGH PRESSURE INSTRUMENTATION WAS NOT INCLUDED IN ANY SURVEILLANCE PROCEDURE. THE REQUIREMENT TO PERFORM A CHANNEL FUNCTIONAL TEST AS PART OF THE CHANNEL CALIBRATION OF THE ELECTRICAL PROTECTION ASSEMBLY BREAKERS HAD NOT BEEN PROPERLY INCORPORATED INTO PROCEDURES. INADEQUATE TESTING OF THE LOSS OF POWER LOGIC WAS FOUND WHICH IMPACTED THE OPERABILITY OF THE EMERGENCY DIESEL GENERATORS. THE PLANT WAS SHUTDOWN AS A RESULT OF THE LAST CONCERN. THE EMERGENCY CORE COOLING SYSTEM RESPONSE TIME TESTING DID NOT ADEQUATELY MEASURE THE TIME TAKEN TO ACHIEVE DESIGN BASIS PUMP PERFORMANCE. THESE CONDITIONS WERE CAUSED BY INCOMPLETE OR INADEQUATE SURVEILLANCE PROCEDURES. THE CORRECTIVE ACTIONS INCLUDE REVISING THE APPROPRIATE PROCEDURES. ALL TECHNICAL SPECIFICATION SURVEILLANCE PROCEDURES ARE SCHEDULED TO BE REVIEWED BY THE END OF 1988 AS PART OF THE TECHNICAL SPECIFICATION IMPROVEMENT PROGRAM.

[75] FERMI 2 DOCKET 50-341 LER 88-009
SAFETY RELIEF VALVES FAIL THEIR SET PRESSURE SURVEILLANCE TOLERANCE TEST.
EVENT DATE: 031188 REPORT DATE: 041188 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 208923) THE MAIN STEAM SYSTEM IS EQUIPPED WITH FIFTEEN SAFETY RELIEF VALVES (SRVS). TECHNICAL SPECIFICATIONS REQUIRE THAT HALF OF THE SRVS BE PROVEN OPERABLE AT LEAST ONCE EVERY EIGHTEEN MONTHS BY PERFORMING A SET PRESSURE TEST. FIFTEEN SRVS WERE REMOVED AND SENT TO WYLE LABORATORIES TO MEET THE SURVEILLANCE REQUIREMENT. WYLE LABORATORIES NOTIFIED DETROIT EDISON THAT NINE OF THE SRVS FAILED THEIR SET PRESSURE TEST. THE CAUSE OF THIS EVENT IS CURRENTLY UNDER REVIEW BY THE SITE AND GENERICALLY BY THE BOILING WATER REACTORS OWNERS GROUP (BWROG) SRV SET POINT DRIFT COMMITTEE. ALL VALVES REMOVED FROM THE PLANT FOR TESTING WERE REFURBISHED, CLEANED, RETESTED AND RECERTIFIED TO BE WITHIN ACCEPTED TOLERANCES PRIOR TO RETURN TO FERMI 2 FROM WYLE LABORATORIES.

[76] FERMI 2 DOCKET 50-341 LER 88-011
FAILURE TO PERFORM SHIFTLY SURVEILLANCE WITHIN THE REQUIRED TIME.
EVENT DATE: 031388 REPORT DATE: 041288 NSSS: GE TYPE: BWR

(NSIC 208937) ON MARCH 13, 1988, THE OPERATING SHIFT FAILED TO PERFORM THE TECH SPEC REQUIRED TWELVE HOUR SURVEILLANCES WITHIN THE ALLOWABLE TIME. REVIEW OF THE ACTIVITIES ON SHIFT INDICATES THAT THE ACTION STATEMENTS WERE MET EXCEPT FOR THE INTERMEDIATE RANGE MONITOR INOPERATIVE CHANNEL. THIS EVENT WAS CAUSED BY A FAILURE TO DESIGNATE WHO WOULD PERFORM THE SURVEILLANCE WHEN PERSONNEL WERE

REASSIGNED. THE SURVEILLANCES WERE SUCCESSFULLY PERFORMED AT MIDNIGHT ON MARCH 14, 1988, 17 HOURS AFTER THEIR LAST PERFORMANCE. AS CORRECTIVE ACTION FOR THIS EVENT, A REQUIREMENT TO COLLECT SURVEILLANCE FORMS AFTER EACH SHIFT HAS BEEN ADDED. AN OPERATING PRACTICE STANDARD WILL BE REVISED TO REQUIRE THAT A RESPONSIBLE INDIVIDUAL IS DESIGNATED FOR THE COMPLETION OF ROUTINE SURVEILLANCES.

[77] FERMI 2 DOCKET 50-341 LER 88-013
INADVERTENT ISOLATION OF SHUTDOWN COOLING WHILE DE-ENERGIZING A MODULAR POWER UNIT.
EVENT DATE: 031888 REPORT DATE: 041888 NSSS: GE TYPE: BWR

(NSIC 208939) ON MARCH 18, 1988, THE MODULAR POWER UNIT (MPU) NUMBER 2 WAS BEING SHUT DOWN FOR PREVENTATIVE MAINTENANCE. AS THE FIRST CIRCUIT WAS OPENED, SHUTDOWN COOLING WAS LOST DUE TO THE CLOSURE OF THE SUCTION ISOLATION VALVE AND SUBSEQUENT TRIPPING OF THE RESIDUAL HEAT REMOVAL PUMP. THE MPU SHUTDOWN WAS HALTED AND SHUTDOWN COOLING WAS RE-ESTABLISHED IN 23 MINUTES. THIS EVENT WAS CAUSED BY PROCEDURAL INADEQUACIES AND PERSONNEL ERROR. THE SYSTEM OPERATING PROCEDURE DID NOT SPECIFICALLY ADDRESS THE EFFECT THAT DE-ENERGIZING MPU NUMBER 2 WOULD HAVE ON THE SUCTION ISOLATION VALVE. ADDITIONALLY, THE LICENSED OPERATOR WHO PERFORMED AN INDEPENDENT REVIEW PRIOR TO THE WORK DID NOT IDENTIFY THE POTENTIAL FOR THIS EVENT. AS LONG TERM CORRECTIVE ACTION, DETAILED LISTS OF THE EFFECTS OF DE-ENERGIZING ANY OF THE MPU'S WILL BE INCORPORATED INTO THE SYSTEM OPERATING PROCEDURE.

[78] FERMI 2 DOCKET 50-341 LER 88-014
INADVERTENT START OF THE DIVISION I EMERGENCY DIESEL GENERATORS DUE TO A PROCEDURAL ERROR.
EVENT DATE: 032088 REPORT DATE: 041988 NSSS: GE TYPE: BWR

(NSIC 209011) ON MARCH 20, 1988 DURING THE PERFORMANCE OF LOGIC FUNCTIONAL TESTING FOR THE CORE SPRAY SYSTEM, THE DIVISION I EMERGENCY DIESEL GENERATORS (EDGS) AUTOMATICALLY STARTED. THE EDGS CAME UP TO SPEED AND PROPERLY MANUALLY LOADED. THEY WERE SECURED BY THE OPERATING SHIFT ONE HOUR AND 47 MINUTES AFTER THE EDGS HAD STARTED. THE STEP FOR RESETTING THE EMERGENCY CORE COOLING SYSTEM (ECCS) START LOGIC HAD BEEN REVERSED WITH THE STEP FOR REMOVING A PORTABLE TEST SWITCH DURING A RECENT PROCEDURE CHANGE. THIS ALLOWED THE EDG START LOGIC TO BE ENABLED PRIOR TO RESETTING THE ECCS START LOGIC. REVISIONS HAVE BEEN MADE TO THE APPROPRIATE LOGIC FUNCTIONAL TESTS. REQUIRED READING ABOUT THIS EVENT WILL BE ISSUED TO PERSONNEL RESPONSIBLE FOR REVIEWING AND PREPARING OF INSTRUMENTATION AND CONTROL PROCEDURE REVISIONS.

[79] FITZPATRICK DOCKET 50-333 LER 87-012
REACTOR TRIPS FROM TURBINE TRIP DUE TO GENERATOR FIELD GROUND.
EVENT DATE: 082887 REPORT DATE: 092887 NSSS: GE TYPE: BWR

(NSIC 209190) AT 1357 ON 8/28/87 WHILE OPERATING AT A POWER LEVEL OF 100% A REACTOR SCRAM OCCURRED AS A RESULT OF A MAIN TURBINE TRIP. THE MAIN TURBINE TRIP WAS INITIATED BY AN UNKNOWN GENERATOR PROTECTIVE RELAY ACTUATION. AT 2237 ON 9/7/87 WHILE OPERATING AT A POWER LEVEL OF 100% A TURBINE TRIP, DUE TO ACTUATION OF THE MAIN GENERATOR FIELD GROUND PROTECTIVE RELAY, OCCURRED WHICH WAS VIRTUALLY IDENTICAL TO THE 8/28/87 EVENT. DURING BOTH EVENTS THE PLANT OPERATORS RESPONDED TO THE TRANSIENT BY UTILIZING APPROVED PLANT PROCEDURES. PLANT RESPONSE FOR BOTH EVENTS WERE WITHIN THE BOUNDS OF TRANSIENT ANALYSIS DISCUSSED IN THE FINAL SAFETY ANALYSIS REPORT. THE ROOT CAUSE OF THE MAIN GENERATOR TRIPS IS BELIEVED TO HAVE BEEN THE PRESENCE OF A CUPRIC OXIDE FILM ON THE INSULATED COOLING WATER TUBES TO THE RECTIFIER BANKS. THIS FILM APPEARS TO HAVE CAUSED A LEAKAGE PATH TO GROUND IN THE GENERATOR FIELD CIRCUIT. AFTER THE INSULATING TUBES WERE CLEANED, THE

FIELD GROUND CURRENT RETURNED TO NORMAL. NO SIMILAR EVENTS INVOLVING GENERATOR FIELD GROUND FAULTS HAVE OCCURRED AT THIS FACILITY.

[80] FITZPATRICK DOCKET 50-333 LER 88-003
ENGINEERED SAFETY FEATURE ACTUATIONS DUE TO TRIP OF REACTOR PROTECTION SYSTEM POWER SUPPLY AS A RESULT OF PROTECTIVE RELAY FAILURE.
EVENT DATE: 041888 REPORT DATE: 051088 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 209271) ON 4/18/88 AT 0500 HOURS DURING NORMAL OPERATION AT 100% RATED POWER, REACTOR PROTECTION SYSTEM (RPS) MOTOR GENERATOR (MG) (EF) A TRIPPED WHEN THE MG DRIVE MOTOR OVERLOAD PROTECTIVE RELAY FAILED. THE TRIP RESULTED IN ISOLATION OF THE REACTOR WATER CLEANUP (CE), PRIMARY CONTAINMENT (NH) DRAIN, REACTOR WATER SAMPLE, AND REACTOR BUILDING VENTILATION (VA) SYSTEMS. ISOLATION OF REACTOR BUILDING VENTILATION CAUSED AN AUTO START OF THE STANDBY GAS TREATMENT (BH) SYSTEM. OPERATING PERSONNEL TRANSFERRED TO ALTERNATE POWER AND RESTORED SYSTEMS TO NORMAL WITHIN 20 MINUTES. REPLACEMENT OF THE RELAY WAS COMPLETED AND THE MG WAS RESTORED TO SERVICE IN APPROXIMATELY 12 HOURS. SYSTEMS PERFORMED AS DESIGNED. CORRECTIVE ACTIONS WERE TO SHIFT TO ALTERNATE POWER, RESTORE EFFECTED SYSTEMS TO NORMAL, REPLACE THE FAILED RELAY, AND RETURN THE MG TO SERVICE. NO SIMILAR EVENTS HAVE BEEN CAUSED BY RELAY FAILURES.

[81] FT. CALHOUN 1 DOCKET 50-285 LER 88-008
FAILURE TO CONDUCT SURVEILLANCE TEST ON CONTAINMENT SPRAY HEADS WITHIN REQUIRED INTERVAL.
EVENT DATE: 050387 REPORT DATE: 041488 NSSS: CE TYPE: PWR

(NSIC 208931) SURVEILLANCE TEST ST-NZ-1 WAS PERFORMED ON MAY 3, 1987 TO VERIFY THAT ALL THE SPRAY HEADS ON THE CONTAINMENT SPRAY SYSTEM WERE UNOBSTRUCTED. THIS TEST IS REQUIRED BY TECHNICAL SPECIFICATION 3.6.2 TO BE PERFORMED EVERY FIVE YEARS. DURING A RECENT REVIEW OF THE TEST RESULTS, IT WAS DETERMINED THAT ELEVEN SPRAY HEADS WERE NOT TESTED BECAUSE OF DIFFICULTY REACHING THEM. ONE SPRAY HEAD WAS DISCOVERED MISSING AND WAS NOT REPLACED. THE EFFECT OF HAVING TWELVE POTENTIALLY DISABLED SPRAY HEADS ON THE CONTAINMENT SPRAY HEADERS WILL BE EVALUATED. BY THE END OF THE 1988 REFUELING OUTAGE EITHER THE TWELVE SPRAY HEADS WILL BE TESTED AS REQUIRED BY ST-NZ-1 OR BASED ON THE EVALUATION RESULTS, A REQUEST TO AMEND THE TECHNICAL SPECIFICATIONS WILL BE SUBMITTED.

[82] FT. CALHOUN 1 DOCKET 50-285 LER 88-004
INSTRUMENT AIR VALVE PCV-1849 OUTSIDE DESIGN BASIS FOR CONTAINMENT ISOLATION CRITERIA.
EVENT DATE: 031188 REPORT DATE: 041188 NSSS: CE TYPE: PWR

(NSIC 208929) ON MARCH 11, 1988, AT 1442 HOURS CST, IT WAS DETERMINED THAT ISOLATION VALVE, PCV-1849, FOR THE INSTRUMENT AIR (IA) SYSTEM CONTAINMENT PENETRATION M-73 DID NOT MEET THE CONTAINMENT ISOLATION CRITERIA SPECIFIED IN THE UPDATED SAFETY ANALYSIS REPORT (USAR) APPENDIX G, CRITERION 53, ITEM 3, BECAUSE IT CANNOT BE ASSURED THAT THE IA SYSTEM PRESSURE WILL BE GREATER THAN THE CONTAINMENT PRESSURE DURING A LOSS OF COOLANT ACCIDENT (LOCA) WITH A CONCURRENT LOSS OF OFFSITE POWER. A PHONE CALL WAS MADE TO THE NRC OPERATIONS CENTER WITHIN ONE HOUR OF THE DETERMINATION TO NOTIFY THE NRC OF THE UNANALYZED CONDITION PURSUANT TO THE REQUIREMENTS OF 10 CFR 50.72 (B)(1)(II)(B). DURING A LOCA A POTENTIAL LEAKAGE PATH THROUGH PCV-1849 COULD EXIST IF IA SYSTEM PRESSURE IS NOT MAINTAINED ABOVE THE CONTAINMENT PRESSURE. PCV-1849 IS DESIGNED TO CLOSE ON RECEIPT OF A LOW IA SYSTEM PRESSURE IN CONJUNCTION WITH A CONTAINMENT ISOLATION ACTUATION SIGNAL (CIAS). HOWEVER, PCV-1849 IS AN AIR OPERATED VALVE THAT HAS AN ACTUATOR WHICH WOULD ALLOW THE VALVE TO OPEN ON LOSS OF AIR. THIS IS NOT

CONSISTENT WITH THE USAR APPENDIX G CRITERION 53 AND THEREFORE CONTAINMENT INTEGRITY CANNOT BE ASSURED DURING A LOCA WITH CONCURRENT LOSS OF OFFSITE POWER.

[83] FT. CALHOUN 1 DOCKET 50-285 LER 88-006
SURVEILLANCE TEST NOT PERFORMED DURING JANUARY 1988.
EVENT DATE: 031588 REPORT DATE: 041488 NSSS: CE TYPE: PWR

(NSIC 208930) ON MARCH 15, 1988 IT WAS DISCOVERED THAT SURVEILLANCE TEST ST-DC-1 F.1, "STATION BATTERIES" HAD NOT BEEN COMPLETED DURING THE MONTH OF JANUARY 1988. THIS IS IN VIOLATION OF TECHNICAL SPECIFICATION 3.7(2)A WHICH REQUIRES A TEST BE PERFORMED "EVERY MONTH". INVESTIGATION REVEALED THAT ELECTRICAL MAINTENANCE PERSONNEL HAD COMPLETED THE JANUARY SURVEILLANCE TEST ON DECEMBER 28, 1987 (SCHEDULED DUE DATE). SINCE THE TEST WAS NOT COMPLETED AGAIN UNTIL FEBRUARY, THE REQUIRED TESTING INTERVAL OF "EVERY MONTH" HAD NOT BEEN MET. THE REASON THE TEST WAS NOT SCHEDULED AND SUBSEQUENTLY COMPLETED DURING JANUARY CAN BE ATTRIBUTED TO THE WAY THE SURVEILLANCE TEST SCHEDULING PROGRAM WAS DESIGNED. TO PREVENT A POSSIBLE RECURRENCE OF THIS PROBLEM, THE TEST WILL BE SCHEDULED FOR THE SECOND WEEK OF EACH MONTH. THIS WILL ENSURE IT IS PERFORMED DURING ITS SCHEDULED CALENDAR MONTH. IN ADDITION OPPD HAS SUBMITTED AN APPLICATION FOR AMENDMENT OF THE TECHNICAL SPECIFICATIONS TO PROVIDE FOR A 25 PERCENT EXTENSION TO SURVEILLANCE INTERVALS. THIS WILL PROVIDE A REASONABLE COMPLETION PERIOD FOR ALL SURVEILLANCE TESTS NOT PRESENTLY COVERED UNDER THIS EXTENSION.

[84] FT. CALHOUN 1 DOCKET 50-285 LER 88-007
INADVERTENT START OF EMERGENCY DIESEL GENERATOR D-1 AS THE RESULT OF SURVEILLANCE TEST ERROR.
EVENT DATE: 032388 REPORT DATE: 042288 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 1 (PWR)
CATAWBA 2 (PWR)

(NSIC 209005) DURING PERFORMANCE OF ST-ESF-6 F.2 APP. C, MONTHLY EMERGENCY DIESEL GENERATOR SURVEILLANCE TEST, ON 4/23/88 AT APPROX. 1113 CST, EMERGENCY DIESEL GENERATOR D-2 TRIPPED RESULTING IN AN AUTO-START OF THE EMERGENCY DIESEL GENERATOR D-1. IT WAS LATER DETERMINED THAT THE D-2 LOCK-OUT RELAY TRIPPED WHILE THE EMERGENCY DIESEL GENERATOR WAS BEING UNLOADED DUE TO A REVERSE CURRENT FLOW ACROSS THE OUTPUT BREAKER. WHEN THE LOCK-OUT RELAY TRIPPED IT RESULTED IN AN AUTO-START OF EMERGENCY DIESEL GENERATOR D-1. THE PRIMARY CAUSE OF THE EVENT WAS DETERMINED TO BE OPERATOR ERROR. THE OPERATOR WHILE UNLOADING THE EMERGENCY DIESEL GENERATOR, ALLOWED THE LOAD TO DROP TOO LOW BEFORE OPENING THE BREAKER RESULTING IN THE REVERSE CURRENT CONDITION. THIS EVENT WAS NOT ORIGINALLY CONSIDERED A REPORTABLE EVENT PURSUANT TO 10 CFR 50.72, BECAUSE THE EMERGENCY DIESEL GENERATORS WERE NOT INCLUDED IN THE ENGINEERED SAFETY FEATURES SECTION OF THE UPDATED SAFETY ANALYSIS REPORT AND BECAUSE THE SIGNAL WHICH STARTED THE DIESEL WAS NOT AN ENGINEERED SAFETY FEATURES SIGNAL. CORRECTIVE ACTION TO PRECLUDE RECURRENCE OF EVENTS OF THIS TYPE INCLUDE A REVISION TO THE UPDATED SAR SECTION 6 TO CLARIFY WHICH COMPONENTS AND SYSTEMS COMPRISE ENGINEERED SAFEGUARDS EQUIPMENT, AND A REVISION TO THE SURVEILLANCE TEST ST-ESF-6 TO INCLUDE GUIDANCE IN THE EMERGENCY DIESEL GENERATOR OPERATING INSTRUCTION FOR UNLOADING THE DIESEL.

[85] FT. CALHOUN 1 DOCKET 50-285 LER 88-009
PNEUMATIC OPERATED VALVES OUTSIDE DESIGN BASIS DURING DESIGN BASIS ACCIDENT.
EVENT DATE: 040688 REPORT DATE: 050688 NSSS: CE TYPE: PWR

(NSIC 209223) ON APRIL 6, 1988 AT 1425 (CDT), THE OMAHA PUBLIC POWER DISTRICT CONCLUDED AN INVESTIGATION, AS PART OF A SELF-CONDUCTED SSFI, WHICH IDENTIFIED CONCERNS ABOUT THE CAPABILITY OF CERTAIN VALVES TO PERFORM THEIR DESIGN FUNCTION DURING A DESIGN BASIS ACCIDENT WITH A CONCURRENT LOSS OF INSTRUMENT AIR. THE VALVES OF CONCERN ARE LCV-383-1&2, SAFETY INJECTION AND REFUELING WATER TANK

(SIWRT) ISOLATION VALVES, HCV-238 & HCV-239 CHARGING PUMP HEADER TO REACTOR COOLANT SYSTEM (RCS) ISOLATION VALVES; AND HCV-240 AUXILIARY PRESSURIZER SPRAY ISOLATION VALVE; AND HCV-438 B & D, ISOLATION VALVES FOR COMPONENT COOLING WATER (CCW) TO REACTOR COOLANT PUMP (RCP) SEAL COOLERS. NRC RESIDENT INSPECTORS WERE NOTIFIED OF THE CONCERNS AND A ONE HOUR REPORT TO THE NRC UNDER 10 CFR 50.72(B)(1)(II)(B) WAS MADE AT 1523 ON APRIL 6, 1988. FURTHER EFFORTS CONCERNED IDENTIFYING ALTERNATE METHODS OF PERFORMING THE FUNCTIONS THAT ARE AFFECTED BY THESE VALVES OR SUPPLEMENTING THEIR MOTIVE FORCE (INSTRUMENT AIR) TO MAINTAIN OPERABILITY.

[86] FT. ST. VRAIN DOCKET 50-267 LER 87-004 REV 01
UPDATE ON UNANALYZED LIQUID WASTE RELEASE.
EVENT DATE: 021587 REPORT DATE: 020188 NSSS: GA TYPE: HTGR

(NSIC 208998) CAUSE - PROCEDURAL DEFICIENCY AND COMMUNICATION ERROR. AT 2049 HOURS ON 2/7/87, WITH THE REACTOR SHUTDOWN AND COOLED DOWN FOR ENVIRONMENTAL QUALIFICATION WORK, A LIQUID EFFLUENT RELEASE WAS INITIATED. FORT ST. VRAIN ELCO 8.1.2(B)(1) REQUIRES THAT PRIOR TO A RELEASE, TWO SAMPLES FROM THE RADIOACTIVE LIQUID WASTE SYSTEM SHALL BE ANALYZED FOR GROSS ALPHA AND BETA ACTIVITY, PRINCIPAL GAMMA EMITTERS AND TRITIUM. CONTRARY TO THE ABOVE, THIS ACTION WAS NOT PERFORMED FOR THIS RELEASE DUE TO AN ASSUMPTION THAT THE LIQUID EFFLUENT PRESENT IN THE MONITOR TANK WAS THE SAME FOR WHICH SAMPLES WERE TAKEN ON 2/1/87. HOWEVER, UNKNOWN TO THE OPERATING CREW ON DUTY AT THE TIME OF THE RELEASE ON 2/7/87, THE LIQUID EFFLUENT FROM WHICH THE SAMPLES HAD BEEN TAKEN WAS RELEASED AT APPROXIMATELY 0235 HOURS ON 2/3/87. THIS ERROR WAS NOT DISCOVERED UNTIL APPROXIMATELY 1645 HOURS ON 2/15/87, AT WHICH TIME NRC NOTIFICATION WAS MADE. SINCE TWO SAMPLES WERE NOT OBTAINED AND ANALYZED PRIOR TO THE RELEASE PERFORMED ON FEBRUARY 7, 1987, THIS CONDITION CONSTITUTES A CONDITION PROHIBITED BY THE PLANT'S TECH SPECS. NEW POLICIES AND PROCEDURE CHANGES HAVE BEEN IMPLEMENTED TO PROVIDE MORE SPECIFIC CONTROLS TO ENSURE THAT EACH EFFLUENT RELEASE IS PROPERLY SAMPLED AND ANALYZED.

[87] FT. ST. VRAIN DOCKET 50-267 LER 87-030
TECH SPEC SURVEILLANCE NOT PERFORMED WITHIN REQUIRED INTERVAL.
EVENT DATE: 121387 REPORT DATE: 011288 NSSS: GA TYPE: HTGR

(NSIC 208997) CAUSE - PERSONNEL ERROR. TECH SPEC SURVEILLANCE ESR 8.11.A-M WAS NOT COMPLETED WITHIN THE REQUIRED INTERVAL. THE CAUSE OF THIS EVENT IS PERSONNEL ERROR. RETEST INFORMATION WAS ATTACHED TO THE REGULAR MONTHLY ISSUE OF ESR 8.1.1.A-M, AND THE SURVEILLANCE WAS DESIGNATED FOR RETEST ON THE COMPUTER GENERATED SURVEILLANCE REPORT. THESE TWO FACTORS LED THE SHIFT SUPERVISORS TO BELIEVE THAT THE SURVEILLANCE HAD BEEN ISSUED FOR RETEST ONLY, AND NOT FOR REGULAR MONTHLY PERFORMANCE. SINCE THE RETEST PORTION OF THE SURVEILLANCE COULD NOT BE PERFORMED DUE TO EQUIPMENT DEFICIENCIES, THE SURVEILLANCE WAS NOT ASSIGNED TO APPROPRIATE OPERATIONS PERSONNEL FOR PERFORMANCE. ALTHOUGH TWO RADIATION MONITORS TESTED BY ESR 8.1.1.A-M WERE INOPERABLE AND COULD NOT BE TESTED, REDUNDANT MONITORS WERE SUCCESSFULLY TESTED ON 12/14/87, AND THE REQUIREMENTS OF ELCO 8.1.1.G.1 WERE MET AT ALL TIMES. ESR 8.1.1.A-M WAS COMPLETED ON 12/14/87. THE SCHEDULING TECHNICIAN RESPONSIBLE FOR THE SURVEILLANCE SCHEDULING PROGRAM HAS BEEN INSTRUCTED NOT TO ATTACH RETEST INFORMATION TO SURVEILLANCES THAT ARE ISSUED FOR PERFORMANCE ON THE REGULARLY SCHEDULED INTERVAL, AND ALSO TO REMOVE THE RETEST DESIGNATION FROM THE COMPUTER SURVEILLANCE REPORT WHEN A SURVEILLANCE IS ISSUED FOR PERFORMANCE ON THE REGULARLY SCHEDULED INTERVAL.

[88] FT. ST. VRAIN DOCKET 50-267 LER 87-031
HOT REHEAT TEMPERATURE HIGH SCRAM DUE TO OPERATOR INATTENTION.
EVENT DATE: 122587 REPORT DATE: 012588 NSSS: GA TYPE: HTGR

(NSIC 208999) CAUSE - FLUX CONTROLLER PLACED IN MANUAL. AT 1328 HOURS ON 12/25/87, WITH THE REACTOR OPERATING AT 29% POWER, A HOT REHEAT TEMPERATURE HIGH SCRAM OCCURRED. THE FLUX CONTROLLER (JD)*, WHICH CONTROLS HOT REHEAT STEAM TEMPERATURE, HAD BEEN PLACED IN MANUAL IN RESPONSE TO A PLANT TRANSIENT AND POWER REDUCTION THE PREVIOUS DAY. A XENON TRANSIENT AS A RESULT OF THE POWER REDUCTION WAS CAUSING CORE HELIUM OUTLET TEMPERATURES, HOT REHEAT, AND MAIN STEAM TEMPERATURES TO SLOWLY INCREASE. THE REACTOR OPERATOR ON DUTY WAS INVOLVED WITH THE ONGOING TROUBLESHOOTING OF THE TRIPPED CIRCULATOR AND WAS NOT ATTENTIVE TO THE TRANSIENT XENON TEMPERATURE EFFECT. HOT REHEAT STEAM TEMPERATURES SLOWLY INCREASED UNTIL THE TRIP OCCURRED, AS DESIGNED. FOLLOWING THE SCRAM ALL PLANT CONTROL SYSTEMS (JA)* FUNCTIONED AS DESIGNED AND ACTIVE CORE COOLING WAS MAINTAINED. THE AUXILIARY BOILER (SA)* WAS STARTED AND HELIUM CIRCULATORS (AB)* WERE MAINTAINED IN OPERATION ON AUXILIARY BOILER STEAM. THE PLANT WAS RESTARTED ON 12/27/87. THIS INCIDENT AND ITS SIGNIFICANCE HAVE BEEN REVIEWED WITH ALL OPERATORS, AND AN AUDIBLE ALARM HAS BEEN ADDED TO THE PLANT DATA LOGGER TO GIVE WARNING OF EXCESSIVE STEAM TEMPERATURES. THIS EVENT DOES NOT APPEAR TO INVOLVE A TRAINING DEFICIENCY.

[89] GINNA DOCKET 50-244 LER 88-004
"B" STEAM GENERATOR TUBE LEAK DUE TO MISCALL OF EDDY CURRENT DATA DURING RECENT REFUELING AND MAINTENANCE OUTAGE CAUSES A PLANT SHUTDOWN.
EVENT DATE: 031488 REPORT DATE: 041288 NSSS: WE TYPE: PWR
VENDOR: HUNTINGTON ALLOY COMPANY

(NSIC 208935) ON MARCH 14, 1988 AT 1409 EST WITH REACTOR POWER AT APPROXIMATELY 85% OF FULL POWER A TUBE LEAK WAS DETECTED IN THE "B" STEAM GENERATOR. SUBSEQUENTLY THE PLANT WAS SHUTDOWN TO INSPECT AND REPAIR THE LEAKY TUBE. BECAUSE THE STEAM GENERATOR TUBE LEAK WAS GREATER THAN .1 GPM (I.E. APPROXIMATELY .14 GPM) AN UNUSUAL EVENT WAS DECLARED. THE UNDERLYING CAUSE OF THE EVENT WAS A MISSED CALL FROM THE PRECENT STEADY GENERATOR EDDY CURRENT INSPECTION PROGRAM, (I.E. DIRECT INTERPRETATION WOULD HAVE HAD THE LEAKY TUBE REPAIRED). CORRECTIVE ACTION TAKEN WAS TO RE-REVIEW PERTINENT INSPECTION DATA FROM THE RECENT EDDY CURRENT EXAMINATION, RE-EXAMINE AN APPROPRIATE SAMPLING OF TUBES AND PERFORM A PRESSURE TEST ON THE "B" STEAM GENERATOR. BASED ON THE RESULTS FROM THESE ACTIONS, 9 TUBES WERE PLUGGED INCLUDING THE LEAKING TUBE. CORRECTIVE ACTION PLANNED TO PREVENT RECURRENCE INCLUDES UPDATING THE EDDY CURRENT DATA ANALYSIS PROCEDURES TO ASSURE THAT INDEPENDENT REVIEWS ARE CONDUCTED FOR ALL STEAM GENERATOR EDDY CURRENT INSPECTION DATA.

[90] GRAND GULF 1 DOCKET 50-416 LER 88-008 REV 01
UPDATE ON MSIV-LCS DILUTION AIR INLET FOUND SEALED WITH TAPE.
EVENT DATE: 020888 REPORT DATE: 042988 NSSS: GE TYPE: BWR

(NSIC 209124) ON 02/08/88 AN OPERATOR ON A ROUTINE TOUR OF THE AUXILIARY BUILDING DISCOVERED THAT THE DILUTION FLOW INLET FOR THE OUTBOARD MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM (MSIV-LCS) WAS SEALED WITH DUCT TAPE. THE MSIV-LCS IS DESIGNED TO BE INITIATED MANUALLY FROM THE CONTROL ROOM FOLLOWING A DESIGN BASIS ACCIDENT IN ORDER TO ENSURE ALL MSIV LEAKAGE IS TREATED BY THE STANDBY GAS TREATMENT SYSTEM. DILUTION FLOW FROM THE AUXILIARY BUILDING GENERAL AREA IS MIXED WITH MSIV-LCS SUCTION FLOW FROM ALL FOUR MAIN STEAM LINES IN ORDER TO LOWER THE TEMPERATURE TO PROTECT THE BLOWERS. THE DUCT TAPE WAS IMMEDIATELY REMOVED AND A WALKDOWN INSPECTION OF THE INBOARD AND OUTBOARD MSIV-LCS SUBSYSTEMS WAS PERFORMED. NO OTHER DISCREPANCIES WERE IDENTIFIED. DILUTION FLOW FOR THE INDEPENDENT INBOARD SYSTEM WAS NOT HAMPERED. THE ROOT CAUSE AND DATE OF OCCURRENCE COULD NOT BE ASCERTAINED. IT IS KNOWN THAT THE DILUTION FLOW INLET WAS NOT TAPED ON DECEMBER 17, 1987 WHEN THE 18 MONTH FUNCTIONAL SURVEILLANCE WAS PERFORMED. SIGNS HAVE BEEN PLACED AT THE DILUTION FLOW INLET TO WARN AGAINST OBSTRUCTING THE OPENING. ADDITIONALLY, THE OPERATIONS MONTHLY FUNCTIONAL TEST AND THE 18 MONTH FUNCTIONAL TEST HAVE BEEN CHANGED TO CHECK THAT THE OPENING IS

CLEAR. A REVIEW OF SAFETY RELATED EQUIPMENT IDENTIFIED NO OTHER PROCESS PIPING OPENINGS THAT COULD BE MISTAKENLY SEALED AND POSSIBLY CAUSE A MALFUNCTION OR FAILURE OF THE COMPONENT.

[91] GRAND GULF 1 DOCKET 50-416 LER 88-011
INADVERTENT RHR PUMP START DUE TO PERSONNEL ERROR.
EVENT DATE: 031788 REPORT DATE: 041588 NSSS: GE TYPE: BWR

(NSIC 209025) ON MARCH 17, 1988 ELECTRICIANS WERE CONDUCTING THE MONTHLY FUNCTIONAL TEST OF THE RESIDUAL HEAT REMOVAL (RHR) PUMP B START TIME DELAY RELAY. THE SURVEILLANCE PROCEDURE REQUIRES THE ELECTRICIAN TO DENERGIZE THE TIME DELAY RELAY TO VERIFY THAT THE RELAY IS FUNCTIONING PROPERLY. ENERGIZING THE TIME DELAY RELAY STARTS THE RHR PUMP. THE PROCEDURE PREVENTS THE PUMP FROM STARTING BY REQUIRING THE ELECTRICIAN TO MANUALLY TRIP THE PUMP BREAKER LOCKOUT RELAY BEFORE ENERGIZING THE TIME DELAY RELAY. THE ELECTRICIAN PERFORMING THIS STEP TRIPPED THE LOCKOUT RELAY FOR RHR PUMP C RATHER THAN FOR RHR PUMP B. WHEN THE TIME DELAY RELAY FOR RHR B WAS ENERGIZED DURING THE SUBSEQUENT PROCEDURE STEPS, RHR PUMP B STARTED BECAUSE ITS LOCK-OUT RELAY HAD NOT BEEN TRIPPED. THE BREAKER CUBICLES FOR RHR B AND RHR C ARE LOCATED CLOSE TOGETHER SINCE BOTH ARE ON THE DIVISION 2 BUS. THE ELECTRICIAN HAD MENTALLY TRANSPOSED THE BREAKER NUMBERS LISTED IN THE PROCEDURE INSTRUCTIONS AND DID NOT RECOGNIZE THIS ERROR AT THE BREAKER PANEL BECAUSE THE LABELS FOR RHR PUMP C002B-B AND RHR PUMP C002C-B WERE EXTREMELY SIMILAR. CORRECTIVE ACTIONS INCLUDE PROCEDURAL ENHANCEMENTS, IMPROVED LABELING, AND RETRAINING OF APPROPRIATE PERSONNEL.

[92] HATCH 1 DOCKET 50-321 LER 88-002
PERSONNEL ERROR DURING BACKFILLING OF INSTRUMENT REFERRENCE LEG CAUSES LOW LEVEL SCRAM.
EVENT DATE: 041088 REPORT DATE: 050688 NSSS: GE TYPE: BWR

(NSIC 209269) ON 4/10/88 AT APPROXIMATELY 1018 CDT, UNIT 1 WAS IN COLD SHUTDOWN WITH AN APPROXIMATE POWER LEVEL OF 0 MWT (APPROXIMATELY 0 PERCENT OF RATED THERMAL POWER). NON-LICENSED INSTRUMENTATION AND CONTROL (I&C) PERSONNEL WERE BACKFILLING AN INSTRUMENT REFERENCE LEG TO CORRECT THE OUTPUT OF A FEEDWATER CONTROL SYSTEM (FCS EIIS CODE SJ) REACTOR WATER LEVEL TRANSMITTER. TWO OTHER LEVEL TRANSMITTERS, WHICH SHARE THE REFERENCE LEG, SENSED LOW LEVEL DUE TO THE BACKFILLING. A REACTOR PROTECTION SYSTEM (RPS EIIS CODE JC) ACTUATION OCCURRED, AND THE OUTBOARD VALVES OF PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS EIIS CODE JM) VALVE GROUP 2 ISOLATED. SINCE ALL THE CONTROL RODS WERE ALREADY FULLY INSERTED, NO ACTUAL SCRAM OCCURRED. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR BY THE I&C PERSONNEL WHO DID NOT ADEQUATELY PREPARE FOR THE POTENTIAL EFFECTS OF BACKFILLING THE REFERENCE LEG. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED REVIEWING THE EVENT WITH INVOLVED PERSONNEL, SELECTING THE EVENT FOR REVIEW IN CONTINUING TRAINING FOR I&C PERSONNEL, AND SCHEDULING DEVELOPMENT OF A DEPARTMENT INSTRUCTION ON BACKFILLING OF INSTRUMENTS.

[93] HATCH 2 DOCKET 50-366 LER 88-004 REV 01
UPDATE ON PRIMARY CONTAINMENT PENETRATIONS FAIL LOCAL LEAK RATE TESTS DUE TO NORMAL EQUIPMENT WEAR.
EVENT DATE: 012088 REPORT DATE: 041988 NSSS: GE TYPE: BWR
VENDOR: FISHER CONTROLS CO.
GENERAL ELECTRIC CO.
LONERGAN, J.E., CO.

(NSIC 209015) ON 1/20/88, UNIT 2 WAS IN THE REFUELING MODE OF OPERATION AT AN APPROXIMATE POWER LEVEL OF 0 MWT (APPROXIMATELY 0 PERCENT OF RATED THERMAL POWER), IN PREPARATION FOR A SCHEDULED REFUELING OUTAGE. AT THAT TIME, PLANT PERSONNEL WERE PERFORMING LEAK RATE TESTING ON SOME PRIMARY CONTAINMENT ISOLATION

SYSTEM (PCIS EIIS JM) PENETRATIONS, HATCHES, AND AIRLOCKS AND DETERMINED THAT SOME OF THE PENETRATIONS AND HATCHES WOULD NOT MEET THE LEAKAGE REQUIREMENTS ALLOWED BY THE PLANT'S TECHNICAL SPECIFICATIONS. THE ROOT CA'SE OF THE LEAKAGE IS ATTRIBUTED TO NORMAL EQUIPMENT USE AND WEAR. THE DETAII ' THE CAUSE FOR EACH VALVE FAILURE AND THE CORRECTIVE ACTIONS PERFORMED ARE TOO NUMEROUS TO LIST IN THIS ABSTRACT. THIS INFORMATION IS INCLUDED IN TABLES ATTACHED TO THE BODY OF THIS LICENSEE EVENT REPORT.

[94] HATCH 2 DOCKET 50-366 LER 88-006
PROCEDURE DEFICIENCY CAUSES SCRAM AND ONE VALVE FAILS TO CLOSE ON GROUP 1 ISOLATION.
EVENT DATE: 031888 REPORT DA E: 041888 NSSS: GE TYPE: BWR

(NSIC 209016) ON 03/18/88 AT APPROXIMATELY 0920 CST, UNIT 2 WAS IN THE STARTUP MODE OF OPERATION AT AN APPROXIMATE POWER LEVEL OF 172 MWT (APPROXIMATELY 5 PERCENT OF RATED THERMAL POWER). THE MAIN TURBINE WAS IN THE TRIPPED CONDITION. PLANT PERSONNEL WERE PERFORMING A FUNCTIONAL TEST OF THE TURBINE CONTROL VALVE FAST CLOSURE SCRAM INSTRUMENTATION. THE REMOVAL OF ELECTRICAL LINKS, TO REMOVE THE LESS THAN 30% POWER SCRAM BYPASS FEATURE, RESULTED IN AN UNPLANNED SCRAM. LATER, ONE MAIN STEAM LINE DRAIN ISOLATION VALVE (EIIS SB) FAILED TO CLOSE ON THE EXPECTED GROUP 1 SIGNAL DUE TO LOSS OF CONDENSER VACUUM. THE VALVE DID NOT RECEIVE THE ISOLATION SIGNAL. THE SCRAM WAS CAUSED BY A DEFICIENCY IN THE PROCEDURE, WHICH FAILED TO PROVIDE PROPER INSTRUCTIONS FOR PERFORMING THE FUNCTIONAL TEST. REMOVING THE BYPASS FOR BOTH CHANNELS AT THE SAME TIME, WHEN THE CONTROL VALVE CLOSURE SIGNAL WAS PRESENT, RESULTED IN THE SCRAM. THE CAUSE FOR THE VALVE NOT RECEIVING THE ISOLATION SIGNAL COULD NOT BE DETERMINED. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED ISSUING A PROCEDURE REVISION, INVESTIGATING THE LACK OF ISOLATION SIGNAL TO THE GROUP 1 VALVE, AND DEMONSTRATING THE VALVE WOULD ISOLATE IN RESPONSE TO THE GROUP 1 SIGNAL.

[95] HATCH 2 DOCKET 50-366 LER 88-008
CALIBRATION PROCEDURAL DEFICIENCY FOR FEEDWATER CONTROLLER CAUSES LOW WATER LEVEL SCRAM.
EVENT DATE: 032188 REPORT DATE: 042088 NSSS: GE TYPE: BWR

(NSIC 209017) ON 3/21/88 AT APPROXIMATELY 1644 CST, UNIT 2 WAS IN THE RUN MODE OF OPERATION AT AN APPROXIMATE POWER LEVEL OF 440 MWT (APPROXIMATELY 18% OF RATED THERMAL POWER). IN THE PROCESS OF TRANSFERRING THE FEEDWATER CONTROL SYSTEM (FCS EIIS CODE SJ) FROM STARTUP CONTROL TO SINGLE ELEMENT CONTROL, FLUCTUATIONS IN REACTOR VESSEL WATER LEVEL WERE EXPERIENCED. OPERATIONS PERSONNEL MANUALLY SCRAMMED THE REACTOR DUE TO DECREASING LEVEL AND ALSO RECEIVED A PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS EIIS CODE JM) VALVE GROUP 2 ISOLATION. LEVEL WAS RESTORED USING THE REACTOR FEED PUMPS AND MANUALLY INITIATING REACTOR CORE ISOLATION COOLING (RCIC EIIS CODE BN). THE IMMEDIATE CAUSE FOR THE FCS INSTABILITY WAS IMPROPER SETTINGS ON THE MASTER CONTROL LOOP CONTROLLER (2C32-K636). THE ROOT CAUSE FOR THESE IMPROPER SETTINGS WAS DETERMINED TO BE A DEFICIENT CALIBRATION PROCEDURE. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED CORRECTING THE SETTINGS ON THE CONTROLLER AND SCHEDULING DEVELOPMENT OF A MORE COMPREHENSIVE CALIBRATION PROCEDURE. AS AN ENHANCEMENT, A COMPARABLE PROCEDURE WILL ALSO BE DEVELOPED FOR UNIT 1.

[96] HATCH 2 DOCKET 50-366 LER 88-009
PERSONNEL ERRORS CAUSE MISSED TESTS RESULTING IN CONDITION PROHIBITED BY TECH SPECS.
EVENT DATE: 032188 REPORT DATE: 041888 NSSS: GE TYPE: BWR

(NSIC 209018) ON 03/21/88 AT APPROXIMATELY 1330 CST, UNIT 2 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 405 MWT (APPROXIMATELY 17 PERCENT OF RATED

THERMAL POWER). AT THAT TIME, A LOCAL LEAK RATE TEST (LLRT) COORDINATOR DETERMINED THAT ONE PRIMARY CONTAINMENT PIPING PENETRATION DID NOT RECEIVE AN AS-LEFT LLRT AFTER MAINTENANCE WAS PERFORMED ON THE VALVE. SUBSEQUENT INVESTIGATIONS DETERMINED AN ELECTRICAL PENETRATION ALSO DID NOT HAVE AN AS-LEFT LLRT PERFORMED ON IT. THE ROOT CAUSE OF THESE EVENTS IS PERSONNEL ERROR. IN THE FIRST EVENT, PLANT PERSONNEL FAILED TO FULLY FOLLOW ALL ADMINISTRATIVE CONTROLS RELATIVE TO LLRT TESTING. IN THE SECOND CASE, THE ADMINISTRATIVE CONTROLS WERE FOLLOWED. HOWEVER, THE TEST WAS INADVERTENTLY NOT PERFORMED. CORRECTIVE ACTIONS FOR THESE EVENTS INCLUDED: 1) PERFORMING LLRTS, 2) REVIEWING THE COMPUTER TRACKING LLRT MAINTENANCE WORK ORDER (MWO) DATABASE, 3) REVIEWING MWOS, 4) ISSUING A LETTER TO FURTHER CONTROL LLRT PERFORMANCE, AND 5) SCHEDULING PROCEDURE REVISIONS OR DEVELOPMENT.

[97] HOPE CREEK 1 DOCKET 50-354 LER 87-036 REV 01
UPDATE ON LOSS OF CONTROL POWER TO HIGH PRESSURE COOLING INJECTION, REACTOR HEAT REMOVAL AND CORE SPRAY LOGIC CIRCUITS DUE TO UNKNOWN CAUSE.
EVENT DATE: 080487 REPORT DATE: 042988 NSSS: GE TYPE: BWR
VENDOR: TOPAZ ELECTRONICS

(NSIC 209211) ON AUGUST 4, 1987 AT 2030 HOURS, THE PLANT WAS IN OPERATIONAL CONDITION (POWER OPERATION) AT 100% POWER GENERATING 1067 MWE WHEN SEVERAL "A" CHANNEL CLASS 1E 125 VDC LOADS FROM THE "A" DISTRIBUTION PANEL WERE DE-ENERGIZED. THE BREAKERS FOR THE AFFECTED SYSTEMS WERE THEN OPENED AND RESET. THESE ACTIONS RESTORED POWER TO ALL SYSTEMS THAT HAD LOST 125 VDC. TESTING PERFORMED DURING THE FIRST MIDCYCLE OUTAGE DID NOT DETERMINE THE ROOT CAUSE OF THIS OCCURRENCE. CORRECTIVE ACTIONS INCLUDE PROCEDURE REVISIONS TO PRECLUDE SIMILAR RECURRENCES.

[98] HOPE CREEK 1 DOCKET 50-354 LER 88-005 REV 01
UPDATE ON REQUIRED TECH SPEC ACTIONS NOT TAKEN DUE TO INADEQUATE DESIGN CHANGE PACKAGE PREPARATION DUE TO PROGRAMMATIC DEFICIENCY.
EVENT DATE: 031088 REPORT DATE: 041388 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 208974) ON MARCH 10, 1988 IT WAS DETERMINED THAT DURING THE COURSE OF PERFORMING THREE WORKORDERS ASSOCIATED WITH A MAJOR DESIGN CHANGE, VARIOUS INPUTS TO THE PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) HAD BEEN RENDERED INOPERABLE, AND THAT THE APPROPRIATE TECH SPEC ACTIONS HAD NOT BEEN TAKEN. THE SUBJECT DESIGN CHANGE AND ASSOCIATED WORKORDERS WERE BEING PERFORMED TO REPLACE RELAYS IN VARIOUS SYSTEMS ON A CHANNELIZED BASIS DUE TO POTENTIAL ENVIRONMENTAL QUALIFICATION CONCERNS AS DESCRIBED IN NRC INFORMATION NOTICE 84-20. THE INOPERABLE PCIS INPUTS WERE DISCOVERED DURING REVIEW OF THE COMPLETED WORKORDERS. FOLLOWING DISCOVERY, ALL REMAINING OPEN WORKORDERS ASSOCIATED WITH THE SUBJECT DESIGN CHANGE WERE REVIEWED TO ENSURE THAT TECH SPEC APPLICABILITY WAS PROPERLY IDENTIFIED. THE PRIMARY CAUSE OF THIS OCCURRENCE WAS AN INADEQUATE DESCRIPTION IN THE DESIGN CHANGE PACKAGE (DCP) OF EQUIPMENT THAT WOULD BE AFFECTED DURING THE REPLACEMENT OF EACH RELAY. THIS LED TO THE INADEQUATE PREPARATION OF WORKORDERS ASSOCIATED WITH THE RELAY REPLACEMENTS, AND IN TURN, DID NOT PROVIDE THE WORK CONTROL CENTER WITH ENOUGH INFORMATION TO DETERMINE TECH SPEC OPERABILITY REQUIREMENTS. CORRECTIVE ACTIONS INCLUDE CONTINUING A ONGOING PHASE-IN OF A NEW DCP PROCESS WHICH PROVIDES DEFINITIVE ACCOUNTABILITY FOR REVIEW OF TECH SPEC REQUIREMENTS.

[99] HOPE CREEK 1 DOCKET 50-354 LER 88-006
INTERMEDIATE RANGE MONITOR SPIKE DUE TO WELDING NEAR IRM CABINET CAUSES FULL ACTUATION OF REACTOR PROTECTION SYSTEM WHILE IN THE NON-COINCIDENT MODE.
EVENT DATE: 032188 REPORT DATE: 042088 NSSS: GE TYPE: BWR

(NSIC 209032) WHILE IN OPERATIONAL CONDITION 5 (REFUELING), WITH THE REACTOR

PROTECTION SYSTEM (RPS) IN THE NON-COINCIDENT MODE (SHORTING LINKS REMOVED), A MOMENTARY UPSCALE SPIKE ON "A" INTERMEDIATE RANGE MONITOR (IRM) RESULTED IN A FULL RPS ACTUATION (SCRAM SIGNAL). THE UPSCALE SPIKE ON "A" IRM WAS CAUSED BY ELECTRONIC NOISE GENERATED WHEN TIG WELDING IN THE VICINITY OF THE "A" IRM PRE-AMP CABINET. IN THE NON-COINCIDENT MODE OF RPS, A TRIP SIGNAL FROM ANY SINGLE IRM WILL RESULT IN A FULL RPS ACTUATION. NO CONTROL ROD MOVEMENT OCCURRED, AS ALL RODS WERE FULL-IN AT THE TIME OF OCCURRENCE, AND NO CORE ALTERATIONS WERE IN PROGRESS. ROOT CAUSE ANALYSIS DETERMINED THAT A VARIETY OF FACTORS CONTRIBUTED TO THIS EVENT, ALL RELATED TO THE LACK OF AWARENESS BY THE WELDERS THAT THEY WERE WELDING IN AN ELECTRONICALLY SENSITIVE AREA. AS CORRECTIVE ACTIONS, THE STATION WILL EVALUATE CURRENT METHODS OF CONTROLLING UNDERVESSEL WORK DURING OUTAGE SITUATIONS, AND DETERMINE WHERE IMPROVEMENTS CAN BE MADE. THIS EVALUATION WILL INCLUDE A REVIEW OF METHODS USED IN NOTIFYING PERSONNEL PERFORMING UNDERVESSEL WORK THAT, WHEN IN THE RPS NON-COINCIDENT MODE, EXTREME CAUTION DURING WORK IS REQUIRED.

[100] HOPE CREEK 1 DOCKET 50-354 LER 88-007
RPS ACTUATION CAUSED BY A SPURIOUS UPSCALE SPIKE IN "G" CHANNEL INTERMEDIATE RANGE MONITOR (IRM) - PERSONNEL ERROR.
EVENT DATE: 033088 REPORT DATE: 042888 NSSS: GE TYPE: BWR

(NSIC 209179) ON MARCH 30, 1988 AT 1258 HOURS, THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) WHEN A CHANNEL "G" INTERMEDIATE RANGE MONITOR (IRM) UPSCALE SPIKE CAUSED A REACTOR PROTECTION SYSTEM (RPS) ACTUATION SINCE A HALF RPS TRIP WAS ALREADY IN. ALL ALARMS/INDICATIONS RECEIVED WERE CONSISTENT WITH AN RPS ACTUATION. NO CONTROL ROD MOVEMENT OCCURRED BECAUSE ALL OPERABLE CONTROL RODS WERE INSERTED AT THE TIME OF THE RPS TRIP AND THERE WERE NO CORE ALTERATIONS IN PROGRESS. NO TECH SPEC ACTION STATEMENTS WERE ENTERED BECAUSE THE PLANT WAS IN OPERATIONAL CONDITION 4 DURING THE FIRST REFUELING OUTAGE. THIS EVENT WAS DETERMINED TO HAVE BEEN CAUSED BY WORKERS INSTALLING A DESIGN CHANGE IN THE AREA NEAR THE IRM PREAMPLIFIER CABINET. CORRECTIVE ACTIONS INCLUDE POSTING SIGNS WARNING OF THE PRESENCE OF SENSITIVE INSTRUMENTATION, THE EVALUATION OF A TECH SPEC CHANGE TO REMOVE THE MSL RAD MONITOR TRIP FROM THE RPS SCRAM LOGIC AND THE ADDITION OF INSTRUCTIONS TO IDENTIFY SENSITIVE INSTRUMENTS DURING DESIGN CHANGE WALKDOWNS.

[101] HOPE CREEK 1 DOCKET 50-354 LER 88-008
CAULKING MATERIAL DISCOVERED AROUND REACTOR BUILDING BLOWOUT PANELS CAUSED VIOLATION OF CONFIGURATION CONTROL DUE TO PERSONNEL ERROR.
EVENT DATE: 040188 REPORT DATE: 050288 NSSS: GE TYPE: BWR

(NSIC 209123) ON APRIL 1, 1988 AT 1416 HOURS, THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) AT 0% POWER GENERATING 0 MWE WHEN A REPORT WAS RECEIVED BY THE CONTROL ROOM STAFF OF THE PRESENCE OF CAULKING AROUND THE REACTOR BUILDING EXTERIOR BLOWOUT PANELS LOCATED ON THE REACTOR BUILDING CYLINDER WALL ABOVE El.136'. IT WAS DETERMINED THAT THIS CAULKING HAD BEEN INSTALLED IN VIOLATION OF THE PLANT CONFIGURATION CONTROL. THE REMOVAL OF THE CAULKING WAS INITIATED ON APRIL 1, 1988 AND COMPLETED ON APRIL 7, 1988. THE ROOT CAUSE OF THIS OCCURRENCE WAS THE INSTALLATION OF CAULKING WITHOUT DESIGN CHANGE AUTHORIZATION AND IN VIOLATION OF THE DESIGN BASIS OF THE PLANT. CORRECTIVE ACTIONS INCLUDE REMOVAL OF THE CAULKING AND TRAINING OF THE NUCLEAR DEPARTMENT ENGINEERS IN THE IDENTIFICATION, EVALUATION AND REPORTING OF ANY DISCOVERED DEFICIENCIES IN THE DESIGN, OPERATION OR CONDITION OF ALL NUCLEAR PLANTS OPERATED BY PSE&G.

[102] INDIAN POINT 2 DOCKET 50-247 LER 88-003
ISOLATION VALVE SEAL WATER SYSTEM LEAKAGE EXCEEDS LIMIT.
EVENT DATE: 032388 REPORT DATE: 042388 NSSS: WE TYPE: PWR
VENDOR: ANCHOR VALVE CO.

(NSIC 209115) ON MARCH 23, 1987, DURING A REVIEW OF COMPLETED TEST RESULTS, IT WAS DETERMINED THAT A TECHNICAL SPECIFICATION LIMIT HAD BEEN EXCEEDED. DURING THE 1987 REFUELING OUTAGE, A SURVEILLANCE TEST OF TYPE "B" AND "C" (APPENDIX J) VALVES WAS CONDUCTED FROM OCTOBER THROUGH DECEMBER OF 1987. WHEN TEST RESULTS WERE TOTALED, IT WAS DISCOVERED THAT THE AGGREGATE OF INDIVIDUAL VALVE LEAKAGE EXCEEDED THAT PERMITTED BY THE TECHNICAL SPECIFICATION PERTAINING TO THE LIMIT FOR THE ISOLATION VALVE SEAL WATER SYSTEM. EXCESSIVE LEAKAGE OCCURRED ACROSS TWO LETDOWN CONTAINMENT ISOLATION VALVES (201 AND 202) AS WELL AS A CONTAINMENT ISOLATION VALVE IN THE SAFETY INJECTION SYSTEM (851A). THE VALVES HAVE NOW BEEN REPAIRED AND THE LEAKAGE IS CURRENTLY WELL WITHIN ACCEPTABLE LIMITS.

[103] INDIAN POINT 3 DOCKET 50-286 LER 88-002
REACTOR TRIP, MAIN BOILER FEED PUMP TRIP DUE TO MAIN BOILER FEED PUMP DISCHARGE VALVE LIMIT SWITCH ACTUATION CAUSED BY WATER HAMMER INDUCED VIBRATION.
EVENT DATE: 033188 REPORT DATE: 042188 NSSS: WE TYPE: PWR
VENDOR: CRANE VALVE CO.
ITT-BARTON

(NSIC 209118) ON MARCH 31, 1988 WITH THE REACTOR AT 100% POWER, MAIN BOILER FEED PUMP (MBFP) NO. 32 TRIPPED. OPERATORS RESPONDING TO THE PUMP LOSS TOOK MANUAL CONTROL OF THE PLANT STARTING AUXILIARY BOILER FEED PUMP (ABFP) NO. 31 AND 33 AND INITIATED A TURBINE RUNBACK. STEAM GENERATOR LEVELS INITIALLY DECREASED BUT TURNED AROUND AND STARTED INCREASING REQUIRING THE SHUTDOWN OF THE ABFPS. THE OPERATORS WERE SUBSEQUENTLY UNABLE TO PREVENT AN AUTOMATIC UNIT TRIP DUE TO HIGH STEAM GENERATOR LEVEL. ALL PLANT SYSTEMS FUNCTIONED PROPERLY FOLLOWING THE UNIT TRIP WITH THE EXCEPTION OF THE NO. 33 ABFP RECIRCULATION VALVE WHICH FAILED TO AUTOMATICALLY CLOSE. THE VALVE WAS MANUALLY CLOSED BY THE CONTROL ROOM OPERATOR. IT WAS DETERMINED SUBSEQUENT TO THE TRIP THAT THE MBFP HAD TRIPPED DUE TO A LIMIT SWITCH ON ITS MOTOR OPERATED DISCHARGE VALVE MAKING UP. THIS LIMIT SWITCH MOVEMENT WAS CAUSED BY "WATER HAMMER" VIBRATION. THE NO.33 ABFP FAILURE WAS DETERMINED TO HAVE OCCURRED DUE TO A DAMAGED FLOW CONTROLLER WHICH WAS CAUSED BY A BROKEN GLASS FACEPLATE. NO PROXIMATE CAUSE WAS IDENTIFIED AND THE FACEPLATE WAS REPLACED WITH PLEXIGLAS AND THE CONTROLLER RETURNED TO SERVICE. THE NO. 32 MBFP DISCHARGE VALVE LIMIT SWITCH WAS ADJUSTED TO PROVIDE LESS SENSITIVE INDICATION OFF THE OPEN SEAT AND THE VALVE WAS RETURNED TO SERVICE.

[104] KEWAUNEE DOCKET 50-305 LER 88-001
INSULATION FAILURE AND DIRT ACCUMULATION CAUSE ELECTRICAL BUS BAR FAILURE AND REACTOR TRIP.
EVENT DATE: 030288 REPORT DATE: 040488 NSSS: WE TYPE: PWR
VENDOR: CALVERT COMPANY INC., THE

(NSIC 208849) AT 1348 ON MARCH 2, 1988, WITH THE PLANT AT 93.3% POWER DUE TO END OF LIFE COAST-DOWN PRIOR TO THE ANNUAL REFUELING OUTAGE, A REACTOR/TURBINE TRIP OCCURRED DUE TO AN UNDERVOLTAGE (UV) CONDITION ON 4160 VOLT ELECTRICAL BUSES 1-1 AND 1-2. THESE BUSES SUPPLY POWER TO THE REACTOR COOLANT PUMPS (RXCP) AND THE MAIN FEEDWATER PUMPS (FWP). DURING NORMAL POWER OPERATION, THE MAIN AUXILIARY TRANSFORMER (MAT) SUPPLIES POWER TO THE 4160 VOLT BUSES 1-1 AND 1-2 FROM ITS "Y" WINDINGS. A SECTION OF THE BUS BAR RUNNING FROM THE MAT TO THE BUS SWITCHGEAR WAS BADLY DAMAGED DUE TO INSULATION FAILURE AND A SUBSEQUENT FAULT. THE FAULT ON THE BUS BAR CAUSED AN UNDERVOLTAGE CONDITION OF LESS THAN 77% OF RATED VOLTAGE ON BUSES 1-1 AND 1-2, RESULTING IN A REACTOR TRIP. THE RESERVE AUXILIARY TRANSFORMER (RAT) ASSUMED THE LOADS FOR BUSES 1-1 AND 1-2. BASED ON THE AMOUNT OF SMOKE PRESENT IN THE TURBINE BUILDING THE SHIFT SUPERVISOR ACTIVATED THE

PLANT'S EMERGENCY SIREN WHICH REQUIRED ALL ON-SITE PERSONNEL TO ASSEMBLE FOR ACCOUNTABILITY. THE DIFFERENTIAL CURRENT PROTECTION FUNCTIONED AS DESIGNED AND OPENED ALL BREAKERS ON THE AFFECTED PROTECTION ZONE. THIS DE-ENERGIZED THE AFFECTED BUS AND TERMINATED THE FIRE. THE FIRE TEAM WAS DISPATCHED TO CONTROL THE SITUATION.

[105] KEWAUNEE DOCKET 50-305 LER 88-002
UNPLANNED ACTUATION OF SAFETY INJECTION DUE TO PROCEDURAL WEAKNESS AND PERSONNEL ERROR.
EVENT DATE: 032888 REPORT DATE: 042788 NSSS: WE TYPE: PWR

(NSIC 209119) AT 0854 (CST) ON MARCH 28, 1988, WITH THE PLANT IN A REFUELING SHUTDOWN CONDITION, TWO UNPLANNED SAFETY INJECTION (SI) SIGNALS WERE GENERATED DURING DIESEL GENERATOR SURVEILLANCE TESTING. THIS RESULTED IN ACTIVATION OF ENGINEERED SAFETY FEATURES (ESF) EQUIPMENT. THE PURPOSE OF THIS SURVEILLANCE IS TO TEST THE AUTOMATIC START OF THE DIESEL GENERATORS, LOAD SHEDDING, AND RESTORATION OF MAJOR EQUIPMENT FOR A SIMULTANEOUS BLACKOUT AND SI INITIATION. THIS IS DONE BY MANUALLY LOCKING OUT THE NORMAL POWER SUPPLY TO THE SAFEGUARD BUS UNDER TEST CONCURRENT WITH INITIATION OF THAT TRAIN OF SI. DURING PERFORMANCE OF THE A TRAIN TEST, THE A TRAIN WAS UNBLOCKED IN ACCORDANCE WITH THE PROCEDURE. HOWEVER, THE OPERATOR ALSO MISTAKENLY UNBLOCKED B TRAIN. THIS RESULTED IN AN UNPLANNED B TRAIN SI ACTUATION. SUBSEQUENTLY, DURING ESTABLISHMENT OF INITIAL CONDITIONS TO RE-PERFORM THE B TRAIN TEST, BOTH SI TRAINS AND ASSOCIATED ESF EQUIPMENT ACTIVATED WHEN AUTOMATIC SI WAS RE-ENABLED WITH BOTH TRAINS OF SI STILL UNBLOCKED. THE EVENT WAS CAUSED BY A COMBINATION OF PROCEDURAL WEAKNESSES, AN INADEQUATE PRETEST BRIEFING, AND PERSONNEL LACK OF UNDERSTANDING OF THE TEST SEQUENCE. THE UNPLANNED ACTUATIONS HAD NO SAFETY CONSEQUENCES SINCE HIGH PRESSURE SI WAS ISOLATED FROM THE REACTOR COOLANT SYSTEM, AND UTILIZES THE RECIRCULATION PATH TO THE REFUELING WATER STORAGE TANK.

[106] KEWAUNEE DOCKET 50-305 LER 88-003
INTEGRANULAR ATTACK AND INTERGRANULAR STRESS CORROSION CRACKING RESULT IN DEFECTIVE STEAM GENERATOR TUBES.
EVENT DATE: 040388 REPORT DATE: 050388 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 209254) AT 0104 ON 4/3/88, WITH THE PLANT IN REFUELING SHUTDOWN, THE INSPECTION, PLUGGING, AND REPAIR OF THE STEAM GENERATOR TUBES WERE COMPLETED FOR THE 1988 REFUELING OUTAGE. THE INSPECTION FOUND THAT 42 TUBES IN SG A AND 51 TUBES IN STEAM GENERATOR B (SG B) MET THE TECHNICAL SPECIFICATION DEFINITION OF DEFECTIVE. SINCE THE NUMBER OF DEFECTIVE TUBES EXCEEDED 1 PERCENT OF THE TOTAL NUMBER OF TUBES INSPECTED, BOTH STEAM GENERATORS WERE CATEGORIZED AS C-3. AS REQUIRED BY KEWAUNEE'S TECHNICAL SPECIFICATIONS, THE NUCLEAR REGULATORY COMMISSION WAS NOTIFIED WITHIN 4 HOURS OF DETERMINING THAT A STEAM GENERATOR WAS IN THE C-3 CATEGORY. THE MAJORITY OF THE DEFECTS ARE ASSUMED TO BE CAUSED BY INTERGRANULAR ATTACK AND INTERGRANULAR STRESS CORROSION CRACKING (IGA/IGSCC). HOWEVER, 3 TUBES WERE DAMAGED BY A VIBRATING TUBE LANE BLOCKING DEVICE. EIGHT OF THE DEFECTIVE TUBES IN SG A AND 8 OF THE DEFECTIVE TUBES IN SG B WERE PLUGGED. THE REMAINING DEFECTIVE TUBES WERE SLEEVED. IN ADDITION 9 TUBES IN SG A AND 18 TUBES IN SG B WERE PLUGGED AS A PREVENTIVE MEASURE. TO MINIMIZE THE EFFECTS OF IGA/IGSCC IN THE FUTURE, 1862 NON-DEFECTIVE TUBES WERE SLEEVED. THE REMAINING ACCESSIBLE TUBES WILL BE SLEEVED DURING THE 1989 REFUELING OUTAGE. IN ORDER TO REDUCE THE AMOUNT OF SLUDGE IN THE SGS KEWAUNEE HAS REPLACED COPPER COMPONENTS IN THE SECONDARY SYSTEM WITH STAINLESS STEEL COMPONENTS.

[107] LA SALLE 1 DOCKET 50-373 LER 88-002
TYPE B AND TYPE C TOTAL LEAKAGE EXCEEDED 0.6 LA DURING LEAK RATE TESTING.
EVENT DATE: 031688 REPORT DATE: 041588 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 208976) ON MARCH 16, 1988, WITH UNIT 1 SHUTDOWN FOR REFUELING, IT WAS DETERMINED THAT THE TOTAL AS-FOUND CONTAINMENT LEAKAGE RATE DUE TO TYPE B AND TYPC C TESTING EXCEEDED THE TECH SPEC LIMIT OF 0.6 LA (231.4 SCFH). THE SPECIFIC TESTS THAT IDENTIFIED LEAKAGES OVER THE 0.6 LA LIMIT INVOLVED PRIMARY CONTAINMENT VENTILATION SYSTEM VALVES AND MAIN STEAM SYSTEM DRAIN VALVES. THE CAUSE OF THE LEAKAGE HAS NOT BEEN DETERMINED AT THIS TIME, BUT WORK REQUESTS HAVE BEEN PREPARED TO DETERMINE THE CAUSE. THE VALVES WILL BE REPAIRED AS NECESSARY UNDER THE CONTROL OF THE SAME WORK REQUESTS. A SUPPLEMENT REPORT WILL BE SUBMITTED AT THE END OF THE REFUEL OUTAGE INCLUDING OTHER LEAK RATE TEST FAILURES WHICH MAY HAVE OCCURRED DURING THE OUTAGE. THIS REPORT IS BEING SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(II).

[108] LA SALLE 1 DOCKET 50-373 LER 88-003
TWO INOPERABLE INTERMEDIATE RANGE MONITORS DURING REFUEL DUE TO PERSONNEL ERROR.
EVENT DATE: 032588 REPORT DATE: 042588 NSSS: GE TYPE: BWR

(NSIC 209164) AT APPROXIMATELY 0005 HOURS ON MARCH 17, 1988, UNIT 1 ENTERED OPERATIONAL CONDITION 5 (REFUEL). AT THAT TIME, CHANNEL B OF THE REACTOR PROTECTION SYSTEM (RPS) HAD ONLY TWO OF FOUR INTERMEDIATE RANGE MONITORS (IRM'S) OPERABLE. TECH SPEC 3.3.1 REQUIRES AT LEAST THREE OF FOUR IRM INSTRUMENT CHANNELS PER TRIP SYSTEM BE OPERABLE WHILE IN REFUEL. THIS TECH SPEC DEVIATION WAS IDENTIFIED BY THE SHIFT CONTROL ROOM ENGINEER (SCRE) AT APPROXIMATELY 0100 HOURS ON MARCH 25, 1988 DURING CORE ALTERATIONS. CORE ALTERATIONS WERE IMMEDIATELY SUSPENDED AND THE B TRIP SYSTEM WAS PLACED IN THE TRIPPED CONDITION RESULTING IN A HALF SCRAM ON CHANNEL B OF THE RPS. THE ROOT CAUSE OF THE EVENT WAS PERSONNEL ERROR, POOR ADMINISTRATIVE CONTROLS AND PROCEDURAL INADEQUACY. THE SCRE WHO AUTHORIZED PLACING THE UNIT IN REFUEL FAILED TO RECOGNIZE THAT TECH SPEC 3.3.1 REQUIRED THREE OF FOUR IRM'S PER TRIP SYSTEM IN REFUEL. ALSO, NO PROCEDURE EXISTED WHICH OUTLINED THE REQUIREMENTS FOR TAKING THE REACTOR FROM OPERATIONAL CONDITION 4 (COLD SHUTDOWN) TO REFUEL. AN OPERATING PROCEDURE WILL BE IMPLEMENTED WHICH WILL PROVIDE A CHECKLIST TO ENSURE ALL REQUIREMENTS ARE MET PRIOR TO ENTERING INTO REFUEL.

[109] LA SALLE 2 DOCKET 50-374 LER 88-004
MISSED TECH SPEC SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 032188 REPORT DATE: 042088 NSSS: GE TYPE: BWR

(NSIC 209035) ON MARCH 21, 1988 AT 1130 HOURS WITH UNIT 2 IN OPERATIONAL CONDITION 1 (RUN) AT 35% POWER, LASALLE OPERATING SURVEILLANCE LOS-MS-M1, "MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM (MSIV LCS) BLOWER AND HEATER OPERABILITY TEST," WAS DISCOVERED TO BE PAST ITS CRITICAL TIME AND DATE OF 0820 HOURS ON MARCH 21, 1988. THE MSIV LCS WAS DECLARED INOPERABLE AND THE ACTION STATEMENT FOR TECH SPEC 3.0.3 WAS ENTERED AT 1245 HOURS. FOLLOWING THE SATISFACTORY PERFORMANCE OF THE INBOARD PORTION OF LOS-MS-M1, TECH SPEC 3.0.3 WAS EXITED AT 1322 HOURS. THE OUTBOARD PORTION OF LOS-MS-M1 WAS SATISFACTORILY COMPLETED AT 1355 HOURS. THE PRIMARY CAUSE FOR THE MISSED SURVEILLANCE WAS THE FAILURE OF THE UNIT 2 SHIFT FOREMEN, OVER A PERIOD OF SEVERAL SHIFTS, TO INITIATE THE PERFORMANCE OF LOS-MS-M1 EVEN THOUGH THE SURVEILLANCE WAS PAST DUE AND APPROACHING ITS CRITICAL DUE DATE. TO PREVENT RECURRENCE OF THIS EVENT, THE SHIFT FOREMEN HAVE BEEN TRAINED ON THEIR RESPONSIBILITY TO REVIEW THE COMPUTERIZED SURVEILLANCE SCHEDULE, THE SHIFT TURNOVER AND THE PLANNING SCHEDULE AT LEAST ONCE EACH SHIFT AND A CHECKOFF BLOCK HAS BEEN ADDED TO THE SHIFT FOREMAN'S SHIFT TURNOVER SHEET TO ENSURE THE REVIEW IS PERFORMED. THIS EVENT IS

REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I) DUE TO THE DEVIATION FROM PLANT TECH SPEC.

[110] LA SALLE 2 DOCKET 50-374 LER 88-005
HIGH PRESSURE CORE SPRAY PUMP MINIMUM FLOW BYPASS DIFFERENTIAL PRESSURE SWITCH FOUND BELOW REJECT LIMIT.
EVENT DATE: 041288 REPORT DATE: 051088 NSSS: GE TYPE: BWR
VENDOR: STATIC-O-RING

(NSIC 209298) ON APRIL 12, 1988, AT 0255 HOURS, DURING PERFORMANCE OF ROUTINE SURVEILLANCE, FLOW SWITCH FS-2E22-N006 WAS FOUND WITH A SETPOINT OUT-OF-TOLERANCE BELOW THE REJECT LIMIT FOR ITS APPLICATION. UNIT 2 WAS IN OPERATIONAL CONDITION 1, RUN, AT 60% POWER. THIS SWITCH FUNCTIONS TO PROVIDE MINIMUM FLOW BYPASS FOR THE HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) UNDER LOW FLOW CONDITIONS. THE UNIT 2 HPCS WAS DECLARED INOPERABLE, AT 0330 HOURS, IN ACCORDANCE WITH TECH SPEC 3.5.1 AND A 14 DAY TIMECLOCK ENTERED. ALL OTHER EMERGENCY CORE COOLING SYSTEMS WERE OPERABLE THROUGHOUT THE EVENT. THE OUT-OF-TOLERANCE SETPOINT WAS CAUSED BY SETPOINT DRIFT AND DID NOT NOT INTERFERE WITH THE ABILITY OF THIS SWITCH TO PROPERLY PERFORM THAT FUNCTION. SINCE THE REJECT LIMIT HAD BEEN EXCEEDED, THE SUBJECT FLOW SWITCH WAS REPLACED WITH A NEW, QUALIFIED SWITCH ON APRIL 14, 1988. THE OLD SWITCH WAS SENT TO ITS MANUFACTURER FOR DISASSEMBLY AND INSPECTION. A REPLACEMENT SWITCH WAS INSTALLED, CALIBRATED AND FUNCTIONALLY TESTED SATISFACTORILY. HPCS WAS DECLARED OPERABLE ON APRIL 4, 1988 AT 1310 HOURS. THIS EVENT IS REPORTED TO THE NUCLEAR REGULATORY COMMISSION AS A LICENSEE EVENT REPORT IN COMPLIANCE WITH 10CFR50.73(A)92)(V) DUE TO AN EMERGENCY CORE COOLING SYSTEM BEING DECLARED INOPERABLE AND THE REQUIREMENTS OF I.E. BULLETIN 86-02, "STATIC-OR-RING DIFFERENTIAL PRESSURE SWITCHES."

[111] LIMERICK 1 DOCKET 50-352 LER 88-008
NON-COMPLIANCE WITH TECH SPECS DUE TO MISSING AND INCORRECTLY INSTALLED FIRE RATED PENETRATION CONDUIT SEALS.
EVENT DATE: 032388 REPORT DATE: 042288 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 209134) ON MARCH 23, 1988 AT 1430 HOURS, IT WAS DISCOVERED THAT AN ELECTRICAL CONDUIT, PENETRATING A FIRE RATED FLOOR SEPARATING SAFE SHUTDOWN FIRE AREAS, LACKED FIRE RATED INTERNAL CONDUIT SEALS. INVESTIGATION REVEALED THAT COMPENSATORY FIREWATCH MEASURES WERE NOT IN PLACE FROM RECEIPT OF THE LICENSE ON OCTOBER 26, 1984, THROUGH NOVEMBER, 1986 AS REQUIRED BY TECHNICAL SPECIFICATION 3.7.7. IN NOVEMBER, 1986 AN HOURLY FIRE WATCH PATROL WAS INSTITUTED FOR CONTROL ENCLOSURE FIRE AREAS DUE TO UNIT 2 CONSTRUCTION ACTIVITIES. THE CONDUIT PENETRATES A CONTROL ENCLOSURE FIRE RATED FLOOR BETWEEN ACCESS CORRIDOR ROOM 437, AND THE UNIT 2 CABLE SPREADING ROOM. SUBSEQUENT INSPECTION REVEALED TWO ADDITIONAL INTERNAL CONDUIT FIRE RATED SEALS IN CONDUITS RUNNING FROM ROOM 437 AND TERMINATING AT TRANSFORMERS IN SWITCHGEAR ROOMS 428 AND 434 TO BE INSTALLED INCORRECTLY. IN THE EVENT OF A FIRE IN ANY OF THE AFFECTED AREAS, SAFE SHUTDOWN METHODS WOULD HAVE REMAINED AVAILABLE TO SAFELY SHUTDOWN UNIT 1. THE MISSING FIRE RATED CONDUIT SEALS WERE INSTALLED ON APRIL 15, 1988. A NONCONFORMANCE REPORT AND STARTUP WORK ORDER WERE INITIATED TO DISPOSITION THE INCORRECT SEAL INSTALLATIONS. INVESTIGATION INTO THE CAUSE OF THIS EVENT IS CONTINUING. THE RESULTS OF THE INVESTIGATION AND ANY SUBSEQUENT CORRECTIVE ACTIONS WILL BE PROVIDED IN A SUPPLEMENT TO THIS LER.

[112] LIMERICK 1 DOCKET 50-352 LER 88-009 REV 01
UPDATE ON REACTOR WATER CLEANUP ISOLATION DUE TO HIGH REGENERATIVE HEAT EXCHANGER ROOM TEMPERATURE CAUSED BY A PRESSURE RELIEF VALVE LIFTING.
EVENT DATE: 032488 REPORT DATE: 051388 NSSS: GE TYPE: BWR
VENDOR: LONERGAN, J.E., CO.

(NSIC 209283) ON MARCH 24, 1988, AT 0448 HOURS, AN ISOLATION OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM OCCURRED. THE ISOLATION OCCURRED WHEN THE REGENERATIVE HEAT EXCHANGER ROOM TEMPERATURE SENSING ELEMENT SENSED A TEMPERATURE ABOVE ITS 122 DEGREE FAHRENHEIT SET POINT AND INITIATED A NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) GROUP III, DIVISION I "STEAM LEAK DETECTION HIGH TEMPERATURE" ISOLATION. UPON RECEIPT OF THE ISOLATION SIGNAL, THE INBOARD ISOLATION VALVE (HV-044-1F001) CLOSED AND THE 'A' AND 'B' RWCU PUMPS TRIPPED AS DESIGNED. THE HIGH ROOM TEMPERATURE CONDITION RESULTED WHEN A PRESSURE RELIEF VALVE (PSV-044-108) LIFTED AND THE WATER FROM THE SHELL SIDE OF THE REGENERATIVE HEAT EXCHANGER (10E207) FLASHED TO STEAM. THE ISOLATION WAS RESET AT 1207 HOURS AND RWCU WAS BLOCKED TO REPLACE THE LEAKING PRESSURE RELIEF VALVE PSV-044-108 (LONERGAN MODEL D72G), LOCATED ON THE REGENERATIVE HEAT EXCHANGER. THERE WERE NO ADVERSE CONSEQUENCES ASSOCIATED WITH THIS EVENT AND THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT. ENGINEERING HAS BEEN REQUESTED TO INVESTIGATE AND DETERMINE THE FAILURE MODE OF THE PRESSURE SAFETY VALVES AND SUGGEST CORRECTIVE ACTIONS TO PREVENT RECURRENCE. A SUPPLEMENTAL REPORT WILL BE ISSUED AFTER THE CAUSE OF THE VALVE FAILURE HAS BEEN DETERMINED.

[113] LIMERICK 1 DOCKET 50-352 LER 88-010
INITIATION OF AN ENGINEERED SAFETY FEATURE DUE TO A REACTOR ENCLOSURE ISOLATION CAUSED BY A PERSONNEL ERROR.
EVENT DATE: 032688 REPORT DATE: 042588 NSSS: GE TYPE: BWR

(NSIC 209181) ON MARCH 26, 1988 AT 0241 HOURS A REACTOR ENCLOSURE ISOLATION OCCURRED ON LOW DIFFERENTIAL PRESSURE BETWEEN THE REACTOR ENCLOSURE AND OUTSIDE AIR. THE ISOLATION CAUSED REACTOR ENCLOSURE RECIRCULATION SYSTEM AND STANDBY GAS TREATMENT SYSTEM, ENGINEERED SAFETY FEATURES, TO INITIATE AS DESIGNED. A VALVING ERROR DURING RETURN TO SERVICE OF THE INSTRUMENT AIR SYSTEM CAUSED SYSTEM PRESSURE TO DROP, ULTIMATELY RESULTING IN THE ISOLATION. AFTER BEING ALERTED BY AIR HEADER LOW PRESSURE ALARMS, A BACK-UP AIR SUPPLY WAS VALVED IN. AT 0247 HOURS, SIX MINUTES AFTER THE ISOLATION OCCURRED, THE ISOLATION WAS RESET AND REACTOR ENCLOSURE VENTILATION RETURNED TO SERVICE. THE CAUSE OF THE VALVING ERROR WAS A PERSONNEL ERROR BY A UTILITY EMPLOYED NON-LICENSED OPERATOR TO PROPERLY FOLLOW AN IMPLEMENTING PROCEDURE. THE OPERATOR INVOLVED WAS COUNSELED REGARDING THE CARE AND DILIGENCE REQUIRED IN FOLLOWING PROCEDURES. IN ADDITION, THIS EVENT AND THE PROCEDURES ASSOCIATED WITH IT WILL BE REVIEWED IN TRAINING FOR NON-LICENSED PERSONNEL BY JUNE, 1988. THERE WERE NO ADVERSE CONSEQUENCES AND NO RELEASE OF RADIATION OCCURRED AS A RESULT OF THIS EVENT.

[114] LIMERICK 1 DOCKET 50-352 LER 88-011
RWCI DUE TO HIGH DIFFERENTIAL FLOW CAUSED BY A PRESSURE TRANSIENT WHILE REMOVING A FILTER DEMINERALIZER FROM SERVICE.
EVENT DATE: 032688 REPORT DATE: 042588 NSSS: GE TYPE: BWR

(NSIC 209182) ON MARCH 26, 1988 AT 0615 HOURS, A REACTOR WATER CLEANUP (RWCU) ISOLATION OCCURRED ON A NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS), AN ENGINEERING SAFETY FEATURE, GROUP III CHANNEL A AND D "HIGH DIFFERENTIAL FLOW" ISOLATION SIGNAL. OPERATIONS PERSONNEL, REMOVING THE 'A' RWCU FILTER DEMINERALIZER FROM SERVICE IN ORDER TO RESTART THE 'C' RWCU PUMP, OPENED THE FILTER DEMINERALIZER BYPASS VALVE HV-44-1F044 IN CONJUNCTION WITH CLOSING THE 'A' FILTER DEMINERALIZER FLOW CONTROL VALVE FV-C-45-1-66A. THE FILTER DEMINERALIZER BYPASS VALVE HV-44-1F044 COULD NOT BE ADJUSTED TO PERMIT THE MINIMUM REQUIRED SYSTEM FLOW. THIS CAUSED THE 'A' RWCU PUMP TO TRIP ON LOW SUCTION FLOW, RESULTING IN A PRESSURE TRANSIENT, WHICH CAUSED A HIGH DIFFERENTIAL FLOW INITIATING AN NSSSS GROUP III, A AND D CHANNEL "HIGH DIFFERENTIAL FLOW" ISOLATION SIGNAL. THE RWCU INBOARD AND OUTBOARD ISOLATION VALVES HV-044-1F001 AND HV-044-1F004 CLOSED ISOLATING RWCU. THE ISOLATION WAS RESET AT 0640 HOURS AND RWCU WAS RETURNED TO NORMAL OPERATION AT 0905 HOURS. THERE WERE NO ADVERSE CONSEQUENCES ASSOCIATED WITH THIS EVENT AND THERE WAS NO RELEASE OF RADIOACTIVE

MATERIAL. THE FILTER DEMINERALIZER BYPASS VALVE HV44-1F044 HAS BEEN SCHEDULED T. BE REPLACED, DURING AN OUTAGE OF SUFFICIENT LENGTH, WITH A VALVE MORE APPROPRIA FOR THIS APPLICATION AFTER THE NEW VALVE HAS BEEN PROCURED.

[115] LIMERICK 1 DOCKET 50-352 LER 88-012
REACTOR PROTECTION SYSTEM ACTUATION ON INTERMEDIATE RANGE MONITOR UPSCALE DUE TO PERSONAL ERROR.
EVENT DATE: 040988 REPORT DATE: 050688 NSSS: GE TYPE: BWR

(NSIC 209280) ON APRIL 9, 1988 AT 0415 HOURS AN ACTUATION OF THE REACTOR PROTECTION SYSTEM, AN ENGINEERED SAFETY FEATURE OCCURRED. A CONTROLLED SHUTDOWN WAS IN PROGRESS AND CONTROL ROD INSERTION WAS STOPPED AT 0400 HOURS TO FACILITATE A GRADUAL REACTOR DEPRESSURIZATION TO MINIMIZE OFFGAS RELEASES. APPROXIMATELY THREE MINUTES PRIOR TO THE SCRAM, POWER BEGAN INCREASING DUE TO THE POSITIVE REACTIVITY EFFECT OF DECREASING MODERATOR TEMPERATURE. REACTOR POWER, INITIALLY 27 ON RANGE 2, ROSE TO 108 ON RANGE 2 WHERE AN INTERMEDIATE RANGE MONITOR HIGH NEUTRON FLUX ALARM INITIATED, AND CAUSED THE SCRAM AT 120 ON RANGE 2. REACTING TO THE ALARM, THE REACTOR OPERATOR WAS UNABLE TO "UPRANGE" ALL THE INTERMEDIATE RANGE MONITORS BEFORE THE SCRAM OCCURRED SIX SECONDS AFTER THE ALARM. ALL SYSTEMS ACTUATED AS DESIGNED AND THE SCRAM WAS RESET AT 0423 HOURS. THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL TO THE ENVIRONMENT AS A RESULT OF THIS EVENT. THE CAUSE OF THE EVENT WAS A COGNITIVE PERSONAL ERROR WHEN THE UTILITY EMPLOYED LICENSED REACTOR OPERATOR DID NOT ADEQUATELY ANTICIPATE AND OBSERVE THE EFFECTS OF DECREASING MODERATOR TEMPERATURE ON CORE REACTIVITY. TO PREVENT RECURRENCE, THE OPERATOR WAS CAUTIONED TO BE CONSTANTLY AWARE OF THOSE CONDITIONS WHICH AFFECT CORE REACTIVITY.

[116] LIMERICK 1 DOCKET 50-352 LER 88-013
FULL SCRAM SIGNAL GENERATED DUE TO AN INTERMEDIATE RANGE MONITOR (IRM) DETECTOR FAILURE.
EVENT DATE: 040988 REPORT DATE: 050388 NSSS: GE TYPE: BWR
VENDOR: REUTER-STOKES ELECTRIC COMPANY

(NSIC 209236) ON APRIL 9, 1988 AT 1536 HOURS WITH THE UNIT IN COLD SHUTDOWN (REACTOR COOLANT TEMPERATURE = 171 DEGREES F) AND A MANUAL HALF-SCRAM IN PLACE DUE TO A TECHNICAL SPECIFICATION REQUIREMENT, A FULL-SCRAM SIGNAL WAS RECEIVED WHEN AN INTERMEDIATE RANGE MONITOR (IRM) DETECTOR FAILED. THE IRM SPIKED UPSCALE AND COMPLETED THE LOGIC REQUIRED FOR A FULL SCRAM. THERE WAS NO CONTROL ROD MOTION AS A RESULT OF THIS EVENT DUE TO ALL CONTROL RODS HAVING BEEN PREVIOUSLY INSERTED WHEN ENTERING THE SHUTDOWN MODE. THE CAUSE OF THIS EVENT WAS DETERMINED TO BE THE FAILURE OF AN IRM DETECTOR. THE DEFECTIVE IRM WAS BYPASSED WHEN IT WAS DETERMINED TO BE DEFECTIVE. AN INSTRUMENTATION AND CONTROL ENGINEER CHECKED THE VICINITY OF THE IRM DRAWER AND PREAMP BUT FOUND NO WORK IN PROGRESS WHICH MAY HAVE CAUSED THE PROBLEM. THERE ARE NO SPECIFIC ACTIONS TO PREVENT RECURRENCE FOR THIS EVENT. REPLACEMENT IS THE NORMAL COURSE OF ACTION FOR A DEFECTIVE IRM DETECTOR.

[117] MAINE YANKEE DOCKET 50-309 LER 88-003
SAFETY SYSTEM MOTOR OPERATED VALVE HAS LESS CONSERVATIVE THRUST VALVES.
EVENT DATE: 032188 REPORT DATE: 042088 NSSS: CE TYPE: PWR
VENDOR: LIMITORQUE CORP.

(NSIC 209170) DURING THE 1987 REFUELING OUTAGE, MAINE YANKEE CONTRACTED MOVATS INCORPORATED TO PERFORM MOTOR-OPERATED VALVE (MOV) TESTING. MAINE YANKEE OBTAINED MINIMUM REQUIRED MOV THRUST VALUES FROM MOVATS AND LIMITORQUE. THE MOST CONSERVATIVE THRUST VALUES WERE USED TO DETERMINE THE TORQUE SWITCH TRIP SETPOINTS. SINCE THE OUTAGE, MOVATS HAS EXPANDED THEIR DATABASE AND IMPROVED THEIR STATISTICAL ANALYSIS METHOD AND PROVIDED NEW MINIMUM THRUST VALUES TO MAINE

YANKEE. THE TWO VOLUME CONTROL TANK OUTLET ISOLATION MOVS, CH-M-1 AND CH-M-87, HAD CLOSING TORQUE SWITCH SETTINGS LESS CONSERVATIVE THAN SPECIFIED BY THE NEW THRUST VALUES. AN ENGINEERING EVALUATION DETERMINED THAT CH-M-1 WOULD HAVE FUNCTIONED PROPERLY DURING A DESIGN BASIS ACCIDENT, BUT COULD NOT DETERMINE WHETHER CH-M-87 WOULD HAVE CLOSED FULLY UNDER THE SAME CONDITION. AS A RESULT, THE REDUNDANCY PROVIDED BY THE TWO VALVES IN SERIES MAY NOT HAVE BEEN MAINTAINED. THE TORQUE SWITCH SETTINGS FOR CH-M-1 AND CH-M-87 WERE RESET TO DEVELOP CLOSING THRUST GREATER THAN THE NEW MOVATS VALUES. THERE WAS NO SIGNIFICANT SAFETY CONCERN FOR THE FOLLOWING REASONS: 1) ACTUAL DIFFERENTIAL PRESSURE WOULD HAVE BEEN MUCH LESS THAN THE ASSUMED MAXIMUM. 2) CH-M-1 AND CH-M-87 ARE IN SERIES SO CH-M-1 CLOSING PERFORMS THE FUNCTION FOR BOTH TRAINS. 3) SINCE THE VALVES ARE IN SERIES, THE DIFFERENTIAL PRESSURE IS DISTRIBUTED WHEN BOTH CLOSE.

[118] MAINE YANKEE DOCKET 50-309 LER 88-004
POTENTIALLY DEGRADED FIRE DOORS.
EVENT DATE: 040188 REPORT DATE: 050288 NSSS: CE TYPE: PWR

(NSIC 209255) ON APRIL 1, 1988, MAINE YANKEE IDENTIFIED SEVEN FIRE BARRIER DOORS WHICH MAY NOT HAVE PERFORMED THEIR INTENDED FUNCTION UNDER ALL CIRCUMSTANCES. THE SEVEN DOORS WERE DESIGNED TO UNLATCH ON LOSS OF POWER TO THE SOLENOID OPERATED LATCHING MECHANISM. THE NATIONAL FIRE CODES REQUIRE THAT FIRE BARRIER DOORS BE EQUIPPED WITH LATCHING MECHANISMS TO ENSURE THAT THE DOORS ARE LATCHED DURING A FIRE. IT IS POSTULATED THAT AN INTENSE FIRE COULD RESULT IN FAILURE OF NON-QUALIFIED WIRING DISRUPTING POWER TO THE SOLENOID LATCHING MECHANISM PERMITTING THE DOORS TO UNLATCH. AS A PRECAUTIONARY MEASURE, MAINE YANKEE ESTABLISHED A ROVING FIRE WATCH, WITH NO OTHER ASSIGNED DUTIES, TO ENSURE THERE WERE NOT FIRE HAZARDS IN THE AFFECTED AREAS. BY APRIL 6, 1988, MODIFICATIONS WERE COMPLETED SO THE DOORS REMAIN LATCHED ON LOSS OF POWER TO THE SOLENOID LATCHING MECHANISM.

[119] MCGUIRE 1 DOCKET 50-369 LER 87-033 REV 01
UPDATE ON FOUR MAIN STEAM TO AUXILIARY EQUIPMENT VALVES WERE OMITTED FROM THE INSERVICE VALVE TESTING PROGRAM DUE TO INCORRECT DETERMINATION CAUSED BY PERSONNEL ERROR.
EVENT DATE: 120887 REPORT DATE: 041888 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 208986) ON 12/08/87, AN EVALUATION OF MCGUIRE WAS BEING PERFORMED BY NRC ANALYSIS AND EVALUATION OF OPERATIONAL DATA PERSONNEL (NRC). NRC FOUND THAT VALVES 1SA-5 AND 2SA-5, MAIN STEAM 1C AND 2C TO AUXILIARY FEEDWATER PUMP TURBINE CHECK, AND 1SA-6 AND 2SA-6, MAIN STEAM 1B AND 2B TO AUXILIARY FEEDWATER PUMP TURBINE CHECK, HAD BEEN OMITTED FROM THE INSERVICE VALVE TESTING PROGRAM. ON 12/09/87 AT 1600, OPERATIONS (OPS) DECLARED UNIT 1 AND 2 TURBINE DRIVEN AUXILIARY FEEDWATER (CA) PUMPS INOPERABLE. WORK REQUESTS WERE WRITTEN BY OPS TO VERIFY THAT THE FOUR VALVES WERE IN THE FULLY CLOSED POSITION. ON 12/09/87, THE FREEDOM OF TRAVEL WAS MEASURED ON ALL FOUR VALVES BY OPERATIONS AND DETERMINED TO BE WITHIN THE VALVE MANUFACTURER'S ACCEPTABLE TOLERANCE. THE TURBINE DRIVEN CA PUMPS WERE RETURNED TO OPERABLE STATUS AT 1900. A CAUSE OF PERSONNEL ERROR IS ASSIGNED TO THIS EVENT BECAUSE THE VALVES WHICH SHOULD HAVE BEEN INCLUDED IN THE INSERVICE VALVE TESTING PROGRAM WERE OMITTED WHEN ENGINEERS, WHO INITIALLY DEVELOPED THE PROGRAM, MISINTERPRETED THE DESIGN FUNCTION OF THE VALVES AND FAILED TO RECOGNIZE THE VALVES PERFORMED A FUNCTION REQUIRED TO ACHIEVE A SAFE SHUTDOWN CONDITION. TEST REQUIREMENTS WILL BE DETERMINED AND NECESSARY PROCEDURE CHANGES WILL BE MADE FOR THESE VALVES.

[120] MCGUIRE 1 DOCKET 50-369 LER 88-003
TRAIN B OF THE COMPONENT COOLING SYSTEM WAS INOPERABLE DUE TO A NUCLEAR SERVICE WATER SYSTEM VALVE BEING IN A NONCONSERVATIVE POSITION DUE TO LOOSE TRAVEL STOPS.
EVENT DATE: 030988 REPORT DATE: 040888 NSSS: WE TYPE: PWR
VENDOR: FISHER CONTROLS CO.

(NSIC 208942) ON 03/09/88 AT 1330, PERFORMANCE PERSONNEL WERE INSPECTING THE PROGRESS OF SOME VALVE WORK AND DISCOVERED THE COMPONENT COOLING (KC) HEAT EXCHANGER (HX) 1B CONTROL, VALVE 1RN-190B, WITH THE TRAVEL STOPS LOOSE AND IN A NONCONSERVATIVE POSITION. PERFORMANCE IMMEDIATELY NOTIFIED THE CONTROL ROOM SENIOR REACTOR OPERATOR (SRO) THAT VALVE 1RN-190B WOULD NOT ALLOW THE PROPER AMOUNT OF NUCLEAR SERVICE WATER (RN) SYSTEM FLOW THROUGH THE 1B KC HX. THE SRO DECLARED KC TRAIN 1B INOPERABLE AT 1336 ON 03/09/88. PERFORMANCE RESET THE TRAVEL STOPS TO THE POSITION THAT WAS DETERMINED DURING THE MOST RECENT RN SYSTEM FLOW BALANCE ON JANUARY 29, 1988. PERFORMANCE TIGHTENED THE HEX NUTS FOR THE TRAVEL STOP ADJUSTMENT TO ENSURE THE HEX NUTS WOULD STAY IN POSITION. AT 1420 ON 03/09/88 THE SRO RETURNED KC TRAIN 1B TO OPERABLE STATUS. A CAUSE OF UNKNOWN HAS BEEN ASSIGNED TO THIS EVENT BECAUSE THIS INVESTIGATION COULD NOT DETERMINE WHEN OR HOW THE HEX NUTS HAD COME LOOSE FROM THE SECURED POSITION. LOCTITE THREAD SEALANT WILL BE APPLIED TO THE THREADS OF THE HEX NUTS FOR THE TRAVEL STOP ADJUSTMENTS ON VALVE 1RN-190B AND AN EVALUATION WILL BE PERFORMED REGARDING THE NEED FOR LOCTITE THREAD SEALANT ON VALVES 2RN-190B, AND 1 AND 2RN-89A. APPROPRIATE MAINTENANCE PERSONNEL WILL BE REQUIRED TO READ A MEMO WARNING PERSONNEL.

[121] MCGUIRE 1 DOCKET 50-369 LER 88-004
UNIT 1 AND UNIT 2 ENTERED TECH SPEC 3.0.3 WHEN EVCB VITAL BATTERY CHARGER WAS DEENERGIZED DUE TO UNKNOWN REASONS WHILE EVCA VITAL BATTERY WAS INOPERABLE.
EVENT DATE: 030988 REPORT DATE: 040888 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 208943) ON 03/09/88 AT APPROXIMATELY 1050, OPERATIONS (OPS) ATTEMPTED TO SWITCH THE POWER SUPPLY FOR EVCB VITAL BATTERY CHARGER FROM UNIT 1 TO UNIT 2 SO THAT EMERGENCY POWER SUPPLY TO EVCB WOULD BE FROM DIESEL GENERATOR (D/G) 2B INSTEAD OF D/G 1B. (D/G 1B WAS TO BE DECLARED INOPERABLE FOR SCHEDULED MAINTENANCE ON NUCLEAR SERVICE WATER (RN) SYSTEM TRAIN 1B.) DURING THE ATTEMPTED SWITCH, THE UNIT 2 POWER SUPPLY BREAKER TRIPPED. OPS RESET THE BREAKER BUT COULD NOT GET THE POWER RESTORED TO EVCB THROUGH THE UNIT 2 BREAKER OR BY RECLOSING THE UNIT 1 POWER SUPPLY BREAKER. OPS DECLARED THE EVCB CHARGER INOPERABLE AT 1100 AND ENTERED TECHNICAL SPECIFICATION (TS) 3.0.3 DUE TO EVCA BATTERY ALREADY BEING INOPERABLE FOR TESTING. OPS DISCOVERED THAT THE EVCB CHARGER ALTERNATING CURRENT (AC) INPUT BREAKER HAD ALSO TRIPPED. AFTER OPS RESET THE UNIT 1 EVCB POWER SUPPLY BREAKER AND EVCB CHARGER AC INPUT BREAKER AND THE CHARGER OUTPUT VOLTAGE WAS DECREASED SLIGHTLY, OPS SUCCESSFULLY STARTED THE EVCB CHARGER AT 1112. OPS DECLARED THE EVCB CHARGER OPERABLE AT 1114. THIS EVENT IS ASSIGNED A CAUSE OF OTHER BECAUSE TS 3.0.3 WAS ENTERED DUE TO A PROBLEM OF UNKNOWN ORIGIN WITH THE POWER SUPPLY BREAKERS FOR EVCB VITAL BATTERY CHARGER. FURTHER INVESTIGATION WILL BE COMPLETED AND A SUPPLEMENTAL REPORT WILL BE SUBMITTED IF THE ORIGIN OF THE PROBLEM IS DETERMINED.

[122] MCGUIRE 1 DOCKET 50-369 LER 88-005
SPURIOUS TRAIN A SAFETY INJECTION AND MAIN STEAM SYSTEM ISOLATION RESULTING IN A REACTOR TRIP DUE TO UNKNOWN CAUSES.
EVENT DATE: 032388 REPORT DATE: 042288 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 209180) ON 03/23/88 AT APPROXIMATELY 1051, INSTRUMENTATION AND ELECTRICAL (IAE) PERSONNEL, WHILE PERFORMING A MONTHLY FUNCTIONAL TEST, CLOSED THE UNIT 1 SOLID STATE PROTECTION SYSTEM (SSPS) TRAIN A LOGIC CABINET DOOR AND INSTANTLY

HEARD A RELAY ENERGIZE. OPERATIONS (OPS) WAS IMMEDIATELY ALERTED TO A TURBINE TRIP/REACTOR TRIP ANNUNCIATOR AND INDICATION OF CONTROL ROD INSERTION. AMONG THE NUMEROUS ALARMS AND INDICATIONS, A SAFETY INJECTION (SI) ACTUATION AND A MAIN FEEDWATER (CF) SYSTEM ISOLATION HAD OCCURRED. THE CF ISOLATION TRIPPED CF PUMPS 1A AND 1B DUE TO HIGH DISCHARGE PRESSURE, WHICH LED TO THE UNIT 1 TURBINE TRIP/REACTOR TRIP. ALSO, A UNIT 1 MAIN STEAM (SM) SYSTEM ISOLATION OCCURRED, BUT WITH NO SUPPORTING ISOLATION SIGNALS PRESENT. OPS IMPLEMENTED REACTOR TRIP, SAFETY INJECTION, AND NOTIFICATION OF UNUSUAL EVENT PROCEDURES. THE TRAIN A SI AND SM ISOLATION SIGNALS WERE RESET. UNIT 1 WAS STABILIZED NEAR NO LOAD CONDITIONS AT APPROXIMATELY 1120 ON 03/23/88. THIS EVENT WAS ASSIGNED A CAUSE OF OTHER SINCE THE CAUSE OF THE UNIT 1 TRAIN A SI AND SM ISOLATION SIGNALS COULD NOT BE DETERMINED. IAE WILL INSPECT THE SSPS TRAIN A LOGIC CABINET MORE CLOSELY DURING AN UPCOMING OUTAGE. THIS EVENT WILL BE REVIEWED DURING SEGMENT 4 OF LICENSED OPERATORS REQUALIFICATION TRAINING.

[123] MCGUIRE 1 DOCKET 50-369 LER 88-006
A NUCLEAR SERVICE WATER SYSTEM ISOLATION VALVE WAS NOT TESTED FOLLOWING MAINTENANCE AS REQUIRED BY TECH SPEC 4.0.5 DUE TO PERSONNEL ERROR.
EVENT DATE: 032888 REPORT DATE: 042788 NSSS: WE TYPE: PWR
VENDOR: PACIFIC VALVES, INC.

(NSIC 209231) ON 03/28/88, MAINTENANCE (MNT) WAS PREPARING TO ADJUST THE PACKING GLANDS ON VALVE 1RN-21A, NUCLEAR SERVICE WATER (RN) STRAINER 1A BACKFLUSH AUTO SUPPLY ISOLATION, AND VALVE 1RN-22A, RN STRAINER 1A BACKFLUSH AUTO DRAIN. WHILE REVIEWING THE WORK REQUESTS (WR) PRIOR TO STARTING WORK, MNT NOTICED THAT ON 02/04/88, THE PACKING GLAND ON VALVE 1RN-21A HAD BEEN ADJUSTED, AND THAT A VALVE STROKE TIMING TEST HAD NOT BEEN PERFORMED AFTER THE JOB WAS COMPLETED. THIS TEST IS REQUIRED TO BE PERFORMED AFTER ANY MAINTENANCE THAT MAY ALTER THE STROKE TIMING OF THE VALVE. CONTROL ROOM PERSONNEL WERE NOTIFIED. AT THE TIME OF THE DISCOVERY, TRAIN A OF THE UNIT 1 RN SYSTEM HAD BEEN DECLARED INOPERABLE BECAUSE PERFORMANCE (PRF) WAS CONDUCTING A FLOW BALANCE TEST. THE WR FOR VALVE 1RN-21A WAS ADDED TO THE UNIT 1 TECH. SPEC. ACTION ITEM LOGBOOK FOR TRAIN A OF THE UNIT 1 RN SYSTEM ON 03/28/88 AT 2300. ON THE MORNING OF 03/29/88, PRF SUCCESSFULLY CONDUCTED A STROKE TIMING TEST OF VALVE 1RN-21A. TRAIN A OF THE UNIT 1 RN SYSTEM WAS RETURNED TO OPERABLE STATUS AT 1330 ON 03/29/88, WHEN THE FLOW BALANCE TEST WAS COMPLETED. A CAUSE OF PERSONNEL ERROR WAS ASSIGNED TO THIS EVENT BECAUSE PLANNER A FAILED TO RECOGNIZE THE NEED FOR VALVE IRN-21A TO HAVE A STROKE TIMING TEST AFTER MAINTENANCE AND DID NOT NOTE THE RETEST ON THE WR.

[124] MCGUIRE 2 DOCKET 50-370 LER 86-001 REV 01
UPDATE ON REACTOR TRIP ON STEAM GENERATOR LOW LOW LEVEL.
EVENT DATE: 011586 REPORT DATE: 041988 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 1 (PWR)

(NSIC 208982) ON JANUARY 15, 1986 AT 1149, THE UNIT 2 REACTOR TRIPPED FROM 80% POWER AS A RESULT OF LOW LOW LEVEL IN STEAM GENERATOR (S/G) "A". THE S/G LOW LOW LEVEL WAS CAUSED BY REDUCED MAIN FEEDWATER (CF) FLOW AFTER CF PUMP TURBINE 2A TRIPPED AT 1147 ON LOW CONDENSER VACUUM. WHEN THE FEEDWATER PUMP TURBINE TRIPPED, A RUNBACK OF THE TURBINE AND REACTOR WAS INITIATED. THE LOSS OF CONDENSER VACUUM WAS DUE TO A MALFUNCTION OF TWO VALVES IN THE VACUUM PRIMING SYSTEM ASSOCIATED WITH CF PUMP 2A. THE CONDENSER OF CF PUMP 2B TURBINE WAS ALSO AFFECTED RESULTING IN A LOSS OF TURBINE SPEED AND A REDUCTION OF PUMP DISCHARGE PRESSURE. S/G LEVELS DECREASED THROUGH THE TRANSIENT UNTIL THE AUTOMATIC REACTOR TRIP SETPOINT ON S/G A WAS REACHED. THE MAIN TURBINE AUTOMATICALLY TRIPPED WHEN THE REACTOR TRIP WAS INITIATED SINCE REACTOR POWER WAS GREATER THAN 48%. UNIT 2 WAS IN MODE 1 AT 100% POWER WHEN CF PUMP TURBINE 2A TRIPPED AND AT 80% POWER WHEN THE REACTOR/TURBINE TRIPS OCCURRED. PLANT SYSTEMS RESPONDED AS EXPECTED FOR THE TRANSIENT. AFTER THE REACTOR TRIP, THE TWO SUBJECT VALVES WERE CLEANED, REPAIRED, REINSTALLED, AND RETURNED TO SERVICE. ALSO, A PERMANENT DRAIN LINE WAS

ADDED TO BOTH UNITS ON ONE VALVE PER CONDENSER TO ALLOW CONTINUOUS WATER FLOW FROM THE VACUUM PRIMING VALVE TO THE FLOOR DRAIN TO PREVENT THIS TYPE OF TRANSIENT.

[125] MCGUIRE 2 DOCKET 50-370 LER 87-009 REV 01
UPDATE ON REACTOR TRIP BREAKER FAILURE DUE TO MECHANICAL FAILURE.
EVENT DATE: 070287 REPORT DATE: 032588 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 1 (PWR)
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 208706) AT 2341 ON JULY 2, 1987 DURING PERFORMANCE OF THE CONTROL ROD DROP TIMING TESTS, PERSONNEL DETECTED SMOKE IN THE AREA OF THE REACTOR TRIP SWITCHGEAR. THE CONTROL ROOM WAS NOTIFIED AND OPERATORS MANUALLY TRIPPED THE REACTOR TRIP BREAKERS (RTBS). CONTROL ROOM STATUS LIGHTS INDICATED BOTH BREAKERS HAD OPENED, THOUGH INVESTIGATION REVEALED RTB 2 TO BE CLOSED. THE BREAKER COULD NOT BE OPENED LOCALLY UNTIL AN ATTEMPT WAS MADE TO MANUALLY TENSION THE BREAKER CLOSURE SPRING. THE OPERATORS WERE NOT HOLDING THE FEEDWATER ISOLATION RESET BUTTON WHEN THE BREAKER DID OPEN SO A TRAIN B FEEDWATER ISOLATION OCCURRED, THOUGH IT DID NOT CAUSE ANY ADVERSE EFFECTS. THE FAILURE OF THE BREAKER HAS BEEN CLASSIFIED AS A MANUFACTURING DEFICIENCY DUE TO A FABRICATION DEFICIENCY CAUSING THE FAILURE OF A WELD INSIDE THE BREAKER. THE INVESTIGATION HAS REVEALED THAT THE BREAKER FAILED TO AUTOMATICALLY OPEN DUE TO A MECHANICAL BINDING OF THE BREAKER. A WELD FAILURE AND WORN COMPONENTS OF THE BREAKER CLOSURE MECHANISM ARE SUSPECTED OF CAUSING THE BINDING, BUT NOTHING CONCLUSIVE HAS BEEN FOUND DURING THE INVESTIGATION AT MCGUIRE WHICH PINPOINTS THE CAUSE. THE BREAKER WILL UNDERGO FURTHER INSPECTIONS AND TESTS AT WESTINGHOUSE.

[126] MILLSTONE 1 DOCKET 50-245 LER 87-029 REV 01
UPDATE ON STANDBY GAS TREATMENT INOPERABILITY.
EVENT DATE: 072487 REPORT DATE: 032988 NSSS: GE TYPE: BWR

(NSIC 208834) AT APPROXIMATELY 0330 ON 7/24/87, DURING REFUELING OPERATIONS, ATMOSPHERIC CONTROL SYSTEM (VA) VALVE 1-AC-10 WAS TAGGED OUT AND REMOVED FOR ROUTINE MAINTENANCE. 1-AC-10 SERVES TO ISOLATE THE STANDBY GAS TREATMENT (SBGT) (BH) SYSTEM FROM PRIMARY CONTAINMENT. THE LIMIT SWITCH ON 1-AC-10 WAS LEFT ELECTRICALLY CONNECTED AND THE CONTACT POSITION OF THE LIMIT SWITCH CORRESPONDS TO THAT OF 1-AC-10 IN THE OPEN POSITION. VALVES 1-SG-1A AND 1-SG-1B ARE ELECTRICALLY INTERLOCKED WITH 1-AC-10 SUCH THAT WITH 1-AC-10 OPEN, 1-SG-1A AND 1-SG-1B REMAIN CLOSED. UPON RECOGNIZING THE PROBLEM WITH 1-SG-1A AND 1-SG-1B, OPERATIONS PERSONNEL IMMEDIATELY TERMINATED FUEL HANDLING OPERATIONS AND POSITIONED THE LIMIT SWITCH TO REFLECT 1-AC-10 IN THE CLOSED POSITION. THE SBGT SYSTEM WAS THEN RETURNED TO SERVICE.

[127] MILLSTONE 1 DOCKET 50-245 LER 88-004
POTENTIAL FOULING OF ECCS SUCTION STRAINERS.
EVENT DATE: 031888 REPORT DATE: 041288 NSSS: GE TYPE: BWR
VENDOR: CHICAGO BRIDGE AND IRON COMPANY

(NSIC 208936) ON MARCH 18, 1988, WHILE OPERATING AT 100% POWER (529 DEGREES FAHRENHEIT, 1032 PSIG) PLANT PERSONNEL WERE NOTIFIED BY NORTHEAST UTILITIES SERVICE COMPANY (NUSCO) OF AN ENGINEERING EVALUATION WHICH HAD DETERMINED THAT THERE WAS A POTENTIAL FOR THE EMERGENCY CORE COOLING SYSTEM (ECCS) PUMP SUCTION STRAINERS TO FOUL WITH FIBROUS INSULATION DEBRIS. FOLLOWING A LOSS OF COOLANT ACCIDENT (LOCA) FIBROUS INSULATION DEBRIS MIGHT FOUL THE TORUS STRAINERS AND REDUCE THE AVAILABLE NET POSITIVE SUCTION HEAD (NPSH) BELOW THE MINIMUM REQUIRED FOR PUMP OPERABILITY. A JUSTIFICATION FOR CONTINUED OPERATION (JCO) CONCLUDED THAT CONTINUED OPERATION IS JUSTIFIED BASED ON AN EXTREMELY LOW PROBABILITY OF A LOCA IN THE DRYWELL DURING THE REMAINDER OF THE CURRENT FUEL CYCLE AND MORE

REALISTIC ASSUMPTIONS FOR PREDICTING INSULATION DEBRIS FOULING OF THE ECCS SUCTION STRAINERS WHICH RESULTS IN AN ACCEPTABLE NPSH FOR THE ECCS PUMPS. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(VI).

[128] MILLSTONE 3 DOCKET 50-423 LER 88-012
FAILURE TO MONITOR AN INOPERABLE FIRE BOUNDARY DOOR.
EVENT DATE: 031888 REPORT DATE: 041888 NSSS: WE TYPE: PWR

(NSIC 209026) ON MARCH 18, 1988, AT 1900 HOURS, WHILE OPERATING IN MODE 1 AT 99% POWER, 2250 PSIA AND 586 DEGREES FAHRENHEIT, A FIRE BOUNDARY DOOR WAS FOUND BLOCKED OPEN BY A TEMPORARY AIR HOSE WITHOUT AN HOURLY FIRE WATCH PATROL BEING ESTABLISHED AS REQUIRED BY THE PLANT'S TECHNICAL SPECIFICATIONS. THE DOOR SEPARATES THE AUXILIARY BUILDING GENERAL AREA FROM THE MOTOR CONTROL CENTER/ROD CONTROL AIR CONDITIONER AREA. THE DISCOVERY WAS MADE BY AN UNLICENSED OPERATOR (PEO) DURING ROUTINE ROUNDS. AFTER THE PEO DETERMINED THAT THE HOSE WAS NO LONGER NECESSARY, HE REMOVED THE HOSE AND CLOSED THE DOOR. ROOT CAUSE OF THE EVENT WAS PROCEDURAL DEFICIENCY. THERE WAS NO MEANS FOR PERSONNEL INVOLVED TO READILY DETERMINE THAT THE DOOR WAS A FIRE RATED ASSEMBLY. AS ACTION TO PREVENT RECURRENCE, THE ADMINISTRATIVE PROCEDURE ON WORK PRACTICES HAS BEEN REVISED TO INCLUDE A LISTING OF THE ATTRIBUTES OF DOORS. A MEMO HAS BEEN DISTRIBUTED TO UNIT PERSONNEL DISCUSSING THE INCIDENT AND THE GUIDELINES WHICH HAVE BEEN ESTABLISHED TO PREVENT RECURRENCE OF THIS EVENT. AS DISCUSSED IN LER 87-048-00, FAILURE TO MONITOR INOPERABLE FIRE ASSEMBLIES, A PROGRAM HAS BEEN INSTITUTED TO MARK ALL PLANT DOORS WITH THEIR ATTRIBUTES BY DECEMBER 31, 1988.

[129] MILLSTONE 3 DOCKET 50-423 LER 88-013
INCOMPLETE INSTALLATION OF DAMPER CIRCUIT IN THE HYDROGEN RECOMBINER SYSTEM.
EVENT DATE: 032888 REPORT DATE: 042788 NSSS: WE TYPE: PWR

(NSIC 209257) ON MARCH 28, 1988, AT 1230 HOURS, WITH THE PLANT IN MODE 1 AT 100%, AN ENGINEERING REVIEW DETERMINED THAT THE INCOMPLETE INSTALLATION OF TWO CIRCUITS IN THE HYDROGEN RECOMBINER SYSTEM WAS A REPORTABLE EVENT. THE CIRCUITS FOR THE RECOMBINER INLET AND EXHAUST AIR DAMPERS WERE NOT INSTALLED IN ACCORDANCE WITH THE LATEST DESIGN DOCUMENT. THE REMAINING WORK CONSISTED OF INSTALLING INTERPOSING RELAYS ON FOUR MOTOR OPERATED DAMPERS TO ENABLE DAMPER OPERATION DURING AN UNDERVOLTAGE CONDITION. THE CONDITION OF NON-COMPLIANCE WITH THE DESIGN EXISTED FROM INITIAL REQUIRED OPERATION OF THE HYDROGEN RECOMBINERS IN JANUARY, 1986, THROUGH NOVEMBER, 1987. THE ROOT CAUSE OF THIS EVENT IS HUMAN ERROR. ENGINEERING ERRONEOUSLY REMOVED THE DESIGN DOCUMENT FROM THE ACTIVE WORK LIST ON THE PLANT'S WORK TRACKING PROGRAM WITHOUT CORRECTLY VERIFYING THAT THE WORK WAS DONE. THE FAILURE TO COMPLETE THE INSTALLATION OF THE INTERPOSING RELAYS IN THE HYDROGEN RECOMBINER SYSTEM PRIOR TO PLANT STARTUP IS NOT CONSIDERED A SERIOUS DEGRADATION OR COMPROMISE TO PLANT SAFETY. BUT THE POSSIBILITY OF AN UNCONTROLLED RADIOACTIVE RELEASE EXISTED IF THE DAMPERS HAD BEEN MANUALLY ALIGNED AFTER A DESIGN BASIS ACCIDENT, COINCIDENT WITH AN UNDERVOLTAGE CONDITION, AND A RADIATION LEAK IN THE HYDROGEN RECOMBINER AIR HEAT EXCHANGERS.

[130] MONTICELLO DOCKET 50-263 LER 88-002
FAILURE TO PROVIDE FIRE WATCH DUE TO INADEQUATE PROCEDURES AND TRAINING.
EVENT DATE: 031488 REPORT DATE: 041388 NSSS: GE TYPE: BWR
VENDOR: YALE

(NSIC 208926) DURING A PLANT INSPECTION, A FIRE DOOR LOCATED IN AN APPENDIX "R" FIRE BARRIER WAS FOUND IN THE OPEN POSITION DUE TO AN INOPERABLE SELF-CLOSING MECHANISM. THE CONDITION OF THIS MECHANISM HAD BEEN NOTED PREVIOUSLY BY MEMBERS OF THE PLANT STAFF, BUT THE REQUIRED FIRE WATCH HAD NOT BEEN ESTABLISHED SINCE IT WAS NOT RECOGNIZED THAT AN OPERABLE SELF-CLOSING MECHANISM WOULD CAUSE A FIRE DOOR TO BE DECLARED INOPERABLE. THE FIRE WATCH IS REQUIRED PER TECH SPEC

3.13.G.2. THE SELF-CLOSING MECHANISM WAS REPAIRED SHORTLY AFTER THE INOPERABLE CONDITION OF THIS DOOR WAS IDENTIFIED. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE INADEQUACIES IN PROCEDURES AND TRAINING. THE DAILY FIRE DOOR INSPECTION PROCEDURE WAS REVISED TO PROVIDE MORE SPECIFIC INSTRUCTIONS TO AID OPERATORS IN IDENTIFYING INOPERABLE FIRE DOORS, A LETTER WAS ISSUED BY THE PLANT MANAGER TO ALL PLANT PERSONNEL TO BRING TO THEIR ATTENTION THE IMPORTANCE OF FIRE DOORS AND TO REMIND THEM OF THEIR RESPONSIBILITIES IN THE USE OF THESE DOORS. ALSO, THE TRAINING DEPARTMENT WILL BE REQUESTED TO PROVIDE FURTHER TRAINING ON WHAT CONSTITUTES AN OPERABLE FIRE DOOR AND A PREVENTATIVE MAINTENANCE PROCEDURE WILL BE PREPARED TO INSURE THAT THESE DOORS ARE KEPT IN A BETTER STATE OF REPAIR.

[131] MONTICELLO DOCKET 50-263 LER 88-003
MISSED SURVEILLANCE OF EDG LOAD SEQUENCING DUE TO PERSONNEL ERROR.
EVENT DATE: 032988 REPORT DATE: 042888 NSSS: GE TYPE: BWR
VENDOR: STRUTHERS DUNN, INC.

(NSIC 209216) IT WAS DETERMINED ON MARCH 29, 1988 THAT THE EMERGENCY FILTRATION TRAIN SYSTEM (EFT) LOADS HAD NOT BEEN VERIFIED TO SEQUENCE ONTO THE EMERGENCY DIESEL GENERATORS (EDG) DURING THE 1984, 1986, AND 1987 REFUELING OUTAGES AS REQUIRED BY TECHNICAL SPECIFICATION 4.9.B.3.C.2 THE ROOT CAUSE WAS PERSONNEL ERROR DURING THE MODIFICATION PROCESS IN NOT IDENTIFYING THE EDG LOADING PROCEDURE TO BE REVISED. THIS IS CONTRARY TO THE MODIFICATION PROCEDURES. CORRECTIVE ACTIONS TAKEN INCLUDE TESTING THE EFT LOAD SEQUENCING LOGIC WHICH IDENTIFIED FIVE FAILED RELAYS. A HOLD WAS PLACED ON THE EDG LOADING SURVEILLANCE TO INSURE IT IS REVISED BEFORE NEXT USAGE. THE EFT MODIFICATION WAS REVIEWED TO ENSURE 11 OTHER TECHNICAL SPECIFICATION REQUIREMENTS WERE PROPERLY IDENTIFIED. THE EDG LOADING WAS REVIEWED TO VERIFY ALL OTHER SEQUENCED LOADS ARE PROPERLY TESTED. A TRAINING REQUEST WAS INITIATED TO PROVIDE TRAINING ON THE LESSON OF THIS EVENT. PLANNED CORRECTIVE ACTIONS INCLUDE PROVIDING ADDITIONAL GUIDANCE IN THE MODIFICATION PROCEDURES TO ENSURE ALL AFFECTED TECH. SPECS. ARE REVIEWED WHEN IDENTIFYING SURVEILLANCE REQUIREMENTS. A SAMPLE OF COMPLETED MODIFICATIONS AFFECTING SAFETY ELATED SYSTEMS WILL BE REVIEWED TO ENSURE THAT ALL TESTING REQUIREMENTS WERE PROPERLY IDENTIFIED. FOLLOW-UP ACTIONS ON THE FAILED RELAYS WILL BE TAKEN WHICH WILL INCLUDE FAILURE ANALYSIS AND ACCELERATED TESTING.

[132] NINE MILE POINT 1 DOCKET 50-220 LER 87-028 REV 01
UPDATE ON MANUAL REACTOR SCRAM INITIATED DUE TO FEEDWATER PIPING VIBRATION.
EVENT DATE: 121987 REPORT DATE: 042188 NSSS: GE TYPE: BWR
VENDOR: EAGLE SIGNAL
FISHER CONTROLS CO.
ITT GRINNELL
WORTHINGTON PUMP CORP.

(NSIC 209125) AT 18:13:56 ON DECEMBER 19, 1987, NINE MILE POINT UNIT 1 (NMP1) WAS MANUALLY SCRAMMED DUE TO EXCESSIVE VIBRATION OBSERVED IN THE HIGH PRESSURE FEEDWATER PIPING. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE FATIGUE FRACTURE OF THE STEM/VALVE PLUG ASSEMBLY OF FLOW CONTROL VALVE 13A DUE TO FLOW-INDUCED VIBRATION. THE VALVE FAILURE CAUSED PRESSURE SURGES WHICH RESULTED IN VIBRATION OF THE FEEDWATER PIPING AND DAMAGE TO ADJACENT PIPE SUPPORTS. THE PRESSURE-RETAINING BOUNDARY OF THE FEEDWATER SYSTEM WAS NOT DAMAGED BY THIS TRANSIENT. A DETAILED VISUAL INSPECTION OF THE FEEDWATER SYSTEM PIPING AND SUPPORTS WAS CONDUCTED. NON-DESTRUCTIVE EXAMINATION OF SELECTED PIPE WELDS IN THE FEEDWATER SYSTEM WHICH WERE SUBJECTED TO THE HIGHEST STRESSES DURING THE TRANSIENT WAS PERFORMED WITH NO DISCREPANCIES NOTED. THE STEM/VALVE PLUG ASSEMBLY HAS BEEN REPLACED WITH A NEW DESIGN WHICH SHOULD PROVIDE GREATER RELIABILITY. THIS SUPPLEMENT IS BEING SUBMITTED TO DESCRIBE THE FINAL DETERMINATION OF THE ROOT CAUSE, THE RESULTS OF THE INSPECTIONS AND, ANALYSES COMPLETED, AND THE CORRECTIVE ACTION TAKEN. AN ADDITIONAL SUPPLEMENT IS NOT EXPECTED TO BE SUBMITTED.

[133] NINE MILE POINT 1 DOCKET 50-220 LER 88-007
AUTOMATIC INITIATION OF REACTOR BUILDING EMERGENCY VENTILATION DUE TO A PROCEDURAL DEFICIENCY.
EVENT DATE: 031388 REPORT DATE: 041288 NSSS: GE TYPE: BWR

(NSIC 208917) ON 3/13/88, WITH NINE MILE POINT UNIT #1 (NMP1) IN A REFUELING OUTAGE, OPERATIONS PERSONNEL WERE REMOVING A LPRM FROM THE REACTOR CORE. WHILE USING THE REFUEL BRIDGE MONORAIL HOIST TO REMOVE AN LPRM ASSEMBLY, THE CABLE TRAVEL LIMIT WAS OVERRIDDEN TO SUCCESSFULLY DRAW THE ASSEMBLY THROUGH THE UPPER CORE GRID. HOWEVER, THE ASSEMBLY WAS INADVERTENTLY DRAWN TOO CLOSE TO THE SURFACE OF THE REACTOR CAVITY POOL AND THE INCREASED RADIATION LEVEL INITIATED THE REACTOR BUILDING EMERGENCY VENTILATION SYSTEM. THE CAUSE OF THIS EVENT WAS DETERMINED TO BE A PROCEDURAL DEFICIENCY AS THE PROCEDURE DID NOT REQUIRE RESETTING OF THE MONORAIL HOIST TRAVEL LIMIT STOP TO PERFORM LPRM REMOVAL. CONTRIBUTING FACTORS INCLUDED HABIT INTRUSION, AN INABILITY TO SEE THE CABLE MARKING TAPE, AND RADIATION PROTECTION PERSONNEL NOT NOTIFIED TO MONITOR POOL SURFACE RADIATION LEVELS. IMMEDIATE CORRECTIVE ACTION CONSISTED OF LOWERING THE LPRM ASSEMBLY INTO THE POOL AND DIRECTING ASSISTING PERSONNEL TO LEAVE THE REFUEL BRIDGE. REACTOR BUILDING EMERGENCY VENTILATION WAS THEN SECURED AND THE REACTOR BUILDING VENTILATION RETURNED TO SERVICE. LONG TERM CORRECTIVE ACTION WILL INCLUDE REVISING THE FUEL HANDLING PROCEDURE, CHANGING THE COLOR OF THE CABLE MARKING TAPE, ISSUING A "LESSONS LEARNED TRANSMITTAL", A TRAINING MODIFICATION REQUEST, AND A PROBLEM REPORT TO MODIFY THE TRAVEL LIMIT MECHANISM.

[134] NINE MILE POINT 1 DOCKET 50-220 LER 88-001 REV 01
UPDATE ON TECHNICAL SPECIFICATION VIOLATION DUE TO ISI PROGRAM DEFICIENCIES.
EVENT DATE: 031588 REPORT DATE: 041488 NSSS: GE TYPE: BWR

(NSIC 208916) ON 1/15/88, WITH NINE MILE POINT UNIT 1 (NMPL) AT 0% POWER AND THE MODE SWITCH IN REFUEL, A REVIEW OF THE FIRST TEN YEAR INSERVICE INSPECTION (ISI) INTERVAL WAS COMPLETED. THIS REVIEW IDENTIFIED SEVERAL INSPECTION DEFICIENCIES REQUIRING RESOLUTION, I.E. FAILURE TO COMPLETE THE FIRST TEN YEAR INTERVAL INSPECTION REQUIREMENTS AND FAILURE TO PROPERLY DISPOSITION DEFICIENCY/CORRECTIVE ACTION (DCA) NOTICES AND OTHER EXAMINATION RESULTS. ON MARCH 15 AND 25, 1988, NIAGARA MOHAWK WAS INFORMED OF ADDITIONAL MISSED EXAMINATIONS FROM THE FIRST TEN YEAR ISI INTERVAL. BY NOT COMPLETING AND RESOLVING ALL THE INSPECTION REQUIREMENTS REQUIRED BY SECTION XI OF THE ASME BOILER AND PRESSURE VESSEL CODE, A VIOLATION OF TECH SPEC 3.2.6 HAS OCCURRED. THE ROOT CAUSE OF THE EVENT IS MANAGEMENT INEFFECTIVENESS IN IMPLEMENTING THE ISI PROGRAM PLAN. INITIAL CORRECTIVE ACTION WAS TO DETERMINE WHAT OMITTED ITEMS ARE NECESSARY TO COMPLETE THE FIRST TEN YEAR ISI INTERVAL. ALL THESE ITEMS WILL BE COMPLETED DURING THE PRESENT REFUELING OUTAGE. ALSO, ALL THE OUTSTANDING DCA'S AND INSPECTION REPORTS WILL BE PROPERLY DISPOSITIONED. THE LONG-TERM CORRECTIVE ACTION IS TO REALIGN THE ORGANIZATIONAL RESPONSIBILITIES SO THAT CLEAR ACCOUNTABILITY IS MAINTAINED. NECESSARY PROCEDURES AND PROGRAM PLAN CHANGES WILL BE MADE TO ALIGN WITH THE ORGANIZATIONAL STRUCTURE.

[135] NINE MILE POINT 1 DOCKET 50-220 LER 88-008
CONTAINMENT PENETRATION FAILS LOCAL LEAK RATE TEST.
EVENT DATE: 031788 REPORT DATE: 041588 NSSS: GE TYPE: BWR
VENDOR: CIRCLE SEAL

(NSIC 208934) ON MARCH 17, 1988, WITH NINE MILE POINT UNIT 1 (NMP1) IN A REFUELING OUTAGE AND PRIMARY CONTAINMENT INTEGRITY NOT REQUIRED, A PRIMARY CONTAINMENT PENETRATION FAILED ITS LOCAL LEAK RATE TEST. THIS TEST WAS BEING PERFORMED IN ACCORDANCE WITH 10 CFR 50 APPENDIX J REQUIREMENTS. THE PENETRATION IS FOR THE TRANSVERSE IN-CORE PROBE (TIP) NITROGEN SUPPLY SYSTEM. THE ROOT CAUSE FOR THE TIP ISOLATION VALVE FAILURES HAS NOT BEEN DETERMINED AT THIS TIME. IMMEDIATE CORRECTIVE ACTIONS FOR THE FAILURE WAS TO DECLARE THE VALVES AND SYSTEM

INOPERABLE, WRITE AN OCCURRENCE REPORT (OR) TO DOCUMENT THE FAILURE, AND GENERATE A WORK REQUEST (WR) TO CORRECT THE PROBLEM. NO ADDITIONAL ACTION HAS BEEN TAKEN ON THE TIP SYSTEM VALVES AS SPARE PARTS ARE NOT AVAILABLE AT THIS TIME. A SUPPLEMENTAL REPORT WILL BE SUBMITTED BY JUNE 30, 1988, AND WILL INCLUDE A DETERMINATION OF THE ROOT CAUSE AND THE CORRECTIVE ACTION TAKEN.

[136] NINE MILE POINT 1 DOCKET 50-220 LER 88-011
SUMMATION OF LOCAL LEAK RATE TESTS EXCEED REGULATORY LIMIT.
EVENT DATE: 032588 REPORT DATE: 042588 NSSS: GE TYPE: BWR
VENDOR: AEROQUIP CORP.
ANCHOR/DARLING VALVE CO.

(NSIC 208995) ON MARCH 25, 1988, WITH NINE MILE POINT UNIT 1 (NMP1) IN A REFUELING OUTAGE, IT WAS DISCOVERED THAT THE SUM OF THE TYPE B AND C TEST RESULTS EXCEEDED THE ALLOWABLE LIMIT (0.60 LA) AS DEFINED IN 10 CFR 50 APPENDIX J. UNDER THIS CONDITION, THE PRIMARY CONTAINMENT IS CONSIDERED INOPERABLE WITH RESPECT TO PROVIDING A LEAKAGE BOUNDARY AND REPRESENTS A DEGRADATION OF A PRINCIPAL SAFETY BARRIER. THE SUM OF THE TEST RESULTS WAS APPROXIMATELY 16,000 SCFD AT 35 PSIG, WHEREAS, 0.60 LA FOR NMP1 IS 9274.84 SCFD AT 35 PSIG. THE ROOT CAUSE OF THIS EVENT IS THE AMOUNT OF LEAKAGE THROUGH THOSE VALVES THAT FAILED THEIR LOCAL LEAK RATE TESTS. THERE WERE 18 VALVES AND ONE PENETRATION THAT FAILED AND SIGNIFICANTLY CONTRIBUTED TO THE OVERALL LEAKAGE RATE VALUE. ALSO DISCOVERED WAS A PROCEDURAL DEFICIENCY EXISTING IN THE CONTROLLING PROCEDURE AS IT DOES NOT REQUIRE A RUNNING TOTAL OF TEST RESULTS BE KEPT. THIS RESULTED IN A DELAY IN IDENTIFYING WHEN THE REGULATORY LIMIT WAS EXCEEDED. CORRECTIVE ACTIONS FOR THE PENETRATION AND VALVE FAILURES CONSISTED OF DECLARING THE COMPONENTS INOPERABLE AND PERFORMING THE ASSOCIATED ADMINISTRATIVE FUNCTIONS. THESE COMPONENTS HAVE BEEN, OR WILL BE, REPAIRED OR REPLACED AND RETESTED DURING THE CURRENT REFUELING OUTAGE.

[137] NINE MILE POINT 1 DOCKET 50-220 LER 88-009
FIRE BARRIER PENETRATIONS FILLED WITH NONQUALIFIED MATERIAL DUE TO PERSONNEL ERROR, INADVERTENT OMISSION.
EVENT DATE: 032688 REPORT DATE: 042588 NSSS: GE TYPE: BWR

(NSIC 208994) ON MARCH 26, 1988, WITH NINE MILE POINT UNIT 1 (NMP1) IN A REFUELING OUTAGE AND THE CORE OFFLOADED, SIX NONFUNCTIONAL FIRE BARRIER PENETRATIONS WERE DISCOVERED. SUBSEQUENT FIELD WALKDOWNS FOUND AN ADDITIONAL FIFTEEN NONFUNCTIONAL TECHNICAL SPECIFICATION (TECH. SPEC.) FIRE BARRIER PENETRATIONS. THE ROOT CAUSE OF THIS EVENT IS PERSONNEL ERROR. THESE PENETRATIONS WERE INADVERTENTLY OVERLOOKED DURING PREVIOUS PENETRATION WORK. IN ACCORDANCE WITH NMP1 TECH. SPECS. SECTION 3.6.10.1C, FIRE DETECTION WAS VERIFIED ON ONE SIDE OF THE PENETRATIONS AND A FIRE WATCH PATROL WAS ESTABLISHED. SUBSEQUENT CORRECTIVE ACTIONS INCLUDED THE ISSUANCE OF A PROBLEM REPORT TO TRACK PROGRESS IN CORRECTING THE PROBLEM. ACTIONS WILL BE TAKEN TO REVIEW ISSUED FIRE BARRIER AS-BUILT TRANSMITTALS AND THE BALANCE OF THE TECH. SPEC. BARRIERS PRIOR TO THE END OF THE 1988 REFUELING OUTAGE. IN ADDITION, REPAIR OF THE TWENTY PENETRATIONS AND ALL NONFUNCTIONAL TECH. SPEC. FIRE BARRIER PENETRATIONS FOUND WILL BE REPAIRED IN ACCORDANCE WITH EXISTING PROCEDURES PRIOR TO THE END OF THE 1988 OUTAGE. THIS REPORT SATISFIES LICENSEE EVENT REPORT AND NMP1 TECH. SPEC. SPECIAL REPORT SUBMISSION REQUIREMENTS.

[138] NINE MILE POINT 2 DOCKET 50-410 LER 87-063
REACTOR WATER CLEANUP SYSTEM ISOLATION ON HIGH DIFFERENTIAL FLOW SIGNAL DUE TO CONSTRUCTION AND DESIGN DEFICIENCIES.
EVENT DATE: 101387 REPORT DATE: 111087 NSSS: GE TYPE: BWR

(NSIC 209196) ON OCTOBER 13, 1987 AT 1234 HOURS, NINE MILE POINT UNIT 2

EXPERIENCED ACTUATION OF AN ENGINEERED SAFETY FEATURE, SPECIFICALLY, ISOLATION OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM. AT THE TIME OF THE EVENT THE PLANT WAS IN A HOT SHUTDOWN CONDITION WITH THE REACTOR MODE SWITCH IN "SHUTDOWN." REACTOR PRESSURE AND TEMPERATURE WERE APPROXIMATELY 477 POUNDS PER SQUARE INCH GAUGE AND 479 DEGREES FAHRENHEIT, RESPECTIVELY. THIS ISOLATION WAS INITIATED BY A DIFFERENTIAL FLOW SIGNAL WHICH OCCURRED AS A RESULT OF ERRATIC FLOW OSCILLATIONS DURING STARTUP OF THE RWCU SYSTEM. THE ROOT CAUSE FOR THIS EVENT IS CONSTRUCTION AND DESIGN DEFICIENCIES. THE CORRECTIVE ACTIONS FOR THIS EVENT ARE: 1. A MODIFICATION IS CURRENTLY SCHEDULED TO REPLACE RWCU SYSTEM FLEX HOSES OUTSIDE PRIMARY CONTAINMENT WITH RIGID TUBING AND TO RELOCATE FLOW TRANSMITTER 2WCS*FT67". MODIFICATION PN2Y87MX165 IDENTIFIES THE REPLACEMENT OF PRINTED CIRCUIT (PC) BOARDS WITH NEWLY DEVELOPED BOARDS WHICH ARE EXPECTED TO FILTER OUT PRESSURE TRANSIENTS. 3. THE SPECIAL TASK FORCE COMMITTEE ASSIGNED TO EVALUATE AND TROUBLESHOOT THE RWCU SYSTEM WILL CONTINUE ONGOING SURVEILLANCES.

[139] NINE MILE POINT 2 DOCKET 50-410 LER 87-074 REV 01
UPDATE ON INOPERABLE FIRE BARRIER DUE TO A BREACHED FLOOR PLUG INSTALLATION AND CONSTRUCTION DEFICIENCY/PERSONNEL ERROR.
EVENT DATE: 121987 REPORT DATE: 043088 NSSS: GE TYPE: BWR

(NSIC 209212) ON 12/19/87 AT 1100 WITH THE REACTOR IN COLD SHUTDOWN (OPERATIONAL CONDITION 4, AN UNSATISFACTORY FLOOR PLUG INSTALLATION WAS DISCOVERED IN A FIRE RATED FLOOR IN THE DIVISION 2 VENTILATION ROOM LOCATED ON CONTROL BUILDING (CB) ELEVATION 306. THE INSTALLATION, DISCOVERED BY THE NINE MILE POINT UNIT 2 (NMP2) FIRE DEPARTMENT, CONSTITUTED AN APPENDIX R VIOLATION. AS A RESULT OF THIS EVENT AN ENGINEERING EVALUATION WAS CONDUCTED WHICH IDENTIFIED TWO OTHER POTENTIALLY UNSATISFACTORY APPENDIX R FLOOR PLUG INSTALLATIONS ON JANUARY 14, 1988. IN EACH CASE, THE FIRE RATED FLOORS WERE DECLARED INOPERABLE AND FIRE WATCH PATROLS WERE IMMEDIATELY ESTABLISHED. REVISION ZERO OF LER 87-74 ORIGINALLY REPORTED THAT THESE INSTALLATIONS WERE DEFICIENT AS A RESULT OF A DESIGN DEFICIENCY. UPON FURTHER ANALYSIS BY A CONTRACTOR'S ENGINEERING DEPARTMENT IT WAS DETERMINED THAT THE INSTALLATION DESIGN WAS ACCEPTABLE. HOWEVER, OF THE THREE APPENDIX R FLOOR PLUGS IDENTIFIED AS POTENTIALLY UNSATISFACTORY, TWO OF THEM WERE ACTUALLY IN A DEFICIENT CONFIQURATION DUE TO SEALANT BEING REMOVED FROM THE INSTALLATIONS. THIS WAS CAUSED BY A CONSTRUCTION DEFICIENCY AND PERSONNEL ERROR. THE CORRECTIVE ACTIONS FOR THIS EVENT ARE: (1) FIRE WATCH PATROLS WERE ESTABLISHED IN THE AFFECTED FIRE AREAS, (2) OTHER FLOOR PLUG INSTALLATIONS HAVE BEEN EVALUATED, (3) THE FLOOR PLUG HAS BEEN RESEALED.

[140] NINE MILE POINT 2 DOCKET 50-410 LER 88-014
REACTOR SCRAM AND EMERGENCY CORE COOLING SYSTEM ACTUATION DUE TO A LOSS OF FEEDWATER FLOW CAUSED BY A DESIGN DEFICIENCY.
EVENT DATE: 031388 REPORT DATE: 041288 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 208945) ON 3/13/88 AT 17:39 WITH THE REACTOR MODE SWITCH IN RUN (OPERATIONAL CONDITION 1) AND AT A POWER LEVEL OF APPROX. 43% (SEE NOTE) RATED THERMAL CAPACITY, NINE MILE POINT UNIT 2 EXPERIENCED AN AUTOMATIC REACTOR SCRAM, AN AUTOMATIC INITIATION OF THE DIVISION 3 EMERGENCY CORE COOLING SYSTEM (ECCS) WITH A SUBSEQUENT COOLANT INJECTION, AND THE AUTOMATIC ACTUATION OF SEVERAL ENGINEERED SAFETY FEATURES. THESE EVENTS WERE THE RESULT OF LOW WATER LEVELS IN THE REACTOR VESSEL CAUSED BY A TOTAL LOSS OF FEEDWATER FLOW. AN UNUSUAL EVENT DECLARED AT 17:45 WAS TERMINATED BY 18:00 THAT DAY. THE ECCS INJECTION WAS MANUALLY TERMINATED AND A NORMAL REACTOR SHUTDOWN WAS COMMENCED BY THE NMP2 OPERATORS. (NOTE: REACTOR POWER WAS AT 98% TWO MINUTES PRIOR TO THIS EVENT. HOWEVER, AN INSTRUMENT FAILURE CAUSED THE REACTOR RECIRCULATION PUMPS TO DOWNSHIFT TO LOW SPEED OPERATION, DECREASING REACTOR POWER TO 43%.) THE IMMEDIATE CAUSE FOR THIS EVENT IS AN EQUIPMENT FAILURE. HOWEVER, THE ROOT CAUSE FOR THIS EVENT IS A DESIGN DEFICIENCY. THE CORRECTIVE ACTIONS FOR THIS EVENT ARE: (1) A

TEMPORARY MODIFICATION HAS BEEN PERFORMED TO BYPASS THE "SEAL IN" LOGIC, (2) PERMANENT MODIFICATION WILL BE IMPLEMENTED TO REROUTE THE PIPING FOR THE SECOND POINT FEEDWATER HEATERS LEVEL SWITCHES, AND (3) THE FAILED PRESSURE TRANSMITTER WILL BE SENT TO THE VENDOR FOR A FAILURE MODE ANALYSIS.

[141] NINE MILE POINT 2 DOCKET 50-410 LER 88-016
PRIMARY CONTAINMENT PURGE PERFORMED WITHOUT OBTAINING SAMPLE ANALYSIS AND ACCEPTABLE PURGE RATE CAUSED BY PROCEDURAL DEFICIENCY.
EVENT DATE: 031888 REPORT DATE: 041588 NSSS: GE TYPE: BWR

(NSIC 208946) ON 3/18/88 AT 1155 HOURS, A TECH SPEC SURVEILLANCE REQUIREMENT WAS DISCOVERED TO HAVE BEEN MISSED AT NINE MILE POINT UNIT 2 (NMP2). THE SURVEILLANCE REQUIREMENT WAS MISSED WHEN PRIMARY CONTAINMENT WAS INSERTED BY A NITROGEN PURGE WITHOUT FIRST OBTAINING AN INITIAL SAMPLE ANALYSIS AND PERMISSIBLE PURGE RATE, AS REQUIRED. AT THE TIME OF THE EVENT NMP2 WAS AT APPROXIMATELY 11% OF RATED THERMAL POWER. THE CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE A PROCEDURAL DEFICIENCY. THE OPERATING PROCEDURE DETAILING THE PURGE OPERATION FAILED TO PROVIDE ADEQUATE INSTRUCTION TO OPERATORS IN THE CONTROL ROOM. THE PROCEDURE DID NOT REQUIRE THAT A GAS SAMPLE ANALYSIS AND PERMISSIBLE PURGE RATE BE OBTAINED PRIOR TO PROCEEDING WITH A PRIMARY CONTAINMENT PURGE OPERATION. IMMEDIATE CORRECTIVE ACTION WAS FOR THE OPERATORS TO DISCONTINUE PURGE OPERATION UNTIL A GAS SAMPLE ANALYSIS AND PERMISSIBLE PURGE RATE WERE OBTAINED FROM THE CHEMISTRY DEPARTMENT. THE DEFICIENT PROCEDURE HAS BEEN REVISED TO REQUIRE THAT A SAMPLE ANALYSIS AND PERMISSIBLE PURGE RATE BE OBTAINED FROM THE CHEMISTRY DEPARTMENT BEFORE PURGING OPERATIONS ARE INITIATED. A LESSONS LEARNED TRANSMITTAL SHALL ALSO BE ISSUED TO REMIND ALL OPERATIONS PERSONNEL OF THE TS REQUIREMENT.

[142] NINE MILE POINT 2 DOCKET 50-410 LER 88-017
REACTOR SCRAM CAUSED BY INADEQUATE PLANT IMPACT ASSESSMENT BEFORE PERFORMING LOOP CALIBRATION ON FEEDWATER FLOW TRANSMITTERS.
EVENT DATE: 032188 REPORT DATE: 042088 NSSS: GE TYPE: BWR

(NSIC 209184) ON MARCH 21, 1988 AT 1027 HOURS WITH THE REACTOR MODE SWITCH IN RUN (OPERATIONAL CONDITION 1) AND AT A POWER LEVEL OF APPROXIMATELY 97.5% RATED THERMAL CAPACITY, NINE MILE POINT UNIT 2 EXPERIENCED AN AUTOMATIC REACTOR SCRAM AS A RESULT OF THE MAIN TURBINE TRIP. THE TURBINE TRIP OCCURRED WHEN A FEEDWATER FLOW TRANSMITTER WAS VALVED OUT OF SERVICE CREATING INCREASED FEED FLOW UNTIL LEVEL 8 WAS REACHED. (NORMAL TURBINE TRIP ON HIGH REACTOR WATER LEVEL.) THE ROOT CAUSE OF THE EVENT IS THAT CURRENT WORK CONTROL PROCEDURES DO NOT ASSURE PROPER ASSESSMENT OF PLANT IMPACT. IMMEDIATE CORRECTIVE ACTION WAS TO RESTORE REACTOR WATER LEVEL TO NORMAL. FURTHER CORRECTIVE ACTIONS INCLUDE REVISION OF REPAIR AND TROUBLE SHOOTING PROCEDURES, PLANT IMPACT POLICY ISSUANCE, LESSONS LEARNED TRANSMITTALS, AND SYSTEM DESIGN REVIEW.

[143] NINE MILE POINT 2 DOCKET 50-410 LER 88-015
REACTOR WATER CLEANUP ISOLATION ON HIGH AIR TEMPERATURE DUE TO DESIGN DEFICIENCY.
EVENT DATE: 040688 REPORT DATE: 050688 NSSS: GE TYPE: BWR

(NSIC 209284) ON APRIL 6, 1988 AT 0510 HOURS, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED AN ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF), SPECIFICALLY, ISOLATION OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM ON A HIGH RWCU PUMP ROOM AIR TEMPERATURE. AT THE TIME OF THE EVENT, THE PLANT WAS OPERATING AT APPROXIMATELY 100% OF ITS RATED THERMAL POWER WITH THE REACTOR MODE SWITCH IN THE "RUN" POSITION. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE A DESIGN DEFICIENCY IN THE REACTOR BUILDING VENTILATION (HVR) SYSTEM WHICH RESULTED IN INADEQUATE VENTILATION IN THE RWCU PUMP ROOM DURING RWCU SYSTEM OPERATION. THE INADEQUATE VENTILATION OCCURS DURING THE OFF NORMAL OPERATING MODE OF THE HVR SYSTEM.

IMMEDIATE CORRECTIVE ACTIONS FOR THE EVENT WERE FOR THE OPERATORS TO VERIFY THE AUTOMATIC RESPONSE OF THE RWCU SYSTEM, VERIFY PLANT STATUS AS NORMAL, DECREASE THE RWCU PUMP ROOM 'A' AIR TEMPERATURE TO A NORMAL LEVEL, RESET THE ISOLATION SIGNAL AND RESTORE THE RWCU SYSTEM TO SERVICE. TEMPORARY VENTILATION WAS THEN PROVIDED TO THE RWCU DUMP ROOMS TO INSURE THAT THERE WAS ADEQUATE COOLING TO THAT AREA. ADDITIONAL CORRECTIVE ACTIONS ARE: (1) CAUTION STATEMENT IN OPERATING PROCEDURE, WARNING ABOUT POSSIBLE HIGH TEMPERATURE AREAS; (2) ENGINEERING ANALYSIS OF REACTOR BUILDING VENTILATION DURING OFF NORMAL OPERATION OF THE HVR SYSTEM.

[144] NORTH ANNA 1 DOCKET 50-338 LER 87-017 REV 01
UPDATE ON STEAM GENERATOR TUBE RUPTURE.
EVENT DATE: 071587 REPORT DATE: 041488 NSSS: WE TYPE: PWR

(NSIC 209010) AT APPROXIMATELY 0635 HOURS ON JULY 15, 1987, WITH UNIT 1 AT 100 PERCENT POWER AND UNIT 2 AT 81 PERCENT POWER, UNIT 1 WAS MANUALLY TRIPPED DUE TO INDICATIONS OF A STEAM GENERATOR TUBE RUPTURE IN THE "C" STEAM GENERATOR. APPROXIMATELY TWENTY SECONDS LATER, THE SAFETY INJECTION SYSTEM AUTOMATICALLY INITIATED. AT 0639 HOURS, A NOTIFICATION OF UNUSUAL EVENT WAS DECLARED AND NOTIFICATIONS TO STATE AND LOCAL GOVERNMENTS WERE COMPLETED BY 0651 HOURS. THE EVENT WAS SUBSEQUENTLY UPGRADED TO AN ALERT AT 0654 HOURS AND ALL NOTIFICATIONS TO OFF-SITE AGENCIES AND THE NUCLEAR REGULATORY COMMISSION (NRC) WERE COMPLETED BY 0702 HOURS. COOLDOWN AND DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM (RCS) WAS CONDUCTED IN ACCORDANCE WITH STATION EMERGENCY OPERATING PROCEDURES. AT 1336 HOURS, THE RCS WAS COOLED DOWN TO BELOW 200 DEGREES F AND THE UNIT WAS PLACED IN MODE 5. AT THIS TIME, PLANT CONDITIONS WERE STABLE AND THE EMERGENCY WAS TERMINATED. THIS EVENT IS REPORTABLE PURSUANT TO 10 CFR 50.73(A)(2)(II), 10 CFR 50.73 (A)(2)(IV), AND WAS THE ELEVENTH ECCS ACTUATION REPORTABLE PURSUANT TO TECHNICAL SPECIFICATION 6.9.2. AT NO TIME DURING THIS EVENT WERE THE HEALTH AND SAFETY OF THE GENERAL PUBLIC AFFECTED.

[145] NORTH ANNA 1 DOCKET 50-338 LER 87-019
EXCESSIVE SKIN EXPOSURE DUE TO CONTAMINATION FROM HOT PARTICLE TRANSFERRED TO INDIVIDUAL FROM LAUNDERED PROTECTIVE CLOTHING.
EVENT DATE: 081987 REPORT DATE: 091887 NSSS: WE TYPE: PWR

(NSIC 209191) ON AUGUST 19, 1987, WITH UNIT 1 IN MODE 5, A HEALTH PHYSICS TECHNICIAN WAS CONTAMINATED BY A PINPOINT SIZED RADIOACTIVE COBALT-60 PARTICLE LOCATED ON THE SKIN OF THE BACK. THE PARTICLE WAS CAPTURED, ANALYZED, AND FOUND TO BE ABOUT 20 TO 30 MICRONS IN DIAMETER WITH AN ACTIVITY OF 1.6 MICROCURIES. A LOCALIZED SKIN DOSE OF 23.6 REM WAS ASSESSED BASED UPON A 3.25 HOUR EXPOSURE TIME. THE 23.6 REM DOSE PLUS THE CURRENT ACCUMULATED QUARTERLY DOSE OF 0.228 REM TOTALED 23.828 REM (JULY 1, 1987 TO AUGUST 19, 1987) WHICH EXCEEDS THE 7.5 REM LIMIT FOR SKIN DOSE SPECIFIED IN 10CFR20.101(A) AND IS REPORTABLE PURSUANT TO 10CFR20.405(A)(I)(I). THE MOST PROBABLE CAUSE OF THE CONTAMINATION WAS THE TRANSFER OF THE PARTICLE FROM LAUNDERED PROTECTIVE CLOTHING TO THE SKIN OF THE HEALTH PHYSICS TECHNICIAN. THE INDIVIDUAL CONCERNED HAS BEEN RESTRICTED FROM ENTRY INTO THE RADIOLOGICAL CONTROL AREA FOR THE REMAINDER OF THE CURRENT CALENDAR QUARTER. THIS EVENT HAD NO EFFECT ON THE PUBLIC HEALTH AND SAFETY.

[146] NORTH ANNA 1 DOCKET 50-338 LER 87-021 REV 01
UPDATE ON LOSS OF ENVIRONMENTAL QUALIFICATION OF SI ACCUMULATOR TANK PRESSURE TRANSMITTERS.
EVENT DATE: 091187 REPORT DATE: 041488 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)
VENDOR: ROSEMOUNT, INC.

(NSIC 208985) ON SEPTEMBER 11, 1987 AT 0900 HOURS WITH UNIT 1 IN MODE 5 (COLD

SHUTDOWN), IT WAS DISCOVERED THAT THE ENVIRONMENTAL SEALS ON THE SENSOR NECKS OF THE SAFETY INJECTION (SI) ACCUMULATOR TANK PRESSURE TRANSMITTERS HAD BEEN BROKEN DURING INSTALLATION IN MAY, 1987. PURSUANT TO GENERIC LETTER 85-15, THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(V). DURING THE INSTALLATION OF ONE OF THE SIX EQ PRESSURE TRANSMITTERS ON THE UNIT 2 SI ACCUMULATOR TANKS ON SEPTEMBER 11, 1987, THE QUALITY CONTROL (QC) INSPECTOR WITNESSED THE ELECTRONICS HOUSING BEING ALLOWED TO TURN ON THE SENSOR NECK. IT WAS THEN DISCOVERED THAT THIS ACTION VOIDS THE ENVIRONMENTAL SEAL WHICH IS BAKED ON THE SENSOR NECK. THE CONSTRUCTION PERSONNEL THEN TOLD THE QC INSPECTOR THAT THE SAME METHOD OF INSTALLATION HAD BEEN USED IN MAY, 1987, DURING THE UNIT 1 REFUELING OUTAGE, CAUSING THE UNIT 1 TRANSMITTERS TO BE ENVIRONMENTALLY UNQUALIFIED. AS A CORRECTIVE ACTION, THE UNIT 1 TRANSMITTERS WERE REMOVED AND THE SENSOR NECKS WERE RESEALED. THE CONSTRUCTION PERSONNEL WERE TRAINED TO BE CAUTIOUS OF THE NECK SEAL AND THE PROPER INSTALLATION WAS COMPLETED ON SEPTEMBER 24, 1987. QC PERSONNEL HAVE BEEN TRAINED TO BE AWARE OF POSSIBLE DAMAGE TO THE NECK SEAL DURING INSTALLATION.

[147] NORTH ANNA 1 DOCKET 50-338 LER 88-017
FAILURE TO TEST CONTAINMENT PERSONNEL AIRLOCK EQUALIZING VALVES.
EVENT DATE: 021688 REPORT DATE: 041488 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)
VENDOR: CHICAGO BRIDGE AND IRON COMPANY

(NSIC 209031) AT 1030 HOURS ON FEBRUARY 16, 1988, WITH UNIT 1 AND UNIT 2 AT 100 PERCENT POWER (MODE 1), IT WAS DISCOVERED THAT THE EMERGENCY EQUALIZING VALVES FOR THE CONTAINMENT ISOLATION INNER AIR LOCK ESCAPE DOORS HAD NOT BEEN ADEQUATELY TESTED FOR LEAKAGE SINCE THE INITIAL STARTUP OF BOTH UNITS. THE CONTAINMENT AIR LOCK IS TESTED EVERY SIX MONTHS IN ACCORDANCE WITH 1/2-PT-62.1 (CONTAINMENT AIR LOCKS - LEAKAGE RATE), BUT THERE IS NO SPECIFIC PROCEDURE DESCRIBED FOR TESTING THE EMERGENCY EQUALIZING VALVES, WHICH FUNCTION AS A PART OF THE OVERALL HATCH PRESSURE BOUNDARY. THE VALVES WERE COVERED BY BLANK FLANGES IN ACCORDANCE WITH THE VENDOR MANUAL AND WERE NOT TESTED WHENEVER THE OVERALL AIR LOCK LEAKAGE TEST WAS PERFORMED. AS AN IMMEDIATE CORRECTIVE ACTION, THE EMERGENCY EQUALIZING VALVES WERE SATISFACTORILY LEAK-TESTED. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B). THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED THROUGHOUT THIS EVENT.

[148] NORTH ANNA 1 DOCKET 50-338 LER 88-011
POST MODIFICATION TESTING NOT PERFORMED AS REQUIRED BY TWO TECHNICAL SPECIFICATIONS.
EVENT DATE: 031188 REPORT DATE: 040788 NSSS: WE TYPE: PWR
VENDOR: ASCO VALVES

(NSIC 208855) ON MARCH 11, 1988, WITH UNIT 1 AT 100% POWER (MODE 1), STATION PERSONNEL IDENTIFIED NINE CONTAINMENT ISOLATION TRIP VALVES THAT HAD NOT BEEN FOLLOWING MODIFICATION. TECH SPEC 3.6.3.1 AND TECH SPEC 4.0.5 (WHICH REQUIRES COMPLIANCE WITH ASME XI) BOTH REQUIRE THAT THESE VALVES BE DEMONSTRATED OPERABLE AFTER MAINTENANCE (MODIFICATION) IS PERFORMED, PRIOR TO RETURNING THE VALVE TO SERVICE. DUE TO THE FAILURE TO COMPLY WITH THESE TECH SPECS, THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I). THE REASON THE VALVES WERE NOT TESTED FOLLOWING MODIFICATION WAS DUE TO THE INADEQUACY OF THE ENGINEERING WORK REQUEST (EWR) USED TO MODIFY THE AIR VENT LINE FOR THE ASSOCIATED SOLENOID OPERATED VALVES (SOVS). AN INVESTIGATION DETERMINED THAT ALL VALVES WERE STROKED WITH SATISFACTORY RESULTS DURING A SUBSEQUENT OUTAGE. NO SIGNIFICANT SAFETY CONSEQUENCES RESULTED FROM THIS EVENT BECAUSE THE VALVES WERE DETERMINED TO BE FULLY OPERABLE FROM THE TIME THE MODIFICATION WAS PERFORMED UNTIL THE VALVES WERE SATISFACTORILY STROKE TESTED. THE HEALTH AND SAFETY OF THE GENERAL PUBLIC WAS NOT AFFECTED.

[149] NORTH ANNA 1 DOCKET 50-338 LER 88-012
LOSS OF RHR CAPABILITY DUE TO FAILED SOLENOID OPERATED VALVE.
EVENT DATE: 031588 REPORT DATE: 040888 NSSS: WE TYPE: PWR
VENDOR: ASCO VALVES

(NSIC 208919) AT 2158 HOURS ON 3/15/88, WITH UNIT 1 AT 100% POWER (MODE 1), BOTH RESIDUAL HEAT REMOVAL (RHR) SUBSYSTEMS WERE DECLARED INOPERABLE FOR APPROXIMATELY 40 MINUTES DUE TO ISOLATION OF COMPONENT COOLING (CC) WATER TO BOTH RHR HEAT EXCHANGERS AND RHR PUMP MECHANICAL SEAL COOLERS. COMPONENT COOLING WATER WAS ISOLATED IN ORDER TO PERFORM REQUIRED MAINTENANCE ON THE SOLENOID OPERATED VALVE (SOV) FOR CONTAINMENT ISOLATION VALVE, 1-CC-TV-103B. 1-CC-TV-103B, THE CC RETURN ISOLATION VALVE FOR THE 'B' RHR HEAT EXCHANGER AND BOTH RHR PUMP SEAL COOLERS, FAILED TO STROKE CLOSED WITHIN THE TIME SPECIFIED BY TECH SPEC 3.6.3.1. SINCE NO REPLACEMENT PARTS WERE AVAILABLE, AND 1-CC-TV-103B COULD NOT BE ISOLATED WITHOUT AFFECTING BOTH RHR SUBSYSTEMS, THE SOV WAS REPLACED WITH THE SOV FROM 1-CC-TV-103A. SINCE 1-CC-TV-103A IS THE CC RETURN ISOLATION VALVE FROM THE 'A' RHR HEAT EXCHANGER, CC TO BOTH RHR SUBSYSTEMS WAS ISOLATED WHEN BOTH VALVES WERE TAGGED OUT TO INTERCHANGE THE SOVS. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B). THE CAUSE OF THIS EVENT WAS THE FAILURE OF THE CONTAINMENT ISOLATION VALVE, 1-CC-TV-103B, TO STROKE CLOSED WITHIN THE REQUIRED TIME LIMIT, COMPOUNDED BY THE LACK OF SPARE PARTS TO REPAIR THE FAILED SOV.

[150] NORTH ANNA 1 DOCKET 50-338 LER 88-014
REACTOR TRIP DUE TO FAILURE TO MAINTAIN 'C' S/G LEVEL ABOVE 25% WHILE BISTABLE WAS IN TRIP.
EVENT DATE: 031888 REPORT DATE: 041588 NSSS: WE TYPE: PWR

(NSIC 208921) AT 1705 HOURS ON MARCH 18, 1988, WITH UNIT 1 AT ZERO PERCENT POWER (MODE 3), AND THE REACTOR TRIP BREAKERS OPEN, A REACTOR TRIP SIGNAL WAS GENERATED WHEN THE 'C' STEAM GENERATOR (S/G) NARROW RANGE LEVEL DECREASED TO 25 PERCENT WHILE THE 'C' S/G CHANNEL IV STEAM FLOW-FEED FLOW MISMATCH BISTABLE WAS IN TRIP. THE 'C' S/G CHANNEL IV STEAM FLOW- FEEDWATER FLOW MISMATCH BISTABLE WAS PLACED IN TRIP FOR TROUBLESHOOTING FI-1495, 'C' S/G CHANNEL IV STEAM FLOW. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV). A FOUR HOUR REPORT WAS MADE IN ACCORDANCE WITH 10CFR50.72(B)(2)(II). THE CAUSE OF THIS EVENT WAS FAILURE TO MAINTAIN THE 'C' S/G LEVEL ABOVE 25 PERCENT WHILE THE 'C' S/G CHANNEL IV STEAM FLOW-FEEDWATER FLOW MISMATCH BISTABLE WAS IN TRIP. AS A CORRECTIVE ACTION, THE 'C' S/G LEVEL WAS RESTORED ABOVE 25 PERCENT. A HUMAN PERFORMANCE EVALUATION SYSTEM (HPES) REVIEW WILL BE PERFORMED WITH REGARD TO THIS EVENT. THE HEALTH AND SAFETY OF THE GENERAL PUBLIC WERE NOT AFFECTED AT ANY TIME DURING THIS EVENT.

[151] NORTH ANNA 1 DOCKET 50-338 LER 88-013
TURBINE TRIP/REACTOR TRIP DUE TO EHC SYSTEM MALFUNCTION.
EVENT DATE: 031988 REPORT DATE: 041488 NSSS: WE TYPE: PWR

(NSIC 208920) AT 0133 HOURS ON MARCH 19, 1988, UNIT 1 EXPERIENCED AN AUTOMATIC REACTOR TRIP FROM APPROXIMATELY 3.5% POWER, 0 MWE. THE REACTOR TRIP OCCURRED DUE TO A SPIKE IN THE TURBINE IMPULSE (FIRST STAGE) PRESSURE WHICH CAUSED A TURBINE TRIP AND ENABLED THE LOGIC FOR A REACTOR TRIP WHEN A TURBINE TRIP CONDITION EXISTED. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73 (A)(2)(IV). THE TURBINE IMPULSE PRESSURE SPIKE OCCURRED WHILE PLACING THE TURBINE IN OPERATION. INTERNAL BYPASS LEAKAGE IN THE ELECTRO-HYDRAULIC FLUID CONTROL SYSTEM (EHC) WAS BELIEVED TO HAVE OCCURRED IN THE SERVO CONTROL VALVES AND VALVE ACTUATORS WHICH CONTROL THE POSITION OF THE TURBINE GOVERNOR VALVES. THE INCREASED BYPASS LEAKAGE ALLOWED THE TURBINE GOVERNOR VALVES TO CLOSE AND RESULTED IN A DECREASE IN TURBINE IMPULSE PRESSURE. A SECOND EHC PUMP WAS STARTED TO INCREASE THE EHC FLUID PRESSURE AND REOPEN THE TURBINE CONTROL VALVES. THE REOPENING OF THE TURBINE CONTROL VALVES CAUSED THE IMPULSE PRESSURE TO EXCEED THE SETPOINTS FOR A TURBINE TRIP IF THE GENERATOR OUTPUT BREAKER IS OPEN AND FOR A REACTOR TRIP FROM

A TURBINE TRIP. AS A CORRECTIVE ACTION, THE SERVO CONTROL VALVES FOR THE FOUR GOVERNOR VALVES WERE REPLACED AND TESTED SATISFACTORILY. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS.

[152] NORTH ANNA 1 DOCKET 50-338 LER 88-015
RHR PUMPS NOT TESTED DURING STEAM GENERATOR TUBE RUPTURE OUTAGE.
EVENT DATE: 032288 REPORT DATE: 041488 NSSS: WE TYPE: PWR

(NSIC 208922) AT 1100 HOURS ON MARCH 22, 1988, WITH UNIT 1 AT 3 PERCENT POWER (MODE 2), IT WAS DISCOVERED THAT THE UNIT 1 RESIDUAL HEAT REMOVAL (RHR) PUMPS HAD NOT BEEN TESTED DURING THE UNIT 1 STEAM GENERATOR TUBE RUPTURE OUTAGE, WHICH LASTED FROM JULY 15, 1987 TO OCTOBER 13, 1987. THE TEST WAS SCHEDULED FOR AUGUST 11, 1987, BUT WAS NOT PERFORMED DUE TO THE NATURE OF THE OUTAGE AND THE EXISTING UNIT CONDITIONS. AS A RESULT OF ADMINISTRATIVE ERROR, THE TEST WAS NOT RESCHEDULED NOR PERFORMED FOR THE REMAINDER OF THE OUTAGE. THE RHR PUMPS ARE REQUIRED TO BE TESTED DURING EACH COLD SHUTDOWN IN ACCORDANCE WITH TECHNICAL SPECIFICATION 4.0.5. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B). THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED THROUGHOUT THIS EVENT.

[153] NORTH ANNA 2 DOCKET 50-339 LER 87-003
IMPROPER DETERMINATION OF QUADRANT POWER TILT RATIO.
EVENT DATE: 052287 REPORT DATE: 061887 NSSS: WE TYPE: PWR

(NSIC 209192) AT APPROXIMATELY 2200 HOURS ON MAY 20, 1987, WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN) AND UNIT 2 IN MODE 1 AT 100 PERCENT POWER, A QUADRANT POWER TILT RATIO WAS CALCULATED USING ONLY ONE SET OF FOUR SYMMETRIC INCORE THIMBLES INSTEAD OF TWO SETS AS REQUIRED BY TECH SPEC 4.2.4.2. ON NOVEMBER 20, 1986, EXCORE DETECTOR N43 FAILED. THIS FAILURE REQUIRED THAT A TWELVE HOUR SURVEILLANCE FOR QUADRANT POWER TILT RATIO BE PERFORMED IN ACCORDANCE WITH TECH SPECS. THIS SURVEILLANCE REQUIRES A CALCULATION OF QUADRANT POWER TILT BASED ON TWO SETS OF FOUR SYMMETRIC INCORE LOCATIONS WHEN REACTOR POWER IS GREATER THAN SEVENTY FIVE PERCENT OF RATED THERMAL POWER. THIS IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B).

[154] NORTH ANNA 2 DOCKET 50-339 LER 88-019
MISSED SURVEILLANCE ON A CONTAINMENT ISOLATION VALVE.
EVENT DATE: 040788 REPORT DATE: 050388 NSSS: WE TYPE: PWR

(NSIC 209227) AT 1000 HOURS ON APRIL 7, 1988, WITH UNIT 2 AT 100 PERCENT POWER (MODE 1), IT WAS DISCOVERED THAT THE SURVEILLANCE TEST FOR THE STEAM GENERATOR (S/G) SURFACE SAMPLE LINE OUTSIDE CONTAINMENT ISOLATION VALVE, TV-SS-212B HAD NOT BEEN PERFORMED WITHIN THE SURVEILLANCE INTERVAL ALLOWED BY TECHNICAL SPECIF ICATION 4.0.5. TECHNICAL SPECIFICATION 4.0.5 REFERS TO ASME SECTION XI, WHICH REQUIRES THIS VALVE TO BE STROKED EVERY THREE MONTHS. SINCE THIS VALVE HAD NOT BEEN SATISFACTORILY STROKED SINCE OCTOBER 26, 1987, THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B). THE CAUSE FOR THE MISSED SURVEILLANCE HAS BEEN ATTRIBUTED TO PERSONNEL ERROR. TO PREVENT RECURRENCE, PROGRAM CHANGES HAVE BEEN IMPLEMENTED TO FURTHER ASSURE THAT THE SURVEILLANCE REQUIREMENTS ARE MET. NO ADVERSE SAFETY CONSEQUENCES RESULTED FROM THIS EVENT BECAUSE THE CAPABILITY TO ISOLATE THE STEAM GENERATOR SURFACE SAMPLE LINE CONTAINMENT PENETRATION WAS NOT AFFECTED. THE HEALTH AND SAFETY OF THE GENERAL PUBLIC WERE NOT ADVERSELY AFFECTED AT ANY TIME DURING THIS EVENT.

[155] OCONEE 1 DOCKET 50-269 LER 87-001 REV 01
UPDATE ON FAILURE TO COMPLY WITH ISI PROGRAM ON POST ACCIDENT LIQUID SAMPLING SYSTEM.
EVENT DATE: 011987 REPORT DATE: 033188 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 208700) ON JANUARY 19, 1987, NON-COMPLIANCE WITH THE INSERVICE INSPECTION (ISI) PROGRAM WAS IDENTIFIED BY CHEMISTRY PERSONNEL. THE MISSED ISI SURVEILLANCE INVOLVED "STROKE TESTS" ON SEVERAL VALVES WHICH ARE IN THE POST ACCIDENT LIQUID SAMPLING (PALS) SYSTEM. UNITS 1 AND 2 WERE AT 95% AND 97% RESPECTIVELY AND UNIT 3 WAS IN A REFUELING OUTAGE WHEN THE INCIDENT WAS IDENTIFIED, BUT UNIT STATUS WAS NOT A FACTOR ASSOCIATED WITH THIS INCIDENT. THIS INCIDENT WAS DETERMINED TO BE REPORTABLE PER THE REQUIREMENTS OF 10 CFR 50.73(A)(2)(I)(B) ON FEBRUARY 24, 1987. THE ROOT CAUSE OF THE MISSED SURVEILLANCE WAS INADEQUATE TURNOVER OF THE ISI PROGRAM REQUIREMENTS/RESPONSIBILITIES. THERE HAVE BEEN NO RADIATION RELEASES FROM THIS SYSTEM TO THE ENVIRONMENT. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED BY THIS INCIDENT.

[156] OCONEE 1 DOCKET 50-269 LER 87-011 REV 01
UPDATE ON CABLE ROOM SPRINKLER SYSTEMS INOPERABLE DUE TO DESIGN DEFICIENCY OF PRESSURE AND FLOW RATES.
EVENT DATE: 120287 REPORT DATE: 041988 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 208983) ON OCTOBER 8, 1987 DUKE POWER'S DESIGN ENGINEERING GROUP IDENTIFIED THAT THE UNIT 3 CABLE ROOM SPRINKLER SYSTEM COULD NOT PROVIDE ITS DESIGN FLOW DUE TO A DESIGN DEFICIENCY. THIS WAS IDENTIFIED WHILE DESIGN ENGINEERING WAS RESPONDING TO A MAY, 1987 FIRE PROTECTION SYSTEM AUDIT. ON MARCH 2, 1988 THE INOPERABILITY OF THE KEOWEE HYDRO STATION MAIN LUBE OIL STORAGE ROOM WATER SPRAY SYSTEM WAS DISCOVERED. THIS INCIDENT WAS DETERMINED TO BE REPORTABLE PURSUANT TO THE REQUIREMENTS OF 10CFR 50.73(A)(2)(I)(B) ON DECEMBER 2, 1987. THE ROOT CAUSE OF THIS INCIDENT WAS DETERMINED TO BE A DESIGN DEFICIENCY. THE INOPERABILITY OF THE UNIT 3 CABLE ROOM SPRINKLER SYSTEM WAS ORIGINALLY MISCLASSIFIED AS NON-REPORTABLE TO THE NRC DUE TO THE FACT THAT IT WAS NOT INITIALLY IDENTIFIED AS A DESIGN DEFICIENCY. CORRECTIVE ACTIONS INCLUDED ESTABLISHING FIRE WATCHES UNTIL THE WATER SPRAY SYSTEMS WERE MADE OPERABLE AND RESTORING THE SYSTEMS TO AN OPERABLE CONDITION BY MODIFICATION. BECAUSE OF THE AVAILABLE SYSTEMS AND PERSONNEL FOR DETECTION AND CONTROL, THERE IS LITTLE CHANCE THAT A FIRE WOULD HAVE BEGUN AND DEVELOPED INTO A THREAT TO PLANT SAFETY. THEREFORE THIS EVENT IS CONSIDERED NOT TO BE SIGNIFICANT WITH RESPECT TO THE HEALTH AND SAFETY OF THE PUBLIC.

[157] OCONEE 1 DOCKET 50-269 LER 88-004
EMERGENCY POWER SWITCHING LOGIC RETRANSFER TO STARTUP LOGIC DEFEATED DUE TO DESIGN DEFICIENCY.
EVENT DATE: 022288 REPORT DATE: 040688 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 208927) ON FEBRUARY 22, 1988, DESIGN ENGINEERING IDENTIFIED A CONDITION WHICH PREVENTED THE EMERGENCY POWER SWITCHING LOGIC (EPSL) FROM PERFORMING A PART OF ITS SAFETY FUNCTION. SPECIFICALLY, THE ISOLATION OF THE KEOWEE STANDBY BUS TRANSFORMER, CT.4, NORMALLY REQUIRED THE REMOVAL OF THE SUPPLY BREAKERS AND THEIR ASSOCIATED CONTROL POWER FUSES. THE REMOVAL OF THESE CONTROL POWER FUSES DEFEATED THE RETRANSFER TO STARTUP LOGIC PORTION OF THE EPSL THIS WAS IN VIOLATION OF TECHNICAL SPECIFICATION 3.7.2(B). WHEN THIS INCIDENT WAS DISCOVERED, UNIT 1 AND UNIT 3 WERE AT 65 AND 100 PERCENT FULL POWER RESPECTIVELY,

WHILE UNIT 2 WAS SHUTDOWN FOR REFUELING. DESIGN ENGINEERING'S IMMEDIATE CORRECTIVE ACTION WAS TO CONTACT THE OPERATIONS GROUP AT THE STATION TO INFORM THEM OF THE PROBLEM. OPERATIONS ASSURED ALL THE BREAKERS AND CONTROL POWER FUSES ASSOCIATED WITH THE EPSL WERE IN PLACE AND OPERATIONAL. THE ROOT CAUSE OF THIS INCIDENT IS CLASSIFIED AS DESIGN DEFICIENCY, BECAUSE DURING THE DESIGN OF THE EPSL, DESIGN ENGINEERING FAILED TO CONSIDER THE MAN/MACHINE INTERFACE ASSOCIATED WITH THE EPSL CIRCUITRY. SPECIFICALLY, THE CONVENTIONAL METHOD OF REMOVING THE SK BREAKERS FROM SERVICE DEFEATED THE RETRANSFER TO STARTUP PORTION OF THE EPSL.

[158] OYSTER CREEK DOCKET 50-219 LER 88-005
DEGRADED STANDBY GAS TREATMENT SYSTEM DUE TO PERSONNEL ERROR.
EVENT DATE: 032488 REPORT DATE: 042588 NSSS: GE TYPE: BWR

(NSIC 209214) ON 3/24/88, AT APPROX. 1525 HOURS, IT WAS DISCOVERED THAT A TAGGING ERROR CAUSED COMPONENTS OF BOTH SGTS TO BE INOPERABLE DUE TO PERSONNEL ERROR. AT THE TIME, THE REACTOR WAS OPERATING AT 100% POWER. SGTS TRAIN 1 HAD ITS FAN BREAKER DE-ENERGIZED WHICH RENDERED IT TOTALLY INOPERABLE WHILE SGTS TRAIN 2 HAD AN INSTRUMENT POWER FUSE PULLED FOR MAINTENANCE WHICH PLACED IT IN A DEGRADED MODE. IT WAS ACTUALLY INTENDED THAT SGTS TRAIN 2 FAN BREAKER BE DE-ENERGIZED BUT THE OPERATOR PREPARING THE PAPERWORK MADE AN ERROR THAT WAS NOT IDENTIFIED BY SUBSEQUENT INDEPENDENT REVIEW. OPERATION OF SGTS TRAIN 2 IN THIS DEGRADED MODE WOULD RESULT IN LOWER AIR FLOW FROM THE REACTOR BUILDING AND LOWER IODINE REMOVAL EFFICIENCY BY THE CHARCOAL FILTER. THE TOTAL TIME THE SYSTEMS WERE IN THIS CONFIGURATION WAS ABOUT 2 HOURS. A PREVIOUS EVALUATION SHOWS THAT, IN THIS DEGRADED MODE, THE CHARCOAL EFFICIENCY IS STILL ADEQUATE TO CONTROL RELEASES OF IODINE WITHIN 10CFR100 LIMITS. THE REDUCTION IN FLOW RATE WOULD HAVE BEEN LIMITED BY RESTRICTING ORIFICES TO APPROX. 100 CUBIC FEET PER MINUTE. EVEN WITH THIS REDUCTION OF FLOW, THE SYSTEM WOULD STILL HAVE BEEN ABOVE THE MINIMUM REQUIRED FLOW TO PERFORM ITS FUNCTION. THEREFORE, THE SAFETY SIGNIFICANCE IS CONSIDERED MINIMAL. CORRECTIVE ACTION INCLUDES MANAGEMENT REINFORCEMENT OF EXPECTED PERFORMANCE AND REQUIRED READING FOR LICENSED OPERATORS.

[159] PALO VERDE 1 DOCKET 50-528 LER 87-015
SURVEILLANCE INTERVAL EXCEEDED FOR THREE CONTAINMENT ISOLATION VALVES DUE TO PERSONNEL ERROR.
EVENT DATE: 010587 REPORT DATE: 042787 NSSS: CE TYPE: PWR

(NSIC 209199) AT APPROXIMATELY 1300 ON MARCH 31, 1987 PALO VERDE UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER WHEN IT WAS DISCOVERED THAT SURVEILLANCE TESTING HAD NOT BEEN CONDUCTED WITHIN THE CORRECT TIME INTERVAL ON TWO STEAM GENERATOR BLOWDOWN ISOLATION VALVES AND ONE STEAM TRAP/BYPASS ISOLATION VALVE. ON NOVEMBER 30, 1986 SURVEILLANCE TESTING HAD BEEN CONDUCTED ON THE THREE VALVES IN ACCORDANCE WITH TECH SPEC 4.0.5 WHICH REQUIRES TESTING IN ACCORDANCE WITH SECTION XI OF THE ASME BOILER AND PRESSURE VESSEL CODE. THE THREE VALVES MET THE REQUIRED ACCEPTANCE CRITERIA HOWEVER, THE STROKE TIMES MEASURED HAD INCREASED BY MORE THAN 50% FROM THE PREVIOUS TESTS. THE VALVES ARE REQUIRED TO BE TESTED ONCE PER 3 MONTHS, HOWEVER, WHEN STROKE TIMES INCREASE BY 50% OR MORE RELATIVE TO THE PREVIOUS TEST, ASME SECTION XI REQUIRES TESTING FREQUENCY TO BE ADJUSTED TO A MONTHLY INTERVAL. THE TESTING SCHEDULE WAS NOT MODIFIED TO MEET THE MONTHLY SURVEILLANCE INTERVAL FOR THE THREE VALVES. ON JANUARY 5, 1987 THE MODIFIED SURVEILLANCE INTERVAL WAS EXCEEDED. THE ROOT CAUSE FOR THE EVENT WAS EVALUATED TO BE A COGNITIVE PERSONNEL ERROR BY THE TEST ENGINEER RESPONSIBLE FOR TRACKING THE COMPLETED TESTS. TO PREVENT RECURRENCE THE ENGINEER RECEIVED APPROPRIATE DISCIPLINARY ACTION.

[160] PALO VERDE 1 DOCKET 50-528 LER 88-002
NONCONSERVATIVE SETPOINTS ON THE HI LOG POWER TRIP.
EVENT DATE: 012288 REPORT DATE: 041888 NSSS: CE TYPE: PWR

(NSIC 208955) ON MARCH 17, 1988, AT APPROXIMATELY 1110 MST PALO VERDE UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 100 PERCENT POWER WHEN ENGINEERING PERSONNEL (UTILITY, LICENSED) THAT THE HI LOG POWER TRIPS (JC) MAY HAVE BEEN SET CONTRARY TO THE ALLOWABLE VALUES IN THE TECHNICAL SPECIFICATIONS, TABLE 2.2.-1. ON JANUARY 22, 1988, UNIT 1 WAS IN MODE 3 (HOT STANDBY) WHEN THE REACTOR TRIP SWITCHGEAR (AA) WERE CLOSED. TECHNICAL SPECIFICATION 3.3.1, TABLE 3.3-1 REQUIRES THE HI LOG TRIP SETPOINT TO BE OPERABLE IN MODE 3 WITH THE REACTOR TRIP SWITCHGEAR CLOSED. PRELIMINARY CALCULATIONS INDICATE THE HI LOG TRIP SETPOINTS MAY HAVE BEEN SET CONTRARY TO THE ALLOWABLE SETPOINT. THE CAUSE OF THE SETPOINTS BEING SET CONTRARY TO THE PVNGS TECHNICAL SPECIFICATIONS ALLOWABLE VALUES IS STILL UNDER INVESTIGATION. A SUPPLEMENT TO THIS REPORT WILL BE ISSUED DESCRIBING THE CAUSES AND NECESSARY CORRECTIVE ACTIONS TO PREVENT RECURRENCE. THE TRIP SETPOINTS HAVE BEEN CONSERVATIVELY RESET.

[161] PALO VERDE 1 DOCKET 50-528 LER 88-006
SURVEILLANCE INTERVAL EXCEEDED FOR INCORE DETECTOR SYSTEM.
EVENT DATE: 032088 REPORT DATE: 041988 NSSS: CE TYPE: PWR

(NSIC 209029) ON MARCH 21, 1988, PALO VERDE UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER WHEN ENGINEERING PERSONNEL (UTILITY, NON-LICENSED) DETERMINED THAT THE ALLOWABLE SURVEILLANCE TEST INTERVAL HAD BEEN EXCEEDED FOR THE INCORE DETECTOR SYSTEM (IG). THIS RESULTED IN THE INCORE DETECTOR SYSTEM AND THE CORE OPERATING LIMIT SUPERVISORY SYSTEM (COLSS) (ID) BECOMING ADMINISTRATIVELY INOPERABLE. THE ROOT CAUSE OF THE EVENT WAS A COGNITIVE PERSONNEL ERROR BY ENGINEERING PERSONNEL (UTILITY, NON-LICENSED) RESPONSIBLE FOR ASSIGNING THE PERFORMANCE OF THE SURVEILLANCE TEST. TO PREVENT RECURRENCE, THE INDIVIDUAL WILL RECEIVE APPROPRIATE COUNSELING AND/OR DISCIPLINARY ACTION. A PRELIMINARY EVALUATION HAS DETERMINED THAT A CONTRIBUTORY CAUSE WAS INADEQUATE PROGRAMMATIC CONTROLS TO ENSURE SURVEILLANCE TESTS ARE CONDUCTED WITHIN THE SPECIFIED INTERVALS. A SUPPLEMENT TO THIS LER WILL BE ISSUED DESCRIBING THE RESULTS OF THE REVIEW OF THE PROGRAMMATIC CONTROLS AND THE NECESSARY CORRECTIVE ACTIONS TO PREVENT RECURRENCE.

[162] PALO VERDE 2 DOCKET 50-529 LER 87-021 REV 01
UPDATE ON PASS INCORRECTLY DECLARED OPERABLE.
EVENT DATE: 120287 REPORT DATE: 040788 NSSS: CE TYPE: PWR

(NSIC 208840) THIS IS A SUPPLEMENT TO LER 2-87-021-00. AT APPROXIMATELY 1440 MST ON DECEMBER 2, 1987 THE POST-ACCIDENT SAMPLING SYSTEM (PASS)(IP) WAS DECLARED INOPERABLE FOLLOWING DISCOVERY OF AN IMPROPER VALVE LINEUP. THE LINEUP WAS PERFORMED ON NOVEMBER 7, 1987 TO PERMIT INSTALLATION OF TWO PASS CHECK VALVES (IP)(V). WORK WAS SUSPENDED AND THE PASS DECLARED OPERABLE AT APPROXIMATELY 1220 MST ON NOVEMBER 9, 1987. BASED ON SUBSEQUENT INVESTIGATION, THE PASS WAS DETERMINED TO HAVE BEEN INOPERABLE SINCE APPROXIMATELY 0405 MST, NOVEMBER 7, 1987, AND TO HAVE EXCEEDED THE 7 DAY LIMIT FOR INOPERABILITY PER TECHNICAL SPECIFICATION (T.S.) 3.3.3.1 AT 0405 MST ON NOVEMBER 14, 1987. THE PREPLANNED ALTERNATE SAMPLING PROGRAM (PASP) WAS INITIATED AT 1420 MST ON DECEMBER 4, 1987, THEREFORE PALO VERDE UNIT 2 OPERATED FOR APPROXIMATELY 20 DAYS IN A CONDITION CONTRARY TO T.S. 3.3.3.1. AS IMMEDIATE CORRECTIVE ACTION THE PASS WAS RESTORED, TESTED FOR OPERABILITY, AND DECLARED OPERABLE AT 1900 MST ON DECEMBER 6, 1987. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE PERSONNEL ERROR CONTRARY TO APPROVED PROCEDURES. THIS EVENT WAS SUBSEQUENTLY INVESTIGATED, AND THE CORRECTIVE ACTIONS REQUIRED TO PREVENT RECURRENCE ARE PROVIDED IN THIS REPORT SUPPLEMENT. THIS LER ALSO PROVIDES A SPECIAL REPORT IN ACCORDANCE WITH T.S. 3.3.3.1 ACTION 28(2) AND 6.9.2. NO SIMILAR EVENTS HAVE BEEN IDENTIFIED.

[163] PALO VERDE 2 DOCKET 50-529 LER 88-005 REV 01
UPDATE ON INADVERTENT SAFETY INJECTION RESULTING FROM PERSONNEL ERROR.
EVENT DATE: 022188 REPORT DATE: 040888 NSSS: CE TYPE: PWR
VENDOR: BORG-WARNER CORP.

(NSIC 208864) ON FEBRUARY 21, 1988, PALO VERDE UNIT 2 WAS IN MODE 5 (COLD SHUTDOWN) AT APPROXIMATELY 170F AND 125 PSIA BEING COOLED-DOWN AND DEPRESSURIZED TO BEGIN A REFUELING OUTAGE. AT APPROXIMATELY 0719 MST AN INADVERTENT SAFETY INJECTION (JE) FROM THE SAFETY INJECTION TANKS (BP)(ACC) OCCURRED AS A RESULT OF LOW PRESSURIZER PRESSURE SIGNALS NOT BEING PROPERLY BYPASSED. THE SAFETY INJECTION WAS ACCOMPANIED BY A CONTAINMENT ISOLATION (BP)(JE) ENGINEERED SAFETY FEATURES (ESF) ACTUATION. THERE WERE NO OTHER ESF ACTUATIONS AND NONE WERE NECESSARY. DURING THE EVENT A HIGH PRESSURE SAFETY INJECTION (HPSI) VALVE (INV) DID NOT FULLY OPEN. ALL OTHER EQUIPMENT OPERATED PER DESIGN. THE ROOT CAUSE OF THE EVENT WAS A COGNITIVE PERSONNEL ERROR ON THE PART OF UTILITY, LICENSED PERSONNEL. ADDITIONALLY DURING THE EVENT, THE HPSI LOOP INJECTION VALVE DID NOT OPEN DUE TO A BLOWN FUSE (FU). AS CORRECTIVE ACTION, APPROPRIATE DISCIPLINARY MEASURES WILL BE TAKEN. THE HPSI LOOP INJECTION VALVE WAS VERIFIED TO OPERATE PROPERLY AFTER REPLACING THE MALFUNCTIONING FUSE. A ROOT CAUSE OF FAILURE WAS INITIATED FOR THE BLOWN FUSE AND THE CAUSE OF THE FUSE OPENING COULD NOT BE DETERMINED. FURTHER TESTING WILL BE CONDUCTED AND THE RESULTS PROVIDED IN A SUPPLEMENT TO THIS REPORT. THERE HAVE BEEN NO PREVIOUS SIMILAR OCCURRENCES.

[164] PALO VERDE 2 DOCKET 50-529 LER 88-008
INCREASED RADIATION LEVEL DURING FUEL INSPECTION CAUSES ESF ACTUATION.
EVENT DATE: 031988 REPORT DATE: 040788 NSSS: CE TYPE: PWR

(NSIC 208980) ON MARCH 19, 1988 AT APPROXIMATELY 2029 MST, PALO VERDE UNIT 2 WAS IN MODE 6 (REFUELING) WHEN A BALANCE OF PLANT ENGINEERED SAFETY FEATURE (BOP ESF) (JE) ACTUATION OCCURRED ON THE FUEL BUILDING ESSENTIAL VENTILATION SYSTEM (FBEVS) (VG) TRAIN "A". THIS ALSO RESULTED IN THE DESIGNED CROSS-TRIP ACTUATION OF FBEVS TRAIN "B" AND THE CONTROL ROOM ESSENTIAL FILTRATION SYSTEM (VI) TRAINS "A" AND "B". THERE WERE NO OTHER ESF ACTUATIONS AND NONE WERE NECESSARY. AS A RESULT OF THE BOP ESF ACTUATIONS, THE ESSENTIAL CHILLED WATER SYSTEM (KM) TRAINS "A" AND "B" ACTUATED. ALL EQUIPMENT OPERATED PER DESIGN. THE BOP ESF ACTUATIONS OCCURRED AS A RESULT OF RADIATION LEVELS INCREASING ABOVE THE "ALARM/TRIP" SETPOINT FOR THE FUEL BUILDING (ND) FUEL POOL AREA RADIATION MONITOR (RU-31) (IL) (RI). THE RADIATION LEVEL INCREASES WERE A RESULT OF FUEL INSPECTION ACTIVITIES IN THE AREA OF RU-31. THERE WERE NO PROCEDURAL DEFICIENCIES OR PERSONNEL ERRORS INVOLVED. AS CORRECTIVE ACTION, THE SETPOINT FOR RU-31 WAS RAISED AND PERSONNEL WORKING IN THE AREA WERE INSTRUCTED TO UTILIZE CAUTION IN THE AREA OF RU-31. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS.

[165] PALO VERDE 3 DOCKET 50-530 LER 88-002
ASME SURVEILLANCE INTERVAL EXCEEDED FOR CONTAINMENT ISOLATION VALVE.
EVENT DATE: 010988 REPORT DATE: 042188 NSSS: CE TYPE: PWR

(NSIC 209030) ON MARCH 24, 1988 PALO VERDE UNIT 3 WAS IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER WHEN IT WAS DISCOVERED THAT SURVEILLANCE TESTING HAD NOT BEEN CONDUCTED WITHIN THE ALLOWABLE TIME INTERVAL FOR A CONTAINMENT ISOLATION VALVE FROM THE CONTAINMENT RADWASTE SUMP (WK). ON DECEMBER 1, 1987 SURVEILLANCE TESTING HAD BEEN CONDUCTED ON THE VALVE IN ACCORDANCE WITH TECHNICAL SPECIFICATION 4.0.5 WHICH REQUIRES TESTING IN ACCORDANCE WITH SECTION XI OF THE ASME BOILER AND PRESSURE VESSEL CODE. THE VALVE MET THE REQUIRED ACCEPTANCE CRITERIA; HOWEVER, THE MEASURED STROKE TIME INCREASED BY MORE THAN 50% FROM THE PREVIOUS TEST. THE VALVE IS REQUIRED TO BE TESTED ONCE PER 3 MONTHS; HOWEVER, WHEN STROKE TIMES INCREASE BY 50% OR MORE RELATIVE TO THE PREVIOUS TEST, ASME SECTION XI REQUIRES THE TESTING FREQUENCY TO BE ADJUSTED TO A MONTHLY INTERVAL. THE TESTING SCHEDULE WAS NOT MODIFIED TO MEET THE MONTHLY SURVEILLANCE INTERVAL

FOR THE VALVE. ON JANUARY 9, 1988 THE MODIFIED SURVEILLANCE INTERVAL WAS EXCEEDED. THE ROOT CAUSE OF THE EVENT WAS EVALUATED TO BE A COGNITIVE PERSONNEL ERROR BY A TECHNICIAN (UTILITY, NON-LICENSED) RESPONSIBLE FOR TRACKING THE COMPLETED TESTS. TO PREVENT RECURRENCE THE INDIVIDUAL WILL RECEIVE APPROPRIATE COUNSELING AND/OR DISCIPLINARY ACTION. A PREVIOUS SIMILAR EVENT OCCURRED AS DESCRIBED IN LER 1-87-002-00.

[166] PALO VERDE 3 DOCKET 50-530 LER 88-004
ENGINEERED SAFETY FEATURE ACTUATION RESULTING FROM LOSS OF POWER.
EVENT DATE: 040688 REPORT DATE: 050588 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 2 (PWR)

(NSIC 209246) AT APPROXIMATELY 1421 MST ON APRIL 6, 1988, PALO VERDE UNIT 3 WAS OPERATING IN MODE 1 (POWER OPERATIONS) AT APPROXIMATELY 100 PERCENT POWER WHEN A LOSS OF POWER EVENT OCCURRED WHICH RESULTED IN A "B" TRAIN ENGINEERED SAFETY FEATURE (ESF) LOAD SEQUENCER SYSTEM ACTUATION (JE). THIS RESULTED IN THE AUTOMATIC START OF THE "B" TRAIN EMERGENCY DIESEL GENERATOR (DG). THERE WERE NO OTHER SAFETY SYSTEM RESPONSES AND NONE WERE NECESSARY. UNIT 3 CONTINUED TO OPERATE AT 100 PERCENT POWER THROUGHOUT THE EVENT. THE EVENT OCCURRED WHEN POST MAINTENANCE TESTING ON A CURRENT TRANSFORMER (XCT) FOR UNIT 2 NON-VITAL BUS (2E-NAN-S05) (BU) (EA) RESULTED IN THE X01 STARTUP TRANSFORMER (2E-NAN-X01) (EA)(XFMR) RELAYING OUT (I.E. TRIPPING). THE X01 STARTUP TRANSFORMER WAS SUPPLYING POWER TO THE UNIT 3 "B" TRAIN VITAL BUS (3E-NAN-S06) PRIOR TO THE EVENT. THE EVENT WAS CAUSED BY AN ERROR IN THE WORK DOCUMENT BEING UTILIZED TO REPLACE THE CURRENT TRANSFORMER. AS IMMEDIATE CORRECTIVE ACTION, THE TESTING WAS STOPPED ON THE UNIT 2 NON- VITAL BUS AND THE APPROPRIATE WORK DOCUMENT REVISIONS INCORPORATED. AS CORRECTIVE ACTION TO PREVENT RECURRENCE, THIS EVENT WILL BE REVIEWED BY PERSONNEL RESPONSIBLE FOR WORK ORDER PREPARATION. THESE PERSONNEL WILL BE RESPONSIBLE FOR DELINEATING INSTRUCTIONS NECESSARY TO ENSURE THAT UNEXPECTED PLANT TRANSIENTS DO NOT OCCUR. THERE HAVE BEEN NO PREVIOUS SIMILAR OCCURRENCES.

[167] PEACH BOTTOM 2 DOCKET 50-277 LER 87-002 REV 01
UPDATE ON LOCAL LEAK RATE TEST LIMIT EXCEEDED DUE TO NORMAL VALVE WEAR.
EVENT DATE: 031387 REPORT DATE: 042188 NSSS: GE TYPE: BWR
VENDOR: ATWOOD & MORRILL CO., INC.
BLACK, SIVALLS & BRYSON, INC.
POWELL-TRINITY EQUIP CO

(NSIC 208984) ON MARCH 13, 1987, THE CONTAINMENT LEAK TEST PROGRAM IDENTIFIED THAT THE TOTAL COMBINED LEAKAGE OF THE TYPE B AND C TESTS EXCEEDED THE ALLOWABLE LEAK RATE LIMIT ESTABLISHED IN PART 50, APPENDIX J, OF THE COMMISSION'S REGULATIONS. AT THE TIME OF THE EVENT, UNIT 2 WAS IN THE PROCESS OF SHUTTING DOWN FOR A REFUELING OUTAGE. SEVERAL VALVES TESTED HAD EXCESSIVE LEAKAGE RATES WHICH CONTRIBUTED TO THE COMBINED LEAKAGE TOTAL. THESE VALVES WERE REPAIRED AND SUCCESSFULLY RETESTED. LOCAL LEAK RATE TEST RESULTS OF THE REDUNDANT ISOLATION VALVES WERE ACCEPTABLE. THEREFORE, CONTAINMENT INTEGRITY WOULD HAVE BEEN MAINTAINED IN THE EVENT OF AN ACCIDENT. THE MAIN STEAM LINE DRAIN VALVES MO-2-2-74. AND MO2-2-77 ALSO HAD LEAKAGE CONCERNS; HOWEVER, THE CONSEQUENCES OF THESE CONCERNS WERE MINIMAL AS EXPLAINED IN LER 2-87-05.

[168] PEACH BOTTOM 2 DOCKET 50-277 LER 87-028 REV 01
UPDATE ON DESIGN DEFICIENCY THAT COULD PERMIT DIESEL GENERATOR TRIPS DURING A SEISMIC EVENT.
EVENT DATE: 121787 REPORT DATE: 042288 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 209167) A DESIGN DEFICIENCY WAS DISCOVERED WHICH COULD RESULT IN DIESEL

GENERATOR TRIPS DURING A LOSS OF OFFSITE POWER (LOOP) EVENT. FOUR RELAYS IN THE DIESEL GENERATOR ROOM CARBON DIOXIDE FIRE SUPPRESSION (CARDOX) SYSTEM CONTROL CIRCUITS, WHICH ARE NOT CLASSIFIED AS SAFETY-RELATED OR SEISMIC, COULD INITIATE DIESEL GENERATOR TRIP SIGNALS DURING A LOOP EVENT IF ACTUATED BY SEISMIC CONDITIONS. THERE ARE FOUR DIESEL GENERATORS COMMON TO UNIT 2 AND UNIT 3, AND EACH DIESEL GENERATOR COULD BE TRIPPED BY ITS RESPECTIVE CARDOX SYSTEM RELAY. THE ORIGINAL DESIGN DOES NOT PREVENT A SEISMIC-INDUCED DIESEL GENERATOR TRIP SIGNAL FROM THESE RELAYS. ON DECEMBER 17, 1987 IT WAS DETERMINED THAT THIS CONDITION WAS REPORTABLE PURSUANT TO 10 CFR 50.73(A)(2)(VI). THE CONDITION WAS DISCOVERED APPROXIMATELY ONE MONTH EARLIER. THIS CONDITION COMPROMISED THE ABILITY TO SAFELY SHUT DOWN THE PLANT DURING A LOOP EVENT CONCURRENT WITH A SEISMIC EVENT. BOTH PEACH BOTTOM UNITS ARE SHUTDOWN. TO CORRECT THIS CONDITION, TWO DIFFERENT RESOLUTIONS ARE BEING EVALUATED. THE FIRST OPTION IS TO UPGRADE THOSE AFFECTED COMPONENTS OF THE CARDOX SYSTEM THAT COULD CAUSE A TRIP DUE TO A SEISMIC EVENT. THE OTHER OPTION IS TO RELOCATE THE DIESEL GENERATOR COMBUSTION AIR INTAKE TO DRAW OUTSIDE AIR, AND REMOVE THE CARDOX INITIATION DIESEL TRIP FEATURE.

[169] PEACH BOTTOM 2 DOCKET 50-277 LER 88-004 REV 01
UPDATE ON RELAY TRIP CAUSING BREAKER TRIP THAT RESULTED IN CONTAINMENT ISOLATIONS AND INOPERABILITY OF FIRE PUMPS.
EVENT DATE: 030288 REPORT DATE: 040888 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 208843) ON MARCH 2, 1988 AT 2104 HOURS WHILE IN COLD SHUTDOWN, THE E-224 EMERGENCY LOAD CENTER TRANSFORMER SUPPLY BREAKER TRIPPED DUE TO THE UNEXPECTED TRIP OF AN UNDERVOLTAGE RELAY, WHICH CAUSED OUTBOARD GROUP II AND III PRIMARY CONTAINMENT ISOLATIONS. THE SYSTEMS INVOLVED INCLUDE SHUTDOWN COOLING, REACTOR WATER CLEANUP, REACTOR BUILDING/REFUEL FLOOR VENTILATION AND STANDBY GAS TREATMENT. BOTH FIRE PUMPS WERE INOPERABLE AS A RESULT OF THE BREAKER TRIP. A 'B' CHANNEL HALF SCRAM SIGNAL WAS ALSO GENERATED. THE BREAKER TRIP WAS CAUSED BY THE UNEXPECTED TRIP OF THE UNDERVOLTAGE RELAY (GENERAL ELECTRIC COMPANY, MODEL NO. 12HGA14AH6A) ON THE E-22 BUS, WHICH FEEDS THE E-224 LOAD CENTER. THE RELAY COIL SUSTAINED SEVERE HEAT DAMAGE WHICH APPARENTLY ALLOWED THE RELAY CONTACTS TO CLOSE AGAIN 30 MINUTES LATER, INITIATING CLOSURE OF THE BREAKER. THE RELAY, OR THE RELAY COIL WILL BE SHIPPED TO THE MANUFACTURER FOR A FAILURE ANALYSIS. THIS LER WILL BE UPDATED TO PROVIDE MORE DETAILS ABOUT THE FAILURE WITHIN 45 DAYS FOLLOWING RECEIPT OF THE ANALYSIS RESULTS. THERE WERE NO ADVERSE SAFETY CONSEQUENCES ASSOCIATED WITH THIS EVENT. A TEMPORARY FEED WAS INSTALLED FOR THE E-224 LOAD CENTER. THE FAILED RELAY WAS REPLACED ON MARCH 6, 1988 AT APPROXIMATELY 1700 HOURS, AND THE E-22 BUS WAS RETURNED TO SERVICE.

[170] PEACH BOTTOM 2 DOCKET 50-277 LER 88-008
FAILURE TO SUBMIT A SPECIAL REPORT CONCERNING INOPERABILITY OF CARDOX SYSTEM IN CONTROL ROOM DUE TO A LACK OF ADEQUATE PROGRAMMATIC CONTROLS.
EVENT DATE: 032888 REPORT DATE: 042688 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 209117) ON MARCH 28, 1988 WHILE UNIT 2 AND UNIT 3 WERE SHUTDOWN, IT WAS DETERMINED BY THE PEACH BOTTOM ATOMIC POWER STATION (PBAPS) REGULATORY GROUP THAT A SPECIAL REPORT CONCERNING THE INOPERABILITY OF THE CARDOX SYSTEM IN THE CONTROL ROOM WAS NOT SUBMITTED WITHIN THE 31 DAY TIME LIMIT AS REQUIRED BY TECHNICAL SPECIFICATION 3.14.B.4.B. THIS CONSTITUTES A FAILURE TO COMPLY WITH THE ACTION STATEMENT OF THE TECHNICAL SPECIFICATIONS. THIS NONCONFORMING CONDITION HAS EXISTED SINCE NOVEMBER 15, 1987. THE SYSTEM WAS REMOVED FROM SERVICE ON OCTOBER 1, 1987 AS A RESULT OF A DISCOVERY BY THE PBAPS FIRE PROTECTION COORDINATOR THAT THE CARDOX HOSE IN THE CONTROL ROOM WAS PRESSURIZED AND HAD BLISTERED. THE CAUSE OF THE FAILURE TO REPORT WAS THE LACK OF ADEQUATE PROGRAMMATIC CONTROLS TO ENSURE PROPER IDENTIFICATION AND COMMUNICATION OF CONDITIONS WHICH REQUIRE REPORTING.

THE CARDOX SYSTEM REMAINS OUT-OF-SERVICE PENDING COMPLETION OF AN EVALUATION OF CONTROL ROOM HABITABILITY UPON A DISCHARGE OF THE CARDOX SYSTEM INTO THE CONTROL ROOM. THE FAILURE TO SUBMIT A SPECIAL REPORT PER TECHNICAL SPECIFICATION 3.14.B.4.B IS A SERIOUS ADMINISTRATIVE DEFICIENCY. TO PREVENT RECURRENCE, APPROPRIATE PROGRAMMATIC CONTROLS WILL BE ESTABLISHED.

[171] PEACH BOTTOM 2 DOCKET 50-277 LER 88-009
PERSONAL ERROR RESULTS IN TREATED LIQUID RADWASTE RELEASE WITHOUT PRIOR SAMPLING.
EVENT DATE: 040488 REPORT DATE: 050288 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 209220) ON APRIL 4, 1988, TREATED LIQUID RADWASTE EFFLUENT WAS RELEASED FROM THE 'A' WASTE SAMPLE TANK (AWST) WITHOUT PRIOR SAMPLING. THIS EVENT OCCURRED AFTER A CHEMISTRY TECHNICIAN WAS DIRECTED TO SAMPLE THE (AWST) BUT MISTAKENLY SAMPLED THE FLOOR DRAIN SAMPLE TANK (FDST). THE SAMPLE WAS ANALYZED AND RECORDED FOR THE AWST. THE AWST WAS THEN DISCHARGED ON THE BASIS OF THIS SAMPLE FROM THE FDST. THE CONSEQUENCES OF THIS EVENT ARE JUDGED TO BE MINIMAL SINCE THE ACTIVITY LEVELS RELEASED WERE BELOW THE LEVELS ORIGINALLY RECORDED AND APPROVED FOR RELEASE. A SAMPLE FROM THE INSTRUMENT LINE ON THE AWST WAS ANALYZED AND INDICATED ACTIVITY LEVELS ABOUT 200 TIMES LESS THAN THOSE OF THE FDST. THE LEVELS OF BOTH THE INSTRUMENT LINE AND THE FDST WERE BELOW THE MAXIMUM PERMISSIBLE CONCENTRATION (MPC) LIMITS DEFINED IN 10 CFR 20, APPENDIX B, TABLE II. THE CAUSE OF THE EVENT WAS A PERSONAL ERROR BY THE CHEMISTRY TECHNICIAN. PROCEDURAL DEFICIENCIES OR INADEQUATE LABELING WERE NOT CONTRIBUTORS TO THE CAUSE. THIS INDIVIDUAL WAS COUNSELED ON THE IMPORTANCE OF ATTENTION TO DETAIL. THE DETAILS OF THIS EVENT WERE DISCUSSED AT A MEETING WITH CHEMISTRY PERSONNEL, AND MINUTES OF THE MEETING WERE DISTRIBUTED TO ALL CHEMISTRY TECHNICIANS. NO FURTHER CORRECTIVE ACTIONS ARE PLANNED. THIS EVENT IS REPORTABLE PURSUANT TO 50.73(A)(2)(I)(B).

[172] PEACH BOTTOM 3 DOCKET 50-278 LER 88-001
ACTUATION OF PRIMARY CONTAINMENT ISOLATION SYSTEM DURING SWITCHING OF A 4KV EMERGENCY BUS FEED DUE TO LACK OF PROCEDURAL GUIDANCE.
EVENT DATE: 040888 REPORT DATE: 050688 NSSS: GE TYPE: BWR

(NSIC 209221) AT 0209 HOURS ON APRIL 8, 1988, THE UNIT 3 PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) GENERATED ISOLATION SIGNALS AND THE REACTOR PROTECTION SYSTEM (RPS) GENERATED A HALF SCRAM SIGNAL. THE ACTUATIONS WERE CAUSED BY THE DE- ENERGIZATION OF THE 'A' RPS BUS DURING THE MANUAL TRANSFER OF A 4KV EMERGENCY BUS FEED. THE PROCEDURES THE OPERATORS USED REQUIRED THE OPERATION OF A DIESEL GENERATOR, BUT DID NOT PROVIDE SUFFICIENT GUIDANCE FOR ITS LOADING. AS A RESULT, THE OPERATORS SELECTED A LOAD WHICH WAS TOO HIGH. DURING THE LOAD TRANSFER, THE "A" RPS BUS, WHICH WAS BEING SUPPLIED BY ITS ALTERNATE FEED, SENSED AN OVER-VOLTAGE CONDITION THEREBY TRIPPING THE OVER-VOLTAGE RELAYS AND DEENERGIZING THE RPS BUS. THE ACTUATION OF THE PCIS, AN ENGINEERED SAFETY FEATURE, MAKES THIS EVENT REPORTABLE PURSUANT TO 10CFR50.73 (A)(2)(IV). PERFORMANCE OF THE 4KV EMERGENCY BUS FEED TRANSFER PROCEDURE WAS HALTED, THE LOGICS WERE RESET, AND THE EVENT WAS REVIEWED. PROCEDURE CHANGES WERE MADE TO FACILITATE THE TRANSFERS, AND ADDITIONAL OPERATOR TRAINING IS BEING REQUIRED.

[173] PERRY 1 DOCKET 50-440 LER 88-011
A MOMENTARY DECREASE OF THE DIESEL GENERATOR CONTROL TACHOMETER RESULTED IN AN UNEXPECTED START OF THE DIESEL GENERATOR BUILDING VENTILATION SYSTEM.
EVENT DATE: 040488 REPORT DATE: 042988 NSSS: GE TYPE: BWR
VENDOR: AIRPAX ELECTRONICS INC.

(NSIC 209239) ON APRIL 4, 1988 AT 10:40, THE 1A TRAIN OF THE DIESEL GENERATOR BUILDING VENT ILATION SYSTEM (DGBVS) FOR DIVISION 1 DIESEL GENERATOR (DG)

UNEXPECTEDLY STARTED. THE DIVISION 1 DG START AND LOAD SURVEILLANCE INSTRUCTION WAS BEING PERFORMED AT THE TIME. THE DGBVS UNEXPECTED START WAS DUE TO A MOMENTARY DECREASE IN THE CONTROL TACHOMETER INDICATION. THE CAUSE OF THE INDICATED TACHOMETER DECREASE HAS NOT BEEN DETERMINED. EXTENSIVE TROUBLESHOOTING HAS BEEN PERFORMED WHICH INCLUDED TESTING OF THE CONTROL TACHOMETER, DIRECT CURRENT POWER SUPPLY, ASSOCIATED CONTROL AIR COMPONENTS AND ASSOCIATED WIRING. HOWEVER, NO DISCREPANCIES HAVE BEEN IDENTIFIED. SINCE THIS EVENT, NO FURTHER UNEXPECTED STARTS OF THE DGBVS HAVE OCCURRED. NO ADDITIONAL CORRECTIVE ACTIONS ARE PLANNED AT THIS TIME; HOWEVER, THE ROUTINE SYSTEM PERFORMANCE MONITORING WILL CONTINUE AS REQUIRED AND WILL IDENTIFY ANY FURTHER PROBLEMS.

[174] PILGRIM 1 DOCKET 50-293 LER 88-010
AUTOMATIC CLOSING OF A PRIMARY CONTAINMENT SYSTEM GROUP 6 ISOLATION VALVE DUE TO PERSONNEL ERROR.
EVENT DATE: 031188 REPORT DATE: 041188 NSSS: GE TYPE: BWR

(NSIC 208932) ON MARCH 11, 1988 AT 2220 HOURS, AN AUTOMATIC ACTUATION OF A PORTION OF THE PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM (PCIS) OCCURRED. THE ACTUATION RESULTED IN THE AUTOMATIC CLOSING OF THE INBOARD PRIMARY CONTAINMENT SYSTEM (PCS) GROUP 6/REACTOR WATER CLEANUP (RHCU) SYSTEM ISOLATION VALVE. FOLLOWING IMMEDIATE INVESTIGATION AND CORRECTIVE ACTION, THE PCIS LOGIC CIRCUIT WAS RESET. THE ISOLATION VALVE WAS RE- OPENED AT APPROXIMATELY 2240 HOURS. THE CAUSE FOR THE ACTUATION WAS A BLOWN FUSE IN THE INBOARD PCIS LOGIC CIRCUITRY. THE FUSE WAS BLOWN DURING THE REMOVAL OF A TEMPERATURE SWITCH FOR A ROUTINE CALIBRATION PER WRITTEN PROCEDURE. THE ROOT CAUSE FOR THE ACTUATION WAS UTILITY TECHNICIAN PERSONNEL ERROR. A CRITIQUE OF THE EVENT IDENTIFIED INTERIM AND LONG TERM CORRECTIVE ACTIONS. THE PROCEDURE USED IS BEING REVISED TO REMOVE THE RWCU SYSTEM FROM SERVICE FOR TEMPERATURE SWITCH CALIBRATIONS. A POSSIBLE CHANGE TO THE FREQUENCY OF CALIBRATIONS AND MODIFICATION OF THE TEMPERATURE SWITCHES OR CONNECTIONS HAS BEEN IDENTIFIED. THE ACTUATION OCCURRED WHILE IN COLD SHUTDOWN. THE REACTOR MODE SWITCH WAS IN THE SHUTDOWN POSITION. THE CONTROL RODS WERE IN THE INSERTED POSITION. THE REACTOR WATER TEMPERATURE WAS 95 DEGREES FAHRENHEIT WITH NEGLIGIBLE CORE DECAY HEAT.

[175] PILGRIM 1 DOCKET 50-293 LER 88-011
INADVERTENT ACTUATION OF SECONDARY CONTAINMENT AND STANDBY GAS TREATMENT SYSTEMS DUE TO PERSONNEL ERROR.
EVENT DATE: 033188 REPORT DATE: 050288 NSSS: GE TYPE: BWR

(NSIC 209224) ON MARCH 31. 1988 AT 1242 HOURS AN INADVERTENT ACTUATION OF THE REACTOR BUILDING ISOLATION CONTROL SYSTEM (RBIS) OCCURRED. THE ACTUATION RESULTED IN THE AUTOMATIC CLOSING OF THE VENTILATION DAMPERS OF THE SECONDARY CONTAINMENT SYSTEM (SCS) AND THE AUTOMATIC START OF THE SCS/STANDBY GAS TREATMENT SYSTEM (SGTS). FOLLOWING IMMEDIATE INVESTIGATION AND CORRECTIVE ACTIONS THE AFFECTED SYSTEMS WERE RESTORED TO NORMAL SERVICE AT 1258 HOURS. THE CAUSE WAS LICENSED UTILITY OPERATOR PERSONNEL ERROR. THE OPERATOR INCORRECTLY PERFORMED A PORTION F A ROUTINE SURVEILLANCE OF THE REACTOR BUILDING REFUEL FLOOR RADIATION MONITORS. A CRITIQUE OF THE EVENT IDENTIFIED THE NEED FOR IMPROVEMENTS TO THE PROCEDURE USED FOR THE SURVEILLANCE. THE IMPROVEMENTS HAVE NOT BEEN COMPLETED AT THE TIME OF SUBMITTAL OF THIS REPORT BUT ARE BEING TRACKED. THIS EVENT OCCURRED DURING AN EXTENDED OUTAGE WHILE IN COLD SHUTDOWN. THE REACTOR MODE SELECTOR SWITCH WAS IN THE SHUTDOWN POSITION. THE CONTROL RODS WERE IN THE INSERTED POSITION. THE REACTOR WATER TEMPERATURE WAS 92 DEGREES FAHRENHEIT WITH NEGLIGIBLE CORE DECAY HEAT. THE REACTOR PRESSURE WAS ZERO PSIG. THE REACTOR POWER LEVEL WAS ZERO MEGAWATTS - THERMAL. THIS EVENT POSED NO THREAT TO THE HEALTH AND SAFETY OF THE PUBLIC.

[176] QUAD CITIES 1 DOCKET 50-254 LER 86-025 REV 01
UPDATE ON TORUS ATTACHED SMALL BORE PIPING DOES NOT MEET CODE ALLOWABLE LIMITS DUE TO DESIGN ERROR.
EVENT DATE: 082786 REPORT DATE: 032388 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 208956) DURING A RE-ANALYSIS OF THE IEB 79-141 MARK I PROGRAM, IT WAS DISCOVERED THAT CERTAIN SMALL BORE TORUS ATTACHED PIPING (FOUR INCHES OR LESS) DID NOT MEET FSAR REQUIREMENTS TO MEET CODE ALLOWABLE STRESS LIMITS FOR SEISMIC AND MARK I LOADING CONDITIONS. PIPING ON BOTH UNITS ONE AND TWO IS AFFECTED. THE CAUSE OF THIS OCCURRENCE IS ATTRIBUTED TO INADEQUATE DESIGN REVIEW. THE AE FIRM ONLY QUALIFIED ADDED OR MODIFIED SUPPORTS ON TORUS ATTACHED PIPING DURING THE INITIAL DESIGN REVIEW. SUPPORTS THAT DID NOT REQUIRE MODIFICATION WERE NOT VERIFIED TO BE QUALIFIED AND WERE SUBSEQUENTLY FOUND TO EXCEED CODE STRESS ALLOWABLES. AN OPERABILITY ASSESSMENT HAS DETERMINED THAT THE SYSTEMS IN QUESTION ARE FUNCTIONAL AND THAT THE SAFETY OF THE PLANTS HAS NOT BEEN JEOPARDIZED. FORTY-FOUR SUPPORTS WERE MODIFIED ON UNIT ONE AND THIRTY-TWO SUPPORTS ARE BEING MODIFIED ON UNIT TWO. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH THE REQUIREMENTS OF 10 CFR 50.73(A)(2)(II).

[177] QUAD CITIES 1 DOCKET 50-254 LER 87-022 REV 01
UPDATE ON REACTOR SCRAM WHILE SHUTDOWN DUE TO NEUTRON MONITOR SPIKED HIGH HIGH DUE TO SHORT IN CONNECTOR.
EVENT DATE: 110987 REPORT DATE: 042888 NSSS: GE TYPE: BWR
VENDOR: AMPHENOL

(NSIC 209209) ON NOVEMBER 9, 1987, QUAD CITIES UNIT ONE WAS IN THE REFUEL MODE WITH FUEL BEING LOADED INTO THE CORE. AT 1941 HOURS, INTERMEDIATE RANGE MONITOR (IRM) 14 SPIKED HIGH HIGH, RESULTING IN A HALF SCRAM ON CHANNEL A OF THE REACTOR PROTECTION SYSTEM (RPS). RPS CHANNEL B ALREADY HAD A HALF SCRAM MANUALLY INSERTED DUE TO MAINTENANCE ON IRMS ON THAT RPS CHANNEL. THEREFORE A FULL REACTOR SCRAM OCCURRED. AT 2000 HOURS, NRC WAS NOTIFIED VIA THE EMERGENCY NOTIFICATION SYSTEM OF THIS EVENT PER 10CFR50.72. THE CAUSE OF THIS EVENT COULD NOT BE DETERMINED BUT COULD HAVE BEEN DUE TO A SHORT CIRCUIT WHICH DEVELOPED DUE TO METAL PARTICLES AND DIRT FOUND IN THE IRM WIRING CONNECTORS. THE METAL PARTICLES ARE LIKELY A PRODUCT OF OXIDATION. PROCEDURES WILL BE REVISED TO INCLUDE INSPECTION AND CLEANING OF THE CONNECTORS WHEN THE SIGNAL CABLE IS DISCONNECTED. THIS REPORT IS PROVIDED PER 10CFR50.73(A)(2)(IV).

[178] QUAD CITIES 1 DOCKET 50-254 LER 87-025 REV 01
UPDATE ON CONTROL ROOM HABITABILITY STUDY VERSUS TECH SPEC DISCREPANCY CONCERNING ALLOWABLE FILTER EFFICIENCY DUE TO ANALYSIS DEFICIENCY.
EVENT DATE: 112587 REPORT DATE: 042188 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 209039) ON NOVEMBER 25, 1987, QUAD CITIES UNIT ONE WAS IN THE REFUEL MODE AT 0 PERCENT POWER AND UNIT TWO WAS IN THE RUN MODE AT 98% POWER. AT 1505 HOURS, THE STATION WAS NOTIFIED OF A DESIGN BASIS ASSUMPTION USED IN THE CONTROL ROOM HABITABILITY STUDY WHICH CONFLICTS WITH CURRENT TECH SPEC REQUIREMENTS. TECH SPECS REQUIRE 90% METHYL IODIDE REMOVAL EFFICIENCIES FOR BOTH STANDBY GAS TREATMENT SYSTEMS (SBGTS) AND THE CONTROL ROOM VENTILATION AIR FILTRATION UNIT. THE STUDY PERFORMED FOR CONTROL ROOM HABITABILITY ASSUMED 99% OVERALL DECONTAMINATION EFFICIENCIES FOR THESE TRAINS. NRC NOTIFICATION OF THIS CONDITION WAS COMPLETE AT 1528 HOURS. THE CAUSE OF THIS CONDITION IS DUE TO AN ANALYSIS DEFICIENCY IN THAT INADEQUATE DESIGN REVIEW WAS PERFORMED. A REVIEW OF RECORDS INDICATES THAT CURRENTLY THESE TRAINS MEET THE REQUIREMENTS OF THE CONTROL ROOM HABITABILITY STUDY. FURTHER ANALYSIS HAS SHOWN THAT THE MAXIMUM THYROID DOSE ALLOWED FOR CONTROL ROOM PERSONNEL WOULD NOT HAVE BEEN EXCEEDED FOR ANY TIME PERIOD IN QUESTION. THIS REPORT IS PROVIDED PER 10CFR50.73(A)(2)(II).

[179] QUAD CITIES 1 DOCKET 50-254 LER 88-006
MISSED TECHNICAL SPECIFICATIONS REQUIRED SURVEILLANCE DUE TO SCHEDULING LAPSE.
EVENT DATE: 032588 REPORT DATE: 041288 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 209000) ON MARCH 25, 1988. UNITS ONE AND TWO WERE BOTH IN THE RUN MODE AT 87 AND 84 PERCENT POWER RESPECTIVELY. AT 1600 HOURS IT WAS DETERMINED THAT THE INSTRUMENT MAINTENANCE (IM) DEPARTMENT'S WEEKLY POWER OPERATION FUNCTIONAL TEST HAD EXCEEDED BOTH CRITERIA OF THE TECHNICAL SPECIFICATIONS REQUIRED INTERVAL, WEEKLY PLUS 25 PERCENT AND 3 CONSECUTIVE SURVEILLANCES WITHIN 3.25 PERCENT OF THE SURVEILLANCE INTERVAL. THE FUNCTIONAL TESTING WAS SATISFACTORILY COMPLETED ON BOTH UNITS WITHIN 35 MINUTES OF DISCOVERY. THE CAUSE FOR THIS EVENT IS MANAGEMENT DEFICIENCY BECAUSE THERE WAS NOT ADEQUATE GUIDANCE OR OTHER MEANS IN PLACE TO ENSURE THESE REQUIRED SURVEILLANCES WERE PERFORMED WHEN KEY IM PERSONNEL WERE ABSENT. A TASK FORCE IS BEING ESTABLISHED TO REVIEW THE EXISTING STATION SURVEILLANCE PROGRAM TO VERIFY ADEQUATE CONTROLS ARE IN PLACE. CORRECTIVE ACTIONS WILL BE IMPLEMENTED AS DEEMED APPROPRIATE. THIS REPORT IS PROVIDED TO COMPLY WITH THE REQUIREMENTS OF 10CFR50.73(A)(2 (I)(B).

[180] QUAD CITIES 1 DOCKET 50-254 LER 88-008
GROUP I ISOLATION IN STARTUP/HOT STANDBY MODE CAUSED BY REACTOR MODE SWITCH ROTATIONAL PLAY.
EVENT DATE: 040888 REPORT DATE: 042688 NSSS: GE TYPE: BWR

(NSIC 209215) ON APRIL 8, 1988, QUAD CITIES UNIT ONE WAS IN THE STARTUP/HOT STANDBY MODE AT APPROXIMATELY TWO (2) PERCENT THERMAL POWER. AT 1800 HOURS, A GROUP I ISOLATION (AND CHANNEL A 1/2 SCRAM) OCCURRED AS REACTOR PRESSURE WAS DECREASING (AT ABOUT 840 PSIG). THIS WAS CAUSED BY BYPASS RELAYS THAT DID NOT ENERGIZE BECAUSE OF ROTATIONAL PLAY IN THE MODE SWITCH WHEN THE MODE SWITCH WAS PLACED FROM RUN TO STARTUP/HOT STANDBY AT 1615 HOURS. THIS FAILURE TO ENERGIZE RESULTED IN PROTECTIVE ACTIONS NORMALLY BYPASSED IN STARTUP/HOT STANDBY TO ACTUALLY BE IN EFFECT. NRC NOTIFICATION WAS COMPLETED AT 1840 HOURS TO COMPLY WITH 10CFR50.72. TO CORRECT THIS FAILURE, THE MODE SWITCH WAS MOVED TOWARD THE REFUEL POSITION AND THEN BACK TO STARTUP/HOT STANDBY. THIS CORRECTED THE PROBLEM IDENTIFIED. TEMPORARY PROCEDURES ARE ADMINISTRATIVELY CONTROLLING THE MOVEMENT OF THE MODE SWITCH AND VERIFICATION OF APPROPRIATE RELAY POSITION. THIS WILL REMAIN IN PLACE UNTIL THE MODE SWITCH CAN BE REPLACED (PER MODIF ICATION M4-1(2)-86-26). THIS REPORT IS PROVIDED PER 10CFR50.73(A)(2)(IV).

[181] QUAD CITIES 2 DOCKET 50-265 LER 87-002 REV 01
UPDATE ON RCIC INOPERABLE DUE TO MODIFICATION DESIGN ERROR.
EVENT DATE: 012287 REPORT DATE: 032288 NSSS: GE TYPE: BWR
VENDOR: TERRY STEAM TURBINE COMPANY

(NSIC 208903) ON 1/22/87, AT 0905 HOURS, QUAD CITIES UNIT TWO WAS OVERSPEED TESTING THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM WHEN IT WAS NOTED THAT THE RCIC TURBINE OUTBOARD BEARING WAS FLOODED WITH OIL AND THE TURBINE INBOARD BEARING OIL LEVEL WAS TOO LOW. THE RCIC SYSTEM WAS ISOLATED AND DECLARED INOPERABLE. THE INBOARD AND OUTBOARD BEARING DRAIN PIPING WAS THEREFORE REPLACED PER MODIFICATION M-4-2-86-28 AND THIS CORRECTED THE RCIC OIL SYSTEM PROBLEM. AT 0350 HOURS ON 1/26/87, RCIC OVERSPEED TESTING WAS AGAIN IN PROGRESS. IT WAS OBSERVED DURING THIS TESTING THAT THE OUTBOARD BEARING OIL SLINGER RING WAS NOT SPINNING AT LOWER TURBINE SPEEDS AND THEREFORE RCIC WAS AGAIN ISOLATED AND DECLARED INOPERABLE. THE BEARING WAS REPLACED, AND DURING REPLACEMENT, IT WAS NOTED THAT AN OIL INLET ORIFICE WAS PLUGGED. THE ORIFICE WAS CLEANED AND THEN REINSTALLED. AT 1135 HOURS, THE RCIC TURBINE WAS OVERSPEED TESTED SUCCESSFULLY AND AT 1935 HOURS, RCIC OPERABILITY TESTING WAS SATISFACTORILY COMPLETED. RCIC WAS THEN DECLARED FULLY OPERABLE. THE CAUSE FOR THE OIL FLOODING OF THE OUTBOARD BEARING WAS A MODIFICATION DESIGN DEFICIENCY THAT WAS INSTALLED DURING THE REFUEL

OUTAGE AND AN ORIGINAL INSTALLATION OF A THERMOWELL THAT CAUSED A PARTIAL OIL FLOW BLOCKAGE. THE SLINGER RING PROBLEM WAS DUE TO A MANUFACTURING ERROR. THE PLUGGED OIL ORIFICE WAS DUE TO POOR OIL CLEANLINESS.

[182] QUAD CITIES 2 DOCKET 50-265 LER 88-005
REACTOR SCRAM DUE TO FEEDWATER REGULATING VALVE PACKING FAILURE.
EVENT DATE: 032088 REPORT DATE: 041188 NSSS: GE TYPE: BWR
VENDOR: CONTROL COMPONENTS

(NSIC 209003) ON MARCH 20, 1988, QUAD-CITIES UNIT TWO WAS IN THE RUN MODE AT 45 PERCENT THERMAL POWER. AT 0123 HOURS, A REACTOR SCRAM (AND RELATED ENGINEERED SAFETY FEATURE ACTUATIONS) OCCURRED DUE TO A SPURIOUS TURBINE TRIP CAUSED BY A PACKING LEAK ON 2B FEEDWATER REGULATING VALVE (FRV). THE LEAKAGE CAUSED A SHORT ACROSS TERMINAL CONTACTS IN BUSSES 21 AND 22 THAT CAUSE THE TURBINE TO TRIP. A GROUP I ISOLATION ALSO OCCURRED DURING THIS EVENT. NRC NOTIFICATION VIA THE EMERGENCY NOTIFICATION SYSTEM WAS COMPLETED AT 0215 HOURS PER 10 CFR 50.72. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO COMPONENT FAILURE (PACKING ON 2B RV). CONTRIBUTING TO THIS EVENT IS THE LACK OF ADMINISTRATIVE CONTROLS TO ENSURE VALVE PACKING IS REPLACED PRIOR TO FAILURE. CORRECTIVE ACTIONS INCLUDED REPAIRS AND INSPECTION OF BOTH FRVS AND MOISTURE REMOVAL AND INSPECTION OF AFFECTED EQUIPMENT. ADDITIONAL CORRECTIVE ACTIONS INCLUDE: 1) DISASSEMBLY AND REPAIR OF THE 2B FRV DURING THE REFUEL OUTAGE, 2) VIBRATION ISOLATOR INSTALLATION ON GROUP I INSTRUMENT RACKS, 3) DEVELOPMENT OF GUIDELINES FOR PACKING REPLACEMENT, 4) EVALUATION OF AREA FLOOR DRAIN SYSTEM AND AREA CABLE PAN COVERS, AND 5) ROUND SHEETS REVISION TO ENSURE ONCE/SHIFT INSPECTION OF FRV. THIS REPORT IS SUPPLIED PER 10 CFR 50.73 (A)(2)(IV).

[183] QUAD CITIES 2 DOCKET 50-265 LER 88-006
FLUED HEAD ANCHORS OUTSIDE SAFETY ANALYSIS DESIGN REQUIREMENTS DUE TO ANALYSIS DEFICIENCY.
EVENT DATE: 040488 REPORT DATE: 050288 NSSS: GE TYPE: BWR

(NSIC 209217) ON APRIL 4, 1988, QUAD-CITIES UNIT TWO WAS IN THE RUN MODE AT 93 PERCENT THERMAL POWER. AT 1410 HOURS, THE STATION WAS NOTIFIED BY THE BWR ENGINEERING DEPARTMENT THAT ELEVEN FLULD HEAD ANCHORS DID NOT MEET THE DESIGN REQUIREMENTS OF THE FINAL SAFETY ANALYSIS REPORT (FSAR). NRC NOTIFICATION OF THIS CONDITION WAS COMPLETED AT 423 HOURS TO SATISFY 10 CFR 50.72. THE CAUSE FOR THIS CONDITION WAS DUE TO MISINTERPRETATION OF SCOPE IN THAT THESE STRUCTURES WERE NOT REASSESSED FOR DESIGN BASE REQUIREMENTS BASED ON IE BULLETIN 9-02 AND 79-14 PROGRAMS. MODIFICATION 04-02-88-017 HAS BEEN INITIATED TO REVISE THE STRUCTURES TO COMPLY WITH SAR REQUIREMENTS. A PROGRAM IS IN PLACE TO ANALYZE THE UNIT ONE STRUCTURES IN A SIMILAR MANNER. THIS REPORT IS PROVIDED TO COMPLY WITH THE REQUIREMENTS OF 10 CFR 50.73(A)(2)(II)(B .

[184] RANCHO SECO DOCKET 50-312 LER 87-040
INOPERABLE REGENERANT HOLD-UP TANK DISCHARGE MONITOR.
EVENT DATE: 081787 REPORT DATE: 091687 NSSS: BW TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 209203) AMENDMENT 53 TO THE TECH SPECS, DATED AUGUST 28, 1984, REVISED TABLE 4.19-1 "RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS." ON AUGUST 17, 1987, IT WAS REPORTED VIA OCCURRENCE DESCRIPTION REPORT 87-863 THAT THE DISTRICT IS NOT COMPLYING WITH TECH SPEC TABLE 4.19-1, NOTE 5. THIS TECH SPEC REQUIRES THAT "DURING PERIODS OF KNOWN ACTIVITY IN THE REGENERANT TANK, PERFORM A SOURCE CHECK DAILY DURING RELEASES VIA THIS PATHWAY." THE DISTRICT HAS NOT BEEN PERFORMING THE REQUIRED SOURCE CHECK ON THE APPLICABLE RADIATION MONITOR (R-15020). AS CORRECTIVE ACTION A TEMPORARY CHANGE TO ADMINISTRATIVE PROCEDURE AP 305.13 "ENVIRONMENTAL RELEASES OF LIQUID

RADIOACTIVITY," WAS MADE TO REQUIRE INDEPENDENT CHECKS OF EACH RELEASE. A TEMPORARY CHANGE TO SURVEILLANCE PROCEDURE SP.200.02 "INSTRUMENTATION SURVEILLANCE PERFORMED EACH DAY" WILL BE MADE BY SEPTEMBER 18, 1987, TO INCORPORATE THE TECH SPEC REQUIRED SOURCE CHECK. IN ADDITION, THE SOURCE CHECKS HAVE BEEN INCORPORATED INTO SURVEILLANCE PROCEDURE SP.002 "DAILY INSTRUMENT CHECKS AND SYSTEM VERIFICATION." SP.002 IS BEING WRITTEN TO SUPERSEDE SP.200.02 AND IS CURRENTLY IN THE REVIEW CYCLE.

[185] RIVERBEND 1 DOCKET 50-458 LER 86-051 REV 02
UPDATE ON SPURIOUS REACTOR WATER CLEANUP SYSTEM ISOLATION DURING TEMPERATURE READING SURVEILLANCE.
EVENT DATE: 081086 REPORT DATE: 041188 NSSS: GE TYPE: BWR
VENDOR: RILEY COMPANY, THE - PANALARM DIVISION

(NSIC 208902) ON 8/10/87 AT 0110, WITH THE UNIT AT FULL RATED POWER, A REACTOR WATER CLEANUP (RWCU) ISOLATION OCCURRED WHEN THE DAILY SURVEILLANCE ON THE LEAK DETECTION TEMPERATURE MODULES WAS CONDUCTED. THE SURVEILLANCE CONSISTS OF TAKING READINGS ON RILEY TEMPERATURE MODULES BY PLACING A "READ-SET" SWITCH TO THE "READ" POSITION. AS HAS BEEN REPORTED IN PREVIOUS LERS (SEE LERS 85-009, 85-031 AND 85-048) THIS ACTION CAN PRODUCE SPURIOUS TRIPS, AS WAS THE CASE ON THIS DATE. GULF STATES UTILITIES HAS INSTALLED A PERMANENT "CAUTION" PLAQUE NEAR THE "READ-SET" SWITCHES WHICH CAUTIONS THE OPERATORS TO PLACE THE BYPASS SWITCHES IN THE "BYPASS" POSITION PRIOR TO TAKING ANY TEMPERATURE READINGS AND TO RESET ANY ISOLATIONS PRIOR TO AND AFTER TAKING ANY TEMPERATURE READINGS. AS EVIDENCED BY THE FACT THAT THERE HAVE BEEN NO SUBSEQUENT RWCU ISOLATIONS DUE TO BYPASS SWITCH MISOPERATION, STRICT ADHERENCE TO THE PERMANENT "CAUTION" PLAQUE SHOULD PREVENT FUTURE RWCU ISOLATIONS OF THIS TYPE. THERE WAS NO ADVERSE IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT SINCE THE SYSTEM ISOLATED THE PRIMARY COOLANT PRESSURE BOUNDARY AT THE CONTAINMENT AS DESIGNED.

[186] RIVERBEND 1 DOCKET 50-458 LER 87-004 REV 01
UPDATE ON DIVISION 1 DIESEL GENERATOR OUTPUT BREAKER FAILURE.
EVENT DATE: 020687 REPORT DATE: 041188 NSSS: GE TYPE: BWR
VENDOR: GOULD BROWN BOVERI COMPANY

(NSIC 208912) AT 1735 ON 02/05/87, WITH THE UNIT AT 100% POWER, THE DIVISION I DIESEL GENERATOR (DG) OUTPUT BREAKER FAILED TO CLOSE DURING THE PERFORMANCE OF A WEEKLY SURVEILLANCE TEST. AN IMMEDIATE INSPECTION OF THE OUTPUT CIRCUIT BREAKER REVEALED THAT A MOUNTING BOLT HAD FALLEN OUT OF THE CLOSING SPRING CHARGING MOTOR RENDERING THE MOTOR INOPERABLE. THROUGH AN INVESTIGATION IT WAS CONCLUDED THAT TECH SPEC 3.8.1.1 HAD BEEN VIOLATED AS A RESULT OF THE DG OUTPUT BREAKER AUTOMATIC CLOSURE FUNCTION BEING INOPERABLE DURING THE PERIOD OF 01/29/87 TO 02/07/87. MODIFICATION REQUEST 87-0141 WAS IMPLEMENTED DURING THE REFUELING OUTAGE WHICH PROVIDED FOR THE APPLICATION OF A THREAD ADHESIVE AND SPECIFIED THE TORQUING REQUIREMENTS FOR THE CHARGING MOTOR MOUNTING BOLTS. THIS MODIFICATION WILL ENSURE THAT A SIMILAR CONDITION WILL NOT DEVELOP IN THE FUTURE. DURING THE 8 DAYS THAT THE DIVISION I DG SHOULD HAVE BEEN DECLARED INOPERABLE THERE WERE NO DEMANDS FOR ITS INITIATION. IF A LOSS OF OFFSITE POWER HAD BEEN EXPERIENCED, THE DIVISION II DG AND HIGH PRESSURE CORE SPRAY DG WERE OPERABLE TO PROVIDE INDEPENDENT A.C. ELECTRICAL POWER SOURCES. ADDITIONALLY, THE OUTPUT CIRCUIT BREAKER COULD HAVE BEEN MANUALLY CLOSED. THEREFORE, THERE WAS NO IMPACT ON THE SAFE OPERATION OF THE PLANT OR TO THE HEALTH AND SAFETY OF THE PUBLIC.

[187] RIVERBEND 1 DOCKET 50-458 LER 87-008 REV 02
UPDATE ON CONTROL ROOM CHARCOAL FILTRATION START DUE TO RADIATION MONITOR SPIKE.
EVENT DATE: 052587 REPORT DATE: 041588 NSSS: GE TYPE: BWR

(NSIC 208987) AT 1550 ON 5/25/87 WITH THE UNIT AT APPROXIMATELY 70 PERCENT POWER,

AN AUTOMATIC INITIATION OF THE "B" TRAIN OF THE MAIN CONTROL ROOM CHARCOAL FILTRATION SYSTEM OCCURRED. THE INITIATION WAS CAUSED BY SPURIOUS SIGNALS FROM MAIN CONTROL ROOM LOCAL INTAKE RADIATION MONITOR 1RMS*RE13B. THE OPERATORS DETERMINED THAT NO ACTUAL HIGH RADIATION CONDITION EXISTED AND RETURNED THE SYSTEM TO ITS NORMAL CONFIGURATION. RADIATION MONITOR 1RMS*RE13B AND OTHER RADIATION MONITORS HAVE PREVIOUSLY SHOWN SUSCEPTIBILITY TO ELECTRICAL NOISE. THESE OCCURRENCES WERE PREVIOUSLY REPORTED IN LERS 86-020 (1RMS*RE11A), 86-040 (1RMS*RE13B AND 1RMS*RE14B), 86- 052 (1RMS*RE13B) AND 86-062 (1RMS*RE11A) AND 88-008 (1RMS*RE11B). THE RESULTS OF AN ENGINEERING STUDY DEMONSTRATE THAT THE NOISE SUSCEPTIBILITY CAN BE ALLEVIATED BY INCREASING THE DETECTOR GAIN AND READJUSTING THE DISCRIMINATOR. THE SAFE OPERATION OF THE PLANT AND HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED AS A RESULT OF THIS EVENT. THE CONTINUED OPERATION OF THE PLANT WILL HAVE NO IMPACT ON SAFETY SINCE SYSTEMS WHICH ACTUATE PLACE THE PLANT IN A MORE CONSERVATIVE CONFIGURATION BY FILTERING THE AIR PRIOR TO RELEASING IT.

[188] RIVERBEND 1 DOCKET 50-458 LER 87-033 REV 01
UPDATE ON LOSS OF SHUTDOWN COOLING DUE TO SPURIOUS EPA BREAKER TRIP.
EVENT DATE: 121187 REPORT DATE: 041588 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 208988) AT APPROXIMATELY 1030 ON 12/11/87 WITH THE UNIT IN COLD SHUTDOWN (MODE 4), A LOSS OF THE DIVISION II REACTOR PROTECTION SYSTEM (RPS) BUS OCCURRED DUE TO THE TRIP OF NORMAL POWER SUPPLY ELECTRICAL PROTECTION ASSEMBLY (EPA) BREAKER C71-S003D. THE LOSS OF THE RPS BUS RESULTED IN ALL REQUIRED DIVISION II ISOLATIONS WHICH CAUSED A LOSS OF RHR "A" SHUTDOWN COOLING. THE BUS WAS RE-ENERGIZED FROM THE ALTERNATE POWER SOURCE, ALL ISOLATIONS RESET AND SHUTDOWN COOLING RESTARTED. THE LOGIC CARD IN THE EPA BREAKER WAS FOUND TO BE MALFUNCTIONING AND TWO INTEGRATED CIRCUIT (IC) CHIPS WERE REPLACED. THE SURVEILLANCE TEST PROCEDURE (STP) WAS SUCCESSFULLY PERFORMED AND THE BREAKER WAS RETURNED TO SERVICE. AT APPROXIMATELY 1857 ON 12/19/87 WITH THE UNIT AGAIN IN MODE 4 FOLLOWING A MANUAL UNIT SHUTDOWN, A SECOND TRIP OF EPA BREAKER C71-S003 OCCURRED. AGAIN, THE BREAKER LOGIC CARD WAS FOUND TO BE DEFECTIVE. THREE INTEGRATED CIRCUIT (IC) CHIPS WERE REPLACED, THE STP SUCCESSFULLY PERFORMED AND THE BREAKER WAS RETURNED TO SERVICE. THE OUTPUT VOLTAGE OF THE MOTOR-GENERATOR SET WAS SUBSEQUENTLY INCREASED SLIGHTLY TO HELP PREVENT RECURRENCE. IN EACH CASE, THE UNIT WAS ALREADY IN COLD SHUTDOWN AND ALL SAFETY SYSTEMS FUNCTIONED AS DESIGNED. FOLLOWING THE FIRST EVENT SHUTDOWN COOLING WAS RESTORED WITHIN TEN MINUTES BY UTILIZING THE ALTERNATE POWER SUPPLY.

[189] RIVERBEND 1 DOCKET 50-458 LER 88-009 REV 01
UPDATE ON UNSEALED FIRE BARRIER PENETRATIONS.
EVENT DATE: 030188 REPORT DATE: 042988 NSSS: GE TYPE: BWR

(NSIC 209240) AT 1000 HOURS ON 3/1/88 WITH THE UNIT AT 100% POWER (OPERATING CONDITION 1), AN UNSEALED PENETRATION WAS DISCOVERED IN A CONTROL BUILDING FIRE WALL ON THE 70' ELEVATION. A SECOND UNSEALED PENETRATION WAS DISCOVERED IN THE 98' ELEVATION OF THE DIESEL GENERATOR BUILDING AT 0830 HOURS ON 3/11/88. ON 3/17/88 AT 1100, AN UNSEALED PENETRATION AND AN UNCOATED STEEL BEAM (FORMING PART OF A FIRE BARRIER ASSEMBLY) WAS DISCOVERED IN AN AUXILIARY BUILDING FIRE WALL IN THE "D" TUNNEL ON THE 70' ELEVATION. ON 3/24/88 AT 1630, INADEQUATELY SEALED PENETRATIONS BETWEEN THE AUXILIARY BUILDING AND THE ANNULUS WERE DISCOVERED. ON 4/12/88 AT 1100, AN OPENING BETWEEN A CONTROL BUILDING FIRE DOOR FRAME AND THE FIRE WALL WAS DISCOVERED. ON 4/14/88 AT 0800, ENGINEERING IDENTIFIED AN OPEN HATCHWAY BETWEEN FIRE AREAS, AN OPEN TRENCH BETWEEN FIRE AREAS AND AN UNQUALIFIED PENETRATION SEAL. A FIRE WATCH WAS IN EFFECT FOR THESE AREAS, EXCEPT FOR THE AREA OF THE CONTROL BUILDING DOOR AT THE TIME OF DISCOVERY. A FIRE WATCH WAS ESTABLISHED IN THE CONTROL BUILDING, SATISFYING THE ACTION STATEMENT IN TECH SPEC 3/4.7.7. THE PENETRATIONS WERE FOUND DURING INSTALLATION OF SECURITY

COMMUNICATIONS CABLE FOR A PLANT MODIFICATION AND REPLACEMENT OF MISSING TERMINATION CABINET HARDWARE. THE ANNULUS PENETRATIONS HAD BEEN BREACHED, WHILE THE REMAINING PENETRATIONS HAD NEVER BEEN SEALED. THE DOOR FRAME HAD BEEN PREVIOUSLY SEALED TO THE WALL.

[190] RIVERBEND 1 DOCKET 50-458 LER 88-010
TECH SPEC SURVEILLANCE REQUIREMENT NOT MET DUE TO SECONDARY CONTAINMENT DOORS AND HATCH OMITTED FROM SURVEILLANCE TEST PROCEDURE.
EVENT DATE: 032588 REPORT DATE: 042588 NSSS: GE TYPE: BWR

(NSIC 209241) AT APPROXIMATELY 1300 HOURS ON 3/25/88, WITH THE UNIT IN OPERATIONAL CONDITION 1 (POWER OPERATION - 100 PERCENT POWER) IT WAS DISCOVERED THAT THE SURVEILLANCE REQUIREMENT FOR TECHNICAL SPECIFICATION 3.6.5 "SECONDARY CONTAINMENT" HAD NOT BEEN PROPERLY PERFORMED FOR ALL REQUIRED DOORS AND EQUIPMENT HATCH COVERS DUE TO A PROCEDURAL OMISSION. THE CAUSE OF THE OMISSION FROM THE PROCEDURE IS CONSIDERED TO BE A RESULT OF INSUFFICIENT PROCEDURAL GUIDANCE AND INADEQUATE ADMINISTRATIVE CONTROLS DURING PROCEDURE DEVELOPMENT AND TURNOVER FROM SUBCONTRACTORS. AS CORRECTIVE ACTION, THE APPLICABLE SURVEILLANCE TEST PROCEDURES (STPS) WERE REVISED, AND A PROCEDURE HISTORY SAMPLING REVIEW IS BEING PERFORMED ON OTHER OPERATIONS STPS. BECAUSE OTHER INSTRUMENTATION, AND EFFECTIVE ADMINISTRATIVE CONTROLS AND TRACKING SYSTEMS WERE AVAILABLE, IT HAS BEEN CONCLUDED THAT IT IS UNLIKELY THAT SECONDARY CONTAINMENT INTEGRITY WAS VIOLATED AT ANY TIME WHEN IT WAS REQUIRED. THEREFORE, THERE WAS NO SIGNIFICANT IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT.

[191] ROBINSON 2 DOCKET 50-261 LER 87-005
PERSONNEL EXPOSURE INCIDENT.
EVENT DATE: 051287 REPORT DATE: 061187 NSSS: WE TYPE: PWR

(NSIC 209201) ON MAY 12, 1987, A CONTRACT EMPLOYEE RECEIVED 0.045 REM IN EXCESS OF THE 1-1/4 REMS WHOLE BODY OCCUPATIONAL DOSE SPECIFIED FOR A CALENDAR QUARTER BY 10CFR20.101(A) WITHOUT A COMPLETED FORM NRC-4 OR SIMILAR RECORD. THIS INADVERTENT OVEREXPOSURE WAS CAUSED BY THE FAILURE OF THE INDIVIDUAL TO CORRECTLY IDENTIFY 1.201 REMS PREVIOUSLY ACCUMULATED IN THE SECOND QUARTER OF 1987 PRIOR TO FIRST ENTRY INTO THE UNIT 2 PROTECTED AREA. BETWEEN 0358 HOURS AND 1016 HOURS ON MAY 12, THE INDIVIDUAL RECEIVED 0.094 REM EXPOSURE WHILE WORKING ON STEAM GENERATOR "A" MANWAY BOLTS. FOLLOWING THIS, THE INDIVIDUAL'S PRIOR EXPOSURE RECORDS WERE RECEIVED AND THE TOTAL ACCUMULATED OCCUPATIONAL DOSE TO THE WHOLE BODY WAS FOUND TO BE 1.295 REMS. THE CONTRACT EMPLOYEE WAS REMOVED FROM SITE AND THE INDIVIDUAL'S MANAGEMENT NOTIFIED OF THE INCIDENT. A RADIATION SAFETY VIOLATION WAS ISSUED AGAINST THE INDIVIDUAL FOR PROVIDING INCORRECT EXPOSURE INFORMATION. ALSO, A PLANT OPERATING EXPERIENCE REPORT WAS PREPARED TO DETAIL THE INCIDENT FOR SITE MANAGEMENT, TO PRECLUDE THE OCCURRENCE OF A SIMILAR EVENT.

[192] ROBINSON 2 DOCKET 50-261 LER 88-007
HVH-2 BREAKER FAILED TO CLOSE ON SAFEGUARD SEQUENCE DURING PERFORMANCE OF SPECIAL TEST.
EVENT DATE: 021888 REPORT DATE: 042188 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 209002) THIS LER IS SUBMITTED TO PROVIDE INFORMATION OF POTENTIAL INTEREST TO THE INDUSTRY. ON FEBRUARY 12, 1988, DURING THE PERFORMANCE OF A SPECIAL TEST WHILE IN COLD SHUTDOWN, A SAFETY INJECTION SIGNAL WAS INITIATED. CONTAINMENT FAN COOLER (HVH-2) FAILED TO START DURING THE SAFEGUARD SEQUENCE. INVESTIGATION OF THE HVH-2 SUPPLY BREAKER REVEALED THAT THE DB-50 BREAKER ALARM SWITCH OPERATED ERRATICALLY. THE ALARM SWITCH WAS REPLACED AND THE BREAKER TESTED SATISFACTORILY. SUBSEQUENT INVESTIGATION REVEALED INTERMITTENT HIGH RESISTANCE ON 11 ADDITIONAL SEPARATE ALARM SWITCHES ON EMERGENCY BUS SWITCHGEAR. THESE

ALARM SWITCHES OPERATED THE ASSOCIATED BREAKER DURING THE SPECIAL TEST BUT WERE REPLACED DUE TO THE INCONSISTENT RESISTANCE READING. THE HIGH RESISTANCE WAS APPARENTLY CAUSED BY ACCUMULATION OF OXIDATION ON THE SECONDARY CONTACTS IN THE ALARM SWITCHES DUE TO INADEQUATE PERIODIC INSPECTIONS/CLEANING OF THE ALARM SWITCH CONTACTS AS A RESULT OF THE LACK OF SPECIFIC GUIDANCE IN THE PROCEDURES USED TO PERFORM THE REFUELING INTERVAL INSPECTION. THE PREVENTIVE MAINTENANCE PROCEDURE IS BEING REVISED TO PROVIDE MORE SPECIFIC GUIDANCE FOR THE ALARM SWITCH CHECK ON DB-TYPE BREAKERS.

[193] ROBINSON 2 DOCKET 50-261 LER 88-006
DIESEL GENERATOR FUEL OIL STORAGE CAPACITY LOAD PROFILE DISCREPANCY.
EVENT DATE: 031588 REPORT DATE: 042288 NSSS: WE TYPE: PWR

(NSIC 209001) ON MARCH 15, 1988, DIESEL LOAD PROFILE/FUEL OIL CONSUMPTION CALCULATIONS WERE REVIEWED TO RESOLVE A DISCREPANCY BETWEEN THE PLANT TECHNICAL SPECIFICATIONS BASIS AND THE FINAL SAFETY ANALYSIS REPORT (FSAR). THE BASIS REQUIRES 25,000 GALLONS FOR OPERATION OF ONE DIESEL CARRYING "MINIMUM SAFETY FEATURES" LOAD FOR SEVEN DAYS. THE FSAR DESCRIBES 25,000 GALLONS AS SUFFICIENT FOR ONE DIESEL CARRYING "FULL LOAD" FOR SEVEN DAYS. DETAILED CALCULATIONS FOUND THAT 25,000 GALLONS IS ONLY MARGINALLY SUFFICIENT FOR ONE DIESEL CARRYING "MINIMUM SAFETY FEATURE" LOAD FOR SEVEN DAYS SINCE THIS FUEL CONSUMPTION IS HIGHLY DEPENDENT ON SELECTIVE LOAD SHEDDING. ONE DIESEL CARRYING "FULL LOAD" FOR SEVEN DAYS WOULD REQUIRE APPROXIMATELY 30,430 GALLONS. REVIEW OF RECORDS INDICATES THIS CAPACITY HAS BEEN MAINTAINED ONSITE FOR THE OPERATING HISTORY OF THE PLANT, AND ADMINISTRATIVE CONTROLS WERE EFFECTED IN 1987 AS FURTHER ASSURANCE. THE SPECIFICATIONS BASIS AND FSAR WILL BE REVISED TO REFLECT THIS VALUE. THIS LER IS SUBMITTED UNDER 10CFR50.73(A)(2)(VI).

[194] SALEM 1 DOCKET 50-272 LER 88-005
TECH SPEC SURVEILLANCE PERFORMED LATE DUE TO INADEQUATE ADMINISTRATIVE CONTROL.
EVENT DATE: 030788 REPORT DATE: 032988 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 208928) ON 03/07/88 AT 1200 HOURS, IT WAS IDENTIFIED THAT TECHNICAL SPECIFICATION SURVEILLANCE 4.7.8.1.2.A, SEALED SOURCE LEAK CHECKS, WAS NOT PERFORMED WITHIN SIX MONTHS FROM THE PRIOR SURVEILLANCE. THE SURVEILLANCE WAS OVERDUE AS OF 03/01/88. THE MISSED SURVEILLANCE WAS IDENTIFIED AS A RESULT OF THE INVESTIGATIVE CORRECTIVE ACTIONS REQUIRED BY RECENT LERS WHICH DEAL WITH OTHER MISSED SURVEILLANCE CONCERNS (E.G., LER 272/88-004-00). THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INADEQUATE ADMINISTRATIVE CONTROLS ASSOCIATED WITH THE NEW COMPUTER BASED WORK ACTIVITY SYSTEM, MANAGED MAINTENANCE INFORMATION SYSTEM (MMIS). THE SEALED SOURCE LEAK CHECK SURVEILLANCE WAS COMPLETED 03/14/88. NO LEAKING SOURCES WERE FOUND. A MANUAL SYSTEM FOR TRACKING SURVEILLANCES WITHIN THE RADIATION PROTECTION DEPARTMENT HAS BEEN IMPLEMENTED. THIS WILL CONTINUE UNTIL THE MMIS IS UPDATED TO HANDLE THIS SURVEILLANCE. THE RADIATION AND CHEMISTRY SURVEILLANCE REVIEW COMMITTED TO BY LER 272/88-004-00 IS CONTINUING. THE PSE&G NQA EVALUATION OF THE ADMINISTRATIVE CONTROL OF SURVEILLANCE RECURRING TASKS HAS BEEN INITIATED (REFERENCE SALEM UNIT 2 LER 311/88-004-00).

[195] SALEM 1 DOCKET 50-272 LER 88-006
10CFR 50 APPENDIX R CABLE DESIGN DEFICIENCY DUE TO DESIGN ERROR.
EVENT DATE: 031888 REPORT DATE: 041488 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 209004) ON MARCH 18, 1988, PSE&G ENGINEERS IDENTIFIED A DESIGN DEFICIENCY CONCERNING THE CABLE (CDC22-CT) WHICH PROVIDES AN ALTERNATE SOURCE OF CONTROL AND FIELD FLASHING POWER TO THE THREE DIESEL GENERATORS (D/GS) DURING A POSTULATED FIRE THAT REQUIRES ALTERNATE SHUTDOWN MEASURES. THIS CABLE ORIGINATES FROM THE

"C" TRAIN 125 VOLT VITAL BUS, RUNS THROUGH A CEILING CABLE TRAY IN THE 460V SWITCHGEAR ROOM, AND TERMINATES IN THE 1C DIESEL GENERATOR CONTROL ROOM. THE NUCLEAR REGULATORY COMMISSION (NRC) REQUIREMENTS IN THE CODE OF FEDERAL REGULATIONS 10CFR 50 APPENDIX R SECTION III.G.3, STATE THAT THE REQUIRED ALTERNATE SHUTDOWN CAPABILITY IS TO BE "INDEPENDENT OF CABLES, SYSTEMS OR COMPONENTS IN THE AREA, ROOM OR ZONE UNDER CONSIDERATION". HOWEVER, CONTRARY TO THIS REQUIREMENT, THE CDC22-CT CABLE IS NOT PHYSICALLY INDEPENDENT OF THE CEILING AREA WHICH IS THE "ZONE UNDER CONSIDERATION". THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO A DESIGN ERROR. CORRECTIVE ACTION INCLUDES RE-ROUTING THE CDC22-CT CABLE IN ACCORDANCE WITH 10CFR APPENDIX R CRITERIA DURING THE NEXT REFUELING OUTAGE. AN HOURLY ROVING FIRE WATCH (PREVIOUSLY ESTABLISHED FOR THE AREA FOR OTHER FIRE PROTECTION CONCERNS) WILL BE CONTINUED UNTIL COMPLETION OF CABLE RE-ROUTINE AND SATISFACTION OF THE OTHER CONCERNS.

[196] SALEM 1 DOCKET 50-272 LER 88-007
FIRE BARRIER DAMPERS INADEQUATE DUE TO INADEQUATE REVIEW OF PROCUREMENT DOCUMENTS.
EVENT DATE: 032388 REPORT DATE: 042288 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 209116) ON MARCH 23, 1988, IT WAS IDENTIFIED THAT SEVERAL SALEM UNIT 1 FIRE DAMPERS ARE NOT FIRE RATED NOR DID SEVERAL DUCT SECTIONS PENETRATING BARRIERS HAVE APPROVED FIRE BARRIER COATING. IT IS ASSUMED SALEM UNIT 2 COMPARABLE DUCTS AND DAMPERS ARE ALSO INADEQUATE. SUBSEQUENTLY, TECHNICAL SPECIFICATION ACTION STATEMENT 3.7.11 WAS ENTERED UPON DISCOVERY OF THE DAMPER AND DUCT INADEQUACIES FOR BOTH SALEM UNIT 1 AND SALEM UNIT 2. THE DAMPERS IN QUESTION WERE INSTALLED IN 1980. THE INSTALLATION WAS TO BE DONE IN ACCORDANCE WITH AN NRC LETTER DATED JANUARY 19, 1979. THIS LETTER, REQUIRING THE INSTALLATION OF 3 HOUR RATED FIRE DAMPERS, WAS SUPERCEDED BY AN APPROVED EXEMPTION REQUEST DATED SEPTEMBER 16, 1982 WHICH ACCEPTED THE INSTALLATION OF 1.5 HOUR RATED FIRE DAMPERS. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO PERSONNEL ERROR IN THE ENGINEERING REVIEW OF PROCUREMENT DOCUMENTATION. IT COULD NOT BE DETERMINED WHY THE ENGINEERING PERSONNEL INVOLVED IN THE PREPARATION AND REVIEW OF THE DESIGN CHANGE PACKAGES DID NOT IDENTIFY THE DESIGN CONFIGURATION CONCERN OR WHY NOT ALL DUCTS REQUIRING FIRE WRAP WERE IDENTIFIED. A CONTINUOUS OR HOURLY ROVING FIRE WATCH (AS APPROPRIATE) HAS BEEN ESTABLISHED FOR THE AREAS WHERE THE INADEQUATE DAMPERS/DUCTS ARE LOCATED.

[197] SALEM 1 DOCKET 50-272 LER 88-008
HOURLY ROVING FIRE WATCH SURVEILLANCE PERFORMED LATE.
EVENT DATE: 032488 REPORT DATE: 042588 NSSS: WE TYPE: PWR

(NSIC 209129) ON MARCH 24, 1988, APRIL 4, 1988 AND APRIL 15, 1988, IT WAS DISCOVERED THAT A FEW HOURLY ROVING FIRE WATCHES WERE LATE BETWEEN 10 AND 32 MINUTES FOR FOUR AREAS. THIS IS CONTRARY TO THE REQUIREMENTS OF TECHNICAL SPECIFICATION ACTION STATEMENT 3.7.11. THESE AREAS REQUIRE A ROVING FIRE WATCH DUE TO INOPERABLE FIRE DAMPERS IN FIRE BARRIERS (REFERENCE SALEM UNIT 1 LER 272/88-007-00). THE ROOT CAUSE HAS BEEN ATTRIBUTED TO PERSONNEL ERROR. NUCLEAR FIRE & SAFETY DEPARTMENT MANAGEMENT HAVE REVIEWED THESE EVENTS. APPROPRIATE CORRECTIVE DISCIPLINARY ACTION HAS BEEN TAKEN. ADDITIONALLY, NUCLEAR FIRE & SAFETY DEPARTMENT MANAGEMENT HAS REVIEWED THESE EVENTS WITH DEPARTMENTAL PERSONNEL. THE SIGNIFICANCE OF PERFORMING HOURLY FIRE WATCH SURVEILLANCES WITHIN THE REQUIRED TIME FRAME WAS STRESSED. ADMINISTRATIVE CONTROLS HAVE BEEN IMPLEMENTED TO MITIGATE FUTURE RECURRENCE OF LATE FIRE WATCHES INCLUDING REDUCTION OF THE TIME PERIOD BETWEEN ROVING FIRE WATCHES AND MORE STRINGENT COMMUNICATION REQUIREMENTS BETWEEN THE INDIVIDUALS CONDUCTING ROVING FIRE WATCHES AND THEIR SUPERVISION.

[198] SALEM 1 DOCKET 50-272 LER 88-009
MANUAL REACTOR TRIP DUE TO LOSS OF EH PUMPS 11 AND 12 DUE TO POOR COMMUNICATION IN CONJUNCTION WITH EQUIPMENT FAILURE.
EVENT DATE: 033088 REPORT DATE: 042888 NSSS: WE TYPE: PWR
VENDOR: ROCHESTER MANUFACTURING COMPANY

(NSIC 209218) ON MARCH 30, 1988, AT 1143 HOURS, NO. 12 EH PUMP (JJ) TRIPPED AND NO. 11 EH PUMP FAILED TO AUTOMATICALLY START. WITH THE LOSS OF THE EH PUMPS, AND DECREASING PRESSURE IN THE CONTROL OIL SYSTEM, THE TURBINE GOVERNOR VALVES (JJ) BEGAN TO DRIFT SHUT. THE REACTOR (AC) WAS SUBSEQUENTLY MANUALLY TRIPPED DUE TO INCREASING TAVG. PRIOR TO THIS OCCURRENCE, LEAKAGE OF EH CONTROL OIL HAD BEEN NOTED ISSUING FROM THE 12 AND 13 MS29 VALVES (JJ) AND THE EH RESERVOIR HAD BEEN REFILLED SEVERAL TIMES BY THE EOS. DURING THIS OCCURRENCE THE MECHANICAL LEVEL INDICATOR HAD APPARENTLY MALFUNCTIONED AS IT CONTINUED TO INDICATE A NORMAL TO HIGH LEVEL ALTHOUGH THE ACTUAL SUMP LEVEL HAD REACHED THE PUMP LOW LEVEL LOCKOUT SETPOINT. THE APPARENT "ROOT CAUSE" OF THIS OCCURRENCE HAS BEEN ATTRIBUTED TO POOR COMMUNICATION RESULTING IN NOT ADEQUATELY INVESTIGATING THE CAUSE OF THE CONSTANT EH HIGH/LOW RESERVOIR LEVEL ALARM IN CONJUNCTION WITH EQUIPMENT FAILURE.

[199] SALEM 2 DOCKET 50-311 LER 88-005
TECH SPEC ACTION STATEMENT 3.7.11.A NON-COMPLIANCE DUE TO HOURLY ROVING FIREWATCH LATE DUE TO PERSONNEL ERROR.
EVENT DATE: 040488 REPORT DATE: 050388 NSSS: WE TYPE: PWR

(NSIC 209225) ON 4/4/88, IT WAS DISCOVERED THAT THE HOURLY ROVING FIRE WATCH WAS 23 MINUTES LATE FOR THE 122' AUX. BLDG. AREA (FIRE AREA 2FA-AB-122B). THIS AREA REQUIRES A ROVING FIRE WATCH DUE TO INOPERABLE FIRE DAMPERS IN FIRE BARRIERS (REFERENCE SALEM UNIT 1 LER 272/88 007-00). LATE FIRE WATCHES ARE CONTRARY TO THE REQUIREMENTS SPECIFIED BY TECH SPEC ACTION STATEMENT 3.7.1. THE LATE FIRE WATCHES WERE IDENTIFIED BY THE FIRE WATCH INDIVIDUAL INVOLVED. THE INDIVIDUAL REPORTED IT TO SUPERVISION IMMEDIATELY UPON RECOGNITION THAT THE FIRE WATCH WAS PERFORMED LATE. THE ROOT CAUSE OF THE LATE FIRE WATCH HAS BEEN ATTRIBUTED TO PERSONNEL ERROR. THE ROVING FIRE WATCH DID NOT COMPLY WITH THE ESTABLISHED SEQUENCE, FOR THE AREAS TO BE CHECKED, RESULTING IN THE LATE FIRE WATCH. THE FIRE WATCH DID NOT CAREFULLY REVIEW THE IMPAIRMENT CHECKLIST WHICH IDENTIFIES THE SEQUENCE IN WHICH TO CONDUCT THE ROVE. NUCLEAR FIRE & SAFETY DEPARTMENT MANAGEMENT HAVE REVIEWED THIS EVENT. APPROPRIATE CORRECTIVE DISCIPLINARY ACTION HAS BEEN TAKEN. ADDITIONALLY, NUCLEAR FIRE & SAFETY DEPARTMENT MANAGEMENT HAS REVIEWED THIS EVENT WITH DEPARTMENTAL PERSONNEL. THE SIGNIFICANCE OF PERFORMING HOURLY FIRE WATCH SURVEILLANCES WITHIN THE REQUIRED TIME FRAME WAS STRESSED. ADMINIS TRATIVE CONTROLS HAVE BEEN EXPANDED TO PRECLUDE FUTURE RECURRENCE OF LATE FIRE WATCHES INCLUDING MORE STRINGENT COMMUNICATION REQUIREMENTS.

[200] SAN ONOFRE 1 DOCKET 50-206 LER 88-002
TECHNICAL SPECIFICATION CONTINUOUS FIRE WATCH INTERRUPTED DUE TO INADEQUATE POST ORDERS.
EVENT DATE: 031788 REPORT DATE: 041588 NSSS: WE TYPE: PWR

(NSIC 208915) AT 2235 ON 3/17/88, CONTRARY TO TECH SPEC 3.14.B.2.A, A CONTINUOUS FIRE WATCH FOR CONTAINMENT WAS INTERRUPTED WHEN THE CONTAINMENT EQUIPMENT HATCH (EH) WAS CLOSED. THE INTERRUPTION WAS NOT RECOGNIZED UNTIL 0720 ON 3/18/88 DURING REVIEW OF FIRE WATCH (FW) LOG ENTRIES BY THE FW SITE COORDINATOR. UPON RECOGNITION, A CONTINUOUS FW WAS RE-ESTABLISHED. THE FW POST HAD BEEN ESTABLISHED IN THE VICINITY OF THE EH TO OBSERVE BOTH THE TURBINE DECK AND CONTAINMENT INTERIOR. THE POST ORDERS FOR BOTH AREAS SPECIFIED A CONTINUOUS WATCH. HOWEVER, FW PERSONNEL FOR THIS POST HAD BEEN ORALLY INSTRUCTED TO CONDUCT HOURLY CHECKS INSIDE THE SECONDARY SHIELD, SINCE THE TURBINE DECK FW WAS NOT REQUIRED BY TS. DUE TO A LACK OF EXPLICIT POST ORDER INSTRUCTIONS, FW PERSONNEL BELIEVED THAT HOURLY CHECKS OF CONTAINMENT WERE ACCEPTABLE. AS A

RESULT, A SEPARATE FW WAS NOT ESTABLISHED FOR THE CONTAINMENT PRIOR TO CLOSURE OF THE EH. AS CORRECTIVE ACTIONS, FW PERSONNEL RESPONSIBLE FOR GENERATING POST ORDERS HAVE BEEN INSTRUCTED TO INCLUDE EXPLICIT WRITTEN INSTRUCTIONS FOR COMBINED FIRE AREAS TO ENSURE FW REQUIREMENTS ARE CLEARLY UNDERSTOOD. EXISTING COMBINED POST ORDERS HAVE BEEN REVISED, AS NECESSARY, TO BE MORE EXPLICIT. NEW COMBINED POST ORDERS ARE NOW RECEIVING ADDITIONAL REVIEW PRIOR TO THEIR ISSUANCE TO ENSURE NO CONFLICTS IN MAINTAINING THE POST EXISTS.

[201] SAN ONOFRE 1 DOCKET 50-206 LER 88-006
BACKUP NITROGEN SYSTEM NOT IN FULL CONFORMANCE WITH DESIGN CRITERIA.
EVENT DATE: 032188 REPORT DATE: 042088 NSSS: WE TYPE: PWR

(NSIC 208989) ON MARCH 21, 1988, WITH UNIT 1 IN MODE 5, IT WAS DETERMINED THAT THE BACKUP NITROGEN SYSTEM (BNS) (EIIS SYSTEM CODE LE) DESIGN CRITERIA HAD NOT BEEN FULLY IMPLEMENTED IN THE DESIGN AND INSTALLATION OF THE SYSTEM. THE BNSS ARE A SEISMICALLY QUALIFIED AUTOMATIC BACKUP FOR THE INSTRUMENT AIR SYSTEM (IAS) IN THE EVENT THAT THE IAS IS UNAVAILABLE TO OPERATE COMPONENTS REQUIRED TO SHUTDOWN UNIT 1. THE BNSS FOR THE MAIN STEAM DUMP VALVES AND THE AUXILIARY FEEDWATER PUMP TURBINE STEAM SUPPLY PRESSURE REGULATOR ARE DESIGNED TO 1) PROVIDE AT LEAST 16 HOURS OF OPERATION WITHOUT REPLACEMENT OF N2 CYLINDERS AND 2) PROVIDE FOR A SUPPLY OF REPLACEMENT N2 CYLINDERS FOR USE BEYOND 16 HOURS. THE SECOND REQUIREMENT WAS NOT FORMALLY IMPLEMENTED. ALTHOUGH WE BELIEVE THAT THE BOTTLES WOULD HAVE BEEN REPLENISHED IN THE REQUIRED TIME, THIS ACTIVITY WAS NOT PROCEDURALIZED. AS ADDITIONAL INFORMATION, THE REACTOR COOLANT SYSTEM (RCS) POWER OPERATED RELIEF VALVES (PORV) ARE ALSO PROVIDED WITH A BNS IN THE EVENT OF A LOSS OF THE IAS. THE RCS OVER PRESSURE MITIGATION SYSTEM (OMS) UTILIZES THESE PORVS TO RELIEVE RCS OVERPRESSURE DURING LOW PRESSURE OPERATIONS. IT WAS DETERMINED THAT THE SURVEILLANCE OF THE BNS FOR THE PORVS WAS NEITHER PROCEDURALIZED NOR PERFORMED FOR MODE 5 OPERATIONS.

[202] SAN ONOFRE 1 DOCKET 50-206 LER 88-005
TWO FIRE PROTECTION SYSTEM VALVES NOT INCLUDED IN THE TECH SPEC PROGRAM.
EVENT DATE: 032988 REPORT DATE: 042888 NSSS: WE TYPE: PWR

(NSIC 209174) ON MARCH 29, 1988, WITH UNIT 1 IN MODE 5, A REVIEW OF FIRE PROTECTION WATER SYSTEM (FPW) TECHNICAL SPECIFICATION 4.15 SURVEILLANCE PROCEDURES REVEALED THAT TWO VALVES WERE NOT ADEQUATELY ADDRESSED IN THE SURVEILLANCE PROGRAM. THE SURVEILLANCE REQUIRES THAT (1) EACH ISOLATION VALVE BE VERIFIED TO BE IN ITS REQUIRED POSITION AT LEAST ONCE EVERY 31-DAYS AND (2) EACH ISOLATION VALVE BE STROKED AT LEAST ONCE EVERY 12 MONTHS IF TESTABLE DURING OPERATION, OTHERWISE EACH VALVE MUST BE TESTED AT LEAST ONCE EVERY 18-MONTHS. THE REVIEW REVEALED THAT AN ISOLATION VALVE ASSOCIATED WITH A HOSE REEL STATION HAS NOT BEEN EITHER STROKE TESTED OR VERIFIED TO BE OPEN SINCE THE EARLY 1980S. THE REVIEW ALSO FOUND THAT THE SERVICE WATER RESERVOIR (SWR) NON-TESTABLE ISOLATION VALVE, SDW-301, HAD NOT BEEN INCLUDED IN THE 18-MONTH STROKE TEST SURVEILLANCE PROCEDURE. THE SWR IS THE PRIMARY FIRE WATER SUPPLY FOR UNIT 1. VALVE SDW-301 WAS NOT INCLUDED IN THE POSITION VERIFICATION SURVEILLANCE PROCEDURE DUE TO AN OVERSIGHT. VALVE FPW-399 WAS NOT INCLUDED IN APPLICABLE SURVEILLANCE PROCEDURES SINCE THE EARLY 1980S DUE TO THE INADEQUATE DESIGN IMPLEMENTATION PROCESSES WHICH EXISTED AT THAT TIME. IT HAS BEEN DETERMINED THAT THESE TWO VALVES HAVE BEEN CONTINUOUSLY CAPABLE OF PERFORMING THEIR FUNCTIONS AND ARE CONSIDERED TO BE OPERABLE.

[203] SAN ONOFRE 2 DOCKET 50-361 LER 87-027 REV 01
UPDATE ON TECHNICAL SPECIFICATION FIRE DOOR SURVEILLANCE DISCREPANCIES.
EVENT DATE: 113087 REPORT DATE: 041488 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 208906) ON 11/30/87, WITH UNIT 2 IN MODE 5 AND UNIT 3 AT 100% POWER, A QUALITY ASSURANCE AUDIT DETERMINED THAT TECH SPEC 4.7.9.1 SURVEILLANCE REQUIREMENTS HAD NOT BEEN FULFILLED FOR 11 FIRE DOORS. SPECIFICALLY, THE 6-MONTH VISUAL INSPECTION OF THE CLOSING MECHANISM AND LATCHES ON 7 OF THE 11 DOORS HAD NOT BEEN ADEQUATELY PERFORMED AND THE 18-MONTH FUNCTIONAL TESTING HAD NOT BEEN PERFORMED ON ANY OF THE 11 DOORS. IN ADDITION, IT WAS IDENTIFIED THAT THE 6-MONTH AND 18-MONTH SURVEILLANCES ON FIRE DOORS CONTAINING TWO LEAVES (DOUBLE DOORS) WERE IN SOME CASES NOT PERFORMED CORRECTLY. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE SUBSEQUENT SURVEILLANCES OF THE FIRE DOORS INVOLVED DEMONSTRATED THEM TO BE OPERABLE. THE CAUSES OF THESE SURVEILLANCE DISCREPANCIES WERE PROCEDURAL DEFICIENCIES AND INADEQUATE TRAINING. THE FIRE DOOR SURVEILLANCE PROCEDURE DID NOT REQUIRE ADEQUATE VISUAL INSPECTION AND FUNCTIONAL TESTING OF VARIOUS DOORS WHEN ALARA CONSIDERATIONS WERE INVOLVED. THE PROCEDURE ALSO DID NOT REQUIRE CYCLING OF WATER-TIGHT DOORS TO SATISFY FUNCTIONAL TESTING REQUIREMENTS. INCORRECT INTERPRETATION OF TS SURVEILLANCE REQUIREMENTS LEAD TO THE PROCEDURAL DEFICIENCIES. ADDITIONALLY, THE TRAINING PROGRAM FOR FIRE DOOR INSPECTORS WAS NOT SUFFICIENTLY PRESCRIPTIVE TO ENSURE COMPLETE SURVEILLANCE OF BOTH LEAVES ON DOUBLE DOORS.

[204] SAN ONOFRE 2 DOCKET 50-361 LER 87-031 REV 01
UPDATE ON MANUAL REACTOR TRIP DUE TO FEEDWATER ISOLATION VALVE FAILING CLOSED.
EVENT DATE: 121787 REPORT DATE: 041488 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 1 (PWR)
VENDOR: CONTROL COMPONENTS
COOPER INDUSTRIES
MAROTTA SCIENTIFIC CONTROLS, INC.

(NSIC 208907) AT 0831 ON 12/17/87, WITH UNIT 2 AT 75% POWER, ONE OF TWO MAIN FEEDWATER ISOLATION VALVES (MFIV) FAILED CLOSED, CAUSING BOTH MAIN FEEDWATER PUMPS TO TRIP ON DISCHARGE PRESSURE. UNIT 2 WAS MANUALLY TRIPPED IN ACCORDANCE WITH OPERATING PRACTICE TO MINIMIZE THE EFFECTS OF THE LOSS OF FEEDWATER. CONCURRENT WITH THE AUTOMATIC INITIATION OF THE EMERGENCY FEEDWATER ACTUATION SYSTEM FOR STEAM GENERATOR #2 (EFAS 2) AS A RESULT OF LOW LEVEL IN STEAM GENERATOR #2, EFAS 1 (FOR SG #1) AND EFAS 2 WERE MANUALLY ACTUATED. PLANT CONDITIONS WERE STABILIZED, AND RECOVERY PROCEEDED NORMALLY. THIS EVENT HAD NO EFFECT ON THE HEALTH AND SAFETY OF PLANT PERSONNEL OR THE PUBLIC SINCE ALL SAFETY SYSTEMS OPERATED AS DESIGNED. THE THREADED CONDUIT CONNECTION TO THE AFFECTED MFIV SOLENOID WAS FOUND LOOSE, AND THE CABLE PENETRATION AREA INTO THE CONDUIT CONNECTOR WAS NOT SEALED. BY ONE OR BOTH OF THESE PATHS, WATER AND FOREIGN MATERIAL ENTERED THE SOLENOID HOUSING AND CAUSED CORROSION OF THE POWER LEADS AND TERMINAL BLOCK. THIS RESULTED IN FAILURE OF THE POWER LEAD TO THE MFIV SOLENOID AND CLOSURE OF THE MFIV. THE MAINTENANCE PROCEDURE FOR REASSEMBLY OF THE SOLENOID DID NOT PROVIDE SUFFICIENT GUIDANCE REGARDING THE REINSTALLATION OF THE CONDUIT AND SEALING OF THE CABLE PENETRATION TO ENSURE THEIR WATER TIGHTNESS.

[205] SAN ONOFRE 2 DOCKET 50-361 LER 88-004
SPURIOUS CONTROL ROOM ISOLATION SYSTEM ACTUATION DUE TO SPIKE ON RADIATION MONITOR RT-7825.
EVENT DATE: 031088 REPORT DATE: 041188 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 208940) AT 0925 ON MARCH 10, 1988, WITH UNIT 2 AND UNIT 3 AT 100% POWER, A CONTROL ROOM ISOLATION SYSTEM (CRIS) TRAIN B ACTUATION OCCURRED WHEN BOTH CHANNELS OF CRIS TRAIN B RADIATION MONITOR RT-7825 SPIKED, RESULTING IN THE GAS CHANNEL, RT-7825B2, EXCEEDING ITS ACTUATION SETPOINT. ALL CRIS TRAIN B COMPONENTS WERE VERIFIED TO OPERATE IN ACCORDANCE WITH DESIGN. AIR SAMPLE RESULTS VERIFIED THE ACTUATION TO BE SPURIOUS, AND CRIS TRAIN B WAS RESET AND THE VENTILATION LINEUP RETURNED TO NORMAL AT 1110. THE RADIATION MONITOR CABLING AND DETECTOR MODULE WERE INSPECTED, BUT THE CAUSE OF THE ACTUATION COULD NOT BE

DETERMINED. THE INVESTIGATION OF THE ROOT CAUSE IS CONTINUING, AND A SUPPLEMENTAL REPORT WILL BE SUBMITTED UPON THE COMPLETION OF THIS INVESTIGATION. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE RADIATION LEVELS REMAINED NORMAL AND ALL CRIS TRAIN B COMPONENTS OPERATED IN ACCORDANCE WITH DESIGN.

[206] SAN ONOFRE 2 DOCKET 50-361 LER 88-008
COMPONENT COOLING WATER SYSTEM LEAKAGE EXCEEDS DESIGN CRITERIA.
EVENT DATE: 033088 REPORT DATE: 042988 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 209230) FOLLOWING A DESIGN BASIS EVENT, THE COMPONENT COOLING WATER (CCW) SYSTEM MUST EITHER (1) HAVE A SEISMIC CATEGORY 1 MAKEUP WATER SOURCE, OR (2) BE CAPABLE OF OPERATING WITHOUT MAKEUP FOR 7 DAYS WITH PROVISIONS FOR SUPPLYING MAKEUP WATER THROUGH TEMPORARY CONNECTIONS WITHIN 7 DAYS. AS INDICATED IN RESPONSE TO FSAR QUESTION 010.49, WHICH, AS PART OF THE FSAR, WAS ACCEPTED BY THE NRC DURING THE LICENSING PROCESS, THE CCW DESIGN LEAKAGE WAS ANALYZED TO BE SUFFICIENTLY SMALL TO ALLOW OPERATION FOR 122 DAYS WITHOUT MAKEUP; THEREFORE, NO SEISMICALLY QUALIFIED MAKEUP CAPABILITY WAS PROVIDED. IN 1983, IT WAS RECOGNIZED THAT THE ASSUMED DESIGN LEAKAGE WAS UNREALISTICALLY LOW, AND THAT THE CCW SYSTEM WAS NOT SUFFICIENTLY LEAK-FREE TO THE DEGREE NECESSARY TO MEET THIS REQUIREMENT. CONSEQUENTLY, A DESIGN MODIFICATION WAS IMPLEMENTED IN 1984 WHICH PROVIDED THE CAPABILITY TO SUPPLY MAKEUP WATER FROM EXISTING SEISMICALLY QUALIFIED MOBILE TANKERS TO THE CCW SURGE TANKS, AND THE FSAR WAS SUBSEQUENTLY UPDATED TO REFLECT THIS CHANGE. ON 3/30/88, AN EVALUATION OF THE CCW DESIGN CRITERIA WITH RESPECT TO ALLOWABLE SYSTEM LEAKAGE RATES WAS COMPLETED. FROM THIS EVALUATION AND CCW OPERATING HISTORY, IT WAS DETERMINED THAT THE SYSTEM MAY HAVE OPERATED OUTSIDE ITS DESIGN BASIS FOR LEAKAGE PRIOR TO THE 1984 DESIGN MODIFICATION. BECAUSE THIS CONDITION WAS RECOGNIZED IN 1983, THIS LER IS DELINQUENT.

[207] SEQUOYAH 1 DOCKET 50-327 LER 86-022 REV 01
UPDATE ON INADVERTENT CONTAINMENT VENTILATION ISOLATION CAUSED BY RADIATION MONITOR SPIKES.
EVENT DATE: 051586 REPORT DATE: 042888 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 209247) THIS LER IS BEING REVISED TO UPDATE THE CORRECTIVE ACTION SECTION TO ACCURATELY REFLECT THE RECURRENCE CONTROL. AN INADVERTENT CONTAINMENT VENTILATION ISOLATION (CVI) OCCURRED WHILE PLACING THE CONTAINMENT PURGE SYSTEM IN SERVICE FOR THE UNIT 1 CONTAINMENT BUILDING. THE TRAIN "A" RADIATION MONITOR (RM) FOR THE PURGE AIR SYSTEM SPIKED CAUSING THE CVI. THE SPIKE WAS CAUSED BY ELECTROMAGNETIC INTERFERENCE (EMI) GENERATED BY SWITCH CHATTER ON THE RM. THE SWITCH IS FOR THE LOW-FLOW ALARM AND THE HIGH-VACUUM ALARM WHICH HAD ALARMED IN THE MAIN CONTROL ROOM (MCR) SHORTLY BEFORE THE CVI OCCURRED. THE SWITCH WAS PROBABLY VERY CLOSE TO THE ACTUATION SETPOINT AFTER THE PURGE SYSTEM WAS ACTUATED BECAUSE OF THE EXTRA AIR FLOW WHICH WOULD CAUSE MORE VACUUM IN THE RM SAMPLE LINE THAN NORMAL. THE TRAIN "B" RM DID NOT SPIKE, AND NEITHER OF THE RMS INDICATED ANY RADIATION LEVELS ABOVE NORMAL. ANOTHER CVI (UNIT 2) OCCURRED ON THE CONTAINMENT LOWER COMPARTMENT RM LATER THE SAME DAY AS A RESULT OF REMOVING POWER FROM THE RM IN AN EFFORT TO AVOID AN INADVERTENT CVI. NO RADIATION LEVELS ABOVE NORMAL ACTUALLY EXISTED. ALL EQUIPMENT AND PERSONNEL RESPONDED AND PERFORMED AS EXPECTED DURING THE CVIS. THE CVIS WERE RESET, AND ALL EQUIPMENT WAS RETURNED TO NORMAL IN EACH CASE.

[208] SEQUOYAH 1 DOCKET 50-327 LER 87-027 REV 02
UPDATE ON SURVEILLANCE REQUIREMENT NOT FULFILLED BECAUSE POSITION OF FOUR ESSENTIAL RAW COOLING WATER VALVES WERE NOT VERIFIED DUE TO PROCEDURAL ERROR.
EVENT DATE: 060987 REPORT DATE: 042188 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 209040) THIS LER IS BEING REVISED TO (1) DESCRIBE DEFICIENCIES THAT WERE DISCOVERED DURING THE MOST RECENT PERFORMANCE OF SURVEILLANCE INSTRUCTION (SI)-682, "ERCW FLOW BALANCE VALVE POSITION VERIFICATION," AND (2) PROVIDE ADDITIONAL INFORMATION REGARDING THE CORRECTIVE ACTION TAKEN BY TVA. ON JUNE 9, 1987, WITH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN), IT WAS DETERMINED THAT FOUR ESSENTIAL RAW COOLING WATER (ERCW) VALVES, WHICH PROVIDE RAW COOLING WATER TO THE MAIN CONTROL ROOM AND ELECTRICAL BOARD ROOM AIR CONDITIONING SYSTEMS, WERE NOT BEING VERIFIED TO BE IN THEIR CORRECT POSITION WITH THE FREQUENCY REQUIRED BY SURVEILLANCE REQUIREMENT 4.7.4.A. SI-33, "ERCW VALVES SERVICING SAFETY-RELATED EQUIPMENT," IMPROPERLY VERIFIED THAT THE SUBJECT VALVES WERE "OPEN" EVERY 31 DAYS. SI-682 CORRECTLY VERIFIED THESE VALVES WERE "THROTTLED;" HOWEVER, SI-682 WAS PERFORMED ONLY EVERY 92 DAYS. THIS CONDITION WAS DETERMINED TO BE THE RESULT OF A PROCEDURAL DEFICIENCY. BECAUSE THE VALVES WERE BEING VERIFIED IN THE CORRECT POSITION EVERY 92 DAYS, THERE IS A HIGH DEGREE OF CONFIDENCE THAT THE VALVES WERE NEVER PLACED IN THE FULL OPEN POSITION. IN ORDER TO CORRECT THIS DEFICIENCY, SI-682 HAS BEEN REVISED TO FULFILL SR 4.7.4.A, AND SI-33 HAS BEEN REVISED TO DELETE THE SUBJECT VALVES FROM THE PROCEDURE.

[209] SEQUOYAH 1 DOCKET 50-327 LER 88-011 REV 01
UPDATE ON AN INADEQUATE DETERMINATION OF THE EFFECT OF ROUTINE DIESEL GENERATOR TESTING RESULTED IN BOTH TRAINS OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM BEING INOPERABLE.
EVENT DATE: 021688 REPORT DATE: 042188 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 209047) THIS LER IS BEING REVISED TO PROVIDE ADDITIONAL INFORMATION RELATING TO THE CORRECTIVE ACTIONS TAKEN BY TVA TO PREVENT THE RECURRENCE OF THIS EVENT. ON FEBRUARY 16, 1988, AT APPROXIMATELY 0030 EST, SEQUOYAH NUCLEAR PLANT (SQN) UNIT 1 WAS IN MODE 5 (COLD SHUTDOWN) AND SQN UNIT 2 WAS IN MODE 4 (HOT SHUTDOWN) WHEN IT WAS DETERMINED THAT LIMITING CONDITION FOR OPERATION (LCO) 3.0.5 HAD BEEN IN EFFECT SINCE APPROXIMATELY 2340 EST THE PREVIOUS DAY BECAUSE BOTH TRAINS OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) HAD BEEN INOPERABLE SINCE THAT TIME. TRAIN "A" OF CREVS WAS INOPERABLE BECAUSE ITS EMERGENCY POWER SOURCE (I.E., DIESEL GENERATOR (D/G) 1A-A) HAD BEEN TAKEN OUT OF SERVICE TO PERFORM ROUTINE TESTING, AND TRAIN "B" OF CREVS HAD PREVIOUSLY BEEN DECLARED INOPERABLE TO REPLACE FIRE DETECTORS IN THE CREVS DUCTWORK. UPON DISCOVERY OF THIS EVENT, OPERATIONS PERSONNEL WERE IMMEDIATELY DISPATCHED TO THE D/G BUILDING WITH INSTRUCTIONS TO RETURN D/G 1A-A TO SERVICE. AT APPROXIMATELY 0037 EST ON FEBRUARY 16, 1988, D/G 1A-A WAS RETURNED TO SERVICE AND LCO 3.0.5 WAS EXITED. THE EVENT WAS CAUSED BY AN INADEQUATE REVIEW OF THE APPLICABLE TECH SPEC BEFORE REMOVING A D/G FROM SERVICE.

[210] SEQUOYAH 1 DOCKET 50-327 LER 88-015
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT TO SAMPLE CONTAINMENT NOT PROPERLY IMPLEMENTED BEFORE CONTAINMENT PURGE OPERATIONS DUE TO INCOMPLETE PROCEDURE REVIEW.
EVENT DATE: 031788 REPORT DATE: 040888 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 208965) ON 3/17/88, AT 1500 EST WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN) AND UNIT 2 IN MODE 3 (HOT STANDBY), IT WAS DISCOVERED THAT TECH SPEC SURVEILLANCE REQUIREMENT (SR) 4.11.2.1.2 WAS NOT COMPLETELY INCORPORATED INTO THE IMPLEMENTING PROCEDURE. TS SR 4.11.2.1.2 IS IMPLEMENTED, IN PART, BY SURVEILLANCE INSTRUCTION (SI)-410.2, "CONTAINMENT (UPPER, LOWER) PURGE." TS SR 4.11.2.1.2 REQUIRES AN UPPER AND LOWER CONTAINMENT SAMPLE TO BE TAKEN FOR RADIOACTIVITY ANALYSIS BEFORE ALL CONTAINMENT PURGE OPERATIONS. SI-410.2, HOWEVER, ONLY REQUIRED A SAMPLE TAKEN AND ANALYZED OF THE COMPARTMENT TO BE PURGED SINCE THE CONTAINMENT PURGE SYSTEM CAN BE ALIGNED FOR PURGING THE UPPER OR THE LOWER CONTAINMENT COMPARTMENTS. THE CAUSE OF SI-410.2 INCOMPLETELY IMPLEMENTING TECH SPEC SR 4.11.2.1.2 IS ATTRIBUTED

TO AN OVERSIGHT DURING THE SEQUOYAH INITIAL SI-1, "SURVEILLANCE PROGRAM," APPENDIX F REVIEW OF SI-410.2. SI-1, APPENDIX F, IS USED TO REVIEW SIS IN ORDER TO ENSURE TECH SPEC SR ARE PROPERLY IMPLEMENTED. THE OVERSIGHT MADE DURING THE SI-410.2 REVIEW WAS THE FAILURE TO REQUIRE BOTH CONTAINMENT COMPARTMENTS BE SAMPLED AND ANALYZED IN SI-410.2. SI-1, APPENDIX F REVIEWS WERE CONDUCTED ON ALL SIS DURING INITIAL PROGRAM IMPLEMENTED IN 1986. NO IMMEDIATE OPERATOR ACTIONS WERE REQUIRED BECAUSE A CONTAINMENT PURGE WAS NOT IN PROGRESS AT THE TIME OF THE DISCOVERY.

[211] SEQUOYAH 1 DOCKET 50-327 LER 88-016
AN INADVERTENT MAIN STEAM LINE ISOLATION SIGNAL CAUSED BY A HIGH STEAM FLOW SIGNAL FROM AN UNKNOWN SOURCE.
EVENT DATE: 032488 REPORT DATE: 041588 NSSS: WE TYPE: PWR

(NSIC 208966) ON MARCH 24, 1988, AT APPROXIMATELY 1021 EST WITH UNIT 1 IN MODE 5, INADVERTENT MAIN STEAM LINE ISOLATION SIGNAL OCCURRED. THIS SIGNAL WAS GENERATED WHEN AN INADVERTENT ACTUATION OF HIGH STEAM FLOW BISTABLE 1-FS-1-10B OCCURRED. AT THE TIME OF THIS EVENT, ANOTHER HIGH STEAM FLOW BISTABLE (1-FS-1-21A) WAS ALREADY IN THE TRIPPED CONDITION BECAUSE OF ONGOING MAINTENANCE WORK. ALSO, THE LO-LO TAVG (REACTOR COOLANT SYSTEM (RCS) AVERAGE TEMPERATURE BELOW 540 DEGREES F) AND LOW STEAM LINE PRESSURE (BELOW 600 PSIG) SIGNALS WERE PRESENT BECAUSE OF THE PLANT BEING IN MODE 5. THEREFORE, ALL THE REQUIRED LOGIC WAS COMPLETED (HIGH STEAM FLOW IN TWO OUT OF FOUR LOOPS COINCIDENT WITH LO-LO TAVG OR LOW STEAM LINE PRESSURE IN TWO OUT OF FOUR LOOPS) TO GIVE THE ENGINEERED SAFETY FEATURE (ESF) ACTUATION SIGNAL. THE SAFETY INJECTION (SI) SIGNAL ALSO GENERATED FROM THIS LOGIC WAS BLOCKED AS ALLOWED BY TECH SPEC 3.3.2.1 BELOW PERMISSIVE P-12 (TAVG BELOW 540 DEGREES F). THEREFORE, SINCE THE MAIN STEAM ISOLATION VALVES WERE ALREADY CLOSED FOR MODE 5 AND THE AUTOMATIC SI CIRCUITRY WAS BLOCKED AS ALLOWED BY TECH SPEC, NO EQUIPMENT WAS ACTUATED. AN IMMEDIATE INVESTIGATION WAS INITIATED INTO THE ROOT CAUSE OF THE INADVERTENT BISTABLE ACTUATION (1-FS-1-10B), BUT NO ROOT CAUSE HAS BEEN DETERMINED.

[212] SEQUOYAH 1 DOCKET 50-327 LER 88-017
INADEQUATE TAGGING OF A RADIATION MONITOR PUMP SWITCH RESULTS IN A CONTAINMENT VENTILATION ISOLATION.
EVENT DATE: 033188 REPORT DATE: 042188 NSSS: WE TYPE: PWR

(NSIC 209048) ON MARCH 31, 1988, AT 1946 EST, WHILE UNIT 1 WAS IN MODE 5 (0 PERCENT POWER, 6 PSIG, 125 DEGREES F), A TRAIN "A" CONTAINMENT VENTILATION ISOLATION OCCURRED AT SEQUOYAH NUCLEAR PLANT UNIT 1. WHILE PERFORMING ROUTINE AUXILIARY BUILDING ROUNDS, AN ASSISTANT UNIT OPERATOR (AUO) NOTICED THAT THE ABNORMAL FLOW ALARM LIGHT FOR THE CONTAINMENT PURGE EXHAUST RADIATION MONITOR (RM) 1-RM-90-130 (EIIS CODE IL) WAS ILLUMINATED ON THE LOCAL PANEL FOR THE RM. IN ORDER TO CLEAR THE ABNORMAL FLOW INDICATOR, THE AUO JOGGED THE RM SAMPLE PUMP SWITCH. UPON DEPRESSING THE PUMP SWITCH STOP BUTTON, THE AUO RECEIVED INDICATION OF AN INCREASE RADIATION COUNT RATE THAT EXCEEDED THE MONITORS TRIP VALUES RESULTING IN A TRIP ALARM. THE AUO THEN NOTIFIED THE UNIT 1 REACTOR OPERATOR (RO) OF HIS ACTIONS AND OBSERVATIONS. UPON RECEIVING THE AUO REPORT, THE RO CHECKED THE STATUS OF THE CONTAINMENT VENTILATION SYSTEMS BY REVIEW OF THE MAIN CONTROL ROOM ASSOCIATED ISOLATION VALVES INDICATOR LIGHTS AND FOUND THAT A TRAIN "A" CVI HAD OCCURRED. THE RO VERIFIED THAT A HIGH CONTAINMENT RADIATION CONDITION DID NOT EXIST BY REVIEWING THE 1-RM-90-130 TRACE RECORDING IN THE MAIN CONTROL ROOM, THEN PROCEEDED TO RETURN THE CONTAINMENT VENTILATION SYSTEM TO NORMAL IN ACCORDANCE WITH SYSTEM OPERATING INSTRUCTION (SOI)-30.2B, "CONTAINMENT VENTILATION SYSTEM ISOLATION."

[213] SEQUOYAH 2 DOCKET 50-328 LER 88-003 REV 01
UPDATE ON ICE BUILDUP IN THE FLOW PASSAGES OF THE ICE CONDENSER DUE TO SUBLIMITATION WHICH COULD RESULT IN INCREASED CONTAINMENT PRESSURES.
EVENT DATE: 011988 REPORT DATE: 040888 NSSS: WE TYPE: PWR

(NSIC 208967) THIS LER IS BEING REVISED TO DETAIL THE ROOT CAUSES AND ACTIONS TO PREVENT RECURRENCE OF THIS CONDITION. THIS LER IS BEING SUBMITTED AS A VOLUNTARY REPORT TO INFORM NRC OF THE DEGRADATION FOUND WHILE INSPECTING THE UNIT 2 ICE CONDENSER. ON 1/19/88, WITH UNITS 1 AND 2 IN MODE 5 (0% POWER, 4 PSIG, 124 DEGREES F AND 0% POWER, 110 PSIG, 117 DEGREES F, RESPECTIVELY), AN INSPECTION OF THE UNIT 2 ICE CONDENSER REVEALED THAT ACCUMULATION OF ICE OR FROST IN THE FLOW PASSAGES REPRESENTED A DEGRADED CONDITION. TECH SPEC 3.6.5.1 INDICATES MORE THAN ONE RESTRICTED FLOW PASSAGE PER ICE CONDENSER BAY, THERE ARE 24 BAYS IN THE ICE CONDENSER, IS EVIDENCE OF ABNORMAL DEGRADATION OF THE ICE CONDENSER. A SPECIAL MAINTENANCE INSTRUCTION (SMI) WAS WRITTEN BY TVA AND WESTINGHOUSE TO DETERMINE THE EXTENT OF THE DEGRADATION IN ORDER TO JUSTIFY THAT SEQUOYAH NUCLEAR PLANT (SQN) MEET THE INTENT OF TECH SPEC 3.6.5.1. WESTINGHOUSE HAD PREVIOUSLY COMPLETED A COMPUTER ANALYSIS WHICH CONCLUDED WITH 15 PERCENT OF THE ICE CONDENSER FLOW PASSAGES BLOCKED, THE CONTAINMENT PRESSURE WOULD BE WITHIN DESIGN LIMITS. THE SMI WAS WRITTEN WITH AN ACCEPTANCE CRITERIA OF LESS THAN 15 PERCENT TOTAL FLOW PASSAGE BLOCKAGE.

[214] SEQUOYAH 2 DOCKET 50-328 LER 88-005 REV 01
UPDATE ON LOOSENING OF GLAND SEAL BOLTS ON SPEED INCREASER LUBE OIL PUMPS CAUSES A POTENTIAL INOPERABILITY OF BOTH UNIT 2 CENTRIFUGAL CHARGING PUMPS.
EVENT DATE: 021288 REPORT DATE: 040888 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 1 (PWR)
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 208968) ON FEBRUARY 12, 1988, AT APPROXIMATELY 1133 EST, SMOKE WAS DISCOVERED COMING FROM THE SPEED INCREASER UNIT FOR THE 2A-A CENTRIFUGAL CHARGING PUMP (CCP). IMMEDIATELY, THE 2B-B CCP WAS STARTED, AND THE 2A-A CCP WAS STOPPED. UPON DISASSEMBLY OF THE 2A-A CCP SPEED INCREASER, MUCH OF THE INTERNALS WERE FOUND DAMAGED. FURTHER INVESTIGATION FOUND THE TWO GLAND SEAL (GS) RETAINING BOLTS INSIDE THE SPEED INCREASER LUBE OIL PUMP (SILOP) BACKED OUT ALLOWING THE GS TO LOOSEN. THE GS BEING LOOSENED CAUSED REDUCED OIL FLOW TO THE SPEED INCREASER INTERNALS AND ULTIMATE DAMAGE. THE 2B-B AND 1B-B SILOPS WERE INSPECTED, AND THE SAME GS BOLTS AS ON THE 2A-A PUMP WERE FOUND LOOSENED. THE CAUSE OF THE BOLTS BACKING OUT WAS DETERMINED TO BE LACK OF A PERIODIC ADJUSTMENT OF THE GS BOLTS. IT WAS DISCOVERED DURING INVESTIGATION THAT THE ORIGINAL SILOPS FOR 2A-A, 2B-B, AND 1B-B CCPS HAD BEEN REPLACED WITH INCORRECT SILOPS. THE ORIGINAL 1A-A SILOP WAS NOT REPLACED WITH AN INCORRECT SILOP. THE REPLACEMENT SILOPS HAD BEEN ORDERED USING AN INCORRECT PART NUMBER IN APRIL 1985. THE REPLACEMENT SILOPS FOR 1B-B, 2A-A, AND 2B-B WERE RATED FOR 900 RPM AND INCORPORATED A COMPRESSION PACKING SEAL WHICH REQUIRES PERIODIC ADJUSTMENT AS THE PACKING WEARS.

[215] SEQUOYAH 2 DOCKET 50-328 LER 88-013
ROD CONTROL SYSTEM DEFICIENCIES CAUSED INACCURACIES IN THE ROD GROUP DEMAND POSITION INDICATION RESULTING IN THREE MANUAL REACTOR TRIPS.
EVENT DATE: 031688 REPORT DATE: 040888 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 208969) THIS REPORT DESCRIBES THREE EVENTS IN WHICH A MANUAL REACTOR TRIP WAS INITIATED IN ACCORDANCE WITH THE ACTION STATEMENT OF TECH SPEC LIMITING CONDITION FOR OPERATION (LCO) 3.1.3.3. ON MARCH 16 AND 17, 1988, WITH UNIT 2 IN MODE 3 (HOT STANDBY), ROD CONTROL SYSTEM TESTING WAS BEING PERFORMED IN PREPARATION FOR ENTRY INTO MODE 2. AT TWO DIFFERENT TIMES DURING THE PERFORMANCE OF THIS TESTING, IT WAS DETERMINED THAT THE ROD GROUP DEMAND POSITION INDICATION WAS NOT WITHIN +/- 2 STEPS OF THE ACTUAL DEMAND POSITION AS REQUIRED BY TS LCO

3.1.3.3. AS A RESULT, PLANT OPERATORS COMPLIED WITH THE ACTION STATEMENT OF THE SUBJECT LCO AND OPENED THE REACTOR TRIP BREAKERS. ON MARCH 19, 1988, UNIT 2 WAS MAINTAINING MODE 3 CONDITIONS WHILE ADDITIONAL TESTING OF THE ROD CONTROL SYSTEM WAS PERFORMED. DURING THIS TESTING, ROD GROUP MOVEMENT FOR SHUTDOWN BANK B WAS OUT OF SEQUENCE BY THREE STEPS. ALTHOUGH THE DEMAND STEP COUNTERS ACCURATELY INDICATED ROD POSITION, THE OPERATOR CONSERVATIVELY ASSUMED THAT LCO 3.1.3.3 WAS APPLICABLE AND OPENED THE REACTOR TRIP BREAKERS. THE FIRST EVENT WAS CAUSED BY A FAILURE OF THE DEMAND STEP COUNTER CIRCUITRY WHILE THE SECOND EVENT WAS CAUSED BY INTERNAL BINDING OF THE STEP COUNTER ITSELF. IN BOTH OF THESE EVENTS, HOWEVER, THE ROD GROUPS MOVED AS DESIGNED.

[216] SEQUOYAH 2 DOCKET 50-328 LER 88-014
PRESSURE CHANGES IN BALANCE OF PLANT CONDENSATE SYSTEM CAUSED MAIN FEEDWATER PUMPS TO TRIP WHICH RESULTED IN AUTO START OF AUXILIARY FEEDWATER PUMPS.
EVENT DATE: 032088 REPORT DATE: 041588 NSSS: WE TYPE: PWR

(NSIC 208970) ON MARCH 20, 1988, AT 0702 EST WITH UNIT 2 IN MODE 3 (0 PERCENT POWER, 547 DEGREES F, 2235 PSIG), AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION SIGNAL WAS GENERATED WHILE PLACING THE "A" STRING INTERMEDIATE PRESSURE HEATERS IN SERVICE FOR FILLING AND DEAERATING PORTIONS OF THE CONDENSATE SYSTEM. A PRESSURE ANOMALY IN THE CONDENSATE SYSTEM OCCURRED WHILE OPENING 2A INTERMEDIATE HEATER ISOLATION VALVE FROM THE MAIN CONTROL ROOM. THE CONDENSATE PIPING DOWNSTREAM OF INTERMEDIATE HEATERS WAS AT LOW PRESSURE DUE TO VACUUM IN THE MAIN CONDENSER AND THE 2B MAIN FEEDWATER PUMP (MFWP) RECIRCULATION VALVE TO THE CONDENSER BEING PARTIALLY OPEN. THE PRESSURE CHANGE IN THE CONDENSATE SYSTEM CAUSED MFWP GLAND SEAL INJECTION WATER PRESSURE TO DROP BELOW THE MFWP TRIP SETPOINT. SINCE ONLY ONE GLAND SEAL INJECTION PUMP WAS IN SERVICE, THE PRESSURE DROP COULD NOT BE COMPENSATED FOR AND SUBSEQUENTLY TRIPPED BOTH MFWPS. THIS CONDITION GENERATED AN AUXILIARY FEEDWATER PUMP START SIGNAL AS DESIGNED. SINCE BOTH MOTOR-DRIVEN AUXILIARY FEEDWATER PUMPS WERE IN SERVICE AT THE TIME OF THE EVENT (AS PART OF NORMAL PLANT OPERATION DURING MODE 3), ONLY TURBINE-DRIVEN FEEDWATER PUMP STARTED (ESF ACTUATION). THE IMMEDIATE CAUSE OF THE EVENT IS CONCLUDED TO BE THE BALANCE OF PLANT SYSTEM PERTURBATION.

[217] SEQUOYAH 2 DOCKET 50-328 LER 88-015
PERFORMANCE OF AN INADEQUATE MAINTENANCE INSTRUCTION FOR THE INSPECTION OF THE REACTOR TRIP BREAKERS RESULTED IN A MAIN FEEDWATER ISOLATION.
EVENT DATE: 032488 REPORT DATE: 041988 NSSS: WE TYPE: PWR

(NSIC 209049) ON 3/24/88, UNIT 2 WAS IN MODE 3 (HOT STANDBY) WHILE ELECTRICAL MAINTENANCE (EM) PERSONNEL WERE PERFORMING MAINTENANCE ON THE TRAIN "A" REACTOR TRIP BYPASS BREAKER IN ACCORDANCE WITH AN APPROVED MAINTENANCE INSTRUCTION (MI). AT APPROX. 0130 EST, DURING THE PERFORMANCE OF THE TRAIN "A" BYPASS BREAKER CELL SWITCH INSPECTION, AN INADVERTENT TRAIN "A" MAIN FEEDWATER (MFW) ISOLATION OCCURRED. THE MFW ISOLATION WAS THE RESULT OF A REACTOR TRIP BREAKER "OPEN" SIGNAL CONCURRENT WITH A LOW REACTOR COOLANT SYSTEM (RCS) AVERAGE TEMPEATURE. A SUBSEQUENT EVALUATION OF THIS EVENT REVEALED THAT THE TRIP BREAKER "OPEN" SIGNAL WAS THE RESULT OF EM PERSONNEL OPERATING THE CELL SWITCH IN THE TRAIN "A" BYPASS BREAKER COMPARTMENT AS REQUIRED BY MI-10.9.2. PLANT OPERATORS IMMEDIATELY HALTED ALL MAINTENANCE ACTIVITIES ON THE REACTOR TRIP BYPASS BREAKERS AND RECOVERED FROM THE MFW ISOLATION. A REVIEW OF THE SUBJECT MI REVEALED THAT THE PROCEDURE DID NOT HAVE THE NECESSARY STEPS TO PRECLUDE AN MFW ISOLATION WHEN MORE THAN ONE REACTOR TRIP BREAKER WAS "RACKED OUT" OR COMPLETELY REMOVED FROM ITS COMPARTMENT. IN ADDITION, AN EVALUATION THAT HAD BEEN PERFORMED TO ASSESS THE IMPACT OF MI-10.9.2 ON PLANT OPERATION WAS INADEQUATE. INSTRUCTION CHANGE FORMS (ICFS) WERE ISSUED TO PERMANENTLY CHANGE THE MIS FOR THE REACTOR TRIP BREAKERS AS WELL AS THE BYPASS BREAKERS.

[218] SEQUOYAH 2 DOCKET 50-328 LER 88-018
CONTAINMENT INTEGRITY WAS NOT PROPERLY MAINTAINED RESULTING IN A NONCOMPLIANCE WITH TECH SPEC DUE TO AN INADEQUATE PERFORMANCE OF THE RECURRENCE CONTROL FOR A PREVIOUS LER.
EVENT DATE: 033088 REPORT DATE: 042688 NSSS: WE TYPE: PWR

(NSIC 209178) ON 3/30/88, AT 1440 EST WHILE UNIT 2 WAS IN HOT STANDBY, IT WAS DISCOVERED BY PLANT PERSONNEL THAT A THREADED CAP WAS INSTALLED ON A TUBE FITTING TEE IN THE SENSE LINE TO A LOCAL PRESSURE GAUGE (2-PI-63-74). THE PRESSURE GAUGE IS INSTALLED BETWEEN THE INBOARD AND OUTBOARD CONTAINMENT ISOLATION VALVES IN A TEST SYSTEM LINE USED TO MEASURE REACTOR COOLANT SYSTEM BOUNDARY VALVE SEAT LEAKAGE. DUE TO THE LOCATION OF THE CAP, IT SERVES AS A BOUNDARY REQUIRED TO SATISFY CONTAINMENT INTEGRITY REQUIREMENTS. A PREVIOUS NRC POSITION PROVIDED TO TVA STATED THAT THREADED CAPS ARE UNACCEPTABLE TEST CONNECTION BOUNDARIES IN ORDER TO MEET CONTAINMENT INTEGRITY REQUIREMENTS. WHEN OPERATIONS PERSONNEL WERE MADE AWARE OF THE PIPE CAP FINDING, THE ACTION STATEMENT OF TECH SPECS LIMITING CONDITION FOR OPERATION 3.6.1.1 WAS ENTERED FOR APPROPRIATE COMPENSATORY MEASURES. THIS EVENT WAS CAUSED BY AN INADEQUATE REVIEW OF THE PENETRATION WHICH HAD A THREADED CAP FOLLOWING RECEIPT OF THE NRC POSITION. THIS REVIEW HAS SINCE BEEN VERIFIED TO BE COMPLETE. NO ADDITIONAL PENETRATIONS WERE FOUND TO HAVE INADEQUATE CONTAINMENT INTEGRITY DEVICES. THE SUBJECT CAP WAS PART OF THE TYPC C TEST BOUNDARY FOR THIS CONTAINMENT PENETRATION; THEREFORE, ANY ACCOUNTED LEAKAGE WAS ASSOCIATED FOR AND MEASURED TO BE WITHIN THE ALLOWABLE LEAKAGE LIMIT.

[219] SEQUOYAH 2 DOCKET 50-328 LER 88-017
INADVERTENT REACTOR TRIP SIGNAL CAUSED BY MANIPULATION OF A SOURCE RANGE CHANNEL CONTROL POWER FUSE.
EVENT DATE: 040188 REPORT DATE: 042688 NSSS: WE TYPE: PWR
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)

(NSIC 209177) ON APRIL 1, 1988, AN INADVERTENT REACTOR TRIP SIGNAL WAS GENERATED ON UNIT 2 WHEN THE ASSISTANT SHIFT SUPERVISOR (ASST. SS) WAS ATTEMPTING TO SECURE A CONTROL POWER FUSE ON THE SOURCE RANGE, NEUTRON FLUX CHANNEL N-31. THE FUSE LOCKS INTO POSITION BY A METAL CONNECTOR WHICH HAS TWO METAL TABS THAT LOCK INTO THE FUSE HOLDER AS THE CONNECTOR IS INSERTED AND ROTATED. ATTACHED TO THE OUTER PORTION OF THE FUSE CONNECTOR IS A PLASTIC COVER WHICH IS USED TO FACILITATE REMOVAL/INSTALLATION OF THE FUSE. THE PLASTIC COVER HAS TWO FLAT SURFACES WHICH ARE NORMALLY IN THE HORIZONTAL POSITION WHEN THE FUSE CONNECTOR IS LOCKED INTO POSITION CORRECTLY. BEFORE THIS EVENT, THE PLASTIC COVER WAS OBSERVED TO BE ROTATED SLIGHTLY SUCH THAT THE FLAT SURFACES WERE NOT HORIZONTAL. THUS, THE FUSE APPEARED TO BE IN A CONDITION SUCH THAT THE POTENTIAL EXISTED FOR THE FUSE TO DISCONNECT AT ANY TIME AND GENERATE A REACTOR TRIP SIGNAL. WHEN THE ASST. SS ATTEMPTED TO LOCK THE FUSE IN PLACE, CONTROL POWER WAS INTERRUPTED MOMENTARILY, AND SUBSEQUENTLY, THE REACTOR TRIP SIGNAL GENERATED. NO REACTOR TRIP OCCURRED BECAUSE THE REACTOR TRIP BREAKERS WERE OPEN AND THE RODS INSERTED.

[220] SHEARON HARRIS 1 DOCKET 50-400 LER 87-051
AUXILIARY FEEDWATER SYSTEM ACTUATION ON MAIN FEEDWATER PUMP TRIPS DUE TO DESIGN AND EQUIPMENT FAILURE.
EVENT DATE: 083187 REPORT DATE: 093087 NSSS: WE TYPE: PWR
VENDOR: MASONEILAN INTERNATIONAL, INC.

(NSIC 209195) ON AUGUST 31, 1987, THE SHEARON HARRIS NUCLEAR POWER PLANT WAS PERFORMING A TECH SPEC REQUIRED SHUTDOWN DUE TO EXCESS UNIDENTIFIED LEAKAGE (SEE LER-87-050-00). AT 0408 WITH THE REACTOR AT 8% POWER, THE "A" MAIN FEEDWATER PUMP (MFP) TRIPPED DUE TO A FEEDWATER OSCILLATION. THE ROOT CAUSE IS FEEDWATER SYSTEM DESIGN PROBLEMS. BOTH MOTOR-DRIVEN AUXILIARY FEEDWATER (AFW) PUMPS STARTED AS REQUIRED ON LOSS OF FEEDWATER, SINCE THE "A" PUMP WAS THE ONLY MFP IN OPERATION AT THE TIME. MFP "B" WAS STARTED AT 0716 ON AUGUST 31 AND AFW WAS

SECURED. AT APPROXIMATELY 1530 ON AUGUST 31, WHILE THE UNIT WAS IN MODE 3, THE "B" MFP TRIPPED AND AFW ACTUATED. A RESTART ATTEMPT WAS MADE ON THE "B" MFP AT 1703, BUT THE PUMP TRIPPED IMMEDIATELY. AFW ACTUATED AGAIN. THE CAUSE OF THE TRIP WAS A SEPARATED VALVE STEM ON THE "B" MFP RECIRCULATION VALVE. THE VALVE STEM WAS REPAIRED. FEEDWATER SYSTEM CHANGES TO IMPROVE RELIABILITY ARE BEING EVALUATED. BOTH MFPS WERE RETURNED TO SERVICE ON SEPTEMBER 1, 1987.

[221] SHEARON HARRIS 1 DOCKET 50-400 LER 88-006 REV 01
UPDATE ON BOTH EMERGENCY SERVICE WATER SYSTEMS INOPERABLE DUE TO ISOLATION VALVE FAILURES AND DESIGN DEFICIENCY.
EVENT DATE: 020888 REPORT DATE: 041588 NSSS: WE TYPE: PWR
VENDOR: ROCKWELL MANUFACTURING COMPANY
TARGET ROCK CORP.

(NSIC 209022) ON FEBRUARY 8, 1988, DURING SURVEILLANCE TESTING OF THE EMERGENCY SERVICE WATER SYSTEM (ESWS), THE NONSAFETY PORTION OF THE ESW PUMP SEAL WATER SUPPLY SYSTEM FAILED TO ISOLATE AS REQUIRED WHEN TWO SOLENOID VALVES STUCK OPEN AND A CHECK VALVE FAILED TO SEAT. BOTH ESW TRAINS WERE DECLARED INOPERABLE DUE TO THE PIPING CONFIGURATION OF THE SEAL WATER SUPPLY AND THE LOCATIONS OF THE FAILED VALVES. MANUAL VALVES WERE CLOSED TO ISOLATE THE NONSAFETY PIPING AND PERMIT CONTINUED OPERATION WHILE REPAIRS TOOK PLACE. DEBRIS WAS DISCOVERED DURING DISASSEMBLY OF THE FAILED CHECK VALVE, AND IS ALSO SUSPECTED OF CAUSING THE SOLENOID VALVE FAILURES. REPAIRS WERE COMPLETED THE NEXT DAY, THE SURVEILLANCE TESTING WAS COMPLETED AND THE ESW SYSTEM WAS RETURNED TO SERVICE. ON FEBRUARY 12, 1988, A CONCERN REGARDING THE SEAL WATER PIPING CONFIGURATION VULNERABILITY TO SINGLE PASSIVE FAILURES WAS RAISED, AND A MANUAL VALVE WAS IMMEDIATELY CLOSED TO SEPARATE THE TRAINS WHILE AN EVALUATION WAS CONDUCTED. THIS EVALUATION CONCLUDED ON FEBRUARY 25, 1988, THAT THE PIPING CONFIGURATION WAS VULNERABLE TO SINGLE PASSIVE FAILURES WHICH COULD DISABLE BOTH ESW TRAINS, IN CONFLICT WITH THE SYSTEM DESIGN REQUIREMENTS SPECIFIED IN THE FINAL SAFETY ANALYSIS REPORT.

[222] SHEARON HARRIS 1 DOCKET 50-400 LER 88-007
PLANT TRIP DUE TO A LOSS OF FEEDWATER TO 'B' STEAM GENERATOR CAUSED BY A FAILED FUSE.
EVENT DATE: 030988 REPORT DATE: 040888 NSSS: WE TYPE: PWR
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)

(NSIC 208875) THE PLANT WAS OPERATING IN MODE 1, POWER OPERATION, AT 100 PERCENT REACTOR POWER ON MARCH 9, 1988. AT 1654 HOURS, SEVERAL ALARMS WERE RECEIVED BY OPERATORS AT THE MAIN CONTROL BOARD INDICATING LOW WATER LEVEL IN "B" STEAM GENERATOR (SG). OPERATORS OBSERVED THAT THE "B" MAIN FEEDWATER REGULATING VALVE (MFRV) HAD GONE SHUT. THE OPERATORS THEN ATTEMPTED TO TAKE MANUAL CONTROL OF "B" MFRV IN AN EFFORT TO RESTORE WATER LEVEL IN THE "B" SG. HOWEVER, THE OPERATORS EFFORTS COULD NOT RESTORE THE RAPIDLY DROPPING "B" SG WATER LEVEL AS INSTRUMENT AIR WAS LOST TO "B" MFRV AND WITHIN 30 SECONDS THE PLANT TRIPPED DUE TO "B" SG FEEDWATER/ STEAM FLOW MISMATCH WITH LOW SG WATER LEVEL. THERE WERE NO SAFETY CONSEQUENCES DUE TO THIS EVENT AND PLANT RESPONSE TO THE TRIP WAS NORMAL. THE MAIN STEAM ISOLATION VALVES (MSIV) WERE SHUT AT 1657 HOURS IN ORDER TO LIMIT COOLDOWN, SG WATER LEVELS WERE RESTORED WITH THE AUXILIARY FEEDWATER SYSTEM, AND THE PLANT WAS STABILIZED IN MODE 3, HOT STANDBY. THE CAUSE OF THE EVENT WAS A FAILED RENEWABLE FUSE WHICH FED THE SOLENOID FOR "B" MFRV. THE END CAP ON THE FUSE HAD LOOSENED WHICH RESULTED IN A LOSS OF POWER TO A SOLENOID WHICH INTERRUPTED INSTRUMENT AIR TO THE VALVE. THE VALVE FAILED SHUT CUTTING OFF THE FEEDWATER SUPPLY TO "B" SG.

[223] SHOREHAM DOCKET 50-322 LER 87-018
CRASH OF SECURITY COMPUTER RESULTED IN A MODERATE LOSS OF SECURITY EFFECTIVENESS WHICH WAS NOT PROPERLY COMPENSATED FOR.
EVENT DATE: 062987 REPORT DATE: 070287 NSSS: GE TYPE: BWR

(NSIC 209206) AT APPROXIMATELY 1635 HOURS, 6/29/87, A CRASH OF THE SECURITY COMPUTER OCCURRED WHICH AFFECTED ALL INTERIOR BUILDING ALARMS. ALL CONTINGENCY POSITIONS WERE MANNED WITHIN 10 MINUTES WITH THE EXCEPTION OF THE POSITION ON THE STAIRWELL CONTROLLING ACCESS TO THE NORTH DOOR TO THE RELAY ROOM, AND THE WEST DOOR TO THE CONTROL ROOM AT ELEVATION 63'. AT 1710 HOURS, A SECURITY SUPERVISOR, FINDING NO GUARD ON THE STAIRWELL, ASSUMED THE POST HIMSELF. HE DIRECTED A VITAL AREA CHECK AT THIS TIME AND NO UNUSUAL SECURITY CONDITIONS WERE DETECTED. ALL GROUND LEVEL DOORS HAD BEEN CHECKED AT 1638 HOURS. THE SECURITY POST WAS UNMANNED FOR APPROXIMATELY 35 MINUTES. A FIRE WATCH POSTED IN THE RELAY ROOM REPORTED THAT THERE HAD BEEN NO UNUSUAL ACTIVITY. AT 1756, THE SECURITY COMPUTER WAS BACK IN FULL OPERATION. THE GUARD WHO HAD BEEN DISPATCHED VIA RADIO BY THE COMMUNICATIONS SGT. DID NOT RECEIVE THE MESSAGE AND THE SGT. DID NOT FOLLOW-UP ON HIS NON-RECEIPT OF AN ACKNOWLEDGEMENT. THE COMMUNICATIONS SGT. WILL BE RETRAINED IN PROPER RADIOCOMMUNICATIONS PROCEDURES, AS WELL AS ENSURING THAT CONTINGENCY POSTS ARE PROPERLY MANNED.

[224] SHOREHAM DOCKET 50-322 LER 87-019
CONTINUOUS RELAY ROOM FIRE WATCH REQUIRED BY TECH SPECS WAS NOT MET DUE TO BOMB THREATS.
EVENT DATE: 070487 REPORT DATE: 070987 NSSS: GE TYPE: BWR

(NSIC 209207) ON JULY 4, 1987 AT 0640, AND 0655, TWO BOMB THREATS WERE PHONED IN TO THE LILCO SECURITY ORGANIZATION INDICATING THAT A BOMB WAS TO GO OFF AT 12 NOON AT THE SHOREHAM NUCLEAR POWER PLANT. THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) WITH THE MODE SWITCH IN SHUTDOWN AND ALL RODS INSERTED IN THE CORE. A SECURITY ALERT WAS DECLARED AND A SEARCH OF THE SITE WAS CONDUCTED. THE WATCH ENGINEER WAS NOTIFIED AND HE DECLARED AN UNUSLAL EVENT (UE) AT 0657. PLANT MANAGEMENT WAS NOTIFIED OF THE EVENT AND THE NRC WAS NOTIFIED AT 0728. EMERGENCY PLAN UE PROCEDURES WERE IMPLEMENTED. THE SITE WAS NOT EVACUATED. AT 1145, FIRE WATCHES IN THE TURBINE BLDG., REACTOR BLDG. AND CONTROL BLDG. WERE SUSPENDED FOR PERSONNEL SAFETY REASONS. THE SEARCH FAILED TO UNCOVER ANY EXPLOSIVE DEVICES AND THE THREATS WERE ASSESSED TO BE UNFOUNDED. AT 1220, THE FIRE WATCHES WERE RESTORED. ALL FIRE WATCH PATROLS REQUIRED BY TECH SPECS WERE MET WITH THE EXCEPTION OF THE RELAY ROOM CONTINUOUS FIRE WATCH REQUIRED BY 3.7.7.3. THE UE AND THE SECURITY ALERT WERE TERMINATED AT 1317.

[225] SHOREHAM DOCKET 50-322 LER 87-020
BOMB THREAT.
EVENT DATE: 080587 REPORT DATE: 080787 NSSS: GE TYPE: BWR

(NSIC 209208) ON AUGUST 5, 1987 AT 0200 A SECURITY OFFICER POSTED AT THE WEST GATE AT THE SHOREHAM NUCLEAR POWER STATION (SNPS) RECEIVED A TELEPHONE BOMB THREAT. THE MALE CALLER STATED THAT THERE WERE TWO BOMBS STRATEGICALLY PLACED ON THE SITE AND WERE SET TO GO OFF AT 4 A.M. UPON COMPLETION OF THE CALL, THE OFFICER NOTIFIED THE SHIFT SECURITY AGENT WHO THEN DECLARED A SECURITY ALERT AT 0201. AT 0203, THE AGENT NOTIFIED THE WATCH ENGINEER. AT 0215, AN UNUSUAL EVENT WAS DECLARED. PLANT MANAGEMENT WAS NOTIFIED OF THE EVENT AND THE NRC WAS NOTIFIED AT 0237. THE SUFFOLK COUNTY POLICE DEPARTMENT (SCPD) WAS NOTIFIED AND RESPONDED TO THE SITE. THE SITE WAS NOT EVACUATED. A BOMB SEARCH WAS CONDUCTED AND NO BOMB WAS FOUND. IT WAS DETERMINED THAT THERE WAS NO CREDENCE TO THE CALL AND THAT IT WAS STRICTLY A HARRASSMENT CALL. THE UNUSUAL EVENT WAS TERMINATED AT 0500.

[226] SHOREHAM DOCKET 50-322 LER 87-035 REV 01
UPDATE ON ELECTRICAL NOISE BETWEEN GROUNDS FOR TEMPERATURE MONITORING UNITS RESULTS IN HIGH ENERGY LINE BREAK LOGIC INITIATION.
EVENT DATE: 122187 REPORT DATE: 042988 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 209171) ON THREE SEPARATE OCCASIONS, HIGH ENERGY LINE BREAK LOGIC ISOLATIONS OF THE REACTOR WATER CLEANUP (RWCU) AND MAIN STEAM LINE (MSL) DRAINS VALVES (1G33*MOV-034 AND 1B21*MOV-032) OCCURRED MOST LIKELY DUE TO HIGH AMPLITUDE ELECTRICAL NOISE ON THE ISOLATED GROUND SYSTEM (IGS). AT THE TIME OF THE EVENTS, THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) WITH THE MODE SWITCH IN SHUTDOWN AND ALL RODS INSERTED IN THE CORE. THE FIRST TWO EVENTS OCCURRED ON DEC. 21, 1987 AND JAN. 6, 1988. INVESTIGATION INTO THE CAUSE OF THE EVENTS LED TO THE DISCOVERY OF A GROUNDING PROBLEM WITHIN THE PRIMARY CONTAINMENT MONITORING PANEL (PCM) WHERE THE TEMPERATURE MONITORING UNITS (TMU) ARE LOCATED. THIS PARTICULAR PROBLEM WAS ALLEVIATED AS A RESULT OF AN ENGINEERING CHANGE WHICH GROUNDED THE POWER SUPPLIES FOR THE TMUS TO THE IGS. HOWEVER, A THIRD ISOLATION OCCURRED ON MARCH 30, 1988 DUE TO A SIMILAR PROBLEM WITH THE IGS. IN ALL THREE EVENTS, OPERATORS VERIFIED THE SIGNALS AS FALSE AND RETURNED THE SYSTEMS TO THEIR NORMAL CONFIGURATIONS PRIOR TO THE EVENTS. THE ELECTRICAL NOISE LEVEL ON THE ISOLATED GROUND WAS AND IS SUCH THAT PERIODIC VOLTAGE SPIKES OF LARGE MAGNITUDES TRIGGER THE MISOPERATION OF THE TMUS, THEREBY CAUSING THE ISOLATIONS. PLANT MANAGEMENT WAS NOTIFIED OF THE EVENTS AND THE NRC WAS NOTIFIED IN ACCORDANCE WITH 10CFR50.72.

[227] SHOREHAM DOCKET 50-322 LER 88-003
UNPLANNED AUTOMATIC INITIATION OF RBSVS "A" TRAIN DURING AN INSTRUMENT AND CONTROLS SURVEILLANCE PROCEDURE WHEN A TECHNICIAN INADVERTENTLY DEENERGIZED A RELAY.
EVENT DATE: 032288 REPORT DATE: 042188 NSSS: GE TYPE: BWR

(NSIC 209046) ON 03/22/88 AT 1436, AN UNPLANNED AUTOMATIC INITIATION OF THE REACTOR BUILDING STANDBY VENTILATION SYSTEM (RBSVS) "A" SIDE OCCURRED WHEN AN INSTRUMENT AND CONTROLS (I&C) TECHNICIAN INADVERTENTLY DEENERGIZED A RELAY DURING A SURVEILLANCE TEST. THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) WITH THE MODE SWITCH IN SHUTDOWN AND ALL RODS INSERTED IN THE CORE. THE TECHNICIAN, PERFORMING A CHANNEL FUNCTIONAL TEST ON 1T46*PDT-043A (REACTOR BUILDING DIFFERENTIAL PRESSURE TRANSMITTER) BY PROCEDURE, HAD BEGUN TO RETERMINATE THE LEADS LIFTED ON RELAY 3B-1T46019 PRIOR TO THE START OF THE TEST. HOWEVER, WHILE HE WAS PERFORMING THIS STEP ONE OF THE LEADS SLIPPED OUT OF HIS HAND CAUSING THE DEENERGIZATION OF THE RELAY, RESULTING IN A RBSVS "A" TRAIN INITIATION. AFTER THE LEADS WERE RETERMINATED, THE PROCEDURE WAS HALTED. OPERATORS VERIFIED THE SIGNAL AND TOOK THE NECESSARY STEPS TO RETURN THE RBSVS TO ITS NORMAL CONFIGURATION. PLANT MANAGEMENT WAS NOTIFIED OF THE EVENT AND THE NRC WAS NOTIFIED AT 1459 PER 10CFR50.72. BECAUSE THE TERMINALS ARE LOCATED IN THE BOTTOM OF PANEL 1H21*PNL-VX1, ACCESSIBILITY TO PERFORM THIS TEST WAS EXTREMELY DIFFICULT. THE SURVEILLANCE PROCEDURE HAD BEEN PREVIOUSLY REVISED TO ADD CAUTION STATEMENTS PRIOR TO THE STEPS WHEN RETERMINATION OF THE LIFTED LEADS WAS REQUIRED DUE TO PREVIOUS RELATED EVENTS.

[228] SOUTH TEXAS 1 DOCKET 50-498 LER 87-026 REV 01
UPDATE ON DEGRADED UNDERVOLTAGE COINCIDENT WITH A SAFETY INJECTION CIRCUITRY SURVEILLANCE DEFICIENCY DUE TO A DEFICIENT PROCEDURE.
EVENT DATE: 121287 REPORT DATE: 041388 NSSS: WE TYPE: PWR

(NSIC 208913) ON DECEMBER 12, 1987, AT APPROXIMATELY 1857 HOURS WITH UNIT 1 IN MODE 4, PRIOR TO INITIAL CRITICALITY, DURING REVIEW OF WORK INSTRUCTIONS FOR THE REPLACEMENT OF A TIME DELAY RELAY IN THE DEGRADED UNDERVOLTAGE CIRCUIT, IT WAS DETERMINED THAT THE TRIP ACTUATION DEVICE OPERATIONAL TEST (TADOT) ON DEGRADED

UNDERVOLTAGE COINCIDENT WITH SAFETY INJECTION HAD NOT BEEN TESTED AS REQUIRED. ALL THREE ENGINEERED SAFETY FEATURES (ESF) BUSSES WERE DECLARED INOPERABLE. THE PLANT ENTERED TECHNICAL SPECIFICATION 3.0.3 AND A PLANT COOLDOWN TO MODE 5 WAS INITIATED. THE CAUSE OF THE EVENT WAS DETERMINED TO BE A DEFICIENT SURVEILLANCE PROCEDURE RESULTING FROM A PERSONNEL ERROR IN INTERPRETING THE REQUIREMENTS OF THE MONTHLY TADOT. TO PREVENT RECURRENCE, A NEW PROCEDURE WAS WRITTEN AND SATISFACTORILY PERFORMED ON EACH ESF BUS. TESTING WAS COMPLETED AT APPROXIMATELY 1300 ON DECEMBER 13, 1987 PRIOR TO COMPLETING THE COOLDOWN TO MODE 5. ADDITIONALLY, COMPREHENSIVE REVIEWS OF INSTRUMENTATION & CONTROLS AND ELECTRICAL SURVEILLANCE PROCEDURES WERE CONDUCTED TO ENSURE OTHER TESTING REQUIREMENTS WERE COVERED IN SURVILLANCE PROCEDURES. THERE WERE NO ADVERSE SAFETY OR RADIOLOGICAL CONSEQUENCES AS A RESULT OF THIS EVENT.

[229] SOUTH TEXAS 1 DOCKET 50-498 LER 88-014 REV 01
UPDATE ON REACTOR PROTECTION SYSTEM ACTUATION DUE TO A SOFTWARE PROBLEM IN QDPS.
EVENT DATE: 020488 REPORT DATE: 033188 NSSS: WE TYPE: PWR

(NSIC 208878) ON FEBRUARY 4, 1988 AT APPROXIMATELY 1619 HOURS WITH THE UNIT IN MODE 3, COOLING DOWN TO MODE 4, AND PRIOR TO INITIAL CRITICALITY, AN OVER-TEMPERATURE/DELTA-TEMPERATURE (OT/DELTA T) REACTOR TRIP OCCURRED. THE REACTOR TRIP OCCURRED AT A REACTOR COOLANT SYSTEM TEMPERATURE OF 530F. INITIAL OBSERVATION WAS THAT CHANNEL II T-HOT OUTPUT ROSE TO 568F GIVING A OT/DELTA T TRIP COINCIDENT WITH AN OT/DELTA T TRIP ALREADY INSERTED IN CHANNEL IV DUE TO MAINTENANCE WORK. SUBSEQUENT INVESTIGATION RESULTED IN THE IDENTIFICATION OF A SOFTWARE DESIGN ERROR IN THE QUALIFIED DISPLAY PROCESSING SYSTEM (QDPS) TEMPERATURE AVERAGING SYSTEM WHICH ERRONEOUSLY CALCULATED AN AVERAGE HOT LEG TEMPERATURE OF 568F FOR CHANNEL II WHEN TEMPERATURE WAS ACTUALLY 530F. QDPS HAS BEEN MODIFIED TO PREVENT RECURRENCE OF THIS EVENT. NO SAFETY CONSEQUENCES RESULTED FROM THIS EVENT.

[230] SOUTH TEXAS 1 DOCKET 50-498 LER 88-022 REV 01
UPDATE ON REACTOR TRIP FROM SAFETY INJECTION DURING APPROACH TO CRITICALITY.
EVENT DATE: 022888 REPORT DATE: 042588 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 209244) AT APPROXIMATELY 0545 HOURS ON FEBRUARY 28, 1988, WITH UNIT 1 IN MODE 2, REACTOR CONTROL RODS WITHDRAWN, AND REACTOR COOLANT DILUTION IN PROGRESS DURING THE APPROACH TO INITIAL CRITICALITY, A REACTOR TRIP OCCURRED DUE TO AN UNANTICIPATED SAFETY INJECTION SIGNAL. THE SAFETY INJECTION SIGNAL WAS CAUSED BY A SPURIOUS INPUT TO A PROTECTION CHANNEL WHILE SURVEILLANCE TESTING WAS IN PROGRESS IN A DIFFERENT CHANNEL THUS SATISFYING THE COINCIDENT LOGIC FOR SAFETY INJECTION ACTUATION. ENGINEERED SAFETY FEATURES (ESF) EQUIPMENT ACTUATED AND ALL PLANT EQUIPMENT OPERATED AS EXPECTED. THE NRC WAS NOTIFIED AT APPROXIMATELY 0603 HOURS ON FEBRUARY 28, 1988. IN ADDITION, A NOUE WAS DECLARED AT THE SHIFT SUPERVISOR'S DISCRETION. THE CIRCUIT CARDS IN THE CIRCUIT WHERE THE SPURIOUS INPUT OCCURRED HAVE BEEN REPLACED. INVESTIGATION OF THE CAUSE OF THE SPURIOUS SIGNALS IS CONTINUING.

[231] SOUTH TEXAS 1 DOCKET 50-498 LER 88-023
NONPERFORMANCE OF A SCHEDULED SURVEILLANCE TEST FOR ESSENTIAL COOLING WATER SCREEN WASH BOOSTER PUMP.
EVENT DATE: 031188 REPORT DATE: 040788 NSSS: WE TYPE: PWR

(NSIC 208863) ON MARCH 11, 1988, WHILE THE PLANT WAS IN MODE 2, DURING PHYSICS TESTING, THE SYSTEM ENGINEER FOUND THAT THE FREQUENCY OF A REQUIRED SURVEILLANCE TEST ON ESSENTIAL COOLING WATER SCREEN WASH BOOSTER PUMP (ECWSWBP) 1C HAD NOT BEEN DOUBLED AS REQUIRED. AS A RESULT THE REQUIRED SURVEILLANCE WAS NOT PERFORMED WITHIN THE APPROPRIATE INTERVAL. ECW TRAIN 1C WAS DECLARED INOPERABLE.

THE ECWSWBP 1C INSERVICE SURVEILLANCE TEST WAS PERFORMED ON MARCH 11, 1988, AND FAILED DUE TO FLOW BEING TOO HIGH. ON MARCH 12, 1988, AN ECWSWBP 1C REFERENCE VALUES MEASUREMENT TEST WAS PERFORMED. THE PUMP PERFORMANCE DATA FROM THE TEST AGREED WITH THE MANUFACTURER'S PUMP CURVE, AND ECW TRAIN 1C WAS RESTORED TO OPERATIONAL STATUS. THE CAUSE OF THIS MISSED SURVEILLANCE HAS BEEN IDENTIFIED TO BE AN INADEQUATE PROCEDURE. THE SURVEILLANCE SCHEDULING PROCEDURE HAS BEEN REVISED TO CLEARLY DEFINE THE RESPONSIBILITIES AND METHOD FOR CHANGING SURVEILLANCE FREQUENCIES. TRAINING OF SYSTEM ENGINEERS ON THESE CHANGES HAS BEEN CONDUCTED.

[232] SOUTH TEXAS 1 DOCKET 50-498 LER 88-024
DESIGN ERROR WHICH BLOCKS SAFETY INJECTION ACTUATION.
EVENT DATE: 031688 REPORT DATE: 041388 NSSS: WE TYPE: PWR

(NSIC 208954) ON MARCH 16, 1988 AT 1700 HOURS WHILE THE UNIT WAS IN MODE 3, OPERATIONS PERSONNEL WERE NOTIFIED BY ENGINEERING THAT OPERATING THE SAFEGUARDS TEST CABINET MASTER RESET SWITCH, WITH THE REACTOR TRIP BREAKERS OPEN, WOULD RESET AND BLOCK SAFETY INJECTION ACTUATION ON THE ASSOCIATED TRAIN (A, B OR C). THE CAUSE OF THIS CONDITION WAS A DESIGN ERROR BY THE NSSS SUPPLIER, WESTINGHOUSE. NIGHT ORDERS WERE ISSUED TO OPERATIONS PERSONNEL TO ASSURE THAT ADEQUATE PRECAUTIONS ARE TAKEN PRIOR TO TESTING. THE NRC WAS NOTIFIED PURSUANT TO 10CFR50.72. CORRECTIVE ACTIONS INCLUDE MODIFYING THE CIRCUIT TO ELIMINATE THE PROBLEM.

[233] SOUTH TEXAS 1 DOCKET 50-498 LER 88-025
CONTROL ROOM VENTILATION RECIRCULATION ACTUATION DUE TO A RADIATION MONITOR ACTUATION.
EVENT DATE: 032388 REPORT DATE: 041988 NSSS: WE TYPE: PWR

(NSIC 209028) ON MARCH 23, 1988 AT APPROXIMATELY 0353 HOURS, WITH UNIT 1 IN MODE 2, AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION OF THE CONTROL ROOM VENTILATION SYSTEM TO THE RECIRCULATION MODE OCCURRED. SUBSEQUENT INVESTIGATION DETERMINED THAT THE MOST PROBABLE CAUSE OF THIS EVENT WAS AN INADVERTENT ACTUATION OF A CONTROL ROOM VENTILATION RADIATION MONITOR DURING MAINTENANCE ACTIVITIES. DIAGNOSTIC TESTS WERE RUN ON THE MONITOR AND ATTEMPTS WERE MADE TO DUPLICATE THE EVENT, HOWEVER, NO SPECIFIC CAUSE WAS FOUND.

[234] SOUTH TEXAS 1 DOCKET 50-498 LER 88-026
REACTOR TRIP AND SAFETY INJECTION DUE TO LOSS OF OFFSITE POWER CAUSED BY PERSONNEL ERROR.
EVENT DATE: 033088 REPORT DATE: 042988 NSSS: WE TYPE: PWR

(NSIC 209245) AT APPROXIMATELY 2112 HOURS ON MARCH 30, 1988 WITH UNIT 1 IN MODE 1 AT 7% POWER, A PARTIAL LOSS OF OFFSITE POWER RESULTED IN A REACTOR TRIP. A LOW-LOW COMPENSATED T-COLD SIGNAL INITIATED A SAFETY INJECTION (ESF ACTUATION) SEQUENCE. A NOTIFICATION OF UNUSUAL EVENT WAS DECLARED. ALL SAFETY SYSTEMS FUNCTIONED NORMALLY AND OFFSITE POWER WAS RESTORED AT APPROXIMATELY 2202 HOURS ON MARCH 30, 1988. LOSS OF OFFSITE POWER OCCURRED DUE TO PERSONNEL ERROR BY PLANT ELECTRICIANS WHO INADVERTENTLY TRIPPED THE SWITCHYARD SUPPLY BREAKERS WHILE THEY WERE INVESTIGATING A PREVIOUS MAIN GENERATOR TRIP. A DESIGN INADEQUACY IN THE EXCESSIVE COOLDOWN PROTECTION SYSTEM RESULTED IN A SUBSEQUENT SAFETY INJECTION ACTUATION. CORRECTIVE ACTIONS INCLUDE INCORPORATION OF A CASE STUDY OF THIS EVENT INTO TRAINING SESSIONS FOR MAINTENANCE PERSONNEL AND AN EVALUATION OF THE EXCESSIVE COOLDOWN PROTECTION SYSTEM TO PREVENT FUTURE UNNECESSARY SAFETY INJECTION ACTUATIONS.

[235] ST. LUCIE 1 DOCKET 50-335 LER 88-003
REACTOR TRIP ON LOW STEAM GENERATOR LEVEL DUE TO MAIN FEED REGULATING VALVE EQUIPMENT FAILURE.
EVENT DATE: 032888 REPORT DATE: 042788 NSSS: CE TYPE: PWR
VENDOR: FISCHER & PORTER CO.

(NSIC 209133) ON 3/27/88, ST. LUCIE UNIT 1 WAS IN MODE 1, 100% POWER AND AT STEADY STATE CONDITIONS. ALL PLANT CONTROLS WERE IN THEIR NORMAL OPERATING LINE-UP. AT APPROXIMATELY 00:01 EDT, ON MARCH 28, 1988, A MALFUNCTION OF THE MAIN FEEDWATER CONTROL SYSTEM (EIIS:JB) WAS OBSERVED AND THE MAIN FEEDWATER REGULATING VALVE CLOSED. THE CLOSURE OF THE VALVE GENERATED A LOW STEAM GENERATOR LEVEL SIGNAL FROM THE REACTOR PROTECTION SYSTEM AND CAUSED THE REACTOR TO AUTOMATICALLY TRIP. ALL SYSTEMS FUNCTIONED AS DESIGNED AND THE PLANT WAS MAINTAINED AT A STABLE CONDITION. THERE WERE NO RADIOLOGICAL RELEASES OBSERVED DURING THE EVENT. THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT ENDANGERED THROUGHOUT THE EVENT. THE REACTOR TRIP WAS OBSERVED AS AN UNCOMPLICATED REACTOR TRIP. THE ROOT CAUSE OF THE EVENT WAS DUE TO THE MALFUNCTION OF THE FEEDWATER REGULATING VALVE POSITIONER CAUSING THE VALVE TO OVERSHOOT ITS PRE-SET POSITION ON THE CLOSING STROKE. SOME OF THE CONTRIBUTING FACTORS TO THE VALVE CLOSURE WERE (1) THE STEAM FLOW/FEED FLOW MISMATCH CONTROLLER WAS SLUGGISH AND HAD LOW OUTPUT SIGNALS, AND (2) THE ELECTRICAL TO PNEUMATIC CONVERTER SIGNALS WERE NOT CONSISTENT. IMMEDIATE CORRECTIVE ACTIONS WERE (1) REPLACE THE STEAM FLOW/FEED FLOW MISMATCH CONTROLLER, (2) THE ELECTRICAL TO PNEUMATIC CONVERTER WAS CLEANED AND ADJUSTED, AND (3) THE VALVE POSITIONER WAS CLEANED AND ADJUSTED.

[236] ST. LUCIE 2 DOCKET 50-389 LER 88-003
SAFEGUARDS SIGNALS TO ONE CONTAINMENT ISOLATION VALVE OF A REDUNDANT PAIR BYPASSED DUE TO PERSONNEL ERROR.
EVENT DATE: 031388 REPORT DATE: 050688 NSSS: CE TYPE: PWR
VENDOR: ITT-BARTON

(NSIC 209232) ON 04/06/88, IT WAS DISCOVERED THAT WITH THE NORMAL/ISOLATE SWITCH FOR THE LETDOWN ISOLATION VALVE IN THE "ISOLATE" POSITION, THE CONTAINMENT ISOLATION ACTUATION SIGNAL AND SAFETY INJECTION ACTUATION SIGNAL WERE BYPASSED TO ONE CONTAINMENT ISOLATION VALVE. THE NORMAL/ISOLATE SWITCH FOR THE LETDOWN ISOLATION VALVE HAD BEEN PLACED IN THE "ISOLATE" POSITION ON MARCH 13, 1988, BECAUSE FALSE ELECTRICAL SIGNALS WERE GENERATED FROM AN ISOLATED DIFFERENTIAL PRESSURE SWITCH ASSOCIATED WITH THE VALVE RESULTING IN LETDOWN ISOLATING. THE FAILED DIFFERENTIAL PRESSURE SWITCH, LOCATED INSIDE THE CONTAINMENT BUILDING, WAS ISOLATED TO TERMINATE A MINOR PRIMARY COOLANT LEAK. THE ROOT CAUSE OF THE EVENT WAS A COGNITIVE PERSONNEL ERROR BY TWO UTILITY-LICENSED OPERATORS IN THE MISINTERPRETATION OF THE CONTROL WIRING DIAGRAM (CWD) FOR THE CONTAINMENT ISOLATION VALVE; THEREBY UNKNOWINGLY OPERATING IN A CONDITION PROHIBITED BY THE PLANT'S TECH. SPECS. THE OPERATORS WERE COUNSELED ON THE IMPORTANCE OF CONDUCTING A MORE THOROUGH REVIEW, ALL OPERATORS WERE GIVEN NEW GUIDELINES TO FOLLOW PRIOR TO REMOVING EQUIPMENT FROM SERVICE TO ENSURE ALL PLANT SAFETY CONCERNS ARE PROPERLY ADDRESSED, MAINTENANCE WILL REPLACE THE FAILED DIFFERENTIAL PRESSURE SWITCH AS SOON AS A REPLACEMENT BECOMES AVAILABLE, SWITCH LABELING WILL BE UPGRADED, AND ADDITIONAL ADMINISTRATIVE CONTROLS ARE BEING DEVELOPED, AND

[237] SUMMER 1 DOCKET 50-395 LER 88-004
TWO 2 INCH CORE DRILLS FOUND UNSEALED DUE TO UNKNOWN CAUSES.
EVENT DATE: 052681 REPORT DATE: 042288 NSSS: WE TYPE: PWR

(NSIC 209021) TWO 2-INCH CORE DRILLS CONTAINING ONE AND A HALF INCH CONDUIT WERE FOUND UNSEALED. A REVIEW OF CONSTRUCTION DOCUMENTATION IDENTIFIED THAT BOTH CORE DRILLS WERE INSTALLED UNDER A FIELD CHANGE REQUEST "B" THAT WAS INITIATED IN MAY 1981. THE ELECTRICAL CIRCUITS CONTAINED IN THE CONDUITS ARE IDENTIFIED ON THE E-SERIES (ELECTRICAL) DRAWINGS; HOWEVER, THE CORE DRILLS WERE NOT IDENTIFIED ON

THE COMPOSITE (FLOOR/WALL LAYOUT) DRAWINGS. IMMEDIATELY UPON DISCOVERY, THE SHIFT SUPERVISOR WAS NOTIFIED, A CONTINUOUS FIRE WATCH ESTABLISHED, AND A PRIORITY 1 MAINTENANCE WORK REQUEST INITIATED FOR REPAIR. THE CONSEQUENCES DUE TO THIS EVENT WERE MINIMAL. THE CORE DRILL PENETRATED FIFTY-ONE INCHES OF CONCRETE AND EACH TWO INCH CORE DRILL HAS A ONE AND A HALF INCH CONDUIT PASSING THROUGH IT WHICH REDUCES THE PENETRATION OPENING TO ONE QUARTER INCH AROUND THE INSIDE CIRCUMFERENCE. THE FIRE LOADING IN EACH AREA, AS PROVIDED IN THE FIRE PROTECTION EVALUATION REPORT, IS SUCH A SMALL VALUE THAT THE REQUIRED FIRE BARRIER BETWEEN THE AREAS WOULD BE LESS THAN THIRTY MINUTES. IN ADDITION, NO SAFE SHUTDOWN EQUIPMENT IS LOCATED IN THE AFFECTED AREAS. TWO ADDITIONAL WALLS HAVE BEEN INSPECTED WITH THE RESIDENT NRC INSPECTOR PRESENT DURING THE INSPECTION OF ONE AND NO DISCREPANCIES IDENTIFIED. THE LICENSEE CONSIDERS THIS TO BE AN ISOLATED EVENT AND PLANS NO ADDITIONAL ACTION.

[238] SURRY 1 DOCKET 50-280 LER 88-008
LOW HEAD SAFETY INJECTION PUMP INOPERABLE DUE TO FAILED MOTOR LEADS.
EVENT DATE: 030988 REPORT DATE: 040888 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 208844) ON MARCH 9, 1988 AT 2256 HOURS, WITH UNIT 1 AT 98% POWER, ONE OF THE TWO LOW HEAD SAFETY INJECTION PUMPS, 1-SI-P-1B (EIIS-BP P) WAS DECLARED INOPERABLE DUE TO A PHASE TO PHASE FAULT BETWEEN TWO OF ITS MOTOR LEADS. TECHNICAL SPECIFICATION 3.3.B.3 REQUIRES THAT THIS PUMP BE RESTORED TO OPERABLE STATUS WITHIN 24 HOURS OR THE UNIT MUST BE BROUGHT TO HOT SHUTDOWN. PUMP 1-SI-P-1B COULD NOT BE RETURNED TO OPERABLE STATUS WITHIN THE 24 HOUR TIME LIMIT. THEREFORE, A RAMPDOWN OF THE UNIT WAS COMMENCED AT 2203 HOURS ON MARCH 10, 1988. THE UNIT WAS PLACED IN HOT SHUTDOWN AT 0334 HOURS ON MARCH 11, 1988, WHICH SATISFIED THE REQUIREMENTS OF TECHNICAL SPECIFICATIONS. THE REDUNDANT PUMP, 1-SI-P-1A, WAS TESTED AT REGULAR INTERVALS TO ENSURE ITS OPERABILITY WHILE REPAIR WORK COMMENCED ON 1-SI-P-1B. THE DAMAGED MOTOR LEADS WERE REPAIRED AND AN OPERABILITY TEST WAS PERFORMED, DURING WHICH HIGH TEMPERATURES ON ONE OF THE REPAIRED LEADS PREVENTED THE PUMP FROM BEING RETURNED TO OPERABLE STATUS. THE MOTOR FOR 1-SI-P-1B WAS REPLACED WITH A NEW MOTOR, WAS TESTED SATISFACTORILY AND RETURNED TO SERVICE STATUS AT 2252 HOURS ON MARCH 12, 1988. AN ANALYSIS IS BEING PERFORMED TO DETERMINE THE ROOT CAUSE OF THIS FAILURE. ANY FURTHER ACTION WILL BE BASED ON THE RESULTS OF THE ANALYSIS.

[239] SURRY 1 DOCKET 50-280 LER 88-009
IODINE SPIKE DUE TO DEFECTIVE FUEL ELEMENT.
EVENT DATE: 031188 REPORT DATE: 040588 NSSS: WE TYPE: PWR

(NSIC 208845) ON MARCH 11, 1988, AT 0630 HOURS, FOLLOWING A UNIT 1 REACTOR SHUTDOWN FOR REPAIRS TO THE 'B' LOW HEAD SAFETY INJECTION PUMP (EIIS-P), THE SPECIFIC ACTIVITY SAMPLE OF THE REACTOR COOLANT SHOWED A DOSE-EQUIVALENT I-131 LEVEL OF 1.50 MICROCURIES/CC. THIS EXCEEDS THE DOSE EQUIVALENT I-131 TECHNICAL SPECIFICATION OF LESS THAN 1.0 MICROCURIE/CC SPECIFIED IN SECTION 3.1.D.2 AND IS BEING REPORTED IN ACCORDANCE WITH THE SPECIAL REPORTING REQUIREMENTS OUTLINED IN TECHNICAL SPECIFICATION 3.1.D.4. THE IODINE SPIKE WAS CAUSED BY A KNOWN, BUT NOT SPECIFICALLY LOCATED, FUEL ELEMENT DEFECT IN THE REACTOR CORE. POST SHUTDOWN CONDITIONS ENHANCED THE RELEASE OF FISSION PRODUCTS, SPECIFICALLY I-131. THIS CAUSED AN INCREASE IN REACTOR COOLANT SPECIFIC ACTIVITY. THE IMMEDIATE CORRECTIVE ACTION WAS TO IMPLEMENT THE ACTIONS REQUIRED BY TECHNICAL SPECIFICATION TABLE 4.1.2B. SPECIFICALLY, THE LEVEL OF DOSE-EQUIVALENT I-131 WAS MONITORED AT LEAST EVERY FOUR HOURS UNTIL THE LEVEL RETURNED TO LESS THAN 1.0 MICROCURIE/CC.

[240] SURRY 2 DOCKET 50-281 LER 88-005
INOPERABLE CONTROL RODS DUE TO FAILED PHASE CONTROL CARD.
EVENT DATE: 030588 REPORT DATE: 033088 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 208846) AT 0102 HOURS ON MARCH 5, 1988, WITH UNIT 2 OPERATING AT 100% POWER, A CONTROL ROD (EIIS-ROD) URGENT FAILURE ALARM WAS RECEIVED IN THE CONTROL ROOM. IT WAS DETERMINED THAT THE FAILURE WAS IN THE 1BD POWER CABINET. TECHNICAL SPECIFICATION (T.S.) 3.12.C.3 REQUIRES THAT THE CONTROL RODS BE RETURNED TO OPERABLE STATUS WITHIN TWO HOURS OR THE UNIT BE PLACED IN HOT SHUTDOWN. AT 0302 HOURS, A UNIT SHUTDOWN WAS INITIATED IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS. AN UNUSUAL EVENT WAS DECLARED AT 0307 DUE TO A POWER REDUCTION REQUIRED BY T. S. LIMITING CONDITIONS FOR OPERATION. AT 0409 HOURS, THE INSTRUMENT TECHNICIANS REPLACED A CARD IN THE POWER CABINET AND THE URGENT FAILURE ALARM WAS CLEARED. THE RAMP WAS STOPPED AT 92.5% POWER TO FACILITATE TESTING OF THE CONTROL RODS. DURING THE TESTING, THE REPLACED CARD FAILED AND WAS REPLACED AND TESTED SATISFACTORILY AT 0548. THE CARDS WILL BE RETURNED TO THE VENDOR FOR ANALYSIS AND REWORK.

[241] SURRY 2 DOCKET 50-281 LER 88-004
MANUAL REACTOR TRIP DUE TO LOSS OF VITAL BUS CAUSED BY FAILED INVERTER.
EVENT DATE: 032788 REPORT DATE: 042688 NSSS: WE TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 209130) ON MARCH 27, 1988 AT 1621 HOURS, WITH UNIT 2 AT 100% REACTOR POWER, VITAL BUS (VB) (EIIS-ED) 2-III WAS DE-ENERGIZED DUE TO THE LOSS OF 2-III INVERTER (EIIS-INVT). A TURBINE RUNBACK WAS AUTOMATICALLY INITIATED DUE TO THE LOSS OF POWER TO THE POWER RANGE NUCLEAR INSTRUMENT NI-43 (EIIS-IG) WHICH IS POWERED FROM VB 2-III. ABNORMAL PROCEDURE AP-10.2, "LOSS OF VITAL BUS 1-III OR 2-III", WAS ENTERED AND AS REQUIRED BY THE PROCEDURE, THE REACTOR (EIIS-RCT) AND 'A' REACTOR COOLANT PUMP (RCP) (EIIS-P), WERE MANUALLY TRIPPED AT 1622 HOURS. APPROXIMATELY 30 SECONDS LATER, A HIGH STEAM FLOW/LOW REACTOR COOLANT SYSTEM (RCS) (EIIS-AB) TAVG SAFETY INJECTION (SI) (EIIS-BQ) OCCURRED. OPERATORS FOLLOWED APPROPRIATE PLANT PROCEDURES AND QUICKLY STABILIZED THE UNIT FOLLOWING THE REACTOR TRIP/SAFETY INJECTION. AN ENGINEERING EVALUATION OF THE INVERTER DETERMINED THAT AN INTERNAL INDUCTOR FAILED DUE TO AGE, THUS CAUSING A CURRENT SURGE WHICH BLEW THE FUSE AND TRIPPED THE AC OUTPUT BREAKER. ONE OF TWO OF THE STATION VITAL BUS INVERTERS PER UNIT HAVE BEEN REPLACED WITH AN UNINTERRUPTIBLE POWER SUPPLY. THE REMAINING TWO VITAL BUS INVERTERS (2-III AND 1-III) WILL BE REPLACED WITH UNINTERRUPTIBLE POWER SUPPLIES DURING THE UPCOMING REFUELING OUTAGES.

[242] SURRY 2 DOCKET 50-281 LER 88-007
INOPERABLE INDIVIDUAL ROD POSITION INDICATORS DUE TO INSTRUMENT DRIFT.
EVENT DATE: 040588 REPORT DATE: 050588 NSSS: WE TYPE: PWR
VENDOR: MAGNETICS DIV SPANG INDUSTRIES, INC.

(NSIC 209222) ON APRIL 5, 1988 AT 2010 HOURS, FOLLOWING A REACTOR STARTUP, WITH UNIT 2 REACTOR POWER AT 10(-8) AMPS, THE SHIFT SUPERVISOR OBSERVED THAT THE INDIVIDUAL ROD POSITION INDICATORS (IRPIS) (EIIS-ZI) FOR CONTROL RODS (EIIS-ROD) J-9, J-7, AND E-5 IN SHUTDOWN BANK 'B' DIFFERED FROM THE ROD GROUP DEMAND COUNTER BY GREATER THAN 12 STEPS. IT WAS SUBSEQUENTLY DETERMINED THAT THIS CONDITION HAD EXISTED SINCE 1050 HOURS ON APRIL 5, 1988 WHEN THE SHUTDOWN BANKS HAD BEEN WITHDRAWN. THE IRPIS WERE DECLARED INOPERABLE IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.12.E. THE IRPIS WERE CALIBRATED AND RETURNED TO SERVICE AT 2020 HOURS. INCORRECT ROD POSITION INDICATION, DUE TO PRIMARY COOLANT TEMPERATURE CHANGES, IS KNOWN TO BE A GENERIC WESTINGHOUSE PWR CONCERN. PREVIOUS INTERPRETATION OF T. S. 3.12.E, RECOGNIZING THE GENERIC PROBLEM, WAS THAT THE SPECIFICATION APPLIED ABOVE HOT SHUTDOWN, WHERE THE IRPIS ARE CALIBRATED. TECHNICAL SPECIFICATION 3.12.E IS NOW INTERPRETED TO APPLY THE LIMIT OF 12 STEPS

WHENEVER A REACTOR TRIP BREAKER IS CLOSED AND THE CONTROL RODS ARE NOT FULLY INSERTED. A PROPOSED TECHNICAL SPECIFICATION CHANGE IS BEING EVALUATED WHICH WILL RECOGNIZE THE EFFECT OF TEMPERATURE ON THE ROD POSITION INDICATORS. A SETPOINT CHANGE IS BEING EVALUATED WHICH WILL LOWER THE P-250 ROD DEVIATION ALARM TO WARN THE OPERATOR BEFORE THE 12 STEP DEVIATION IS REACHED.

[243] SUSQUEHANNA 2 DOCKET 50-388 LER 88-005
MULTIPLE ESF ACTUATIONS CAUSED BY FAILED RPS EPA BREAKER LOGIC CARD.
EVENT DATE: 031988 REPORT DATE: 041588 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 208944) AT 228 ON MARCH 19, 1988, WITH THE UNIT AT 0% POWER IN THE REFUELING MODE, THE UNIT EXPERIENCED MULTIPLE ENGINEERED SAFETY FEATURE (ESF) ACTUATIONS. THESE ACTUATIONS OCCURRED WHEN THE PRIMARY POWER SUPPLY, TO THE "A" REACTOR PROTECTION SYSTEM (RPS) PANEL 2Y201A WAS LOST DUE TO THE UNPLANNED AUTOMATIC TRIPPING OF THE "C" ELECTRICAL PROTECTION ASSEMBLY (EPA) BREAKER. OPERATIONS PERSONNEL CARRIED OUT THE INSTRUCTIONS OF OFF-NORMAL PROCEDURE ON-259-002, "CONTAINMENT ISOLATION." BY 2235, THE "A" RPS BUS HAD BEEN TRANSFERRED TO ITS ALTERNATE POWER SUPPLY, THE RPS HALF-SCRAM HAD BEEN RESET, AND THE PRIMARY CONTAINMENT ISOLATION HAD BEEN RESET. ONE RWCU PUMP WAS STARTED AT 2252. RHR SHUTDOWN COOLING WAS RESTORED AT 2337. TRIPPING OF THE "C" EPA BREAKER WAS CAUSED BY A FAILURE IN THE VOLTAGE REGULATION CIRCUITRY OF THE BREAKER'S ELECTRICAL PROTECTION LOGIC CARD. THE ESF SYSTEMS ACTUATED AS DESIGNED AND OPERATED PROPERLY. THE FAILED LOGIC CARD WAS REPLACED AND THE NEW CARD WAS CALIBRATED.

[244] SUSQUEHANNA 2 DOCKET 50-388 LER 88-004
UNPLANNED ESF ACTUATION DURING REACTOR LEVEL INSTRUMENT CALIBRATION.
EVENT DATE: 032288 REPORT DATE: 042188 NSSS: GE TYPE: BWR

(NSIC 209020) AT 1518 HOURS ON MARCH 22, 1988, WITH UNIT 2 IN THE REFUELING CONDITION, A FULL REACTOR PROTECTION SYSTEM (RPS) ACTUATION OCCURRED WHEN DIVISION I OF THE RPS WAS INADVERTENTLY TRIPPED, COINCIDENT WITH A DIVISION II RPS TRIP DUE TO 24 VDC BATTERY TESTING. SUBSEQUENT TO THE RELEASE OF A DIVISION I INSTRUMENT TO THE I&C GROUP FOR CALIBRATION, THE 24 VDC BATTERY SYSTEM WAS RELEASED TO PERFORM BATTERY TESTING, WHICH TRIPS RPS DIVISION II. FOLLOWING INSTRUMENT CALIBRATION, AN I&C TECHNICIAN INADVERTENTLY REMOVED A JUMPER OUT OF SEQUENCE, CAUSING THE DIVISION I RPS TRIP, RESULTING IN THE FULL RPS ACTUATION. THIS INCIDENT WAS CAUSED BY THE RELEASE OF WORK CONCURRENTLY ON BOTH DIVISIONS OF RPS AND A COGNITIVE PERSONNEL ERROR ON THE PART OF THE I&C TECHNICIAN. THE INCIDENT WAS REVIEWED WITH ALL I&C PERSONNEL TO RE-EMPHASIZE THE NEED FOR PAYING CLOSE ATTENTION TO DETAIL. A TASK TEAM REVIEW CONCLUDED THAT THE AVOIDANCE OF PROLONGED DIVISIONAL HALF-SCRAMS AND THE RELEASE OF WORK CONCURRENTLY IN SEPARATE DIVISIONS NEED TO BE RE-EMPHASIZED, CONSISTENT WITH THE WORK RELEASE PHILOSOPHY. THE INSERTION OF MANUAL SCRAMS AND/OR ISOLATIONS, WHENEVER POSSIBLE, IS BEING EVALUATED.

[245] SUSQUEHANNA 2 DOCKET 50-388 LER 88-007
UNPLANNED RPS ACTUATION CAUSED BY REMOVAL OF INCORRECT FUSE.
EVENT DATE: 032288 REPORT DATE: 042188 NSSS: GE TYPE: BWR

(NSIC 209037) AT 1950 ON MARCH 22, 1988, WITH THE UNIT IN REFUELING WITH ALL FUEL REMOVED FROM THE REACTOR VESSEL, THE UNIT EXPERIENCED A REACTOR PROTECTION SYSTEM (RPS) ACTUATION WHEN AN INCORRECT FUSE WAS REMOVED FROM SERVICE. THE FUSE WAS BEING REMOVED AS PART OF A PERSONNEL SAFETY PERMIT BLOCKING AND WAS DONE IN ACCORDANCE WITH APPLICABLE ADMINISTRATIVE PROCEDURES. REMOVAL OF THE FUSE RESULTED IN AN RPS DIVISION I TRIP. DIVISION II OF RPS HAD ALREADY BEEN TRIPPED DUE TO PLANNED 24 VDC BATTERY TESTING; THUS, A FULL RPS ACTUATION OCCURRED. THE

FUSE WAS REINSTALLED AND THE SCRAM WAS RESET. CAUSE OF THE EVENT WAS COGNITIVE PERSONNEL ERRORS. THE PERSON REQUESTING THE SAFETY PERMIT, THE PERSON COMPLETING THE SAFETY PERMIT FORM AND THE PERSON APPLYING THE BLOCKING UTILIZED AN ABBREVIATED FUSE IDENTIFICATION. NONE OF THESE PEOPLE REALIZED THE EXISTENCE OF MORE THAN ONE FUSE IN THE PANEL WITH THE SAME ABBREVIATED FUSE IDENTIFICATION. APPLICABLE ADMINISTRATIVE PROCEDURES AND INSTRUCTIONS HAVE BEEN REVISED TO REQUIRE THE SPECIFICATION OF ADDITIONAL INFORMATION ON SAFETY PERMIT BLOCKING REQUESTS WHICH " .L UNIQUELY IDENTIFY THE DESIGNED FUSE.

[246] SUSQUEHANNA 2 DOCKET 50-388 LER 88-006
MSIV ISOLATION LOGIC ACTUATION WHEN WRONG RELAY WAS REMOVED DURING MODIFICATION WORK.
EVENT DATE: 032388 REPORT DATE: 041988 NSSS: GE TYPE: BWR

(NSIC 209036) ON MARCH 23, 1988 WITH UNIT 2 IN CONDITION 5 AT 0% POWER, A FULL MSIV ISOLATION SIGNAL WAS RECEIVED WHEN THE WRONG RELAY WAS REMOVED DURING MODIFICATION WORK TO THE MSIV ISOLATION LOGIC. THE MSIV'S WERE OUT OF SERVICE AND GAGGED CLOSED AT THE TIME OF THE OCCURRENCE SO NO VALVE MOTION RESULTED. THE EVENT WAS CAUSED BY CONFLICTING INFORMATION IN THE WORK INSTRUCTIONS OF A CONSTRUCTION WORK ORDER (CWO) WHICH INFLUENCED THE WORK GROUP TO SELECT THE WRONG RELAY FOR REMOVAL. THIS EVENT WAS DETERMINED TO BE REPORTABLE PER 10CFR50.73(A)(2)(IV), IN THAT ACTUATION OF THE MSIV ISOLATION LOGIC CONSTITUTED AN UNPLANNED ESF ACTUATION. THE EVENT DID NOT POSE ANY SIGNIFICANT SAFETY CONSEQUENCES. THE MSIV ISOLATION TRIP LOGIC IS PROVIDED TO LIMIT THE AMOUNT OF FISSION PRODUCT RELEASE FOR CERTAIN POSTULATED EVENTS. THIS LOGIC IS REQUIRED FOR OPERATIONAL CONDITIONS 1, 2, AND 3 AND DOES NOT APPLY FOR THIS SITUATION WITH UNIT 2 BEING SHUTDOWN IN CONDITION 5 AT 0% POWER. THE MSIV'S WERE ALREADY CLOSED WHICH PLACED THEM IN THE DESIRABLE ISOLATION TRIP POSITION. THE RELAY WAS REINSTALLED AND THE MSIV ISOLATION SIGNAL WAS RESET. TO PREVENT RECURRENCE IN THE FUTURE, WORK INSTRUCTIONS FOR SCHEME CHECKS WILL BE WRITTEN TO IDENTIFY RELAYS BY RELAY NUMBER AND COORDINATE IDENTIFIER WHEN APPLICABLE IN THE SPECIFIC WORK STEPS.

[247] THREE MILE ISLAND 2 DOCKET 50-320 LER 87-005
CONTAMINATED WOODEN PALLET DISCOVERED IN AN UNRESTRICTED AREA.
EVENT DATE: 060787 REPORT DATE: 070787 NSSS: BW TYPE: PWR

(NSIC 209204) AT APPROXIMATELY 1250 HOURS ON SUNDAY, JUNE 7, 1987, DURING A ROUTINE SURVEY BY RADIOLOGICAL CONTROLS PERSONNEL, A CONTAMINATED WOODEN PALLET WAS DISCOVERED IN A REFUSE CONTAINER IN AN UNRESTRICTED AREA. THE RADIOLOGICAL SURVEY OF THE PALLET INDICATED 2 MR/HR GAMMA AND 160 MRAD/HR BETA CONTACT DOSE RATES. THE FIXED CONTAMINATION LEVEL, BASED ON THE CONTACT DOSE RATES, WAS IN EXCESS OF TEN (10) TIMES THE LIMITS ESTABLISHED IN THE GPU NUCLEAR CORPORATION RADIATION PROTECTION PLAN WHICH IS A TMI-2 LICENSING BASIS DOCUMENT. THEREFORE, THIS EVENT IS REPORTABLE PURSUANT TO 10 CFR 20.405(A)(1)(V). THE ROOT CAUSE OF THIS EVENT WAS A LESS THAN ADEQUATE SURVEY PRIOR TO THE TRANSFER OF THE PALLET FROM A CONTROLLED AREA TO AN UNRESTRICTED AREA. HOWEVER, THE CONTAMINATION SOURCE, THE PALLET'S ORIGIN OF LOCATION, AND THE METHOD OF TRANSPORT COULD NOT BE DETERMINED. THE DUMPSTER WAS EMPTY ON JUNE 5, 1987, AND WAS SURVEYED ON A DAILY BASIS. THUS, THIS EVENT OCCURRED DURING THE PERIOD JUNE 5 THROUGH JUNE 7, 1987. THE REMAINING MATERIAL IN THE REFUSE CONTAINER WAS SURVEYED AND NO ADDITIONAL CONTAMINATED MATERIAL WAS FOUND. THE PALLET WAS RELOCATED TO A RADIOLOGICAL CONTROLLED AREA. A SURVEY OF ALL UNRESTRICTED STORAGE AREAS IS BEING PERFORMED. THE CONSOLIDATION OF RADIOACTIVE MATERIALS IS PRESENTLY BEING EVALUATED TO IMPROVE THE CONTROL OF THE MATERIAL.

[248] THREE MILE ISLAND 2 DOCKET 50-320 LER 87-006
CONTAMINATED CHAIN DISCOVERED IN AN UNRESTRICTED AREA.
EVENT DATE: 072887 REPORT DATE: 082787 NSSS: BW TYPE: PWR

(NSIC 209205) ON 7/28/87, DURING A ROUTINE SURVEY OF A VEHICLE THAT CARRIED RADIOACTIVE MATERIAL, A CONTAMINATED CHAIN WAS DISCOVERED IN THE TRUCK BED. THE RADIOLOGICAL SURVEY OF THE CHAIN INDICATED A SMEARABLE CONTAMINATION LEVEL OF 10,000 DPM (SMEAR) AND A FIXED CONTAMINATION LEVEL OF 60,000 DPM (DIRECT READING WITH A PANCAKE PROBE). THE AREA OF CONTAMINATION WAS LIMITED TO 1 LINK OF A 20 FOOT LENGTH OF CHAIN. THE FIXED CONTAMINATION LEVEL, BASED ON THE CONTACT DOSE RATES, WAS IN EXCESS OF 10 TIMES THE LIMITS ESTABLISHED IN THE GPU NUCLEAR CORPORATION RADIATION PROTECTION PLAN WHICH IS A TMI-2 LICENSING BASIS DOCUMENT. THEREFORE, THIS EVENT IS REPORTABLE PURSUANT TO 10 CFR 50.204(A)(1)(V). THE ROOT CAUSE OF THIS EVENT WAS A LESS THAN ADEQUATE SURVEY OF THE CHAIN. THE CHAIN IS NORMALLY "BALLED" TOGETHER AND STORED IN A BAG IN THE TRUCK BED WHEN NOT IN USE. IN THIS CONFIGURATION, IT WOULD BE VERY UNLIKELY THAT THE SMALL AREA OF CONTAMINATION WOULD BE DETECTED DURING ROUTINE SURVEYS. EVEN UNBAGGED, SUCH A SMALL AREA OF CONTAMINATION COULD EASILY GO UNDETECTED DURING ROUTINE SURVEYS. UPON DISCOVERY, THE CHAIN WAS PROPERLY BAGGED/TAGGED AND TAKEN INTO THE WHPF. THE TRUCK THAT THE CHAIN WAS FOUND ON WAS SURVEYED. NO FURTHER CONTAMINATION WAS FOUND. AS A RESULT OF THIS OCCURRENCE, ALL OTHER CHAINS USED TO SECURE RADIOACTIVE MATERIALS TO VEHICLES WERE SURVEYED. SIMILAR LEVERS: 86-06, 87-02, AND 87-05.

[249] TROJAN DOCKET 50-344 LER 86-011 REV 01
UPDATE ON REACTOR TRIP FROM SPURIOUS HIGH VIBRATION SIGNAL ON MAIN TURBINE.
EVENT DATE: 111486 REPORT DATE: 041488 NSSS: WE TYPE: PWR
VENDOR: COPES-VULCAN, INC.
GEN ELEC CO (STEAM TURB/ENGRD PROD)
WESTINGHOUSE ELECTRIC CORP.

(NSIC 208901) ON NOVEMBER 14, 1986 THE PLANT WAS OPERATING AT 100% POWER AT NORMAL OPERATING TEMPERATURE AND PRESSURE. AT 1814 A SPURIOUS HIGH VIBRATION SIGNAL FROM THE NO. 1 BEARING ON THE MAIN TURBINE CAUSED A TURBINE TRIP WHICH RESULTED IN A REACTOR TRIP. DURING THE SHUTDOWN, THE N-35 INTERMEDIATE RANGE NUCLEAR INSTRUMENT FAILED WITH A STEADY INDICATION OF 2 E-9 AMPS DUE TO FAILURE OF ITS COMPENSATED ION CHAMBER. THE SOURCE RANGE INSTRUMENTS DID NOT AUTOMATICALLY ENERGIZE BECAUSE BOTH INTERMEDIATE RANGE INSTRUMENTS DID NOT INDICATE LESS THAN 1 E-10 AMPS. THE SOURCE RANGE DETECTORS WERE ENERGIZED AT 1908 BY PULLING THE POWER SUPPLY FUSES TO THE N-35 INTERMEDIATE RANGE DETECTOR. THE HIGH VIBRATION SIGNAL WAS INITIALLY THOUGHT TO BE DUE TO ABNORMAL WEAR OF THE VIBRATION PROBE. FURTHER INSPECTION CONDUCTED DURING THE 1987 REFUELING OUTAGE REVEALED LOOSE WIRES IN THE VIBRATION PROBE'S JUNCTION BOX WHICH WERE DETERMINED TO HAVE CAUSED THE SPURIOUS TRIP. INITIAL CORRECTIVE ACTION WAS TAKEN TO DISABLE THE TRIP SIGNAL FROM THE VIBRATION PROBE AND TO REPLACE THE FAILED INTERMEDIATE RANGE DETECTOR. FINAL CORRECTIVE ACTION WAS TO TIGHTEN THE LOOSE WIRING. THE EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[250] TROJAN DOCKET 50-344 LER 87-006 REV 02
UPDATE ON DEFICIENCIES IN FLOOD PROTECTION DESIGN PROVISIONS.
EVENT DATE: 030987 REPORT DATE: 041988 NSSS: WE TYPE: PWR

(NSIC 209041) DURING REVIEW OF THE TURBINE BUILDING FLOODING DESIGN BASIS (AS STATED IN THE FINAL SAFETY ANALYSIS REPORT) ON MARCH 9, 1987, IT WAS DETERMINED THAT THE FLOOD RELIEF LOUVERS IN THE WEST WALL OF THE TURBINE BUILDING WERE NOT ADEQUATELY DESIGNED TO RELIEVE A SUFFICIENT QUANTITY OF WATER TO PREVENT FLOODING OF SAFETY RELATED EQUIPMENT IN THE EVENT OF A CIRCULATING WATER LINE FAILURE. THE AFFECTED EQUIPMENT IN THE TURBINE BUILDING INCLUDES THE AUXILIARY FEEDWATER (AFW) PUMPS AND EMERGENCY DIESEL GENERATORS (EDGS). THE APPARENT CAUSE OF THIS

EVENT WAS A DESIGN ERROR. THE FLOOD RELIEF LOUVERS WERE REMOVED AND WILL NOT BE RE-INSTALLED. THEY HAVE BEEN REPLACED BY FLAP-TYPE FLOOD RELIEF PANELS WHICH ARE HINGED AT THE TOP AND OPEN OUTWARD ON WATER PRESSURE. THE FLOOD PROTECTION DIKES WERE RAISED TO ELEVATION 48 FEET TO PREVENT FLOOD WATER FROM ENTERING THE AFW PUMP ROOMS, EDG ROOMS, AND REMOTE SHUTDOWN PANEL ROOM. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[251] TROJAN DOCKET 50-344 LER 87-015 REV 01
UPDATE ON STEAM GENERATOR LEVEL TRANSMITTERS IMPROPERLY CALIBRATED.
EVENT DATE: 041387 REPORT DATE: 040788 NSSS: WE TYPE: PWR

(NSIC 208866) DURING APRIL 13 THROUGH APRIL 15, 1987, ANNUAL CALIBRATION OF STEAM GENERATOR LEVEL TRANSMITTERS WAS PERFORMED. TEN STEAM GENERATOR LEVEL TRANSMITTERS WERE FOUND APPARENTLY OUT-OF-CALIBRATION SUCH THAT THEY WOULD NOT HAVE ACTUATED A TURBINE TRIP AND FEEDWATER ISOLATION ON HIGH-HIGH STEAM GENERATOR WATER LEVEL WITHIN THE TECH SPEC ALLOWED VALUE OF LESS THAN OR EQUAL TO 76%. FURTHER INVESTIGATION REVEALED THAT THE CALIBRATION WAS PERFORMED IMPROPERLY. THE CAUSE OF THIS EVENT WAS PROCEDURE DEFICIENCY. THE PROCEDURE DID NOT SPECIFY THAT WATER SHOULD BE COMPLETELY DRAINED FROM THE TRANSMITTERS PRIOR TO BEGINNING THE CALIBRATION. THE TRANSMITTERS WERE THUS CALIBRATED WITHOUT BEING COMPLETELY DRAINED, AND AN ERROR WAS INTRODUCED INTO THE CALIBRATION PROCESS. THE LEVEL TRANSMITTERS WERE PROPERLY RE-CALIBRATED TO WITHIN THE ALLOWED TOLERANCE. THE CALIBRATION DATA SHEETS FOR THESE INSTRUMENTS HAVE BEEN REVISED TO REQUIRE DRAINING WATER FROM THE TRANSMITTERS PRIOR TO CALIBRATION. I&C TECHNICIANS WERE COUNSELED ON THE NEED TO COMPLETELY DRAIN THE WATER FROM THE LEVEL TRANSMITTERS PRIOR TO PERFORMING CALIBRATION. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[252] TROJAN DOCKET 50-344 LER 87-020 REV 01
UPDATE ON SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE MISSED DUE TO INADVERTENT DELETION FROM SCHEDULE.
EVENT DATE: 080387 REPORT DATE: 040788 NSSS: WE TYPE: PWR

(NSIC 208835) ON AUGUST 3, 1987, DURING A REVIEW OF SURVEILLANCE RECORDS, IT WAS DETERMINED THAT THE MONTHLY CHANNEL CHECK OF THE SEISMIC MONITORING INSTRUMENTATION REQUIRED BY TECHNICAL SPECIFICATION 4.3.3.3.1 WAS NOT PERFORMED IN JULY 1987. THE CAUSE OF THIS EVENT WAS AN INADVERTENT DELETION OF THE SURVEILLANCE FROM THE COMPUTERIZED SURVEILLANCE SCHEDULE DUE TO INADEQUATE UNDERSTANDING OF THE SOFTWARE. AN ERROR IN THE PROGRAM CAUSED A SCHEDULING DATE ERROR WHICH RESULTED IN THE SURVEILLANCE NOT BEING SCHEDULED FOR PERFORMANCE IN JULY. THE REQUIRED SURVEILLANCE WAS PERFORMED AND THE SEISMIC MONITORING INSTRUMENTATION WAS VERIFIED AS OPERABLE. THE COMPUTERIZED SURVEILLANCE SCHEDULING PROGRAM HAS BEEN REVIEWED. THE ERROR WHICH CAUSED THIS EVENT HAS BEEN CORRECTED, AND ADDITIONAL SAFEGUARDS HAVE BEEN INCORPORATED INTO THE PROGRAM TO PROTECT AGAINST INADVERTENT DELETION OF ENTRIES. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[253] TROJAN DOCKET 50-344 LER 87-025 REV 01
UPDATE ON CONTAINMENT VENTILATION ISOLATION DUE TO SPURIOUS ACTUATION OF PRM-1C.
EVENT DATE: 083087 REPORT DATE: 040788 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 208836) ON AUGUST 30, 1987, A CONTAINMENT VENTILATION ISOLATION OCCURRED DURING A CONTAINMENT PRESSURE REDUCTION. THE CONTAINMENT VENTILATION ISOLATION WAS DUE TO A MOMENTARY SPIKE ON THE CONTAINMENT RADIATION MONITORING SYSTEM LOW LEVEL NOBLE GAS MONITOR (PRM-1C) WHICH EXCEEDED THE SETPOINT FOR THE MONITOR. THE CAUSE OF PRM-1C SPIKING HIGH WAS INITIALLY BELIEVED TO BE A SPURIOUS ELECTRONICS FAILURE IN THE DETECTOR ANTI-JAM CIRCUITRY. IT WAS SUBSEQUENTLY

DETERMINED THAT A FAULTY PRM-1C DETECTOR GENERATED INTERMITTENT NOISE THAT RESULTED IN HIGH CHANNEL COUNT RATES. THE PRINTED CIRCUIT CARD FOR THE ANTI-JAM CIRCUITRY WAS REPLACED, AND PRM-1C WAS RETURNED TO SERVICE. SUBSEQUENTLY, THE PRM WAS OBSERVED TO CONTINUE TO SPURIOUSLY SPIKE HIGH, BUT NOT TO THE CONTAINMENT VENTILATION ISOLATION SETPOINT. THE PRM-1C DETECTOR WAS REPLACED FOLLOWING FURTHER TROUBLESHOOTING, AND THE PERIODIC SPIKES IN COUNT RATE DISAPPEARED. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[254] TROJAN DOCKET 50-344 LER 88-003
SINGLE FAILURE MECHANISM DISCOVERED WHICH COULD OVERPRESSURIZE CONTAINMENT ELECTRICAL PENETRATION SEALS.
EVENT DATE: 032488 REPORT DATE: 042288 NSSS: WE TYPE: PWR

(NSIC 209013) DURING A CONTAINMENT PENETRATION RELIABILITY REVIEW ON MARCH 24, 1988, A SINGLE FAILURE MECHANISM WAS DISCOVERED THAT COULD RESULT IN OVERPRESSURIZATION OF CONTAINMENT ELECTRICAL PENETRATION SEALS. THE SEALS WERE OPERATED WITH A CONTINUOUS SUPPLY OF NITROGEN TO THE PENETRATION MODULE. THE NITROGEN SUPPLY SYSTEM TO CONTAINMENT ELECTRICAL PENETRATIONS IS NOT PROVIDING ADEQUATE OVERPRESSURE PROTECTION DOWNSTREAM OF THE PRESSURE REGULATORS. THEREFORE, FAILURE OF A REGULATOR COULD OVERPRESSURIZE THE SEALS AND CAUSE THEIR FAILURE. THE CAUSE OF THIS CONDITION HAS NOT BEEN DETERMINED. CORRECTIVE ACTION WAS TAKEN TO ISOLATE THE NITROGEN SUPPLY TO THE CONTAINMENT ELECTRICAL PENETRATIONS SINCE NITROGEN PRESSURE IS NOT NECESSARY FOR THE SEALS TO FULFILL THEIR DESIGN FUNCTION. NITROGEN PRESSURE IS BEING MANUALLY ADJUSTED FOR THE APPROPRIATE VALUE ONCE PER SHIFT. EVALUATION OF THIS EVENT IS STILL IN PROGRESS. THE STATUS OF THE EVALUATION WILL BE PROVIDED IN A SUPPLEMENTAL REPORT TO BE SUBMITTED BY JUNE 22, 1988. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[255] TROJAN DOCKET 50-344 LER 88-004
PLANT OPERATED WITH MANUAL FEEDWATER LINE VALVES OPEN IN VIOLATION OF TECH SPEC.
EVENT DATE: 032888 REPORT DATE: 042788 NSSS: WE TYPE: PWR

(NSIC 209121) DURING A ROUTINE INSPECTION ON MARCH 28, 1988, IT WAS DISCOVERED THAT THE PLANT WAS BEING OPERATED IN A CONDITION THAT DID NOT COMPLY WITH TROJAN TECHNICAL SPECIFICATION (TTS) REQUIREMENTS. FOUR MANUAL DRAIN VALVES LOCATED ON THE MAIN FEEDWATER LINES WHICH ARE IDENTIFIED AS MANUAL CONTAINMENT ISOLATION VALVES IN TTS TABLE 3.6-1 WERE FOUND TO BE OPEN, WITH PRESSURE TRANSDUCERS CONNECTED TO THE DOWNSTREAM DRAIN LINES. THE FINAL SAFETY ANALYSIS REPORT ASSUMED POSITION FOR THESE VALVES IS LOCKED CLOSED. THE FEEDWATER LINE DRAIN VALVES WERE OPENED IN ACCORDANCE WITH A TEMPORARY MODIFICATION IMPLEMENTED ON AUGUST 7, 1987. THE TTS 4.6.1.1 REQUIREMENT TO PERIODICALLY VERIFY THE VALVES SHUT WAS NOT MET. THE APPARENT CAUSE OF THIS EVENT WAS INADEQUATE DEVELOPMENT AND REVIEW OF THE TEMPORARY MODIFICATION AND ITS SAFETY EVALUATION. IMMEDIATE CORRECTIVE ACTION WAS TO CLOSE THE MANUAL FEEDWATER LINE DRAIN VALVES. AN ACTION PLAN HAS BEEN DEVELOPED TO EVALUATE AND CORRECT CONCERNS REGARDING CONTAINMENT ISOLATION REQUIREMENTS, THE ADEQUACY OF SAFETY EVALUATIONS, AND THE CAUSES OF THIS EVENT. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[256] TROJAN DOCKET 50-344 LER 88-005
SURVEILLANCE INTERVAL FOR CHILLED WATER RETURN VALVES EXCEEDED.
EVENT DATE: 033188 REPORT DATE: 042988 NSSS: WE TYPE: PWR

(NSIC 209228) ON MARCH 31, 1988, DURING A REVIEW OF INSERVICE TEST (IST) PROGRAM RECORDS, IT WAS DISCOVERED THAT THE REQUIRED TEST INTERVAL FOR CHILLED WATER RETURN VALVES CV-10014 AND CV-10015 WAS EXCEEDED. AN INCREASE IN THE VALVES' STROKE TIMES OBSERVED DURING THE NORMAL QUARTERLY TIMING TEST ON JANUARY 12, 1988 REQUIRED THE TESTING FREQUENCY TO BE INCREASED TO MONTHLY. THE NEXT STROKE TIME TEST WAS NOT PERFORMED UNTIL FEBRUARY 25, 1988, WHICH EXCEEDED THE MONTHLY

TESTING INTERVAL. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THE JANUARY 12 TEST WAS LOGGED AS BEING PERFORMED ON JANUARY 19, THUS THE NEXT TEST WAS NOT SCHEDULED FOR PERFORMANCE WITHIN THE REQUIRED INTERVAL. A REVIEW OF PLANT IST RECORDS CONFIRMED THAT VALVES CV-10014 AND CV-10015 WERE TESTED ON FEBRUARY 25. 1988. PLANT ENGINEERING TEST 9-4 WILL BE REVISED BY JUNE 30, 1988 TO PROVIDE IMPROVED GUIDANCE ON SCHEDULING SURVEILLANCE TESTS WHEN TESTING AT INCREASED FREQUENCY IS REQUIRED. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[257] TURKEY POINT 3 DOCKET 50-250 LER 88-003
CONTAINMENT VENTILATION AND CONTROL ROOM VENTILATION ISOLATION WHILE CONTAINMENT PARTICULATE RADIATION MONITOR SETPOINT WAS BEING CHECKED.
EVENT DATE: 030888 REPORT DATE: 040788 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)
VENDOR: NUCLEAR RESEARCH CORP.

(NSIC 208832) ON MARCH 8, 1988, THE CONTAINMENT RADIOACTIVE PARTICULATE MONITOR (R-11) SETPOINT WAS BEING VERIFIED. THE SETPOINT WAS VERIFIED BY PRESSING THE HIGH ALARM PUSHBUTTON, THEN READING THE SETPOINT FROM THE INSTRUMENT'S DIGITAL DISPLAY. AT 1008, AS THE HIGH ALARM WAS BEING PRESSED, AN ALARM SIGNAL WAS GENERATED. THIS RESULTED IN THE CONTROL ROOM VENTILATION AND CONTAINMENT VENTILATION SYSTEMS ISOLATING. THESE SYSTEMS ISOLATED AS REQUIRED, WITH THE CONTROL ROOM VENTILATION SYSTEM SWITCHING INTO THE RECIRCULATION MODE. THE HIGH ALARM INCORPORATES A REDUNDANT DOUBLE-FILAMENT LIGHTED PUSHBUTTON. IN THE EVENT OF A FAILURE OF BOTH LIGHT FILAMENTS, THE HIGH ALARM WILL ACTUATE TO SIGNAL AN INDICATOR MALFUNCTION. TROUBLESHOOTING AFTER THE EVENT FOUND ONE FILAMENT OF THE HIGH ALARM PUSHBUTTON HAD FAILED. THE PUSHBUTTON LIGHT IS NORMALLY OFF. REPEATED PRESSING OF THE HIGH ALARM PUSHBUTTON FOLLOWING THE EVENT DID NOT RESULT IN A REPEAT OF THE ACTUATION. THE PROBABLE CAUSE OF THE ACTUATION WAS A MINOR ELECTRICAL TRANSIENT UPON FAILURE OF ONE FILAMENT OF THE LIGHT WHILE THE HIGH ALARM PUSHBUTTON WAS BEING PRESSED. THE OPERATORS VERIFIED THAT NO VALID ALARM CONDITION EXISTED. THE FAILED DUAL FILAMENT BULB WAS REPLACED. RELIABILITY OF THE DRAWERS IS BEING MONITORED. OTHER ACTIONS TO DECREASE SPURIOUS ACTUATIONS ARE BEING EVALUATED AND SCHEDULED.

[258] TURKEY POINT 3 DOCKET 50-250 LER 88-004
AUXILIARY FEEDWATER INITIATION ON LOW STEAM GENERATOR LEVEL DUE TO INADEQUATE MONITORING OF STEAM GENERATOR LEVEL.
EVENT DATE: 031888 REPORT DATE: 041588 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)

(NSIC 208996) ON MARCH 18, 1988, UNIT 3 WAS IN MODE 3, WITH THE REACTOR COOLANT SYSTEM BEING COOLED USING THE STEAM GENERATORS (SG). SG LEVELS WERE BEING CONTROLLED MANUALLY USING THE BYPASS FEEDWATER CONTROL VALVES (FCV). THE REACTOR CONTROL OPERATOR (RCO) WAS ATTEMPTING TO MAINTAIN LEVEL WITHIN THE APPROXIMATE 50% TO 70% RANGE ON THE NARROW RANGE SG LEVEL INDICATOR. AT APPROXIMATELY 1050 THE 3B SG LEVEL REACHED 68% AND THE RCO DECREASED THE FLOW. BY ABOUT 1120 THE SG LEVEL HAD DROPPED TO THE LO SG LEVEL SETPOINT OF 35%, AND AT 1129 THE SG LEVEL REACHED THE LO-LO SG LEVEL SETPOINT OF 15%. UPON REACHING THE LO-LO SETPOINT ON 2 OUT OF 3 CHANNELS, THE 3 AUXILIARY FEEDWATER PUMPS (AFW) RECEIVED A START SIGNAL AND DELIVERED WATER TO ALL 3 SG'S. UPON THE SG 3B WATER LEVEL RETURNING TO THE OPERATING RANGE, THE AFW PUMPS WERE SECURED AND SG LEVEL CONTINUED TO BE CONTROLLED MANUALLY USING THE BYPASS FCVS. THE CAUSE OF THE AFW ACTUATION WAS PERSONNEL ERROR IN THAT THE RCO FAILED TO ADEQUATELY MONITOR THE SG LEVEL. THE RCO'S ATTENTION WAS FOCUSED ON PERFORMING AN OPERABILITY TEST OF A NIS CHANNEL, AND BECAUSE OF THIS DISTRACTION, HE FAILED TO TAKE CORRECTIVE ACTIONS TO RETURN THE SG LEVEL TO THE OPERATING RANGE UPON THE LO LEVEL ALARM ANNUNCIATING AT THE 35% LEVEL, PRIOR TO THE AUTOMATIC AFW ACT ATION.

[259] VERMONT YANKEE DOCKET 50-271 LER 88-002 REV 01
UPDATE ON MAIN STEAM RELIEF VALVE ABOVE SETPOINT DUE TO STEAM CUTS ON THE PILOT SEAT AND SAFETY VALVE BINDS DUE TO MISALIGNMENT.
EVENT DATE: 030188 REPORT DATE: 041488 NSSS: GE TYPE: BWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV
TARGET ROCK CORP.

(NSIC 208963) DURING 100% POWER OPERATION, ON 02-25-88 AND 02-26-88, TWO MAIN STEAM (EIIS=SB) RELIEF VALVES, REMOVED DURING THE 1987 OUTAGE, AND TWO MAIN STEAM SAFETY VALVES, ONE REMOVED DURING THE 1985/86 OUTAGE AND ONE REMOVED DURING THE 1987 OUTAGE, WERE TESTED AS REQUIRED BY TECH SPEC SECTION 4.6.D.2. ONE OF THE RELIEF VALVES, SERIAL NUMBER 11, ACTIVATED ABOVE THE SETPOINT PRESSURE SPECIFIED IN TECH SPEC SECTION 2.2.B. THE OTHER RELIEF VALVE, SERIAL NUMBER 13, SATISFIED ALL OF THE TESTING REQUIREMENTS. AFTER EVALUATION, THIS EVENT WAS DETERMINED REPORTABLE BY THE SHIFT SUPERVISOR ON 03-01-88. THE CAUSE OF THE RELIEF VALVE FAILURE WAS PILOT STAGE LEAKAGE DUE TO STEAM CUTS ON THE PILOT DISC AND PILOT SEAT. THE VALVE WAS REPAIRED AND ON 03-03-88 WAS SUCCESSFULLY RETESTED. VERMONT YANKEE WILL CONTINUE TO MONITOR THE PERFORMANCE OF THE RELIEF VALVES BASED ON PAST PERFORMANCE. DURING THE TESTS OF THE MAIN STEAM (EIIS=SB) SAFETY VALVES, ONE OF THE VALVES, SERIAL NUMBER B11137, FAILED TO OBTAIN FULL TEST LIFT. THE VALVE APPARENTLY BOUND. UPON DISASSEMBLY AND INSPECTION NO EVIDENCE OF THE CAUSE OF THE BINDING WAS DISCOVERED. AFTER EVALUATION, THIS EVENT WAS DETERMINED REPORTABLE BY THE SHIFT SUPERVISOR ON 03-15-88.

[260] VERMONT YANKEE DOCKET 50-271 LER 88-003
MISSED SURVEILLANCE ON HIGH WATER LEVEL IN SCRAM DISCHARGE VOLUME TRIP CHANNEL AND ALARM DUE TO PROGRAMMATIC DEFICIENCIES.
EVENT DATE: 041288 REPORT DATE: 051088 NSSS: GE TYPE: BWR

(NSIC 209291) ON 4/12/88, DURING NORMAL OPERATION AT 100% POWER IT WAS DISCOVERED THAT FUNCTIONAL TESTING OF THE SCRAM DISCHARGE VOLUME HIGH WATER LEVEL TRIP (EIIS=JC) HAD NOT BEEN TESTED IN ACCORDANCE WITH TECH SPEC TABLE 4.1.1 REQUIREMENTS. TABLE 4.1.1, NOTE 1 REQUIRES THAT TESTING SHALL BE CONDUCTED MONTHLY INITIALLY AND BETWEEN ONE AND THREE MONTH INTERVALS THEREAFTER. A REVIEW OF PAST RECORDS HAS REVEALED THAT THIS TESTING WAS NOT DONE MONTHLY INITIALLY. THE ROOT CAUSE OF THIS EVENT IS THAT THE PROGRAMMATIC TRACKING PROGRAM IN PLACE AT THE TIME AMENDMENT 76 WAS ISSUED DID NOT REQUIRE PROCEDURAL REVIEWS FOR TECH SPECS AMENDMENTS PRIOR TO ISSUANCE. THE CORRECTIVE ACTION INITIATED IN LER 87-03 IS RESPONSIBLE FOR THE DISCOVERY OF THIS EVENT. THE PROGRAMMATIC TRACKING SYSTEM HAS BEEN REVISED AS DESCRIBED IN LER 87-03. THIS CORRECTIVE ACTION IS APPROPRIATE TO PRECLUDE FUTURE PROBLEMS OF THIS TYPE AND WILL CONTINUE TO REVIEW EXISTING PROCEDURAL COMPLIANCE WITH TECH SPECS.

[261] WATERFORD 3 DOCKET 50-382 LER 87-029
FAILURE TO MEET SAFETY TO NON-SAFETY ELECTRICAL DISTRIBUTION SEPARATION CRITERIA DUE TO COGNITIVE PERSONNEL ERROR.
EVENT DATE: 040287 REPORT DATE: 041888 NSSS: CE TYPE: PWR

(NSIC 208908) ON MARCH 22, 1988, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% POWER WHEN A REVIEW CONCLUDED THAT A PREVIOUS REPORTABILITY DETERMINATION WAS IN ERROR. ON APRIL 2, 1987, UTILITY ENGINEERS DISCOVERED THAT TWO NON-SAFETY EMERGENCY LIGHTING PANELS WERE CONNECTED THROUGH SINGLE CIRCUIT BREAKERS TO SAFETY POWER PANELS. IN ADDITION, IT WAS DISCOVERED THAT A NON-SAFETY TELEPHONE CABINET WAS CONNECTED TO A SAFETY POWER PANEL THROUGH A SINGLE CIRCUIT BREAKER. THESE CONFIGURATIONS DO NOT COMPLY WITH NRC REGULATORY GUIDE (RG) 1.75 AS COMMITTED TO IN THE FINAL SAFETY ANALYSIS REPORT (FSAR). THE ROOT CAUSE OF THESE EVENTS WAS COGNITIVE PERSONNEL ERROR IN DEVELOPING DESIGN CHANGES. THE LIGHTING PANEL CIRCUITS WERE ORIGINALLY CORRECT AND WERE LATER MODIFIED BY A DESIGN CHANGE. THE TELEPHONE CABINET INSTALLATION WAS PART OF A

DESIGN CHANGE TO EXPAND THE SYSTEM. BOTH DISCREPANCIES HAVE BEEN CORRECTED. A DESIGN CHANGE IS NOW COORDINATED THROUGH A UTILITY SYSTEM EXPERT AND MUST GO THROUGH AN EXTENSIVE TECHNICAL REVIEW PROCESS. THEREFORE, THERE IS A HIGH LEVEL OF CONFIDENCE THAT AN ERROR OF THIS TYPE WILL NOT BE REPEATED. A REVIEW OF ALL SAFETY TO NON-SAFETY BUS INTERFACES FOR COMPLIANCE WITH RG 1.75 IS SCHEDULED. SINCE THESE SINGLE CIRCUIT BREAKERS WERE WIRED PROPERLY AND OPERABLE, THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT.

[262] WATERFORD 3 DOCKET 50-382 LER 87-030
MISSED GAS DECAY TANK SAMPLE DUE TO PLUGGED SAMPLE INJECTION SYRINGE.
EVENT DATE: 081987 REPORT DATE: 041888 NSSS: CE TYPE: PWR

(NSIC 208909) AT 1330 HOURS ON 8/19/87, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 95% POWER WHEN OPERATIONS PERSONNEL DISCOVERED THAT THE 0900 HOURS GAS DECAY TANK (GDT) HYDROGEN AND OXYGEN SAMPLES HAD NOT BEEN ANALYZED WITHIN FOUR HOURS FROM SAMPLING (I.E, BY 1300 HOURS). ACTION REQUIREMENT 38 OF TECH SPEC TABLE 3.3-13 REQUIRES THE HYDROGEN SAMPLE TO BE ANALYZED WITHIN FOUR HOURS FROM SAMPLE ISOLATION. ACTION REQUIREMENT 40 OF THE SAME TS REQUIRES OXYGEN TO BE SAMPLED AND ANALYZED ONCE PER FOUR HOURS. THE ROOT CAUSE OF THIS EVENT WAS A PLUGGED SAMPLE INJECTION SYRINGE FOR THE GAS CHROMATOGRAPH. CONTRIBUTING TO THIS WAS A FAILURE IN COMMUNICATION BETWEEN CHEMISTRY AND OPERATIONS PERSONNEL WHICH PREVENTED SECURING THE WASTE GAS HOLDUP SYSTEM (WGHS) COMPRESSORS PRIOR TO 1300 HOURS TO PLACE THE PLANT IN A MODE IN WHICH THE SAMPLES WERE NOT REQUIRED. THE SYRINGE WAS REPAIRED. CHEMISTRY PERSONNEL WERE COUNSELED. A STATION MODIFICATION IS BEING IMPLEMENTED TO REPLACE THE INSTALLED AUTOMATIC SAMPLING SYSTEM. THE WGHS WAS SECURED AT 1330 HOURS AND SUBSEQUENT GDT SAMPLES SHOWED NORMAL AND EXPECTED RESULTS. THERE WAS, THEREFORE, NO SAFETY SIGNIFICANCE TO THIS EVENT. SINCE THE WGHS WAS SECURED WITHIN THE ALLOWABLE 25% EXTENSION OF THE SURVEILLANCE INTERVAL, THERE WAS NO CONDITION PROHIBITED BY TECH SPEC, AND THIS REPORT IS SUBMITTED VOLUNTARILY.

[263] WATERFORD 3 DOCKET 50-382 LER 87-031
MISSED VALVE STROKE TEST SURVEILLANCE DUE TO INADEQUATE ADMINISTRATIVE CONTROLS.
EVENT DATE: 102187 REPORT DATE: 041888 NSSS: CE TYPE: PWR

(NSIC 208910) AT 0800 HOURS ON 10/21/87, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% POWER WHEN OPERATIONS PERSONNEL DISCOVERED THAT CONTAINMENT ATMOSPHERE PURGE (CAP) ISOLATION VALVES 103 AND 205 HAD EXCEEDED THEIR REQUIRED STROKE-TIME SURVEILLANCE PERIODICITY BY 22 HOURS AND 47 DAYS RESPECTIVELY. BOTH VALVES SHARE A COMMON POWER SUPPLY WITH CAP-104, AND WERE DEENERGIZED WHEN CAP-104 WAS PLACED OUT-OF-SERVICE ON 6/23/87. WHEN THE ROUTINE SURVEILLANCE WAS PERFORMED ON 7/28/87, CAP-103 AND CAP-205 WERE THEREFORE NOT STROKED, BUT THE SURVEILLANCE WAS TRACKED AS COMPLETE. CAP-103 AND CAP-205 WERE THEREFORE NOT RETESTED WHEN CAP-104 WAS RETURNED TO SERVICE ON 9/3/87. THIS REPORT IS SUBMITTED VOLUNTARILY SINCE THE ACTION REQUIREMENTS OF TS 3.6.3 AND 3.6.1.7 WERE SATISFIED. THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE ADMINISTRATIVE CONTROLS. PROCEDURES HAVE BEEN REVISED TO ADEQUATELY ACCOUNT FOR RESCHEDULING OUT-OF-SERVICE COMPONENTS FOR TS SURVEILLANCE COMPLETION WHEN THEY ARE RETURNED TO SERVICE. SINCE CAP-103 AND CAP-205 SATISFACTORILY PASSED STROKE TESTS AND NO WORK HAD BEEN PERFORMED ON EITHER VALVE SINCE THE PREVIOUS TEST, THERE IS A HIGH LEVEL OF CONFIDENCE THAT THE VALVES WOULD HAVE PERFORMED THEIR SAFETY FUNCTIONS IF NECESSARY. THERE WAS, THEREFORE, NO SAFETY SIGNIFICANCE TO THIS EVENT.

[264] WATERFORD 3 DOCKET 50-382 LER 88-005
BLOWN UNDERVOLTAGE CIRCUIT FUSE REPLACEMENT RESULTS IN TECH SPEC 3.0.3 ENTRY.
EVENT DATE: 032188 REPORT DATE: 042088 NSSS: CE TYPE: PWR

(NSIC 209019) AT APPROXIMATELY 0900 HOURS ON MARCH 21, 1988, WATERFORD STEAM

ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% POWER WHEN A MAINTENANCE TECHNICIAN INADVERTENTLY ALLOWED TWO WIRE CONNECTIONS TO SHORT WHILE REPLACING A VOLTMETER SELECTOR SWITCH, CAUSING A FUSE OF THE 'B' TRAIN 4160V UNDERVOLTAGE (UV) RELAY COIL TO BLOW. REPLACEMENT OF THIS FUSE REQUIRES TEMPORARILY DISABLING ALL THREE CHANNELS OF UNDERVOLTAGE PROTECTION FOR THE 'B' TRAIN 4160V SAFETY BUS. SINCE THESE RELAYS ARE DELTA CONNECTED, THE BLOWN FUSE REDUCED THE UV ACTUATION TO A 1/1 RATHER THAN THE DESIGNED 3/3 LOGIC. SINCE OPERATION IN THIS MODE IS NOT DESIRABLE AND THE DESIGN OF THE INSTALLED TEST CIRCUITRY WOULD NOT ALLOW THE DAILY FUNCTIONAL TEST TO BE PERFORMED, THE RESIDENT INSPECTOR WAS BRIEFED AND TS 3.0.3 WAS ENTERED FOR SIX MINUTES WHILE THE FUSE WAS REPLACED. THE APPLICATION OF TS 3.3.2 ACTION REQUIREMENTS 12 AND 17 TO THIS CIRCUIT IS IMPRACTICAL SINCE REPAIRING ONE CHANNEL HAS THE EFFECT OF RENDERING ALL THREE CHANNELS INOPERABLE. A CHANGE TO THE TS IS BEING PURSUED TO CORRECT THIS DEFICIENCY. THIS CONDITION WOULD NOT PRECLUDE A DIESEL START DUE TO A SAFETY INJECTION ACTUATION SIGNAL (SIAS). DURING THIS EVENT THERE WAS A 290 MINUTE PERIOD WHEN A SINGLE FAILURE COULD HAVE CAUSED A SPURIOUS ESF ACTUATION AND SIX MINUTES WITHOUT UV PROTECTION ON ONE SAFETY BUS.

[265] WOLF CREEK 1 DOCKET 50-482 LER 88-005
ENGINEERED SAFETY FEATURES ACTUATION DUE TO TWO CONTROL ROOM VENTILATION ISOLATION SIGNALS CAUSED BY MALFUNCTIONS OF THE CHLORINE MONITORS.
EVENT DATE: 032688 REPORT DATE: 041488 NSSS: WE TYPE: PWR
VENDOR: M D A SCIENTIFIC, INC.

(NSIC 208979) ON MARCH 26, 1988 AT 1132 CST, AND ON APRIL 2, 1988, AT 1505 CST, CONTROL ROOM VENTILATION ISOLATION SIGNALS (CRVIS) OCCURRED DUE TO CHLORINE MONITOR GK-AITS-3 AND GK-AITS-2, RESPECTIVELY, INDICATING HIGH CHLORINE LEVEL IN THE OUTSIDE AIR MAKEUP TO THE CONTROL BUILDING HEATING, VENTILATING AND AIR CONDITIONING SYSTEM. NO CHLORINE WAS PRESENT IN EACH EVENT AS EVIDENCED BY NORMAL READINGS ON THE REDUNDANT CHLORINE MONITOR. EXAMINATION OF THE MONITOR AFTER THE MARCH 26 EVENT REVEALED THAT THE CHEMICALLY SENSITIVE PAPER TAPE USED TO DETECT CHLORINE HAD BROKEN. THIS CAUSED MORE LIGHT TO BE ABLE TO PASS THROUGH, RESULTING IN A CHANGE IN OPACITY READING SUFFICIENT FOR THE MONITOR TO INITIATE A CRVIS. THE PAPER TAPE WAS REPLACED. NO FURTHER PROBLEMS WITH THE MONITOR WERE FOUND AND IT WAS RETURNED TO OPERATION AT 1350 CST ON MARCH 26, 1988. EXAMINATION OF THE MONITOR AFTER THE APRIL 2 EVENT DETERMINED THAT THIS CRVIS WAS CAUSED BY A SPURIOUS SPIKE. THE CAUSE OF THE SPURIOUS SPIKE COULD NOT BE DETERMINED. AFTER TROUBLESHOOTING, THE MONITOR WAS REPLACED WITH A SPARE AND THE AFFECTED MONITOR TAKEN TO THE INSTRUMENTATION AND CONTROLS (I&C) SHOP FOR FURTHER TROUBLESHOOTING.

[266] WPPSS 2 DOCKET 50-397 LER 85-062
LEAK DETECTION INITIATED RWCU SYSTEM ISOLATION.
EVENT DATE: 121385 REPORT DATE: 010986 NSSS: GE TYPE: BWR

(NSIC 209189) A REACTOR WATER CLEANUP (RWCU) ISOLATION OCCURRED ON DECEMBER 14, 1985, DURING PERFORMANCE OF A SURVEILLANCE PROCEDURE. THE ISOLATION OCCURRED BECAUSE POWER TO THE CONTROL CIRCUITRY WAS INTERRUPTED WHILE INSTALLING A JUMPER WIRE. THE POWER INTERRUPTION RESULTED IN AN AUTO CLOSURE OF RWCU OUTBOARD CONTAINMENT ISOLATION VALVE (RWCU-V-4). THE ROOT CAUSE WAS COGNITIVE PERSONNEL ERROR DUE TO A TECHNICIAN NOT USING THE PROPER JUMPER SPECIFIED IN THE PROCEDURE. THE SYSTEM WAS RETURNED TO SERVICE AND THE SURVEILLANCE COMPLETED. TRAINING WILL BE CONDUCTED FOR ALL I&C TECHNICIANS AND AN ENGINEERING EVALUATION WILL BE COMPLETED TO DETERMINE CHANGES REQUIRED TO IMPROVE TESTABILITY OF THE SYSTEM.

[267] WPPSS 2 DOCKET 50-397 LER 88-008
PLANT TECHNICAL SPECIFICATION FIRE PENETRATION SEALS DISCOVERED IMPAIRED/UNSEALED DURING ONGOING 100% INSPECTION DUE TO UNKNOWN CAUSES.
EVENT DATE: 031888 REPORT DATE: 042288 NSSS: GE TYPE: BWR

(NSIC 209183) ON 3/23/88 AND 3/25/88 IT WAS DETERMINED THAT PROBLEMS DOCUMENTED (ON 3/18/88 AND 3/24/88) WITH REGARD TO 11 PLANT TECHNICAL SPECIFICATION FIRE PENETRATIONS WERE REPORTABLE UNDER 10CFR50.73 (A)(2)(I). THE PENETRATIONS WERE IMPAIRED SUCH THAT THEY WOULD NOT QUALIFY AS A THREE-HOUR FIRE BARRIER. THE PENETRATIONS ARE LOCATED IN THE DIESEL GENERATOR AND REACTOR BUILDINGS. THE PROBLEMS WERE IDENTIFIED DURING THE PERFORMANCE OF ONGOING PHYSICAL WALKDOWNS OF ALL ACCESSIBLE TECHNICAL SPECIFICATION WALLS AND PENETRATIONS AS FURTHER CORRECTIVE ACTION COMMITTED TO IN LER 87-030. AS REQUIRED BY THE PLANT TECHNICAL SPECIFICATIONS, THE PENETRATIONS WERE PLACED ON HOURLY FIRE TOUR. FURTHER CORRECTIVE ACTIONS INCLUDE 1) PREPARING MAINTENANCE WORK REQUESTS (MWRS) TO REPAIR/SEAL THE PENETRATIONS, AND 2) COMPLETING THE 100% REVIEW OF DOCUMENTATION AND PHYSICAL WALKDOWNS OF ALL ACCESSIBLE TECHNICAL SPECIFICATION WALLS AND PENETRATIONS. AT THE COMPLETION OF THE REVIEW AND WALKDOWNS, A SUPPLEMENTAL REPORT DESCRIBING ANY FURTHER DEFICIENCIES DISCOVERED WILL BE SUBMITTED. THE CAUSE OF THIS EVENT IS CURRENTLY UNDER INVESTIGATION. IT WAS DETERMINED THAT THIS EVENT DID NOT AFFECT THE HEALTH AND SAFETY OF EITHER THE PUBLIC OR PLANT PERSONNEL DUE TO THE FIRE PROTECTION DESIGN OF THE FIRE AREAS INVOLVED.

[268] WPPSS 2 DOCKET 50-397 LER 88-009
TWO STANDBY GAS TREATMENT SYSTEM TECHNICAL SPECIFICATION SURVEILLANCES NOT PERFORMED WITHIN TIME LIMITS DUE TO PERSONNEL ERROR.
EVENT DATE: 040188 REPORT DATE: 050288 NSSS: GE TYPE: BWR

(NSIC 209233) ON 4/1/88 IT WAS DISCOVERED THAT THE FOLLOWING 18-MONTH STANDBY GAS TREATMENT (SGT) SYSTEM SURVEILLANCE PROCEDURES (PPMS) HAD NOT BEEN COMPLETED WITHIN THE REQUIRED TIME FRAME PLUS 25% AS REQUIRED BY THE PLANT TECH SPECS: PPM 7.4.6.5.3.5, "STANDBY GAS TREATMENT SYSTEM HEPA DOP TEST AND VISUAL INSPECTION." AND PPM 7.4.6.5.3.6, "STANDBY GAS TREATMENT SYSTEM ADSORBER BYPASS LEAKAGE TEST." THE PROCEDURES WERE DUE TO BE PERFORMED ON 11/12/87 AND WERE OVERDUE 3/23/88 (PPM 7.4.6.5.3.6) AND 3/25/88 (PPM 7.4.6.5.3.5). THE PROCEDURE WAS SUCCESSFULLY COMPLETED ON 4/1/88. THE CAUSE OF THE EVENT IS PERSONNEL ERROR IN THAT A PLANT SENIOR HEALTH PHYSICIST FAILED TO ROUTE THE SURVEILLANCE MONITORING SYSTEM (SMS) COMPUTER TRACKING CARDS TO THE PLANT SYSTEM ENGINEER RESPONSIBLE FOR THE PERFORMANCE OF THE SURVEILLANCES. THE SMS CARDS SERVE AS A REMINDER TO PERFORM THE SURVEILLANCES. THE EFFECT WAS THAT AN SGT SYSTEM LCO WAS NOT MET IN THAT BOTH SGT TRAINS WERE TECHNICALLY (THOUGH NOT IN FACT) INOPERABLE. FURTHER CORRECTIVE ACTIONS INCLUDE COUNSELING THE SENIOR HEALTH PHYSICIST ON THE IMPORTANCE OF DISTRIBUTING SMS CARDS IN A PROMPT MANNER, COUNSELING PLANT HEALTH PHYSICS MANAGEMENT/SUPERVISION ON EFFECTIVE UTILIZATION OF SMS STATUS REPORTS, AND REVISING A PLANT PROCEDURE TO PROVIDE ADDITIONAL GUIDANCE ON ACTIONS TO BE TAKEN WHEN SURVEILLANCE LATE DATE HAS BEEN EXCEEDED.

[269] YANKEE ROWE DOCKET 50-029 LER 87-015 REV 01
UPDATE ON MAIN STEAM LINE PRESSURE SWITCHES INOPERABLE.
EVENT DATE: 120387 REPORT DATE: 042688 NSSS: WE TYPE: PWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)

(NSIC 209172) ON DECEMBER 3, 1987, WITH THE PLANT IN MODE 1 AT 100% POWER, FIVE OF THE TWELVE MAIN STEAM LINE (EIIS-SB) ISOLATION VALVE PRESSURE SWITCHES (IEEE-PS) WERE FOUND DURING ROUTINE MONTHLY SURVEILLANCE, TO BE OUT OF TECH SPEC TOLERANCE (TECH SPEC TABLE 3.3-2, ITEM 3.A). OF THE FIVE PRESSURE SWITCHES THAT FAILED THE SURVEILLANCE, THREE WERE ADJUSTED BACK TO THE REQUIRED TRIP SETPOINT. THE REMAINING TWO (MS-PS-13 AND MS-PS-31) WERE RELACED IN KIND BECAUSE OF THEIR FAILURE TO OPERATE PROPERLY. THE CAUSE OF THE OUT-OF-TOLERANCE SETPOINT SETTINGS

FOR THE THREE READJUSTED SWITCHES WAS, AT THE TIME OF THE EVENT, ATTRIBUTED TO SETPOINT DRIFT, AND THEREFORE LEFT IN SERVICE. THE CAUSE OF THE REMAINING TWO FAILURES WAS, AFTER INVESTIGATION BY THE MANUFACTURER, DETERMINED TO BE EXTRUSION OF THE POLYURETHANE DISC INTO THE AREA BETWEEN THE PISTON AND CYLINDER WALL OF THE PRESSURE SWITCH TRANSDUCER. A PLANT SHUTDOWN WAS INITIATED AT 1140 HOURS IN ACCORDANCE WITH TECH SPEC 3.0.3 AND TERMINATED AT 1145 HOURS UPON INSTALLATION OF A BYPASS JUMPER TO ACCOMPLISH THE TECH SPEC ACTION STATEMENT BY PLACING THE SWITCH IN THE TRIP POSITION. THE PLANT RETURNED TO 100% POWER AT 1210 HOURS ON THE SAME DAY. THE NRC WAS NOTIFIED VIA ENS AT 1340 HOURS DEC. 3, 1987.

[270] YANKEE ROWE DOCKET 50-029 LER 88-001
PROCEDURE INADEQUACY THAT COULD ALLOW INSUFFICIENT LOAD TESTING OF EMERGENCY GENERATORS.
EVENT DATE: 012688 REPORT DATE: 041588 NSSS: WE TYPE: PWR

(NSIC 208958) ON 3/14/88, AT 1530 HOURS, DURING NORMAL STEADY STATE OPERATION, OPERATIONAL MODE 1 AT 100 PERCENT POWER, AN EVALUATION OF A PLANT PROCEDURE DETERMINED THAT A REPORTABLE PROCEDURAL INADEQUACY EXISTED. THE ERROR WAS DISCOVERED ON 1/26/88, DURING THE BIENNIAL REVIEW PROCESS AND AN EVALUATION WAS REQUESTED. THE PROCEDURE WAS OP-4209, "EMERGENCY DIESEL GENERATOR TEST DURING REFUELING INTERVALS - EMERGENCY DIESEL NO ", WHICH IS USED TO FUNCTIONALLY TEST THE EMERGENCY DIESEL GENERATORS PER TECH SPEC 4.8.1.1.2.D.4. THE TECH SPEC REQUIRED THAT EACH OF THE EMERGENCY DIESEL GENERATORS OPERATE FOR GREATER THAN/ OR EQUAL TO 60 MINUTES WHILE LOADED TO GREATER THAN OR EQUAL TO 400 KW. THE PROCEDURE ALLOWED TESTING THE DIESEL TO 400 KW OR 500 KVA. IF THE SURVEILLANCE WAS CONDUCTED AT 500 KVA AND A POWER FACTOR OF LESS THAN 0.8 THE GENERATOR WOULD BE PRODUCING LESS THAN THE REQUIRED 400 KW. PURSUANT TO 10CFR50.72(B)(2)(III) A FOUR HOUR ENS NOTIFICATION WAS MADE AT 1910 HOURS. A PROCEDURE CHANGE NOTICE (PCN) WAS PREPARED TO CORRECT THE PROCEDURE, ON ALL THREE EMERGENCY DIESEL GENERATORS AT 2340 HOURS. THIS IS THE FIRST EVENT OF THIS NATURE AT THIS FACILITY. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO AN INADEQUATE PROCEDURE REVIEW.

[271] YANKEE ROWE DOCKET 50-029 LER 88-002
LOSS OF POWER TO NUCLEAR INSTRUMENT CABINET "A".
EVENT DATE: 032288 REPORT DATE: 042188 NSSS: WE TYPE: PWR
VENDOR: SOLA ELECTRIC COMPANY

(NSIC 209043) ON 3/22/88, AT 0042, WITH THE PLANT OPERATING IN MODE 1 AT 100% POWER, AN AUTOMATIC REACTOR SCRAM OCCURRED WHEN POWER TO THE SCRAM AMPLIFIERS IN THE NUCLEAR INSTRUMENT (NI) CABINET "A" WAS LOST. THE LOSS OF POWER TO NI CABINET "A" WAS A RESULT OF A FAILED SOLA TRANSFORMER (TYPE CV) WITHIN THE NI CABINET. AN ENS PHONE CALL WAS MADE AT 0141 ON 3/22/88 IN ACCORDANCE WITH 10 CFR 50.72(B)(2)(II). THE PLANT EMERGENCY DIESEL GENERATOR NO. 2 STARTED AS REQUIRED WITH THE EXPECTED LOSS OF POWER TO THE #1-2400 VOLT AND #4-480 VOLT BUSSES FOLLOWING THE TURBINE GENERATOR TRIP. THE ROOT CAUSE OF THIS EVENT IS THE FAILURE OF ONE OF TWO CAPACITORS IN THE SOLA TRANSFORMER. BOTH CAPACITORS WERE REPLACED IN KIND AND THE CABINET WAS SATISFACTORILY TESTED. THERE WAS NO ADVERSE AFFECT ON THE HEALTH OR SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT. THIS IS THE FIRST REPORT OF THIS NATURE. NO OTHER CORRECTIVE ACTIONS ARE DEEMED NECESSARY AT THIS TIME.

[272] YANKEE ROWE DOCKET 50-029 LER 88-003
REACTOR SCRAM/TURBINE TRIP ON LOW STEAM GENERATOR LEVEL.
EVENT DATE: 032688 REPORT DATE: 042588 NSSS: WE TYPE: PWR
VENDOR: FOXBORO CO., THE

(NSIC 209173) ON MARCH 26, 1988, WHILE AT 85% POWER AND RETURNING TO NORMAL FULL

POWER OPERATION, THERE WAS AN ALARM FOR SG HIGH-LOW LEVEL, IMMEDIATELY FOLLOWED BY AN AUTOMATIC REACTOR SCRAM AND TURBINE TRIP ON LOW SG LEVELS. JUST PRIOR TO THIS EVENT, LOOP NO. 2 FEEDWATER FLOW BEGAN OSCILLATING AND LOOP NO. 4 FEEDWATER FLOW STARTED DECREASING. AS A RESULT, THE LEVELS IN NO. 2 AND 4 STEAM GENERATORS WERE DECREASING RAPIDLY TOWARDS THE TRIP SETPOINT. THE REACTOR SCRAM AND TURBINE TRIP OCCURRED AT 0529 HOURS. THE NRC WAS NOTIFIED VIA ENS AT 0603 HOURS. AS PART OF THE EMERGENCY PROCEDURES, THE OPERATORS RESTARTED THE FEEDWATER PUMPS. ALL AUTOMATIC SAFETY SYSTEMS FUNCTIONED AS DESIGNED AND THE PLANT RESPONDED AS EXPECTED. THE ROOT CAUSE OF THIS EVENT IS THE FAILURE OF THE POWER SUPPLY FOR SG LEVEL CONTROL CHANNELS ON NO. 2 AND 4 STEAM GENERATORS. CORRECTIVE ACTION INVOLVED THE REPLACEMENT OF THE FAILED EQUIPMENT. NO OTHER ACTION IS DEEMED NECESSARY AT THIS TIME. AS A RESULT OF THIS EVENT, THERE WAS NO ADVERSE EFFECT TO THE PUBLIC HEALTH OR SAFETY. THIS IS THE FIRST OCCURRENCE OF THIS NATURE AT THIS FACILITY.

[273] YANKEE ROWE DOCKET 50-029 LER 88-004
INADEQUACY IN STEAM GENERATOR TUBE RUPTURE PROCEDURE.
EVENT DATE: 033188 REPORT DATE: 042988 NSSS: WE TYPE: PWR

(NSIC 209248) ON 3/31/88, WITH THE PLANT IN NORMAL STEADY STATE OPERATION (MODE 1, 100% POWER), LICENSED OPERATORS ATTENDING A REQUALIFICATION TRAINING CLASS NOTED A PROCEDURAL INADEQUACY IN OP-3107, "STEAM GENERATOR TUBE RUPTURE." THE PROCEDURE FAILED TO PROVIDE ADEQUATE INSTRUCTION FOR COMPLETE ISOLATION OF THE AFFECTED NO. 2 OR NO. 3 STEAM GENERATOR FOLLOWING A TUBE RUPTURE. SPECIFICALLY, THE PROCEDURE DID NOT PROVIDE FOR SHUTTING THE ISOLATION VALVES ON 1-1/2 INCH STEAM SUPPLY LINES FOR THE EMERGENCY FEEDWATER PUMP AND ATMOSPHERIC STEAM DUMP. IMMEDIATE ACTIONS WERE TAKEN TO CORRECT THE PROCEDURE. THE PROCEDURE INADEQUACY WAS DETERMINED TO BE REPORTABLE ON 4/1/88 AT 0930 HOURS BASED ON MANAGEMENT REVIEW OF THE REASONS FOR THE PROCEDURE CHANGE. NOTIFICATION VIA ENS WAS MADE ON 4/1/88 AT 1026 HOURS PURSUANT TO 10CFR50.72(B)(2)(II)(C). THE SYSTEM DESIGN BASIS WAS REVIEWED. THE REVISED PROCEDURAL STEPS WERE APPROPRIATE AND CONSISTENT WITH THE SYSTEM DESIGN. THIS EVENT IS BOUNDED BY THE ANALYSIS OF RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE; THEREFORE, THERE WAS NO ADVERSE AFFECT TO THE PUBLIC HEALTH OR SAFETY.

[274] YANKEE ROWE DOCKET 50-029 LER 88-005
DEGRADED FIRE DOORS.
EVENT DATE: 040188 REPORT DATE: 043088 NSSS: WE TYPE: PWR

(NSIC 209249) ON 4/1/88, AT 1055 HOURS, DURING NORMAL STEADY STATE OPERATION, (MODE 1 - 100 PERCENT POWER) AN EVALUATION OF SIX FIRE DOORS RESULTED IN THE DOORS BEING DECLARED DEGRADED. THE FIRE RATING DEGRADATION RESULTED WHEN THE DOORS WERE MODIFIED FOR ACCESS CONTROL IN 1975. THE MODIFICATION ALLOWED THE DOORS TO UNLATCH ON A LOSS OF ELECTRICAL POWER TO THE DOOR. THIS CONDITION COMPROMISED THE UNDERWRITERS LABORATORY FIRE RATING. THIS IS CONTRARY TO TECH SPEC 3.7.11 WHICH REQUIRES THAT FIRE BARRIERS BE FUNCTIONAL AT ALL TIMES OR THAT A CONTINUOUS FIRE WATCH BE STATIONED WITHIN ONE HOUR. A CONTINUOUS FIRE WATCH WAS STATIONED IN THE AFFECTED AREAS AT 1120 HOURS. THE HARDWARE WAS CHANGED, THE DOORS TESTED IN A LOSS OF POWER CONDITION, AND THE FIRE WATCHES REMOVED AT 1315 HOURS ON 4/2/88. A PREVIOUS FIRE BARRIER DEGRADATION WAS REPORTED AS LER 85-05. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO AN INADEQUATE REVIEW OF THE POSSIBLE DOOR FAILURE MODES WHILE PREPARING OUR FIRE HAZARDS ANALYSIS. TO CORRECT THE DEGRADATION THE DOORS WERE MODIFIED TO PROVIDE A POSITIVE LATCH DURING LOSS OF POWER CONDITIONS. THERE WAS NO ADVERSE EFFECT ON THE PUBLIC HEALTH OR SAFETY AS A RESULT OF THIS EVENT.

[275] ZION 1 DOCKET 50-295 LER 88-003 REV 01
UPDATE ON QUARTERLY COMPOSITE SAMPLE UNABLE TO MEET LOWER LIMIT OF DETECTION REQUIREMENTS.
EVENT DATE: 030288 REPORT DATE: 033188 NSSS: WE TYPE: PWR

(NSIC 209251) FOR THE FIRST, SECOND, AND THIRD QUARTERS OF 1987, ZION STATION SHIPPED QUARTERLY COMPOSITE SAMPLES FOR THE LAKE DISCHARGE AND FIRE SUMP TO AN OFFSITE LABORATORY FOR ISOTOPIC ANALYSIS AS REQUIRED BY TECH SPEC 4.11-1. FOR ONE OF THE ISOTOPES, STRONTIUM 89 (SR-89), THE SAMPLES DID NOT MEET THE LOWER LIMIT OF DETECTION (LLD) REQUIRED BY THE TECH SPECS. THIS WAS CAUSED BY UNEXPECTED DELAYS IN THE SHIPMENT, DUE TO A CHANGE IN THE OFFSITE LABORATORY BEING USED. THE DELAY ALLOWED SUFFICIENT RADIOACTIVE DECAY OF SR-89 SO THAT THE SR-89 LEFT IN THE SAMPLE COULD NOT MEET THE LLD REQUIRED. THE STATION RECEIVED PRELIMINARY NOTIFICATION OF THIS ON 1/26/88, AND RECEIVED CONFIRMATION ON 3/02/88. THE UNIT WAS AT FULL POWER DURING MOST OF THIS PERIOD. THERE WAS NO SAFETY SIGNIFICANCE DUE TO THE LOW LEVELS OF SR-89 PRESENT. THE LABORATORY LLD VALUE FOR THE SR-89 WAS USED IN OFFSITE DOSE CALCULATION, WHICH RESULTED IN LEVELS WELL BELOW APPLICABLE LIMITS. THE CHEMISTRY PROCEDURE WILL BE CHANGED TO INCREASE SAMPLE VOLUME, WHICH WILL ALLOW THE OFFSITE LAB TO MEET LLD REQUIREMENTS.

[276] ZION 1 DOCKET 50-295 LER 88-006
SERVICE WATER PUMP FLOW RATE BELOW REQUIRED DILUTION FLOW DURING A RADIOACTIVE LIQUID WASTE RELEASE.
EVENT DATE: 032588 REPORT DATE: 042288 NSSS: WE TYPE: PWR

(NSIC 209169) 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 RELAXING THE 44,000 GPM REQUIREMENT WILL BE PROPOSED. IN THE INTERIM, A PROCEDURE CHANGE HAS BEEN MADE TO THE ZION CHEMISTRY PROCEDURES TO PREVENT RECURRENCE.

[277] ZION 1 DOCKET 50-295 LER 88-007
FAILURE OF HYDRAULIC STEAM GENERATOR SNUBBERS DUE TO CONTAMINATED HYDRAULIC FLUID.
EVENT DATE: 032688 REPORT DATE: 042588 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 2 (PWR)
VENDOR: BERGEN-PATTERSON PIPE SUPPORT CORPORATION

(NSIC 209252) WITH UNIT 1 IN REFUELING AND UNIT 2 AT POWER, UNIT 1 STEAM GENERATOR (S/G) HYDRAULIC SNUBBER 1DS13 FAILED IN PLACE AS FOUND BLEED RATE TESTING REQUIRED BY TECH SPECS. THE FAILURE WAS CONFIRMED BY BENCH TESTING ON 3/25/88. TWO OTHER S/G SNUBBERS WERE TESTED PER TECH SPECS ON 3/26/88 AND ALSO FAILED. HYDRAULIC FLUID DRAWN FROM THE FAILED SNUBBERS SHOWED SIGNS OF CONTAMINATION AND METAL FLAKES. ALL 16 S/G SNUBBERS WERE TESTED BY 4/01/88, WITH A TOTAL OF 5 FAILURES. FOLLOWING THE TESTING, ALL SNUBBERS WERE FLUSHED AND THE HYDRAULIC FLUID WAS REPLACED. ADDITIONALLY, SNUBBER BLEED FLOW CONTROL VALVES WERE REPLACED WITH VALVES HAVING A LARGER ORIFICE. ALL SNUBBERS PASSED AS LEFT TESTING. STRUCTURAL ANALYSES PERFORMED ON THE AFFECTED PIPING AND SUPPORT STRUCTURES ASSUMING SNUBBERS IN THE LOCKED CONDITION DURING A POSTULATED ACCIDENT

SHOWED THAT SYSTEM INTEGRITY WAS NOT AFFECTED BY THE SNUBBER FAILURES. ROOT CAUSE OF THE FAILURES WAS HYDRAULIC FLUID CONTAMINATION, PARTICULARLY THE METALLIC SHAVINGS. THESE ARE ATTRIBUTED TO INADEQUATE FLUSHING FOLLOWING SNUBBER REBUILDING DURING THE PREVIOUS OUTAGE. THE SNUBBER OVERHAUL PROCEDURE WILL BE REVISED TO INCLUDE IMPROVED FLUSHING PROCEDURES. THIS REPORT IS BEING SUBMITTED BECAUSE OF THE LARGE NUMBER OF FAILURES AND THE APPARENT COMMON CAUSE.

[278] ZION 2 DOCKET 50-304 LER 88-002
UNIT 2 STEAM GENERATOR SAFETIES INOPERABLE.
EVENT DATE: 031288 REPORT DATE: 041188 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 1 (PWR)
VENDOR: CROSBY VALVE

(NSIC 208960) DURING A REVIEW OF THE MAINTENANCE RECORDS FOR THE MAIN STEAM SAFETY VALVE (MSSV) OVERHAULS, IT WAS DISCOVERED THAT THE AS LEFT SETPOINT FOR 3 MSSV'S EXCEEDED THE 1% TOLERANCE SPECIFIED IN THE TECH SPECS TABLE 4.7-1. THE AFFECTED MSSV'S WERE DECLARED INOPERABLE ON MARCH 12, 1988, AND THE UNIT WAS RAMPED DOWN FROM FULL POWER TO APPROXIMATELY 65%. ON MARCH 19, 1988, THE INOPERABLE MSSV'S WERE RETESTED AND THEIR SETPOINTS RESET TO WITHIN 1% OF THE SETPOINT STAMPED ON THE VALVE. THE VALVES WERE DECLARED OPERABLE AND THE UNIT WAS RETURNED TO FULL POWER OPERATIONS. THE CAUSE OF THE EVENT WAS THE MAINTENANCE PROCEDURE, P/M003=5N, WHICH STATES "FINAL SET PRESSURE TOLERANCE SHALL BE +/- 2%". THIS CAME FROM AN INTERPRETATION OF ASME SECTION XI. A REVIEW OF THE FSAR SHOWED THAT FOR THE MOST LIMITING ACCIDENT THAT DEMANDS THE MSSV OPERATION, DESIGN BASIS MSSV FLOW CAPACITY WAS MAINTAINED AT ALL TIMES. THE PROCEDURE WAS CHANGED TO REFLECT TECH SPECS REQUIREMENTS.

[279] ZION 2 DOCKET 50-304 LER 88-003
INITIATION OF "PHASE A" CONTAINMENT ISOLATION DURING SAFEGUARDS TESTING DUE TO OPERATOR ERROR.
EVENT DATE: 032988 REPORT DATE: 042888 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 209176) ON MARCH 29, 1988, ZION UNIT 2 WAS AT 99% POWER AND SAFEGUARDS ACTUATION TEST PT-10 WAS IN PROGRESS. AT 08:45, AN UNPLANNED CONTAINMENT PHASE A ISOLATION OCCURRED. APPROX. 5 MINUTES LATER, 2A CONTROL ROD DRIVE MOTOR (CRDM) VENT FAN TRIPPED OFF, WITH 2B CRDM VENT FAN ALREADY OUT OF SERVICE. AN IMMEDIATE RAMP DOWN OF THE UNIT WAS BEGUN. BOTH PRESSURIZER RELIEF VALVES LIFTED AND RESEATED DURING THE RAMP DOWN SINCE NORMAL PRESSURIZER SPRAY WAS NOT AVAILABLE DUE TO THE PHASE A ISOLATION. ATTEMPTS TO RESTART THE CRDM VENT FAN WERE UNSUCCESSFUL. BY 09:34 THE VENT FAN WAS RESTARTED, THE CONTAINMENT ISOLATION WAS CLEARED, AND THE RAMP DOWN WAS STOPPED AT 47% POWER. THE SECTION OF PT-10 WAS REPEATED 3 TIMES WITHOUT INCIDENT. THE CONTAINMENT ISOLATION WAS CAUSED WHEN THE OPERATOR PRESSED THE WRONG TEST BUTTON DURING PT-10. THE VENT FAN FAILURE WAS UNRELATED TO THE CONTAINMENT ISOLATION. ITS CAUSE IS NOT KNOWN. THE DIFFICULTY IN RESTARTING THE VENT FAN WAS DUE TO A DEFECTIVE CONTROL RELAY. PLANT SAFETY WAS UNAFFECTED BECAUSE THERE WAS NO COMPONENT FAILURE, THE PLANT WAS SUCCESSFULLY STABILIZED AT 47% POWER, AND BECAUSE THE CRDM VENT FANS ARE NOT SYSTEMS REQUIRED FOR SAFE SHUTDOWN. TO PREVENT RECURRENCE, PLACEMENT OF COVERS OVER THE INVOLVED TEST SWITCHES IS BEING EVALUATED.

COMPONENT INDEX

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

COMPONENT INDEX

COMPONENT INDEX

SYSTEM INDEX

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

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KEYWORD INDEX

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

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SEE INSTRUCTIONS ON THE REVERSE

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11a. TYPE OF REPORT
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12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

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

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