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Licensee Event Report (LER) Compilation

For month of December 1990

Oak Ridge National Laboratory

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

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Abstract

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

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

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[1] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 90-010
INOPERABLE CONTROL ROOM HABITABILITY SYSTEMS DUE TO LEAKAGE IN AIR SUPPLY SYSTEM TO CONTROL ROOM ISOLATION DAMPERS CAUSED BY FAILURE TO ADEQUATELY TEST AND MAINTAIN SYSTEM FOLLOWING MODIFICATION.
EVENT DATE: 092190 REPORT DATE: 102290 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 2 (PWR)

(NSIC 219894) ON 9/15/90, IT WAS DETERMINED THAT 4 CHECK VALVES LOCATED IN AIR SUPPLY SYSTEM TO 2 DAMPERS USED FOR ISOLATION OF THE ANO-1 CONTROL ROOM NORMAL VENTILATION SYSTEM HAD NOT BEEN PREVIOUSLY TESTED. DAMPERS ARE PNEUMATIC VALVES REQUIRING AIR PRESSURE FOR INFLATION. AIR SUPPLY IS PROVIDED BY NON-SAFETY RELATED INSTRUMENT AIR (IA) SYSTEM AND 2 REDUNDANT RESERVE AIR ACCUMULATORS WHICH SERVE AS A BACKUP RESERVOIR OF CLOSING AIR. CHECK VALVES FUNCTION TO PREVENT A LOSS OF ACCUMULATOR AIR PRESSURE DUE TO BACKLEAKAGE SHOULD IA SYSTEM BECOME UNAVAILABLE. ON 9/21/90, TESTING OF AIR SUPPLY SYSTEM REVEALED SEVERAL AREAS OF DEGRADED SYSTEM INTEGRITY INCLUDING CHECK VALVE LEAKAGE. IMMEDIATE CORRECTIVE ACTIONS INCLUDED REPAIRS OF LEAKAGE AT PIPING JOINTS AND THREADED CONNECTIONS. THE CHECK VALVES WERE REPLACED AND THE SYSTEM WAS MODIFIED BY ADDING A MANUAL ISOLATION VALVE IN EACH LINE TO PROVIDE ISOLATION OF THE ACCUMULATORS FROM THE IA SYSTEM. PROCEDURE CHANGES WERE IMPLEMENTED REQUIRING MAINTAINING THE MANUAL VALVES IN A CLOSED POSITION EXCEPT FOR PERIODIC CYCLING FOR RECHARGING THE ACCUMULATORS. ROOT CAUSE OF THIS CONDITION WAS DETERMINED TO BE AN INADEQUATE DESIGN CHANGE PROCESS THAT WAS IN PLACE IN 1978 WHEN SYSTEM WAS MODIFIED TO ITS CURRENT CONFIGURATION.

[2] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 90-012
DEFECTS IN ASME III CLASS 1 STAINLESS STEEL PIPE DISCOVERED DURING PREFABRICATION WELDING WHICH IF INSTALLED COULD HAVE RESULTED IN FAILURE OF THE HIGH PRESSURE INJECTION SYSTEM DURING OPERATION.
EVENT DATE: 092990 REPORT DATE: 102990 NSSS: BW TYPE: PWR
VENDOR: SANDVIK SPECIAL METALS CORP.

(NSIC 219896) ON 9/29/90, WHILE PREPARING TO MAKE A PREFABRICATION WELD ON A CLASS 1 STAINLESS STEEL PIPE TO BE USED IN A HIGH PRESSURE INJECTION SYSTEM MODIFICATION, A WELDER VISUALLY IDENTIFIED AN APPARENT DEFECT IN A SECTION OF 2 1/2 INCH PIPE. THE PIPING WAS ULTRASONICALLY TESTED AND THREE SIGNIFICANT DEFECTS RANGING FROM 1/4 TO 1/3 WALL THICKNESS WERE DETECTED. ALL OF THE SUSPECT PIPE (MANUFACTURED BY SANDVIK) WAS PUT ON QUALITY CONTROL HOLD PENDING AN ENGINEERING DETERMINATION OF ITS ACCEPTABILITY. ON 10/3/90, IT WAS DETERMINED THAT THE DEFECTS RENDERED THE PIPING UNSUITABLE FOR USE AT ANO. REPLACEMENT PIPE MANUFACTURED BY COMBUSTION ENGINEERING WAS PURCHASED. THE ROOT CAUSE OF THIS EVENT LIES WITH THE VENDOR AND/OR ITS SERVICE SUPPLIERS IN THAT THE NONCONFORMING PIPE WAS SUPPLIED TO ANO WITH DOCUMENTATION INDICATING THAT IT WAS ASME III CLASS 1 PIPING. THE SUPPLIER DELIVERED NONCONFORMING PIPE TO ANO WITH DOCUMENTATION INDICATING THAT IT WAS ASME III CLASS 1. ANO HAS REMOVED RADNOR ALLOYS FROM ITS QUALIFIED VENDORS LIST PENDING AN INVESTIGATION BY RADNOR OF THIS EVENT. ANO WILL VISIT RADNOR ALLOYS TO VERIFY THE IMPLEMENTATION OF CORRECTIVE ACTIONS ADEQUATE TO PREVENT RECURRENCE OF SIMILAR EVENTS. THIS LER ALSO SATISFIES THE REPORTABILITY REQUIREMENTS OF 10CFR21.

[3] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 90-011
INADVERTENT ACTUATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM INITIATED BY A TRIP OF A CHLORINE MONITOR MOST LIKELY CAUSED BY RADIO FREQUENCY INTERFERENCE.
EVENT DATE: 093090 REPORT DATE: 103090 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 2 (PWR)

(NSIC 219895) ON 9/30/90, AT APPROX. 0050, AN UNEXPECTED ACTUATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) OCCURRED. INVESTIGATION INTO THE CAUSE OF THE ACTUATION REVEALED THAT CHLORINE MONITOR 2CLS-8762-2 WAS TRIPPED. HOWEVER, IMMEDIATE CAUSE OF THE MONITOR TRIP COULD NOT BE POSITIVELY DETERMINED. SINCE NO ACTUAL HIGH CHLORINE CONDITION EXISTED, THE MONITOR WAS RESET AND THE CONTROL ROOM VENTILATION LINEUP WAS RETURNED TO NORMAL AT 0058 HOURS. THE MOST LIKELY CAUSE OF THE ACTUATION WAS RADIO FREQUENCY INTERFERENCE (RFI) CAUSED BY

THE KEYING OF A HAND HELD RADIO IN THE VICINITY OF THE MONITOR. ROOT CAUSE OF THIS EVENT IS DIRECTLY RELATED TO SYSTEM DESIGN. EXTREME SENSITIVITY OF THE CHLORINE MONITORS COUPLED WITH THE ACTUATION LOGIC CONFIGURATION, WHICH REQUIRES ONLY ONE MONITOR TO TRIP IN ORDER TO INITIATE THE CREVS, MAKES THE SYSTEM HIGHLY SUSCEPTIBLE TO INADVERTENT ACTUATIONS. ACTION WAS INITIATED TO BETTER MARK AREAS IN THE PLANT WERE RADIO USAGE IS PROHIBITED. ADDITIONALLY, SINCE THE USE OF CHLORINE AS A BIOCIDE IN THE SERVICE WATER SYSTEM IS BEING DISCONTINUED AND REPLACED WITH AN ALTERNATE METHOD, A CHANGE WILL BE PURSUED TO REMOVE THE CHLORINE MONITORS FROM THE TECH SPECS, AND IF APPROVED, A PLANT CHANGE WILL BE IMPLEMENTED TO REMOVE THE MONITORS FROM THE CREVS ACTUATION LOGIC.

[4] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 90-021
PRESSURIZER AND STEAM GENERATOR LOW PRESSURE VARIABLE SETPOINT TRIPS NOT CALIBRATED IN ACCORDANCE WITH TECHNICAL SPECIFICATION REQUIREMENTS DUE TO INADEQUATE PROCEDURES.
EVENT DATE: 092790 REPORT DATE: 102690 NSSS: CE TYPE: PWR

(NSIC 219913) ON 9/27/90, DURING A REVIEW OF SURVEILLANCE PROCEDURES IT WAS DISCOVERED THAT SETPOINT TOLERANCES SPECIFIED IN THE PLANT PROTECTION SYSTEM (PPS) CALIBRATION AND MONTHLY TESTING PROCEDURES WOULD ALLOW THE PRESSURIZER PRESSURE-LOW VARIABLE SETPOINT (VSP) CARDS AND THE STEAM GENERATOR PRESSURE-LOW VSP CARDS TO BE SET SUCH THAT TECH SPEC (TS) REQUIREMENTS COULD BE EXCEEDED DURING A PLANT COOLDOWN AND DEPRESSURIZATION. PROCEDURES WERE CHANGED TO SPECIFY NEW TOLERANCES AND CALIBRATION CHECKS WERE PERFORMED ON ALL FOUR PPS CHANNELS. SOME OF THE VALUES FOR THESE PARAMETERS WERE FOUND TO BE SLIGHTLY (APPROXIMATELY 1 PSIA) GREATER THAN ALLOWED BY THE TS. THE TRIP SETPOINTS WERE CHANGED WHILE PERFORMING THE CALIBRATION PROCEDURES. THE ROOT CAUSE WAS DETERMINED TO BE PERSONNEL ERRORS RESULTING IN THE PROCEDURAL DEFICIENCIES. A REVIEW OF OTHER ANO-1 AND ANO-2 PROCEDURES USED TO ADJUST AND VERIFY TRIP SETPOINTS WILL BE PERFORMED. THERE WAS NO SAFETY SIGNIFICANCE RELATED TO THIS EVENT.

[5] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 90-020
FAILURE OF MOTOR OPERATED VALVE ON MAIN CONDENSER CIRCULATING WATER PUMP TO CLOSE RESULTS IN LOSS OF VACUUM AND SUBSEQUENT MANUALLY INITIATED REACTOR TRIP.
EVENT DATE: 092890 REPORT DATE: 102690 NSSS: CE TYPE: PWR
VENDOR: LIMITORQUE CORP.

(NSIC 219912) AT 2142 HOURS ON 9/28/90, DURING A PLANNED POWER REDUCTION FROM FULL POWER, MAIN CONDENSER CIRCULATING WATER PUMP 2P-3B WAS SECURED. THE PUMP DISCHARGE VALVE FAILED TO AUTOMATICALLY CLOSE ALLOWING A FLOWPATH FOR CIRCULATING WATER FLOW TO BYPASS THE MAIN CONDENSER. AT 2143 HOURS AN AUTOMATIC MAIN TURBINE TRIP ON HIGH CONDENSER PRESSURE OCCURRED AND CONTROL ROOM PERSONNEL MANUALLY TRIPPED THE REACTOR IN ANTICIPATION OF AN AUTOMATIC REACTOR TRIP. THE EMERGENCY FEEDWATER SYSTEM, ACTUATED AUTOMATICALLY AND WAS USED TO RESTORE AND MAINTAIN NORMAL STEAM GENERATOR WATER LEVELS. THE PLANT WAS SUBSEQUENTLY STABILIZED IN MODE 3 (HOT STANDBY) CONDITIONS. INVESTIGATIONS REVEALED THAT THE VALVE FAILED TO CLOSE DUE TO A MECHANICAL KEY WHICH DISENGAGED FROM THE MOTOR SHAFT ALLOWING THE MOTOR PINION GEAR TO TURN FREELY ON THE SHAFT. VIBRATION CAUSED A SETSCREW USED TO SECURE THE KEY TO LOOSEN. THE ROOT CAUSE WAS DETERMINED TO BE INADEQUATE WORK INSTRUCTIONS LEADING TO THE PREVIOUS INSTALLATION OF A SETSCREW THAT WAS TOO SMALL TO ALLOW PROPER LOCKWIRING. THE PROCEDURE FOR MOTOR PINION GEAR INSTALLATION HAS BEEN CHANGED TO INCLUDE DETAILED INSTRUCTIONS ON MOTOR PINION GEAR INSTALLATION. HOWEVER, THE PROCEDURE WILL BE EVALUATED TO DETERMINE IF ADDITIONAL GUIDANCE CONCERNING THE SELECTION OF SETSCREWS IS WARRANTED.

[6] ARNOLD DOCKET 50-331 LER 90-015
MANUAL SCRAM FOLLOWING LOSS OF AIR SYSTEM PRESSURE DUE TO POORLY SOLDERED JOINT.
EVENT DATE: 091390 REPORT DATE: 101290 NSSS: GE TYPE: BWR

(NSIC 219740) ON SEPTEMBER 13, 1990, WITH THE REACTOR AT APPROXIMATELY 37% POWER, OPERATIONS PERSONNEL MANUALLY SCRAMMED THE REACTOR (REACTOR PROTECTION SYSTEM INITIATION) WHEN RAPIDLY DECREASING INSTRUMENT AIR PRESSURE RESULTED IN REACTOR VESSEL LEVEL CONTROL DIFFICULTIES. THE FEEDWATER REGULATING VALVES, WHICH

RECEIVE THEIR MOTIVE POWER FROM INSTRUMENT AIR, "LOCKED UP" IN THEIR CURRENT POSITION, RESULTING IN A GRADUAL INCREASING TREND IN REACTOR VESSEL LEVEL. PRIMARY CONTAINMENT ISOLATION SYSTEM GROUPS TWO THROUGH FIVE INITIATED AS DESIGNED ON REDUCED LEVEL FOLLOWING THE SCRAM. THE PLANT WAS RETURNED TO A STABLE CONDITION WITH NO FURTHER PROBLEMS. THE CAUSE OF THE INSTRUMENT AIR PRESSURE LOSS WAS THE FAILURE OF A THREE-INCH SOLDERED COPPER FITTING JOINT AT AN AIR DRYER INLET. EXAMINATION OF THE JOINT FOUND INADEQUATE SOLDER COVERAGE. LARGER JOINTS ARE DIFFICULT TO SOLDER, AND CODE-REQUIRED VISUAL EXAMINATION AND IN-SERVICE TESTING MAY NOT DETECT REDUCED SOLDER COVERAGE. AS CORRECTIVE ACTIONS, ALL OTHER AIR SYSTEM JOINTS SOLDERED IN THE SAME TIME PERIOD WERE INSPECTED AND REPAIRED AS NECESSARY, AND A SAMPLING INSPECTION OF OTHER SOLDERED JOINTS IS ONGOING. INSPECTION CRITERIA FOR CERTAIN DIFFICULT JOINTS IS BEING MODIFIED TO INCLUDE ULTRASONIC TESTING.

[7] ARNOLD DOCKET 50-331 LER 90-017
REACTOR WATER CLEANUP SYSTEM ISOLATION AND STANDBY FILTER UNIT INITIATION DUE TO INADVERTENT LOSS OF THE 'B' INSTRUMENT AC SYSTEM BUS.
EVENT DATE: 092090 REPORT DATE: 101790 NSSS: GE TYPE: BWR
VENDOR: ELGAR, CORP.

(NSIC 219873) ON SEPTEMBER 20, 1990, WITH THE PLANT SHUTDOWN, POWER WAS LOST TO THE 'B' INSTRUMENT AC SYSTEM BUS. THIS RESULTED IN A PRIMARY CONTAINMENT ISOLATION SYSTEM GROUP V ISOLATION OF THE REACTOR WATER CLEANUP SYSTEM AND ALSO INITIATED OPERATION OF THE 'B' CONTROL BUILDING VENTILATION STANDBY FILTER UNIT. THE LOSS OF THE BUS WAS DUE TO THE 'B' INSTRUMENT AC INVERTER TRANSFER OVERRIDE SWITCH INADVERTENTLY SELECTED TO PREVENT THE AUTO-TRANSFER OF BUS LOAD TO THE STANDBY 'B' INSTRUMENT AC REGULATING TRANSFORMER. THE ROOT CAUSE FOR THE OVERRIDE OF THE AUTO-TRANSFER IS UNKNOWN. ACTIONS WERE TAKEN TO RESTORE THE AFFECTED SYSTEMS. SUBSEQUENT CORRECTIVE ACTION REMOVED THE TRANSFER OVERRIDE SWITCH TO PREVENT RECURRENCE OF THIS EVENT.

[8] BEAVER VALLEY 1 DOCKET 50-334 LER 90-015
ESF ACTUATION DUE TO CLOSURE OF MAIN STEAM TRIP VALVE DURING PARTIAL STROKE TESTING.
EVENT DATE: 100690 REPORT DATE: 110590 NSSS: WE TYPE: PWR

(NSIC 219874) ON 10/06/90, WITH THE UNIT IN POWER OPERATION (OPERATING MODE 1) AT 27% REACTOR POWER, OPERATIONS PERSONNEL WERE PERFORMING PARTIAL STROKE TESTS OF THE MAIN STEAM ISOLATION VALVES (MSIV,) DURING THE PERFORMANCE OF THE SURVEILLANCE TEST ON THE LOOP B MSIV, TV-MS-101B, THE VALVE DID NOT STOP AT THE 3-DEGREE CLOSED POSITION AS EXPECTED. THE VALVE CONTINUED TO TRAVEL FULLY CLOSED. THIS CAUSED THE TWO REMAINING STEAM GENERATORS, LOOPS A AND C, TO ASSUME THE TURBINE LOAD, WITH THE B STEAM GENERATOR PRESSURE RISING TO 1050 PSIG. THE LOOP B ATMOSPHERIC STEAM DUMP VALVE OPENED AS A RESULT OF THE PRESSURE INCREASE IN THE B STEAM GENERATOR. CONTROL ROOM PERSONNEL COMMENCED AN EMERGENCY SHUTDOWN IN ACCORDANCE WITH ABNORMAL OPERATING PROCEDURE (AOP) 1.51 AT 0440 HOURS. AT 0445 HOURS, THE LOOP B REACTOR COOLANT PUMP WAS SHUTDOWN. THE MAIN TURBINE WAS TRIPPED AT 0451 HOURS, AND THE REACTOR WAS TAKEN SUBCRITICAL AT 0455 HOURS. INSTRUMENT AND CONTROL (I&C) PERSONNEL INVESTIGATED, BUT NO CAUSE FOR THE VALVE CLOSURE COULD BE DETERMINED. I&C REPLACED THE RUPTURE DISKS ON TV-MS-101B. PROPER LIMIT SWITCH OPERATION WAS ALSO VERIFIED. THERE WERE NO SAFETY IMPLICATIONS AS A RESULT OF THIS PROGRESS AT THE TIME OF THE EVENT. A CONTROLLED SHUTDOWN WAS A EVENT AND PLANT CONDITIONS ALLOWED THE REMOVAL OF ONE PRIMARY SECONDARY LOOP IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS.

[9] BEAVER VALLEY 2 DOCKET 50-412 LER 90-013
CONTAINMENT PURGE ISOLATION DURING REACTOR CAVITY FILL.
EVENT DATE: 091590 REPORT DATE: 101590 NSSS: WE TYPE: PWR

(NSIC 219928) ON 9/15/90, THE REACTOR CAVITY WAS FILLED FOR REFUELING. AT 0717 HOURS, THE CONTAINMENT VENTILATION RADIATION MONITOR ALARMED HIGH, INITIATING AN AUTOMATIC CONTAINMENT PURGE ISOLATION. TRAIN B CONTAINMENT PURGE ISOLATION DAMPERS CLOSED. THE A CONTAINMENT VENTILATION RADIATION MONITOR INDICATED NO

SIGNIFICANT INCREASE IN ACTIVITY AT THIS TIME. A LOCAL EVACUATION OF THE AFFECTED AREAS WAS INITIATED AS A PRECAUTIONARY MEASURE. NO PERSONNEL CONTAMINATION OCCURRED. RADIATION CONTROL TECHNICIANS INSPECTED THE B MONITOR AND IDENTIFIED THAT ITS DETECTOR HAD BECOME CONTAMINATED. THE DETECTOR WAS CLEANED AND THE MONITOR RETURNED TO SERVICE AT 1812 HOURS. THE CONTAMINATION OCCURRED DURING AN INCREASE IN CONTAINMENT AIRBORNE ACTIVITY RESULTING FROM WATER TURBULENCE DURING THE REACTOR CAVITY FILL. INDICATION FROM THE UNCONTAMINATED A MONITOR SHOWED THE MAXIMUM AIRBORNE ACTIVITY INCREASE WAS 17% ABOVE BACKGROUND. THE HIGH ALARM SETPOINT WAS 300% ABOVE BACKGROUND. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. ISOLATION OF CONTAINMENT VENTILATION IN RESPONSE TO A HIGH RADIATION SIGNAL IS DESCRIBED IN UNIT 2 UFSAR SECTION 9.4.7.3, "CONTAINMENT PURGE AIR SYSTEM".

[10] BEAVER VALLEY 2 DOCKET 50-412 LER 90-016
SERVICE WATER SYSTEM FLOW BLOCKAGE DUE TO BUILDUP OF CORROSION PRODUCTS.
EVENT DATE: 092190 REPORT DATE: 102590 NSSS: WE TYPE: PWR

(NSIC 219929) ON 9/21/90 WITH THE UNIT, IN REFUELING, THE SERVICE WATER SYSTEM (SWS) PIPING TO THE "B" TRAIN MOTOR CONTROL CENTER (MCC) CUBICLE COOLING COIL, 2HVP*CLC265B, WAS DIASSEMBLED FOR INSPECTION AND CLEANING IN RESPONSE TO GENERIC LETTER 89-13. IT WAS DISCOVERED THAT THE LINES WERE BLOCKED BY A BUILDUP OF CORROSION PRODUCTS. ON 9/25/90, THE SWS PIPING TO THE "A" TRAIN MCC CUBICLE COOLING COIL, 2HVP*CLC265A, WAS DIASSEMBLED FOR INSPECTION AND CLEANING. THE LINES WERE DISCOVERED TO BE PARTIALLY BLOCKED. RADIOGRAPHY REVEALED BLOCKAGE OF THE INLET AND OUTLET, PIPING TO BOTH TRAINS OF MCC CUBICLE COOLING COILS. THE "B" TRAIN MCC CUBICLE COOLING COIL WAS RESTORED TO PROPER FLOW CONDITIONS ON 9/21/90 BY A COMBUSTION OF CHEMICAL AND HIGH PRESSURE WATER CLEANING. THE 'A' TRAIN MCC CUBICLE COOLING COIL WAS RESTORED ON 9/28/90. ADDITIONALLY, SAFETY RELATED SERVICE WATER LOADS WILL BE TESTED FOR REQUIRED DESIGN BASIS SWS FLOW. THERE WERE NO SAFETY IMPLICATIONS AS A RESULT OF THIS EVENT. THE COOLING COILS ARE LOCATED WITHIN THE CUBICLES AND ARE DESIGNED TO MAINTAIN THE AMBIENT AREA TEMPERATURE FOR THE MCC CUBICLES LESS THAN 104F WITH THE REQUIRED SERVICE WATER FLOW. THESE COMPONENTS ARE DESCRIBED IN THE UPDATED FINAL SAFETY ANALYSIS REPORT, SECTION 9.2.1 "STATION SERVICE WATER SYSTEMS".

[11] BEAVER VALLEY 2 DOCKET 50-412 LER 90-017
RECIRCULATION SPRAY PUMPS' TIMER FAILURES.
EVENT DATE: 092790 REPORT DATE: 102590 NSSS: WE TYPE: PWR
VENDOR: AUTOMATIC TIMING AND CONTROL INC.

(NSIC 219930) ON 9/27/90, DURING THE SECOND REFUELING OUTAGE AT BEAVER VALLEY UNIT 2, THE RECIRCULATION SPRAY PUMP TEST (2BVT 1.13.5) WAS PERFORMED. THIS TEST VERIFIES THAT THE RECIRCULATION SPRAY PUMPS START 623 TO 653 SECONDS AFTER A CONTAINMENT ISOLATION PHASE B (CIB) SIGNAL. DURING THIS TEST, THE B, C AND D PUMPS DID NOT START WITHIN THE REQUIRED TIME PERIOD. THE B PUMP STARTED AFTER 663 SECONDS, THE C AFTER 650 SECONDS AND THE D AFTER 648 SECONDS. THE MECHANICAL DELAY TIMERS FOR THESE PUMPS ARE BEING REPLACED WITH MORE PRECISE SOLID STATE TIMERS. THESE REPLACEMENTS WILL BE COMPLETED PRIOR TO PLANT RESTART. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. ENGINEERING ANALYSIS PERFORMED DURING UNIT 2'S FIRST REFUELING OUTAGE SHOWED THAT START DELAYS AS LONG AS 668 SECONDS WOULD HAVE NO ADVERSE AFFECTS ON THE PUMPS' OPERATION OR ABILITY TO FULFILL THEIR SAFETY FUNCTION. THIS ANALYSIS BOUNDS ALL "AS-FOUND" START DELAYS.

[12] BRAIDWOOD 1 DOCKET 50-456 LER 90-018
SPURIOUS TRAIN B SOLID STATE PROTECTION SYSTEM ACTUATIONS DUE TO COMPONENT FAILURE, PERSONNEL ERROR, AND COMPONENT DESIGN INTERFACE DEFICIENCY.
EVENT DATE: 092990 REPORT DATE: 102990 NSSS: WE TYPE: PWR
VENDOR: RELIANCE ELECTRIC COMPANY
WESTINGHOUSE ELECTRIC CORP.

(NSIC 219947) AT 0345 ON 9/29/90 THE TRAIN B SOLID STATE PROTECTION SYSTEM (SSPS) INITIATED A CONTAINMENT VENTILATION ISOLATION SIGNAL. NO COMPONENTS REPOSITIONED AS ALL WERE IN THEIR REQUIRED STATE. AT 1735 A SPURIOUS TRAIN B SAFETY INJECTION

SIGNAL (SI) OCCURRED CAUSING A REACTOR TRIP AND CONTAINMENT ISOLATION TO OCCUR AS WELL AS STARTING TRAIN B ECCS COMPONENTS. THE B TRAIN OF SSPS WAS DECLARED INOPERABLE AND A PLANT COOLDOWN WAS INITIATED. AT 0013 ON 9/30/90 A REACTOR OPERATOR WAS PERFORMING AN SSPS TEST PROCEDURE WHEN THE TRAIN B SSPS MEMORIES TEST SWITCH WAS INADVERTENTLY ROTATED FROM OFF TO POSITION 23. THIS ENABLED THE PRESSURIZER AND STEAMLINE LOW PRESSURE SI AND STEAMLINE ISOLATION CIRCUITS WHICH HAD BEEN BLOCKED. WITH BOTH PRESSURES BELOW THEIR RESPECTIVE SETPOINTS, A TRAIN B SI AND A STEAMLINE ISOLATION OCCURRED. DUE TO THE STEAMLINE ISOLATION, THE RCS TEMPERATURE INCREASED ABOUT 12 DEGREES FROM 340 TO 352 F OVER THE NEXT 14 MINUTES UNTIL COOLING WAS RE-ESTABLISHED. 350F IS THE LOWER LIMIT OF MODE 3 OPERATION. AT 1020 ON 10/3/90 A SPURIOUS TRAIN B FEEDWATER ISOLATION OCCURRED WITH THE UNIT IN COLD SHUTDOWN. THE CAUSES OF THE EVENT WERE COMPONENT FAILURE, PERSONNEL ERROR, AND COMPONENT DESIGN INTERFACE. TRAIN B SSPS WAS REPAIRED, TRAINING WILL BE PROVIDED, AND THE MEMORIES TEST SWITCH WAS RE-ORIENTED.

[13] BROWNS FERRY 1 DOCKET 50-259 LER 86-021 REV 02
UPDATE ON REACTOR BUILDING FLOOD LEVEL SWITCHES NOT SEISMICALLY QUALIFIED DUE TO DESIGN DEFICIENCIES.
EVENT DATE: 062086 REPORT DATE: 102990 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)
VENDOR: ROBERTSHAW CONTROLS COMPANY

(NSIC 219796) THIS LER REVISION IS BEING SUBMITTED TO PROVIDE SUPPLEMENTAL INFORMATION AND IS A VOLUNTARY REPORT. DURING THE DESIGN PROCESS, DESIGN ENGINEERS NOTICED THAT THE REACTOR BUILDING FLOOD LEVEL SWITCHES WERE LISTED IN THE FINAL SAFETY ANALYSIS REPORT (FSAR) AS ONE OF AT LEAST TWO SEISMICALLY QUALIFIED MEANS OF FLOOD DETECTION INSIDE THE REACTOR BUILDING. HOWEVER, DOCUMENTATION AND DESIGN DRAWINGS DO NOT INDICATE THAT THIS MEANS OF INDICATION IS SEISMICALLY QUALIFIED. THE FLOOD LEVEL SWITCH CIRCUIT PROVIDES ANNUNCIATION ONLY AND PERFORMS NO CONTROL FUNCTIONS. THEREFORE, SHOULD THESE CIRCUITS FAIL DURING A SEISMIC EVENT IT WOULD REDUCE THE QUANTITY AND DIVERSITY OF INDICATION OF FLOODING IN THE REACTOR BUILDING. NO ACTUAL FAILURES HAVE OCCURRED. TENNESSEE VALLEY AUTHORITY'S (TVA'S) SAFETY EVALUATION CONCLUDED THAT THE NONCOMPLIANCES DO NOT CONSTITUTE AN UNREVIEWED SAFETY QUESTION AND FURTHERMORE THAT NO NUCLEAR SAFETY CONCERNS EXIST WITH THE CURRENT NONSEISMIC CONFIGURATION. THE BROWNS FERRY NUCLEAR PLANT (BFN) FSAR HAS BEEN REVISED IN THE ANNUAL FSAR SEISMIC DESCRIPTION AND TAKE CREDIT FOR EXISTING OPERATOR ACTIONS TO DETECT AND MITIGATE FLOODING EVENTS. ALSO, THE BROWNS FERRY EARTHQUAKE PROCEDURE HAS BEEN REVISED TO CORRECTLY IDENTIFY THESE FLOOD SWITCHES AS NOT BEING SEISMICALLY QUALIFIED.

[14] BROWNS FERRY 1 DOCKET 50-259 LER 90-012 REV 01
UPDATE ON HIGH-PRESSURE FIRE PROTECTION SYSTEM IN VIOLATION OF TECHNICAL SPECIFICATIONS BECAUSE FUNCTIONAL TEST NOT PERFORMED.
EVENT DATE: 080190 REPORT DATE: 101590 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 219827) ON AUGUST 1, 1990 DURING A REVIEW OF THE FIRE PROTECTION TECHNICAL SPECIFICATIONS (TS) TVA DISCOVERED THAT THE HIGH-PRESSURE FIRE PROTECTION (HPFP) SYSTEM FIRE PUMP START LOGIC WAS NOT BEING TESTED TO VERIFY THAT THE FIRE PROTECTION SYSTEM SWITCH SETPOINT ACTUATE AT 120 PSIG AFTER AN INITIAL PUMP STARTUP AS REQUIRED BY CURRENT TS. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. DURING PREPARATION OF THE SURVEILLANCE INSTRUCTION (SI) THAT VERIFIED TS COMPLIANCE, THE REQUIREMENT TO VERIFY THAT AFTER INITIAL HIGH-PRESSURE FIRE PUMP ACTUATION EACH SUBSEQUENT HIGH-PRESSURE FIRE PUMP STARTS SEQUENTIALLY TO MAINTAIN THE HPFP SYSTEM EQUAL OR GREATER THAN 120 PSIG WAS NOT INCLUDED. THE CORRECTIVE ACTION FOR THIS EVENT WAS TO RAISE THE SETPOINT OF THE PRESSURE SWITCH THAT CONTROLLED THE FIRE PUMP SEQUENCING FROM 100 PSIG TO 120 PSIG. FURTHER CORRECTIVE ACTIONS WILL INCLUDE: REVISING THE APPROPRIATE SI TO FULFILL THE REQUIREMENTS OF THE TS, RE-EVALUATING THE PRESSURE SWITCH SETPOINT, AND CORRECTING THE ERROR CREATED DURING THE ENGINEERING CHANGE NOTICE CLOSURE.

[15]   BROWNS FERRY 2   DOCKET 50-260   LER 90-005
DEENERGIZATION OF REACTOR PROTECTION SYSTEM DUE TO MOTOR GENERATOR SET OPERATION CAUSED BY THE MOTOR OVERLOAD RELAY MISOPERATION.
EVENT DATE: 100290   REPORT DATE: 103190   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 1 (BWR)
BROWNS FERRY 3 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219828) ON OCTOBER 2, 1990 AT 0455 HOURS UNIT 2 REACTOR PROTECTION SYSTEM (RPS) BUS 2B WAS DEENERGIZED WHEN THE ASSOCIATED RPS MOTOR GENERATOR (MG) SET TRIPPED. THIS RESULTED IN THE ACTUATION LOGIC OF PLANT ENGINEERED SAFETY FEATURES INCLUDING A PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) GROUP 6 ISOLATION OF THE UNIT 2 REACTOR BUILDING VENTILATION AND PCIS GROUP 3 (REACTOR WATER CLEANUP) ISOLATION. AN INVESTIGATION INTO THE CAUSE OF THE RPS BUS DEENERGIZATION REVEALED THAT CIRCUIT PROTECTORS 2B1 AND 2B2 TRIPPED AND RPS MG SET 2B COASTING DOWN. A VISUAL INSPECTION OF THE MG SET CONTROL PANEL FOUND NO FAILED OR BURNED COMPONENTS WITHIN THE MG SET CONTROL CIRCUITRY. THE PHASE A MOTOR OVERLOAD RELAY WAS FOUND TO HAVE AN ABNORMALLY HIGH RESISTANCE. THE ROOT CAUSE OF THE EVENT WAS RANDOM COMPONENT FAILURE. THE HIGH RESISTANCE OF THE MOTOR OVERLOAD RELAY CONTACT IN THE MG SET CAUSED A MG MOTOR STARTER COIL CONTROL RELAY TO DROP OUT. THIS TRIPPED THE MG SET AND DEENERGIZED THE RPS BUS. THE HIGH RESISTIVITY OF THE MOTOR OVERLOAD RELAY CONTACT WAS REDUCED BY REPEATED ACTUATION OF THE CONTACT RESET MECHANISM. THE CONTACT WAS CLEANED TO ENSURE THAT GOOD CONTACT CONTINUITY WAS ESTABLISHED. THE MOTOR WAS BRIDGED AND MEGGER TESTED TO ENSURE NO PHASE IMBALANCE OR WINDING INSULATION DAMAGE EXISTED.

[16]   BRUNSWICK 1   DOCKET 50-325   LER 90-016
OPERATION PROHIBITED BY PLANT TECHNICAL SPECIFICATIONS DURING SCRAM DISCHARGE VOLUME MAINTENANCE AND SURVEILLANCE ACTIVITIES.
EVENT DATE: 091790   REPORT DATE: 101790   NSSS: GE   TYPE: BWR

(NSIC 219864) DURING REVIEW OF MAINTENANCE THAT HAD BEEN PERFORMED ON THE C11-LSH- 4516C SCRAM DISCHARGE LEVEL SWITCH IT WAS DETERMINED THAT, AS A RESULT OF A JUMPER BEING USED DURING THE EVOLUTION TO DISABLE THE HALF-SCRAM INPUT SIGNAL FROM THE AC CHANNEL THE TECHNICAL SPECIFICATION REQUIREMENT FOR OPERABLE CHANNELS IN THE "A" TRIP SYSTEM WAS NOT MET SUBSEQUENTLY. THE TECHNICAL SPECIFICATION ACTION STATEMENT FOR PLACING THE TRIP SYSTEM IN THE TRIPPED CONDITION WITHIN ONE HOUR OF NOT HAVING THE MINIMUM NUMBER OF CHANNELS OPERABLE, WAS NOT MET. THE CAUSE OF THE EVENT WAS THE FAILURE OF THE SENIOR REACTOR OPERATOR INVOLVED IN THE EVENT TO RECOGNIZE THE SCOPE OF THE INVOLVED JUMPER. IT WAS NOT RECOGNIZED THAT THE JUMPER INSTALLED BY THE SURVEILLANCE PROCEDURE USED FOR THE MAINTENANCE DISABLED BOTH INSTRUMENTS IN THE A2 INSTRUMENT CHANNEL, THEREFORE THE CHANNEL AND TRIP SYSTEM DID NOT MEET OPERABILITY REQUIREMENTS. CORRECTIVE ACTIONS INCLUDE TRAINING OF BOTH OPERATIONS AND I&C PERSONNEL ON THE EVENT, AND FURTHER CLARIFYING THE SURVEILLANCE PROCEDURE NOTATION AS TO THE INSTRUMENTS BEING MADE INOPERABLE BY THE JUMPER. THE SAFETY SIGNIFICANCE OF THE EVENT IS CONSIDERED MINIMAL. THE REDUNDANT A1 CHANNEL WAS OPERABLE DURING THE EVOLUTION, AS WELL AS THE CORRESPONDING "B" TRIP SYSTEM CHANNELS. THE SYSTEM WAS STILL FULLY CAPABLE OF PROVIDING THE REQUIRED LOGIC RESPONSE.

[17]   BRUNSWICK 1   DOCKET 50-325   LER 90-015
INCORRECT LOCAL POWER RANGE MONITOR ASSIGNMENT DUE TO REVERSED CABLES.
EVENT DATE: 092690   REPORT DATE: 102690   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)

(NSIC 219899) WHILE REVIEWING THE UNIT 2 ON-DEMAND COMPUTER PRINTOUTS OF THE LOCAL POWER RANGE MONITOR (LPRM) READINGS FOR THE CURRENT REFUEL OUTAGE STARTUP SEQUENCE, TWO LPRMS WERE FOUND BY THE SYSTEM ENGINEER TO BE READING INCONSISTENT WHEN COMPARED TO OTHER LPRMS IN THE SAME GENERAL AREA. INVESTIGATION HAS DETERMINED THAT THE CABLE REVERSAL OCCURRED DURING LPRM HARDLINE CABLE MODIFICATION INSTALLATION. EXACT CAUSE FOR THE REVERSAL HAS NOT BEEN DETERMINED AT THIS TIME DUE TO LACK OF ACCESSIBILITY TO THE AREA WHERE THE CABLES ARE REVERSED. CORRECTIVE ACTIONS INCLUDE DEVELOPMENT OF A PROCEDURE TO VERIFY LPRM DETECTOR CONFIGURATION FOLLOWING EACH REFUEL OUTAGE, ADDITIONAL TAGGING CONTROLS

OF UNDERVESSEL CABLES, RESTORING THE CABLES TO THEIR PROPER CONFIGURATION DURING UPCOMING REFUEL OUTAGES, REVIEW OF HISTORICAL DATA FOR OTHER POTENTIAL PAST OPERABILITY CONCERNS, AND FURTHER INVESTIGATION UPON ACCESSIBILITY OF THE DRYWELL AREA. AN ANALYSIS PERFORMED BY GENERAL ELECTRIC HAS DETERMINED THAT NO POTENTIAL OPERABILITY CONCERNS EXISTED FOR THERMAL LIMITS, APRM CONFIGURATION, OR THE ROD BLOCK MONITOR SYSTEM. THE EVENT WAS DETERMINED NOT TO BE REPORTABLE UNDER 10CFR50.73; HOWEVER, THIS REPORT IS BEING SUBMITTED AS A VOLUNTARY LER IN ACCORDANCE WITH NUREG-1022 DUE TO THE POTENTIAL SAFETY SIGNIFICANCE OF THE ISSUE, AND THE POTENTIAL FOR SIMILAR CONCERNS AT OTHER UTILITIES.

[18] BRUNSWICK 1 DOCKET 50-325 LER 90-017
HIGH PRESSURE REACTOR SCRAM WHILE PERFORMING PT-40.2.10 DUE TO ERRONEOUS PROCEDURE GUIDANCE AND DEFECTIVE TURBINE STOP VALVE SWITCHES.
EVENT DATE: 092790 REPORT DATE: 102690 NSSS: GE TYPE: BWR

(NSIC 219865) DURING A SCHEDULED UNIT 1 SHUT DOWN FOR A REFUEL/MAINTENANCE OUTAGE ON SEPTEMBER 27, 1990, THE REACTOR SCRAMMED ON HIGH PRESSURE AT 0348, DURING THE PERFORMANCE OF PERIODIC TEST (PT) 40.2.10, TURBINE CONTROL/STOP VALVES (TCV/TSV) LEAK TIGHTNESS TESTING. PRIOR TO THE EVENT, THE REACTOR WAS AT APPROXIMATELY 22% POWER AND THE EMERGENCY CORE COOLING SYSTEMS (ECCS) WERE OPERABLE IN STANDBY READINESS. EVENT RECOVERY WAS IN ACCORDANCE WITH SITE EMERGENCY OPERATING PROCEDURES, NO ECCS OR ENGINEERED SAFETY FEATURE ACTUATIONS OR ISOLATIONS OTHER THAN SCRAM SIGNALS OCCURRED. THE EVENT WAS OCCURRED BY ERRONEOUS PROCEDURAL GUIDANCE, INCORPORATED INTO THE PT FROM A VENDOR DOCUMENT, AND DEFECTIVE SWITCHES ON THE TSVS WHICH ALLOWED THE TCVS TO OPEN WHEN THE TSVS WERE CLOSING. THIS RESULTED IN THE TURBINE BYPASS VALVES (BPV) OPEN DEMAND SIGNAL BEING LIMITED BY THE MAXIMUM COMBINE FLOW CIRCUITRY OF THE TURBINE CONTROL SYSTEM. THE CLOSURE OF THE TBVS OCCURRED REACTOR PRESSURE TO INCREASE TO THE SCRAM SETPOINT. MAXIMUM POWER ATTAINED DURING THE SCRAM WAS 28%. THIS EVENT HAD MINIMAL SAFETY SIGNIFICANCE AS THE REACTOR IS ANALYZED FOR A HIGH PRESSURE SCRAM FROM FULL POWER. PAST HIGH PRESSURE SCRAM EVENTS WERE REVIEWED AND FOUND NOT TO BE RELATED TO THIS EVENT. THE PROCEDURE WILL BE REWRITTEN AND THE SWITCHES WILL BE REPAIRED.

[19] BRUNSWICK 1 DOCKET 50-325 LER 90-018
PRIMARY CONTAINMENT ISOLATION SYSTEM GROUP 6 ISOLATION AND STANDBY GAS TREATMENT SYSTEM AUTO START DUE TO LOSS OF POWER TO MAIN STACK RADIATION MONITOR.
EVENT DATE: 092790 REPORT DATE: 102690 NSSS: GE TYPE: BWR

(NSIC 219866) ON 9/27/90, UNIT 1 RECEIVED A PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) GROUP 6 ISOLATION AND STANDBY GAS TREATMENT SYSTEM (SBGT) AUTOSTART AS A RESULT OF A MOMENTARY LOSS OF POWER TO THE MAIN STACK RADIATION MONITOR. THE LOSS OF POWER TO THE MAIN STACK RADIATION MONITOR WAS DUE TO MOMENTARY ELECTRICAL SYSTEM PERTURBATIONS FOLLOWING A REACTOR SCRAM ON UNIT 2. SYSTEMS RESPONDED AS EXPECTED. THE UNIT 2 SCRAM CAUSAL FACTORS AND CORRECTIVE ACTIONS ARE REPORTED IN LER 2-90-15. NO FURTHER CORRECTIVE ACTIONS RELATIVE TO THIS EVENT ARE CONSIDERED NECESSARY. THE SAFETY SIGNIFICANCE OF THIS EVENT RELATIVE TO UNIT 1 IS CONSIDERED MINIMAL.

[20] BRUNSWICK 1 DOCKET 50-325 LER 90-019
PRIMARY CONTAINMENT ISOLATION SYSTEM GROUP 6 ISOLATION WITH STANDBY GAS TREATMENT SYSTEM AUTO START AND REACTOR BUILDING VENTILATION ISOLATION WHILE PERFORMING MAINTENANCE SURVEILLANCE TEST.
EVENT DATE: 100290 REPORT DATE: 110190 NSSS: GE TYPE: BWR

(NSIC 219867) ON 10/2/90, AT 0938, WHILE PERFORMING A SURVEILLANCE TEST ON VENTILATION RADIATION MONITOR DETECTOR CHANNEL "A", A TECHNICIAN MISTAKENLY PLACED THE "B" CHANNEL DETECTOR TO THE RADIATION SOURCE. THIS RESULTED IN REACTOR BUILDING OUTBOARD VENTILATION DAMPER AND GROUP 6 VALVE ISOLATIONS. AND STANDBY GAS TREATMENT (SBGT) SYSTEM AUTO START. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR, FAILURE BY THE INVOLVED TECHNICIAN TO PROPERLY VERIFY THE CHANNEL BEING TESTED. A HUMAN PERFORMANCE EVALUATION PERFORMED FOR THE EVENT DETERMINED THAT COMPONENT LABELING, PERSONNEL MINDSET, AND COMMUNICATION WERE POTENTIAL CONTRIBUTING FACTORS. CORRECTIVE ACTIONS INCLUDE DISCIPLINARY ACTION

WITH THE INVOLVED INDIVIDUALS LABELING THE DETECTORS, AND TRAINING OF I&C PERSONNEL ON COMMUNICATION GUIDLINES. THE EVENT HAD MINIMAL SAFETY SIGNIFICANCE.

[21] BRUNSWICK 1 DOCKET 50-325 LER 90-021
TECHNICAL SPECIFICATION 3.0.3 ENTRY DUE TO LOSS OF ALL DIESEL GENERATORS.
EVENT DATE: 100/90 REPORT DATE: 110190 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)
VENDOR: ALLEN-BRADLEY CO.

(NSIC 219900) UNIT 1 WAS IN REFUEL WITH ITS NUCLEAR SERVICE WATER (NSW) HEADER UNDER CLEARANCE FOR MAINTENANCE, AND UNIT 2 WAS OPERATING AT 100% POWER. DURING PERFORMANCE OF A DIESEL GENERATOR (DG) PERIODIC TEST (PT), DG #4 WAS DECLARED INOPERABLE. BASED ON REVIEWS OF TECH SPECS AND A TECH SPEC INTERPRETATION (TSI 84-06, REV. 10), THE 2B NSW PUMP WAS DECLARED INOPERABLE DUE TO LOSS OF EMERGENCY POWER SOURCE. AS A RESULT, PER THE BASES OF TSI 84-06, REV. 10, IT WAS DETERMINED THAT BOTH UNITS WERE IN AN UNANALYZED CONDITION RELATIVE TO THE ADEQUACY OF SERVICE WATER FLOW UNDER CERTAIN DESIGN CONDITIONS. AT 0420 ON 10/8/90, ALL FOUR DGS WERE DECLARED INOPERABLE, AND TECH SPEC 3.0.3 ENTERED FOR UNIT 2. UNIT 1 SUSPENDED CORE ALTERATIONS AND ACTIVITIES WHICH HAD THE POTENTIAL FOR DRAINING THE VESSEL. THE FAILURE OF DG #4 WAS A RESULT OF A FAILED RELAY IN THE PRE-START CIRCUITRY. THE RELAY WAS REPLACED, AND DG #4 WAS TESTED AND RETURNED TO SERVICE. UNIT 2 COMMENCED A POWER INCREASE TO RATED CONDITIONS, AND OUTAGE ACTIVITIES ON UNIT 1 RESTARTED. NO GENERIC CONCERNS WERE IDENTIFIED WITH THIS FAILURE. THE SAFETY SIGNIFICANCE OF THE EVENT WAS MINIMAL, AS THE SHUTDOWN REQUIREMENTS WAS PREDICATED ON A TSI WHICH EVALUATES SW SYSTEM REQUIREMENTS DURING A DESIGN BASIS LOCA WITH A CONCURRENT LOSS OF OFF-SITE POWER AND A FAILURE OF AN EMERGENCY BUS. THE PROBABILITY OF THESE SIMULTANEOUS EVENTS IS EXTREMELY LOW.

[22] BRUNSWICK 2 DOCKET 50-324 LER 90-012 REV 01
UPDATE ON SCRAM CAUSED BY FAILURE OF THE START-UP LEVEL CONTROL VALVE RESULTING IN A LOW LEVEL RPS ACTUATION.
EVENT DATE: 083090 REPORT DATE: 110190 NSSS: GE TYPE: BWR
VENDOR: FISHER FLOW CONTROL DIV (ROCKWELL INT)

(NSIC 219862) ON AUGUST 30, 1990, UNIT 2 REACTOR START-UP WAS IN PROGRESS. THE REACTOR WAS AT APPROXIMATELY 8% POWER AND 300 PSIG. THE EMERGENCY CORE COOLING SYSTEMS WERE OPERABLE IN STANDBY LINE UP EXCEPT FOR THE HIGH PRESSURE COOLANT INJECTION SYSTEM WHICH WAS INOPERABLE AWAITING THE PERFORMANCE OF THE HPCI SYSTEM OPERABILITY TEST. AT 1656 THE START-UP LEVEL CONTROL VALVE (SULCV) FAILED CLOSED RESULTING IN A LEVEL TRANSIENT. AT 1657 THE REACTOR PROTECTION SYSTEM (RPS) LOW LEVEL #1 SETPOINT (165") WAS REACHED CAUSING A REACTOR SCRAM. PRIMARY CONTAINMENT ISOLATION SYSTEM GROUPS 2, 6 AND 8 ALSO RECEIVED AN ISOLATION SIGNAL AND ACTUATED PER DESIGN. SCRAM RECOVERY WAS IN ACCORDANCE WITH THE EMERGENCY FLOWCHARTS AND PROCEDURES. APPROXIMATELY 20 MINUTES AFTER THE SCRAM, LEVEL WAS STABILIZED. THE CAUSE OF THE SULCV FAILURE IS BELIEVED TO BE WORN O-RING SEALS. THE SEALS HAVE BEEN REPLACED AND THE SULCV IS OPERATING PROPERLY. THE WORN SEALS ARE BEING ANALYZED TO DETERMINE THE CAUSE OF THE FAILURE. THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL. LEVEL WAS RECOVERED WITHOUT THE NEED FOR SAFETY SYSTEM INJECTION AND THE UNIT IS DESIGNED FOR A LEVEL TRANSIENT FROM FULL POWER.

[23] BRUNSWICK 2 DOCKET 50-324 LER 90-013
HPCI DECLARED INOPERABLE DUE TO ERRATIC PERFORMANCE DURING PERIODIC TESTING.
EVENT DATE: 090690 REPORT DATE: 100490 NSSS: GE TYPE: BWR
VENDOR: WOODWARD GOVERNOR COMPANY

(NSIC 219732) UNIT 2 WAS OPERATING AT 95% POWER. ON 9/6/90, DURING PERFORMANCE OF THE QUARTERLY HIGH PRESSURE COOLANT INJECTION (HPCI) OPERABILITY TEST, THE HPCI TURBINE EXHIBITED ERRATIC OPERATION AND DID NOT STABILIZE UPON REACHING RATED CONDITIONS. THE SYSTEM WAS TRIPPED AND DECLARED INOPERABLE, AND AN LIMITED CONDITION FOR OPERATION (LCO) INITIATED. DURING INVESTIGATION OF THIS EVENT, THE NEEDLE VALVE ON THE GOVERNOR SPEED CONTROL SYSTEM WAS FOUND TO BE MORE OPEN THAT

THE NORMAL POSITION. THE NEEDLE VALVE WAS ADJUSTED AND THE HPCI SYSTEM RESTARTED WITH NO NOTICEABLE SPEED OSCILLATIONS. THE PERIODIC TEST WAS COMPLETED AND THE LCO WAS CANCELLED. THE CAUSE OF THE EVENT WAS DETERMINED TO BE THE RESULT OF A DEFICIENT NEEDLE VALVE POSITION. REVISIONS WILL BE MADE TO MAINTENANCE PROCEDURES TO ENHANCE CONTROLS FOR EGR NEEDLE VALVE ADJUSTMENTS. DUE TO SUCCESSFUL PERFORMANCE OF THE SYSTEM DURING RECENT EVENTS (UP TO 8/20/90), IT IS REASONABLE TO ASSUME THE SYSTEM WOULD HAVE PERFORMED ITS INTENDED FUNCTION ON OTHER OCCASIONS IF NECESSARY. IN ADDITION, THE ADS SYSTEM WAS OPERABLE BETWEEN 8/20/90 AND 9/6/90.

[24] BRUNSWICK 2 DOCKET 50-324 LER 90-015
REACTOR SCRAM DUE TO ERRATIC VOLTAGE REGULATOR RESPONSE.
EVENT DATE: 092790 REPORT DATE: 102690 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 1 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219863) ON 9/27/90, WHILE OPERATING AT 100% POWER UNIT 2 EXPERIENCED GENERATOR VOLTAGE OSCILLATIONS WHICH RESULTED IN A LOSS OF THE MAIN GENERATOR DUE TO LOSS OF EXCITATION. THIS RESULTED IN A REACTOR SCRAM DUE TO CONTROL VALVE FAST CLOSURE SIGNAL. THE PRIMARY CAUSE OF THE SCRAM WAS A VOLTAGE REGULATOR THAT HAD BECOME POTENTIALLY UNSTABLE DUE TO PAST IMPROPER ADJUSTMENTS. SYSTEMS FUNCTIONED AS DESIGNED. POTENTIAL CONCERNS WERE NOTED WITH A CRD THAT DID NOT FULLY INSERT (02 POSITION VERSUS 00 POSITION) AND DUAL POSITION INDICATION ON AN MSIV. CAUSE OF THE SCRAM WAS INADEQUATE CONFIGURATION CONTROL ON THE VOLTAGE REGULATOR ADJUSTMENTS AND URAL CIRCUIT SETTINGS. POTENTIALLY CONTRIBUTING TO THE EVENT WAS THE GRID SYSTEM CONFIGURATION ON THAT DATE, THE SYSTEM VOLTAGE SCHEDULE, AND PROCEDURE CONTROLS FOR MAINTAINING GENERATOR EXCITATION. CORRECTIVE ACTIONS INCLUDE ADJUSTMENTS OF THE VOLTAGE REGULATOR, UPDATING THE VENDOR TECHNICAL MANUAL TO ENSURE PROPER FUTURE SYSTEM PERFORMANCE, AND EVALUATIONS OF THE GRID STABILITY REQUIREMENTS AND DESIGN BASIS RELATIVE TO SYSTEM CAPACITANCE VALUES. THE EVENT SAFETY SIGNIFICANCE IS CONSIDERED MINIMAL. TRANSIENT PARAMETERS EXPERIENCED WERE WELL WITHIN ANALYZED TRANSIENT PARAMETERS FOR THIS TYPE EVENT.

[25] BYRON 2 DOCKET 50-455 LER 90-007
ECCS THROTTLE VALVE FOUND CLOSED DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 092890 REPORT DATE: 102690 NSSS: WE TYPE: PWR

(NSIC 219939) ON 9/28/90 AT 2130, DURING PERFORMANCE OF TECH SPEC SURVEILLANCE 2BVS 0.5-2.SI.2-1, "SAFETY INJECTION PUMP COLD LEG INJECTION FLOW TEST", IT WAS FOUND THAT THE RCS LOOP "A" BRANCH INDICATED ZERO FLOW WHILE THE ASSOCIATED RCS LOOPS "B, C AND D" BRANCHES INDICATED HIGHER THAN EXPECTED FLOWRATES. THE TEST WAS SUSPENDED AND A PRELIMINARY INVESTIGATION REVEALED THAT THE LOOP "A" COLD LEG SAFETY INJECTION (SI) (BQ) LINE THROTTLE VALVE, 2SI8822A WAS CLOSED. THIS VALVE IS AN EMERGENCY CORE COOLING SYSTEM (ECCS) THROTTLE VALVE AND IS REQUIRED TO BE LOCKED IN A PRE-SET THROTTLED OPEN POSITION TO PROVIDE A BALANCED FLOW TO EACH OF THE RCS COLD LEGS. THE VALVE IS BELIEVED TO HAVE BEEN CLOSED TO SUPPORT MAINTENANCE WORK ON A DIFFERENT SI VALVE DURING THE PREVIOUS OUTAGE AND THE VALVE WAS NOT PLACED IN THE CORRECT THROTTLE POSITION AT THE CONCLUSION OF THIS WORK. A SPECIAL PROCEDURE WILL BE WRITTEN TO BALANCE THE BRANCH LINE FLOWS FOR INTERMEDIATE HEAD SI TO THE COLD LEGS. SUBSEQUENT TO THE FLOW BALANCE ALL FOUR OF THE VALVE'S LOCKING DEVICES WILL BE WELDED INTO POSITION TO PREVENT IMPROPER POSITIONING IN THE FUTURE. THIS EVENT IS BEING REPORTED PURSUANT TO 10CFR50.73(A)(2I)(B) AS ANY OPERATION OR CONDITION PROHIBITED BY TECH SPECS.

[26] BYRON 2 DOCKET 50-455 LER 90-008
DROPPED FUEL ASSEMBLY DURING RECONSTITUTION DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 092990 REPORT DATE: 102990 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 1 (PWR)

(NSIC 219945) ON 9/29/90, AT APPROXIMATELY 2130, BYRON UNIT TWO WAS IN A REFUELING OUTAGE WITH THE REACTOR VESSEL DEFUELED. WESTINGHOUSE AND FUEL HANDLING PERSONNEL WERE PERFORMING BOTTOM NOZZLE FUEL RECONSTITUTION ACTIVITIES IN THE SPENT FUEL POOL IN PREPARATION FOR FUEL RELOAD. RECONSTITUTION OF FUEL

ASSEMBLY T77K HAD BEEN COMPLETED AFTER REMOVING THE INDICATED FUEL RODLET AND REPLACING IT WITH A STAINLESS STEEL DUMMY RODLET. THE RECONSTITUTION BASKET LID WAS CLOSED AND THE STEPS WERE PERFORMED TO SECURE THE LID IN PLACE PRIOR TO ROTATING THE BASKET TO THE UPRIGHT POSITION. DURING THE ROTATION PROCESS, THE FUEL ASSEMBLY SLIPPED OUT OF THE BASKET AND CAME TO REST ON THE TOP OF AN EMPTY FUEL RACK WHILE REMAINING PARTIALLY INSERTED IN THE RECONSTITUTION BASKET. THE ASSEMBLY HAS BEEN TRANSFERRED TO A FAILED FUEL STORAGE CANISTER IN THE SPENT FUEL POOL. PROCEDURAL AND WORK ACTIVITY CHANGES HAVE BEEN MA^E. THIS EVENT WAS DETERMINED TO BE REPORTABLE IN ACCORDANCE WITH 10CFR20. ?(A)(4).

[27] CALLAWAY 1 DOCKET 50-483 LER 90-010
TECHNICAL SPECIFICATION 3.3.3.1 ACTION 27 WAS NOT MET WITHIN ONE HOUR DUE TO COGNITIVE PERSONNEL ERROR.
EVENT DATE: 091890 REPORT DATE: 101890 NSSS: WE TYPE: PWR

(NSIC 219951) ON 09/18/90 AT 0715 CDT, THE OPERATING CREW DETERMINED THE TECH SPEC (T/S) 3.3.3.1 ACTION 27 ONE HOUR LIMIT TO INITIATE A CONTROL ROOM VENTILATION ISOLATION SIGNAL (CRVIS) "HAD NOT BEEN MET UPON DISCOVERING BOTH CONTROL BUILDING VENTILATION EXHAUST FANS AND THE SUPPLY FAN HAD BEEN TURNED OFF AT 0410 TO PERFORM PREVENTIVE MAINTENANCE AND TO ALLEVIATE A PROBLEM WITH OPENING THE CONTROL ROOM DOOR. TURNING OFF THE FANS RESULTED IN A STATIC AIR CONDITION AROUND THE RADIATION MONITOR (GK-RE-4 & 5) WHICH COULD PREVENT AN AUTOMATIC CONTROL ROOM VENTILATION ISOLATION SIGNAL. THE EXHAUST FAN WAS IMMEDIATELY STARTED UPON DISCOVERY. PLANT WAS IN MODE 1, 80% REACTOR POWER. THE LICENSED OPERATORS DID NOT RECALL THAT THE CONTROL BUILDING EXHAUST FANS AND THE SUPPLY FAN SUPPORTED THE RADIATION MONITORS. SINCE THE CONTROL BUILDING SUPPLY FAN AND EXHAUST FANS ARE NOT SAFETY-RELATE NOR SPECIFICALLY REQUIRED IN T/S, THE OPERATORS DID NOT CONNECT THEIR OPERATION TO T/S 3.3.3.1. NOTES WERE ADDED TO THE APPROPRIATE WORK DOCUMENTS TO CONNECT THE FANS TO THE RADIATION MONITOR'S T/S 3.3.3.1. THE ADDITION OF AN ANNUNCIATOR TO ALERT THE OPERATORS WHEN THE FANS HAVE BEEN SECURED WILL BE EVALUATED IN THE ABSENCE OF SUCH AN ANNUNCIATOR. OPERATOR AIDS WILL BE PLACED AT THE FAN SWITCHES TO REFER THE OPERATOR TO T/S. SHIFT SUPERVISORS WILL REVIEW THIS EVENT WITH THEIR RESPECTIVE OPERATING CREWS.

[28] CALLAWAY 1 DOCKET 50-483 LER 90-011
ENGINEERED SAFETY FEATURES ACTUATIONS DUE TO A SPURIOUS SIGNAL ON A FUEL BUILDING RADIATION MONITOR.
EVENT DATE: 100390 REPORT DATE: 110290 NSSS: WE TYPE: PWR

(NSIC 219952) ON 10/3/90 AT 0850 CDT, ENGINEERED SAFETY FEATURES CONTROL ROOM VENTILATION ISOLATION AND FUEL BUILDING VENTILATION ISOLATION ACTUATIONS WERE RECEIVED DUE TO A SPURIOUS SIGNAL ON FUEL BUILDING EXHAUST RADIATION MONITOR GG-RE-27. PROPER ACTUATION OF DAMPERS AND FANS WAS VERIFIED BY THE LICENSED OPERATORS. UTILITY HEALTH PHYSICS TECHNICIANS CONDUCTED A RADIATION SURVEY IN THE FUEL BUILDING AND FOUND NO ABNORMAL RADIATION LEVELS. GG-RE-27 WAS DECLARED INOPERABLE AND THE "A" TRAIN EMERGENCY EXHAUST SYSTEM WAS ISOLATED. THE PLANT WAS IN MODE 6 - REFUELING. THE PROBABLE ROOT CAUSE OF THIS EVENT WAS A SPIKE ON GG-RE-27. A CONTRIBUTING FACTOR WAS A MISCALIBRATED GG-RE-27 DUE TO A TAPED OVER CALIBRATION SOURCE. UTILITY INSTRUMENT AND CONTROL (I&C) TECHNICIANS HAVE REMOVED THE TAPE AND RECALIBRATED GG-RE-27 AND ONE OTHER MONITOR AFFECTED BY USE OF THE TAPED OVER SOURCE. IN ADDITION, UTILITY I&C TECHNICIANS WILL BE TRAINED ON PROPER USAGE OF CALIBRATION SOURCES. AN EVALUATION TO DETERMINE THE CAUSE OF THE GG-RE-27 SPIKE IS ONGOING. GG-RE-27 WAS RETURNED TO SERVICE AT 1643 ON 10/5/90 AND THE "A" EMERGENCY EXHAUST SYSTEM WAS RETURNED TO OPERATION.

[29] CALLAWAY 1 DOCKET 50-483 LER 90-012
CORE ALTERATION PERFORMED WHEN A SOURCE RANGE NUCLEAR INSTRUMENT WAS INOPERABLE DUE TO PERSONNEL ERROR.
EVENT DATE: 100390 REPORT DATE: 110190 NSSS: WE TYPE: PWR

(NSIC 219953) ON 10/3/90 AT 0900 CDT, A CREW HEADED BY AN UTILITY LICENSED SENIOR REACTOR OPERATOR BEGAN TO REMOVE AN IRRADIATION SPECIMEN FROM THE REACTOR VESSEL. PER TECHNICAL SPECIFICATION (T/S) DEFINITION THIS IS A CORE ALTERATION. AT 1020,

A SOURCE RANGE NUCLEAR INSTRUMENT (SRNI) WAS DECLARED INOPERABLE TO ALLOW PREVENTIVE MAINTENANCE (PM). PER T/S 3.9.2, NO CORE ALTERATIONS ARE ALLOWED WITH AN INOPERABLE SRNI. THE CORE ALTERATION WAS CEASED AT APPROXIMATELY 1200. THE PLANT WAS IN MODE 6 - REFUELING WITH 53 OF 193 FUEL ASSEMBLIES IN THE REACTOR VESSEL CORE. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTABLE TO COGNITIVE PERSONNEL ERROR. KNOWING THAT THE SPECIMEN REMOVAL WAS OUTSIDE THE CORE, THE LICENSED SHIFT SUPERVISOR FAILED TO REALIZED THAT THIS WAS A CORE ALTERATION. BOTH THE SPECIMEN REMOVAL AND THE SRNI PM HAD BEEN ADVANCED IN THE SCHEDULE WHEN FUEL MOVEMENT HAD BEEN HALTED DUE TO MECHANICAL PROBLEMS WITH THE FUEL TRANSFER CART. PROCEDURE OTS-KE-00009 FOR REMOVING THE SPECIMEN DID NOT IDENTIFY THIS AS A POTENTIAL CORE ALTERATION. THE DEFINITION AND SPECIFIC EXAMPLES OF CORE ALTERATION WILL BE EMPHASIZED IN INITIAL OPERATOR TRAINING AND FUTURE REFUEL TRAINING. OTS-KE-00009 HAS BEEN REVISED TO DEFINE THE SPECIMEN REMOVAL EVOLUTION AS A CORE ALTERATION WHEN FUEL IS IN THE VESSEL.

[30] CALVERT CLIFFS 1 DOCKET 50-317 LER 89-025 REV 01
UPDATE ON MISSED FIRE WATCH TOUR DUE TO PERSONNEL ERROR.
EVENT DATE: 121189 REPORT DATE: 101990 NSSS: CE TYPE: PWR

(NSIC 219802) ON DECEMBER 8, 1989, THE ACTION STATEMENT FOR UNIT 1 TECHNICAL SPECIFICATION (TS) 3.7.12 WAS ENTERED AS A PRECAUTIONARY MEASURE. IT HAD BEEN DETERMINED THAT VENTILATION DUCTS PENETRATING A TS FIRE BARRIER COULD NOT BE ACCESSED TO DETERMINE IF FIRE DAMPERS WERE INSTALLED. FOR ONE BARRIER, THERE IS SMOKE DETECTION ON BOTH SIDES OF THE BARRIER, BUT AUTOMATIC SPRINKLERS ONLY ON ONE SIDE. THEREFORE, AN HOURLY FIRE WATCH TOUR WAS INITIATED. ON DECEMBER 11, 1989, THE HOURLY FIRE WATCH TOUR WAS MISSED AT MIDNIGHT. THE HOURLY TOUR AT 2300 ON DECEMBER 10, 1989 AND THE HOURLY TOUR AT 0100 ON DECEMBER 11, 1989 WERE PERFORMED. UNIT 1 WAS IN COLD SHUTDOWN (MODE 5) DURING THE INCIDENT. THE CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE COGNITIVE PERSONNEL ERROR. LACK OF MANAGEMENT OVERSIGHT CONTRIBUTED TO THE EVENT. A CONTRACT FIRE BRIGADE MEMBER WAS ASSIGNED TO PERFORM THIS TOUR AT MIDNIGHT. THE TOUR WAS NOT PERFORMED. THERE WAS NO PROGRAM IN PLACE TO INSURE THAT THIS ASSIGNMENT WAS CARRIED OUT. TO PREVENT FUTURE EVENTS OF THIS TYPE, A SINGLE CONTRACT FIRE BRIGADE MEMBER IS ASSIGNED, AS THEIR PRIMARY RESPONSIBILITY, TO PERFORM ALL HOURLY FIRE WATCH TOURS ON HIS SHIFT.

[31] CALVERT CLIFFS 1 DOCKET 50-317 LER 90-026
TILTED EXCORE DETECTORS CAUSED BY INADEQUATE PROCEDURAL GUIDANCE.
EVENT DATE: 082490 REPORT DATE: 102290 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 2 (PWR)

(NSIC 219858) ON AUGUST 24, 1990, IT WAS DISCOVERED THAT TWO EXCORE NUCLEAR INSTRUMENTATION (NI) DETECTOR WELLS ON UNIT 1 WERE TILTED SIX DEGREES. ON SEPTEMBER 20, 1990, IT WAS DETERMINED THAT THESE DETECTORS WOULD BE CONSIDERED NOT OPERABLE FOR THE AXIAL POWER DISTRIBUTION (APD) TRIP WHILE TILTED. THIS CONSTITUTES A VIOLATION OF TECHNICAL SPECIFICATION (TS) 3.3.1.1. THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE PROCEDURAL GUIDANCE. TO CORRECT THIS CONDITION, THE TILTED DETECTORS WERE CORRECTLY POSITIONED. THE PROCEDURES FOR INSTALLING THE DETECTORS ARE BEING REVISED AND APPROPRIATE PERSONNEL WILL BE TRAINED. POWER ASCENSION FOR UNIT 1 WAS HELD AT 30 PERCENT POWER TO PERFORM EXCORE ASI CALIBRATIONS. WE ALSO STOPPED TO CALIBRATE EXCORE ASI BETWEEN 85 AND 90 PERCENT POWER AS PART OF OUR NORMAL STARTUP PROCESS FOLLOWING AN EXTENDED OUTAGE. THIS PROCESS WILL BE REPEATED DURING UNIT 2 STARTUP. THIS ISSUE WAS FOUND DURING THE DEVELOPMENT OF ROUTINE TRAINING. IT WAS ALSO INDEPENDENTLY IDENTIFIED DURING A PROCEDURE REVIEW AS PART OF THE PROCEDURE UPGRADE PROJECT UNDERWAY AS PART OF OUR PERFORMANCE IMPROVEMENT PLAN. SIMILAR DEFICIENCIES, IF THEY EXIST, WILL BE FOUND AS THIS PROJECT CONTINUES.

[32] CALVERT CLIFFS 1 DOCKET 50-317 LER 90-025
POWER LOST TO SAMPLE PUMP FOR GASEOUS EFFLUENT MONITORING.
EVENT DATE: 091190 REPORT DATE: 101190 NSSS: CE TYPE: PWR

(NSIC 219726) ON SEPTEMBER 11, 1990, THE GASEOUS EFFLUENT MONITORING SAMPLING

PUMP FOR IODINE AND PARTICULATE WAS INADVERTENTLY DE-ENERGIZED FOR APPROXIMATELY ONE HOUR. NO GASEOUS EFFLUENT RELEASES WERE IN PROGRESS OTHER THAN AUXILIARY BUILDING VENTILATION. THE PUMP COLLECTS SAMPLES WHICH ARE COUNTED WEEKLY AND USED TO CALCULATE CUMULATIVE OFFSITE DOSE AND DOES NOT PROVIDE ALARMS OR CONTINUOUS INDICATION OF RELEASE RATES. THE CAUSE OF THE EVENT IS INADEQUATE DESIGN COUPLED WITH A PERSONNEL ERROR. THE PUMP IS NOT PROVIDED WITH A DEDICATED POWER SUPPLY. AN INEXPERIENCED HEALTH PHYSICS TECHNICIAN MISTAKENLY DISCONNECTED AN EXTENSION CORD BEING USED TO POWER THE PUMP TEMPORARILY WHILE MAINTENANCE WAS BEING PERFORMED IN THE AREA. A DESIGN CHANGE HAS BEEN PREPARED TO PROVIDE A DEDICATED POWER SUPPLY FOR THE GASEOUS EFFLUENT MONITORING EQUIPMENT TO MINIMIZE THE POTENTIAL FOR LOSING POWER TO THE CIRCUIT. UNTIL THE DESIGN CHANGE IS IMPLEMENTED, ADDITIONAL INFORMATION TAGS HAVE BEEN INSTALLED TO MINIMIZE THE POTENTIAL FOR INADVERTENT DE-ENERGIZATION.

[33] CALVERT CLIFFS 2 DOCKET 50-318 LER 89-007 REV 02
UPDATE ON EVIDENCE OF LEAKAGE FROM PRESSURIZER HEATER PENETRATIONS DUE TO INTERGRANULAR STRESS CORROSION CRACKING CAUSED BY RESIDUAL FABRICATION STRESS.
EVENT DATE: 050589 REPORT DATE: 101990 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 1 (PWR)
VENDOR: COMBUSTION ENGINEERING, INC.

(NSIC 219803) AT 0820 HOURS MAY 5, 1989, AN IN-SERVICE INSPECTION OF THE UNIT 2 PRESSURIZER DISCOVERED EVIDENCE OF REACTOR COOLANT LEAKAGE FROM 28 OF THE 120 PRESSURIZER VESSEL HEATER PENETRATIONS AND ONE UPPER LEVEL NOZZLE. UNIT 2 WAS IN A REFUELING OUTAGE (MODE 6) AT THE TIME OF THE DISCOVERY. AT 0430 HOURS ON MAY 6, 1989, UNIT 1 WAS SHUTDOWN FROM 100 PERCENT POWER (MODE 1) TO ALLOW INSPECTION OF ITS PRESSURIZER. NO SIGNS OR EVIDENCE OF LEAKAGE WERE FOUND ON THE UNIT 1 PRESSURIZER HEATER PENETRATIONS OR PRESSURE/LEVEL PENETRATIONS. ADDITIONAL INSPECTIONS, INCLUDING DYE PENETRANT AND EDDY CURRENT TESTS, OF 28 UNIT 2 AND 12 UNIT 1 HEATER SLEEVES WERE CONDUCTED. THREE SLEEVES FROM UNIT 2 WERE DESTRUCTIVELY EXAMINED. THE CAUSE OF LEAKAGE WAS INTERGRANULAR STRESS CORROSION CRACKING OF INCONEL 600. ALL CRACKS WERE AXIAL AND DETERMINED TO HAVE MINIMAL SAFETY SIGNIFICANCE. REAMING AND REPAIR OPERATIONS ASSOCIATED WITH FABRICATING THE UNIT 2 PRESSURIZER APPEAR TO HAVE CONTRIBUTED TO THE CAUSE. ALL UNIT 2 PENETRATIONS USING J-WELDS AND INCONEL 600 WERE VISUALLY INSPECTED. ALL UNIT 1 PRESSURIZER PENETRATIONS WERE VISUALLY INSPECTED. THE UNIT 2 PRESSURIZER HEATER SLEEVES AND UPPER LEVEL NOZZLES WERE REPLACED.

[34] CATAWBA 1 DOCKET 50-413 LER 89-027 REV 01
UPDATE ON TECHNICAL SPECIFICATION REQUIRED UNIT SHUTDOWN DUE TO AN INOPERABLE CHEMICAL AND VOLUME CONTROL CENTRIFUGAL CHARGING PUMP.
EVENT DATE: 112089 REPORT DATE: 102590 NSSS: WE TYPE: PWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV
WESTINGHOUSE ELECTRIC CORP.

(NSIC 219805) ON NOVEMBER 20, 1989, AT APPROXIMATELY 1650 HOURS, WITH UNIT 1 IN MODE POWER OPERATION, AT 100% POWER, CHEMICAL AND VOLUME CONTROL (NV) SYSTEM CENTRIFUGAL CHARGING PUMP 1B WAS DECLARED INOPERABLE DUE TO ITS INABILITY TO MAINTAIN A SUFFICIENT CHARGING FLOW. SUBSEQUENTLY, A WORK REQUEST WAS INITIATED TO INVESTIGATE AND REPAIR NV PUMP 1B. ON NOVEMBER 22 AT APPROXIMATELY 1400 HOURS, WITH UNIT 1 IN MODE 1 AT 100% POWER, THE CONTROL ROOM OPERATORS (CROS) COMMENCED A POWER REDUCTION AS REQUIRED BY TECHNICAL SPECIFICATIONS DUE TO THE INOPERABILITY OF NV PUMP 1B. ON NOVEMBER 28, WITH UNIT 1 IN MODE 4, HOT SHUTDOWN, NV PUMP 1B REPAIR WORK WAS COMPLETED AND THE PUMP WAS DECLARED OPERABLE. THE REPAIR WORK INVOLVED REPLACING THE PUMP'S ROTATING ELEMENT. THE CROS THEN PROCEEDED TO PLACE THE UNIT IN MODE 5, COLD SHUTDOWN, FOR FURTHER OUTAGE RELATED WORK. ON NOVEMBER 30 AT 2246 HOURS, UNIT 1 ENTERED MODE 1 AND THE CROS BEGAN INCREASING REACTOR POWER TO 100%. THIS INCIDENT HAS BEEN ATTRIBUTED TO EQUIPMENT FAILURE. A FAILURE ANALYSIS WAS PERFORMED ON THE PUMP ROTATING ELEMENT. THIS EVALUATION DID NOT CONCLUSIVELY DETERMINE THE ROOT CAUSE OF THIS INCIDENT. HOWEVER, GAS ENTRAINMENT WAS POSTULATED AS A POSSIBLE CAUSE, AND ACTIONS HAVE BEEN TAKEN TO PRECLUDE THE POSSIBILITY OF GAS ENTRAINMENT.

[35] CATAWBA 1 DOCKET 50-413 LER 90-017
TECHNICAL SPECIFICATION 3.0.3 ENTRY DUE TO LOSS OF CONTROL ROD POSITION INDICATIONS DUE TO EQUIPMENT FAILURE.
EVENT DATE: 092690 REPORT DATE: 102390 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 219931) ON SEPTEMBER 26, 1990, AT 0535 HOURS, WITH UNIT 1 IN MODE 1, POWER OPERATION, THE ABILITY OF THE DIGITAL ROD POSITION INDICATION SYSTEM TO PROPERLY INDICATE THE POSITIONS OF SIX CONTROL RODS WAS LOST. ANNUNCIATORS INDICATING ROD POSITION INDICATION FAILURE WERE RECEIVED IN THE CONTROL ROOM. TECHNICAL SPECIFICATION 3.0.3 WAS ENTERED BECAUSE ROD POSITION INDICATION FOR MORE THAN ONE ROD IN A BANK WAS INOPERABLE. THE APPROPRIATE ABNORMAL PROCEDURE WAS ENTERED, AND CONTROL ROOM OPERATORS VERIFIED ACTUAL CONTROL ROD POSITIONS USING OTHER CONTROL ROOM INDICATIONS. A HIGH PRIORITY WORK REQUEST WAS INITIATED, AND INSTRUMENTATION AND ELECTRICAL PERSONNEL FOUND THAT A MALFUNCTIONING PRINTED CIRCUIT BOARD (DISPLAY CARD A130, FOR CONTROL ROD F-10) WAS CAUSING THE ERRONEOUS INDICATIONS. THE CARD WAS REMOVED BY 0600 HOURS, AT WHICH TIME NORMAL INDICATIONS WERE RESTORED FOR ALL RODS EXCEPT F-10, WHICH WAS RESTORED TO NORMAL INDICATION AT 0625 HOURS WHEN THE DISPLAY CARD WAS REPLACED. NEITHER CONTROL ROD POSITIONS NOR CONTROLS WERE AFFECTED BY THE MALFUNCTION. THIS INCIDENT IS ATTRIBUTED TO EQUIPMENT FAILURE/MALFUNCTION, DUE TO THE FAILURE OF DISPLAY CARD A130. CORRECTIVE ACTIONS INCLUDED ENTRY INTO THE APPROPRIATE ABNORMAL PROCEDURE, AND REPLACEMENT OF THE CARD.

[36] CATAWBA 2 DOCKET 50-414 LER 88-026 REV 02
UPDATE ON TECHNICAL SPECIFICATION REQUIRED SHUTDOWN DUE TO AN INOPERABLE CHEMICAL AND VOLUME CONTROL CENTRIFUGAL CHARGING PUMP.
EVENT DATE: 071388 REPORT DATE: 102590 NSSS: WE TYPE: PWR
VENDOR: PACIFIC PUMPS

(NSIC 219798) ON JULY 13, 1988, AT APPROXIMATELY 2145 HOURS, THE CONTROL ROOM OPERATOR (CRO) NOTED THE TREND RECORDERS INDICATED THAT VOLUME CONTROL TANK (VCT) LEVEL WAS INCREASING AND THE PRESSURIZER (PZR) LEVEL WAS DECREASING. CHEMICAL AND VOLUME CONTROL (NV) SYSTEM CENTRIFUGAL CHARGING PUMP 2A (NV PUMP 2A) WAS IN SERVICE AT THIS TIME. AFTER THE CRO STABILIZED THE PZR AND VCT LEVELS BY PLACING THE POSITIVE DISPLACEMENT CHARGING PUMP 2 (NV PUMP 2) IN SERVICE, NV PUMP 2A WAS SECURED AND NORMAL OPERATING CONDITIONS WERE RESTORED. THE UNIT WAS AT 100% POWER AT THE TIME OF THE INCIDENT. CONTROL ROOM PERSONNEL DETERMINED THAT THE MAXIMUM CHARGING FLOW ACHIEVABLE BY NV PUMP 2A WAS APPROXIMATELY 50 GPM, ALTHOUGH NORMAL CHARGING FLOW IS APPROXIMATELY 87 GPM. THEREFORE, NV PUMP 2A WAS DECLARED INOPERABLE AT 2248 HOURS, ON JULY 13, AND THE UNIT ENTERED A 72 HOUR ACTION STATEMENT. PUMP REPLACEMENT WAS SUBSEQUENTLY INITIATED. WHEN IT BECAME APPARENT THAT PUMP REPLACEMENT WOULD NOT BE ACHIEVED WITHIN 72 HOURS, A SHUTDOWN ON UNIT 2 WAS INITIATED AT APPROXIMATELY 1930 HOURS, ON JULY 16. SUBSEQUENTLY, ON JULY 20, NV PUMP 2A WAS REPLACED AND TESTED. AT 0937 HOURS, THE PUMP WAS DECLARED OPERABLE. THIS INCIDENT HAS BEEN ATTRIBUTED TO AN EQUIPMENT FAILURE DUE TO NV PUMP 2A INABILITY TO DELIVER THE REQUIRED CHARGING FLOW.

[37] COMANCHE 1 DOCKET 50-445 LER 90-031
FAILURE TO COMPLY WITH TECHNICAL SPECIFICATION ACTION STATEMENT DUE TO NON-CONSERVATIVE ALARM SETPOINTS.
EVENT DATE: 082290 REPORT DATE: 101790 NSSS: WE TYPE: PWR

(NSIC 219942) ON 7/23/90, THE CONTAINMENT ATMOSPHERE GASEOUS RADIOACTIVITY MONITORING SYSTEM (CAGRM) WAS DECLARED INOPERABLE DUE TO STEADY STATE BACKGROUND ACTIVITY LEVELS ABOVE THE ALERT SETPOINT. ON 8/10/90, THE CONTAINMENT AIR COOLER (CARCS) CONDENSATE FLOW RATE ALARM SETPOINTS WERE DETERMINED TO BE NON-CONSERVATIVE, AND WAS DECLARED INOPERABLE. WITH BOTH CARCS AND CAGRM INOPERABLE OPERATIONS PERSONNEL ENTERED TECH SPEC (TS) ACTION STATEMENT 3.4.5.1. CONTAINMENT ATMOSPHERE SAMPLING BEGAN AT THE TIME OF DISCOVERY. ON 8/27/90, A NEW CARCS CONDENSATE FLOW RATE ALARM SETPOINTS WERE INSTALLED AND CARCS DECLARED OPERABLE. RECONSTRUCTION OF THE EVENT, PRIOR TO DISCOVERY, SHOWS THAT CARCS HAD BEEN INOPERABLE DUE TO NON-CONSERVATIVE ALARM SETPOINTS SINCE 2/8/90. FURTHERMORE, CONTAINMENT ATMOSPHERE GRAB SAMPLES BETWEEN 7/23/90, AND 8/10/90

WERE NOT TAKEN, AND PLANT SHUTDOWN AFTER 30 DAYS (8/22/90) WAS NOT ACCOMPLISHED. NON-COMPLIANCE WITH THE TS ACTION STATEMENT WAS DUE TO CARCS INOPERABILITY NOT BEING DISCOVERED UNTIL 8/10/90. THE ROOT CAUSE WAS DETERMINED TO BE NON-CONSERVATIVE ASSUMPTIONS MADE IN ESTABLISHING THE ORIGINAL AND SUBSEQUENT CARCS CONDENSATE FLOW RATE ALARM SETPOINTS. CORRECTIVE ACTIONS INCLUDE A REVISION OF THESE ALARM SETPOINTS AND A MEMO TO ENGINEERING PERSONNEL ADDRESSING THESE CONCERNS.

[38] COMANCHE 1 DOCKET 50-445 LER 90-027
MANUAL REACTOR TRIP DUE TO SHEARING OF FEEDWATER FLOW CONTROL VALVE FEEDBACK LINKAGE ARM.
EVENT DATE: 090790 REPORT DATE: 100990 NSSS: WE TYPE: PWR
VENDOR: BAILEY CONTROLS CO.
COPES-VULCAN, INC.

(NSIC 219940) ON 9/7/90, AT 0033, COMANCHE PEAK STEAM ELECTRIC STATION (CPSES) UNIT 1 STEAM GENERATOR (SG) NUMBER (NO.) 2 FEEDWATER FLOW CONTROL VALVE (FCV) FAILED FULL OPEN DUE TO SHEARING OF THE POSITIONER FEEDBACK LINKAGE ARM. THE FAILED VALVE OVERFEED SG NO. 2 AND THE REACTOR WAS MANUALLY TRIPPED AT 0034 WITH SG NO. 2 LEVEL AT APPROXIMATELY 80% NARROW RANGE INDICATED LEVEL. THE PLANT WAS STABILIZED AT 0043 IN MODE 3. AT 0130, A BALANCE OF PLANT REACTOR OPERATOR (RO) DECREASED AUXILIARY FEEDWATER FLOW TO SG NO. 4 SINCE THE LEVEL WAS INCREASING FASTER THAN THE OTHER THREE. AT 0232, A RELIEF RO NOTED THAT SG NO. 4 LEVEL WAS APPROACHING THE LO-LO LEVEL SETPOINT AND INCREASED AUXILIARY FEEDWATER FLOW. THE LOW LEVEL COMBINED WITH THE INCREASED FLOW WHICH CAUSED A "SHRINK AND SWELL" EFFECT IN SG NO. 4 RESULTED IN LO-LO LEVEL SIGNAL WHICH GENERATED AN AUTOMATIC START SIGNAL FOR THE AUXILIARY FEEDWATER SYSTEM. THE CAUSE OF THE LINKAGE ARM FAILURE IS ATTRIBUTED TO FATIGUE RESULTING FROM FLOW INDICED OSCILLATIONS. CORRECTIVE ACTIONS INCLUDE REPAIR OF SG NO. 2 FCV AND A DESIGN MODIFICATION TO MODIFY THE VALVE INTERNALS TO REDUCE FLOW INDUCED OSCILLATIONS.

[39] COMANCHE 1 DOCKET 50-445 LER 90-030
REACTOR TRIP RESULTING FROM IMPROPER ROUTINE OF HEATER DRAIN PUMP CABLE SHIELD GROUND LEAD.
EVENT DATE: 091590 REPORT DATE: 101590 NSSS: WE TYPE: PWR

(NSIC 219941) ON 9/15/90, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 WAS IN MODE 1, POWER OPERATION, WITH REACTOR POWER AT 94%. OPERATIONS PERSONNEL WERE PREPARING TO RAISE REACTOR POWER TO SUPPORT SECONDARY SYSTEM PERFORMANCE EVALUATION AND TUNING. WHEN A SECOND CONDENSATE PUMP WAS STARTED, THE OPERATING HEATER DRAIN PUMP AND MAIN FEEDWATER PUMP TRIPPED. AS A RESULT OF DECREASING STEAM GENERATOR LEVELS, A MANUAL REACTOR TRIP WAS INITIATED. THE CAUSE OF THE EVENT WAS AN IMPROPERLY ROUTED HEAT DRAIN PUMP SHIELD GROUND LEAD WHICH CAUSED A FALSE GROUND OVERCURRENT SIGNAL. CORRECTIVE ACTION INCLUDED REWORK OF THE AFFECTED COMPONENT AND REVIEW OF SIMILAR COMPONENTS FOR THE SAME PROBLEM.

[40] COMANCHE 1 DOCKET 50-445 LER 90-032
FAILURE TO IDENTIFY PROPER DESIGN BASES RESULTED IN OPERATION AND TESTING OF THE CONTAINMENT PERSONNEL AIR LOCK HYDRAULIC SYSTEM INCONSISTENT WITH THE EXISTING DESIGN.
EVENT DATE: 091990 REPORT DATE: 102990 NSSS: WE TYPE: PWR

(NSIC 219943) ON 9/19/90, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 WAS IN MODE 1, POWER OPERATION, WITH REACTOR POWER AT 100%. AN ENGINEERING EVALUATION DETERMINED THAT PLANT DOCUMENTATION DID NOT ADEQUATELY ADDRESS THE QUALIFICATION OF THE PERSONNEL AIRLOCK (PAL) HYDRAULIC SYSTEM AND AS A RESULT THE DESIGN REQUIREMENTS FOR THE PAL HYDRAULIC SYSTEM TO FUNCTION AS A CONTAINMENT ISOLATION BARRIER WERE NOT ADEQUATELY IDENTIFIED AND ADDRESSED IN ENGINEERING AND OPERATING DOCUMENTATION. INSUFFICIENT ADMINISTRATIVE CONTROLS FOR OPERATION OF THE PAL HYDRAULIC SYSTEM HAD EXISTED SINCE 2/8/90, UNTIL 9/20/90, WHEN HYDRAULIC, SAFETY CLASS 2, ISOLATION VALVES WERE UTILIZED AS CONTAINMENT ISOLATION VALVES, TESTED SUCCESSFULLY AND LOCKED SHUT. CORRECTIVE ACTION INCLUDED ENGINEERING ANALYSIS TO PROVIDE DOCUMENTATION FOR THE HYDRAULIC SYSTEM AS MEETING SAFETY CLASS 2 AND

SEISMIC CATEGORY I REQUIREMENTS. A DESIGN MODIFICATION IS BEING PERFORMED TO ADD ADDITIONAL ISOLATION VALVES AND TO REPOSITION OTHER ISOLATION VALVES IN THE PAL HYDRAULIC SYSTEM. PLANT DOCUMENTATION IS BEING REVISED TO REFLECT THE NEW CONFIGURATION AND IS BEING REVIEWED FOR SIMILAR PROBLEMS.

[41] COMANCHE 1 DOCKET 50-445 LER 90-034
MISSED SURVEILLANCE FOR CHANNEL RESPONSE TIME DUE TO PERSONNEL ERROR.
EVENT DATE: 092890 REPORT DATE: 102990 NSSS: WE TYPE: PWR

(NSIC 219944) ON 1/16/90, DURING REPLACEMENT OF STEAM GENERATOR NUMBER 3 LEVEL TRANSMITTER, A TECHNICIAN FAILED TO TAKE ALL DATA REQUIRED TO SATISFY THE RELATED TECH SPEC SURVEILLANCE REQUIREMENTS. ON 9/28/90, WHILE PREPARING FOR A RESPONSE TIME SUMMATION TEST ON THE SAME CHANNEL, A TEST ENGINEER DISCOVERED THE OMISSION. THE CAUSES OF THE EVENT WERE PERSONNEL ERROR AND PROCEDURAL WEAKNESS. CORRECTIVE ACTIONS INCLUDED REVIEW OF SIMILAR TEST DATA, EVENT REVIEW BY AFFECTED PERSONNEL, AND PROCEDURAL ENHANCEMENT.

[42] CONNECTICUT YANKEE DOCKET 50-213 LER 89-006 REV 01
UPDATE ON HEATING STEAM CONTAINMENT ISOLATION VALVES FAILED SURVEILLANCE TEST.
EVENT DATE: 041489 REPORT DATE: 102490 NSSS: WE TYPE: PWR
VENDOR: CONTROMATICS CORP.

(NSIC 219799) ON APRIL 14, 1989, AT 0803, WITH THE PLANT IN MODE 1 AT 100% POWER, THE TWO CONTAINMENT ISOLATION VALVES FOR HEATING STEAM TO CONTAINMENT (HS-TV-380 AND 381) FAILED TO OPERATE DURING QUARTERLY SURVEILLANCE TESTING. THESE FAILURES CONSTITUTED A LOSS OF CONTAINMENT INTEGRITY. THE OPERATORS IMMEDIATELY CLOSED THE MANUAL ISOLATION VALVES FOR THIS PENETRATION AND COMMENCED A LOAD REDUCTION AT 0903. ONE OF THE VALVES WAS VERIFIED OPERABLE AT 0905 AND THE LOAD REDUCTION WAS TERMINATED AT 0911. AT THE TIME OF THE ORIGINAL LER, THE CAUSE OF THIS EVENT WAS UNKNOWN. AN ENGINEERING EVALUATION HAS SINCE DETERMINED THE CAUSE OF THE EVENT TO BE A COMBINATION OF AN UNDERSIZED AIR OPERATOR AND THE USE OF A SEAT MATERIAL THAT REQUIRES INCREASED OPERATING TORQUE AT HIGHER TEMPERATURES. SHORT TERM CORRECTIVE ACTION CONSISTED OF LOCKING THE TWO CONTAINMENT TRIP VALVES IN THE CLOSED POSITION. LONG TERM CORRECTIVE ACTION CONSISTS OF EVALUATING WHETHER THE VALVES SHOULD BE ELIMINATED, MODIFIED OR REPLACED.

[43] CONNECTICUT YANKEE DOCKET 50-213 LER 90-019
TWO OF THREE PRESSURIZER LEVEL CHANNELS DECLARED INOPERABLE.
EVENT DATE: 091090 REPORT DATE: 100990 NSSS: WE TYPE: PWR

(NSIC 219692) ON 9/10/90, AT 1130 HOURS, WITH THE PLANT IN MODE 1 AT 80% POWER, TWO OF THREE PRESSURIZER LEVEL CHANNELS WERE DECLARED INOPERABLE BASED ON CHANNELS 1 AND 2 READING 4% AND 5%, RESPECTIVELY, LOWER THAN CHANNEL 3. IN ACCORDANCE WITH PLANT TECH SPECS A LOAD REDUCTION WAS COMMENCED AT 1230. AT APPROXIMATELY 1400 THE PRESSURIZER HIGH LEVEL REACTOR TRIP SETPOINTS WERE LOWERED, THE TWO CHANNELS WERE DECLARED OPERABLE AND THE LOAD REDUCTION WAS TERMINATED. THE CAUSE OF THE EVENT WAS DETERMINED TO BE SUBTLE DIFFERENCES IN THE CALIBRATION METHODS USED FOR EACH CHANNEL. DURING AN UNSCHEDULED SHUTDOWN ON 9/20/90 THE TRANSMITTER FOR CHANNEL 1 WAS REPLACED AND CHANNELS 2 AND 3 WERE RECALIBRATED BASED ON THE MANUFACTURER'S RECOMMENDATIONS. LONG TERM CORRECTIVE ACTION CONSISTS OF REVISING CALIBRATION PROCEDURES, CONTINUED CHANNEL MONITORING AND ASSESSING THE NEED TO MODIFY THE DESIGN OF THE REFERENCE LEGS. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(VII)(A) SINCE TWO CHANNELS WERE INOPERABLE IN A SINGLE SYSTEM DESIGNED TO SHUT DOWN THE REACTOR.

[44] CONNECTICUT YANKEE DOCKET 50-213 LER 90-020
MANUAL PLANT TRIP DUE TO CONDENSATE PUMP DEGRADATION AND PROCEDURAL NONCOMPLIANCE.
EVENT DATE: 092090 REPORT DATE: 101990 NSSS: WE TYPE: PWR
VENDOR: PROTECTIVE COATINGS

(NSIC 219809) ON SEPTEMBER 20, 1990, AT 0715 HOURS, WITH THE PLANT IN MODE 1 AT 50 PERCENT POWER THE "B" CONDENSATE PUMP WAS SHUT DOWN FOR TROUBLESHOOTING. THE

"A" CONDENSATE PUMP WAS UNABLE TO PROVIDE SUFFICIENT PRESSURE TO THE SUCTION OF THE MAIN FEED PUMP REQUIRING CONTROL ROOM OPERATORS TO MANUALLY TRIP THE PLANT. THE ROOT CAUSES OF THE EVENT CONSISTED OF THE DEGRADATION OF THE RUBBER FLEXIBLE COUPLINGS IN THE CONDENSATE PUMP SUCTION PIPING TO THE POINT THAT FLOW WAS RESTRICTED AND PERSONNEL ERROR SINCE THE PUMP WAS SHUT DOWN AT A HIGHER POWER LEVEL THAN THAT REQUIRED BY PROCEDURE. CORRECTIVE ACTION INCLUDED REPLACING THE FLEXIBLE RUBBER COUPLINGS WITH STAINLESS STEEL COUPLINGS AND RE-EMPHASIZING THE IMPORTANCE OF PROCEDURAL COMPLIANCE TO ALL OPERATORS. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(IV) SINCE IT RESULTED IN MANUAL ACTUATION OF THE REACTOR PROTECTION SYSTEM.

[45] CONNECTICUT YANKEE DOCKET 50-213 LER 90-021
SURVEILLANCE FREQUENCY EXCEEDED FOR TURBINE BUILDING HEAT DETECTOR TEST.
EVENT DATE: 100290 REPORT DATE: 103090 NSSS: WE TYPE: PWR

(NSIC 219810) ON OCTOBER 2, 1990, AT 0930 HOURS, WITH THE PLANT IN MODE 1 AT 100 PERCENT POWER MAINTENANCE DEPARTMENT PERSONNEL, WHILE PERFORMING A MONTHLY CHECK ON THE STATUS OF SURVEILLANCES, DETERMINED THAT THE MONTHLY TURBINE BUILDING DELUGE SPRAY SYSTEM HEAT DETECTOR SURVEILLANCE TEST WAS NOT PERFORMED FOR THE MONTH OF SEPTEMBER AS REQUIRED BY THE PLANT'S TECHNICAL SPECIFICATIONS. THE ROOT CAUSE WAS PERSONNEL ERROR IN THAT ALTHOUGH THE TEST WAS SCHEDULED TO BE PERFORMED IT WAS TEMPORARILY POSTPONED DUE TO MANPOWER CONSTRAINTS AND NOT RESCHEDULED. THE SURVEILLANCE TEST WAS IMMEDIATELY STARTED AND THE TEST RESULTS WERE SATISFACTORY. CORRECTIVE ACTION CONSISTS OF REVISING THE MAINTENANCE DEPARTMENT SCHEDULING CHART TO INCLUDE SPECIFIC COMPLETION DUE DATES FOR SURVEILLANCES AND COUNSELING APPROPRIATE MAINTENANCE PERSONNEL ON THE IMPORTANCE OF COMPLETING ALL SURVEILLANCES PRIOR TO THEIR DUE DATE. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B) SINCE IT RESULTED IN A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[46] COOK 1 DOCKET 50-315 LER 90-010
10CFR50 APPENDIX R DEFICIENCIES RESULTING IN POTENTIAL LOSS OF LOCAL SHUTDOWN INDICATION PANEL FUNCTION.
EVENT DATE: 082490 REPORT DATE: 092190 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)

(NSIC 219723) ON AUGUST 24, 1990 WITH UNIT ONE OPERATING AT 100 PERCENT POWER AND UNIT TWO IN MODE SIX, IT WAS DISCOVERED THAT THE ROUTING OF CABLE ASSOCIATED WITH THE UNIT ONE LOCAL SHUTDOWN INDICATION (LSI) PANELS WAS NOT IN COMPLIANCE WITH 10CFR50 APPENDIX R. ON SEPTEMBER 6, 1990, IT WAS SUBSEQUENTLY DISCOVERED THAT A SIMILAR CONDITION EXISTED FOR THE UNIT TWO LSI PANELS. THE IMMEDIATE CORRECTIVE ACTION TAKEN WAS INITIATION OF PLANT MODIFICATIONS TO BRING THE SUBJECT PLANT CABLING INTO COMPLIANCE WITH 10CFR50 APPENDIX R. ALL IDENTIFIED AREAS ARE PROVIDED WITH ADEQUATE FIRE DETECTION AND SUPPRESSION TO SUBSTANTIALLY MITIGATE THE IMPACT OF A FIRE ON NORMAL AND LSI INSTRUMENTATION. THEREFORE, THE EVENT SHOULD NOT HAVE PRESENTED A SIGNIFICANT HAZARD TO THE PUBLIC HEALTH AND SAFETY.

[47] COOK 1 DOCKET 50-315 LER 90-011
MISSED STEAM GENERATOR PRESSURE CHANNEL CHECK SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 100490 REPORT DATE: 110290 NSSS: WE TYPE: PWR

(NSIC 219857) ON OCTOBER 4, 1990, AN OPERATOR CONDUCTING THE SHIFTLY SURVEILLANCE INCORRECTLY LOGGED THE STEAM GENERATOR LEVELS IN THE STEAM GENERATOR PRESSURE CHANNEL SECTION OF THE SURVEILLANCE FOR THE LOOP 3 AND LOOP 4 CHANNELS. THE REVIEW OF THE SURVEILLANCE BY THE OPERATOR AND THE UNIT SUPERVISOR DID NOT IDENTIFY THE ERROR. AS A RESULT OF THE FAILURE OF THE OPERATOR AND THE UNIT SUPERVISOR TO RECOGNIZE THE LOGGING ERROR, THE TECHNICAL SPECIFICATION REQUIREMENT TO PERFORM A CHANNEL CHECK OF THE REDUNDANT STEAM GENERATOR PRESSURE CHANNELS WAS NOT COMPLETED. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THE INVOLVED OPERATOR AND UNIT SUPERVISOR FAILED TO CONFIRM THE PROPER SURVEILLANCE READINGS DURING THEIR REVIEW TO ENSURE THE TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS WERE SATISFIED. WHEN THE ERROR WAS DISCOVERED, A CHANNEL CHECK OF THE REDUNDANT STEAM GENERATOR PRESSURE CHANNELS WAS COMPLETED. THIS CHANNEL

CHECK CONFIRMED THAT THE REDUNDANT CHANNELS OF THE AFFECTED S/G PRESSURES WERE WITHIN REQUIRED LIMITS DURING THE TIME OF THE SURVEILLANCE.

[48] COOK 2 DOCKET 50-316 LER 89-008 REV 01
UPDATE ON REACTOR CABLE TUNNEL CO(2) FOR QUADRANTS 1, 3 AND 4 INOPERABLE WITHOUT REQUIRED FIREWATCH DUE TO FALSE INDICATION OF STATUS CAUSED BY WORN ISOLATION SWITCH KEY.
EVENT DATE: 031189 REPORT DATE: 092890 NSSS: WE TYPE: PWR

(NSIC 219800) ON 3/11/89, AT 0821 HOURS, IT WAS IDENTIFIED THAT AUTOMATIC CARBON DIOXIDE (CO2) ACTUATION SYSTEM FOR REACTOR CABLE TUNNEL QUADRANTS 1, 3, AND 4, HAD BEEN ISOLATED SINCE 0403 HOURS WITHOUT THE REQUIRED FIREWATCH. CO2 AUTOMATIC ACTUATION WAS ISOLATED VIA LOCAL KEY LOCK ISOLATION SWITCH AT 0357 TO ALLOW PERSONNEL ENTRY. AFTER PERSONNEL EXITED THE AREA, KEY LOCK SWITCH WAS RESTORED TO WHAT WAS THOUGHT TO BE NORMAL POSITION. THIS WAS VERIFIED BY THE LOCAL INDICATING LIGHT EXTINGUISHING WHEN THE KEY LOCK SWITCH WAS TURNED. A SECOND PERSON VERIFIED BY VISUAL OBSERVATION THAT THE SWITCH HAD BEEN RESTORED TO THE NORMAL POSITION. AT 0831, PERMISSION WAS REQUESTED TO ISOLATE THE REACTOR CABLE TUNNEL CO2 SYSTEM. CONTROL ROOM OPERATOR IDENTIFIED THAT ASSOCIATED "CO2 ISOLATED" ANNUNCIATOR ALARM WAS ALREADY IN. IT WAS IDENTIFIED THAT THE KEY LOCK SWITCH WAS NOT FULLY IN THE NORMAL POSITION. ROOT CAUSE OF THIS WAS AN EXCESSIVELY WORN KEY WHICH ALLOWED THE KEY TO BE REMOVED PRIOR TO THE SWITCH BEING FULLY IN THE NORMAL POSITION AND WITH ONLY ONE OF THE TWO SWITCH CONTACTS MADE UP (ONE SET OF CONTACTS IS FOR THE LOCAL INDICATING LIGHT, THE OTHER SET IS FOR THE CO2 ISOLATION AND CONTROL ROOM ANNUNCIATOR). THE WORN KEY WAS REPLACED AND A POLICY HAS BEEN INSTITUTED TO REPLACE AFFECTED KEYS ON A PERIODIC BASIS.

[49] COOK 2 DOCKET 50-316 LER 89-017 REV 02
UPDATE ON LOSS OF TURBINE DRIVEN AUXILIARY FEED PUMP FLOW RETENTION DUE TO INACCURATE FLOW MEASUREMENT.
EVENT DATE: 101989 REPORT DATE: 110290 NSSS: WE TYPE: PWR
VENDOR: BARTON INSTRUMENT CO., DIV OF ITT
VICKERY SIMMS, INC.

(NSIC 219801) THIS REVISION IS BEING SUBMITTED TO PROVIDE ADDITIONAL INFORMATION REGARDING THE CORRECTIVE ACTIONS TAKEN AND UPDATE THE CAUSE DESCRIPTION. ON OCTOBER 19, 1989 WITH UNIT 2 IN MODE 1 (POWER OPERATION) AT 100 PERCENT RATED THERMAL POWER, DURING SURVEILLANCE TESTING, AN NRC INSPECTOR CONDUCTING AN IST AUDIT DISCOVERED AN INSTRUMENT DISCREPANCY BETWEEN THE TURBINE DRIVEN AUXILIARY FEEDPUMP (TDAFP) TEST LINE FLOW INDICATION AND THE PROCESS FLOW INDICATION. THE PROCESS FLOW INSTRUMENT INDICATED A FLOW OF 500 GPM WHILE ACTUAL FLOW WAS 700 GPM. THE PROCESS FLOW INSTRUMENTATION ACTUATES A FLOW RETENTION SIGNAL WHEN THE TDAFP FLOW REACHES 975 GPM TO PREVENT PUMP RUNOUT. THE FLOW RETENTION FUNCTION WOULD HAVE ACTUATED AT A TDAFP FLOW OF APPROXIMATELY 1225 GPM AND WOULD NOT HAVE PREVENTED PUMP RUNOUT, IN THE EVENT OF AN ACCIDENT SUCH AS A FEEDWATER LINE BREAK. THE FLOW AND PROCESS INSTRUMENTATION FOR THE OTHER UNIT 1 AND 2 AUXILIARY FEEDWATER PUMPS WAS CHECKED, NO SIMILAR DEFICIENCIES EXIST. THE FLOW RETENTION ACTUATION SETPOINT WAS RESET TO AN ACCEPTABLE VALUE. THE CAUSE FOR THE FLOW INSTRUMENT ERROR CANNOT BE DETERMINED AT THIS TIME AND THE INVESTIGATION IS CONTINUING THROUGH THE UNIT 1 REFUELING OUTAGE.

[50] COOK 2 DOCKET 50-316 LER 90-009
POTENTIAL LOSS OF CONTROL ROOM HVAC DURING POSTULATED FIRE, WITHOUT COMPENSATORY ACTION, DUE TO OVERSIGHT IN APPENDIX R LOSS OF HVAC STUDY.
EVENT DATE: 092190 REPORT DATE: 101990 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 1 (PWR)

(NSIC 219897) ON 9/21/90 DURING AN INVESTIGATION CONCERNING THE NEED FOR CONTROL ROOM HEATING, VENTILATION AND AIR-CONDITIONING (HVAC) SYSTEM FOR SAFE SHUTDOWN OF EITHER UNIT, IT WAS DETERMINED THAT A SINGLE FIRE IN FIRE ZONES 44N/44S, 51/52 OR 69 COULD CAUSE A LOSS OF BOTH PRIMARY AND REDUNDANT HVAC SYSTEMS FOR BOTH CONTROL ROOMS WHICH IS NOT CURRENTLY COVERED BY PLANT PROCEDURES. IMMEDIATELY UPON DETERMINATION THAT THE PROBLEM EXISTED, FIRE WATCHES WERE POSTED FOR THE AFFECTED

AREAS. THE LONG-TERM CORRECTIVE ACTION IS TO INSTITUTE PROCEDURES TO COPE WITH FIRE-INDUCED LOSS OF NORMAL CONTROL ROOM HVAC. THE PRIMARY CAUSE OF THE CONDITION WAS AN OVERSIGHT IN THE HVAC SYSTEMS EVALUATED FOR THE APPENDIX R LOSS OF HVAC STUDY. WITHOUT IN-PLACE PROCEDURES AND TRAINING, THE EXACT COURSE OF EVENTS FOR THE POSTULATED FIRE CANNOT BE DETERMINED. HOWEVER, IF REASONABLE OPERATOR ACTIONS ARE TAKEN TO MITIGATE A RISE IN CONTROL ROOM TEMPERATURE FOLLOWING A FIRE, THE CONTROL ROOM(S) WOULD NOT REQUIRE EVACUATION DUE TO HABITABILITY OR EQUIPMENT OPERABILITY CONCERNS FOR A NUMBER OF HOURS. BASED ON THIS, WE BELIEVE THIS CONDITION DOES NOT REPRESENT A SIGNIFICANT HAZARD TO THE HEALTH AND SAFETY OF THE PUBLIC.

[51] COOK 2 DOCKET 50-316 LER 90-010
PLANT OUTSIDE DESIGN BASIS DUE TO DOWNGRADING OF POLAR CRANE HOISTS.
EVENT DATE: 100590 REPORT DATE: 110290 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 1 (PWR)

(NSIC 219898) ON 10/5/90, IT WAS DETERMINED THAT A PART 21 NOTIFICATION FILED BY THE WHITING CRANE COMPANY RESULTS IN A REPORTABLE CONDITION PER 10 CFR 50.73(A)(2)(II)(B) AS A "CONDITION THAT WAS OUTSIDE THE DESIGN BASIS OF THE PLANT." THE PART 21 NOTIFICATION INFORMED THE PLANT THAT AS A RESULT OF A DESIGN ERROR, THE POLAR CRANES WERE DERATED FROM 250 TONS TO 55 TONS FOR THE MAIN HOISTS AND FROM 35 TONS TO 6 TONS FOR THE AUXILIARY HOISTS. THE DESIGN ERROR WAS DISCOVERED DURING AN ANALYSIS WHICH REVEALED SIGNIFICANT OVERSTRESS ON CONNECTION POINTS OF THE CRANE TROLLEY. THE OVERSTRESSED CONNECTIONS CAN RESULT IN METAL FATIGUE AND SUBSEQUENT FAILURE. THE CORRECTIVE ACTIONS INVOLVED THE REMOVAL OF THE SUBJECT CRANES FROM GENERAL SERVICE UNTIL THE TIME REPAIRS COULD BE MADE. REPAIRS HAVE BEEN COMPLETED TO RETURN THE CRANES TO THEIR ORIGINAL RATING. THE UNIT 1 MODIFICATIONS WERE COMPLETED ON 10-11-90, AND THE UNIT 2 MODIFICATIONS WERE COMPLETED ON 9/10/90.

[52] CRYSTAL RIVER 3 DOCKET 50-302 LER 90-014
LACK OF KNOWLEDGE CAUSES AUXILIARY NUCLEAR OPERATOR TO DE-ENERGIZE CONTAINMENT ISOLATION VALVE PRIOR TO BEING FULLY SEATED.
EVENT DATE: 091890 REPORT DATE: 101890 NSSS: BW TYPE: PWR
VENDOR: LIMITORQUE CORP.

(NSIC 219853) ON 9/18/90 AT 1845, WHILE CRYSTAL RIVER UNIT 3 WAS OPERATING IN MODE I (POWER OPERATION) AT 95% REACTOR POWER, VALVE DHV-43 WAS DISCOVERED PARTIALLY OPEN. THIS VALVE ISOLATES THE REACTOR BUILDING SUMP FROM THE SUCTION HEADER OF THE "B" DECAY HEAT REMOVAL SYSTEM. THE PARTIALLY OPEN VALVE WAS A VIOLATION OF TECHNICAL SPECIFICATION 3.6.1.1 WHICH SPECIFIES THAT CONTAINMENT INTEGRITY MUST BE MAINTAINED IN MODE I. DHV-43 WAS NOT FULLY SEATED FOLLOWING ITS USE AS A DRAIN PATH AT 0430 ON 09/18/90. WHEN THE VALVE "CLOSED" INDICATION WAS RECEIVED, THE AUXILIARY NUCLEAR OPERATOR OPENED THE BREAKER, REMOVING CLOSING POWER FROM THE MOTOR-OPERATOR PRIOR TO THE VALVE BEING FULLY SEATED. THE "CLOSED" LIGHT IS OPERATED BY A GEARED LIMIT SWITCH BUT THE MOTOR-OPERATOR CLOSING POWER IS CONTROLLED BY A TORQUE SWITCH. THE VALVE MOTOR-OPERATOR HAD NOT DEVELOPED SUFFICIENT TORQUE TO BE FULLY SEATED. CONTAINMENT INTEGRITY WAS RESTORED WITHIN 15 MINUTES OF DISCOVERY OF THE NON-CONFORMANCE. CORRECTIVE ACTION INCLUDES ENHANCING THE TRAINING GIVEN TO LICENSED AND NON-LICENSED OPERATORS.

[53] CRYSTAL RIVER 3 DOCKET 50-302 LER 90-015
DECAY HEAT VALVE ENCLOSURES AS BUILT DO NOT MATCH FSAR DESCRIPTION.
EVENT DATE: 092590 REPORT DATE: 102590 NSSS: BW TYPE: PWR

(NSIC 219855) ON SEPTEMBER 25, 1990, CRYSTAL RIVER UNIT 3 WAS IN MODE 1 (POWER OPERATIONS) AT 97% REACTOR POWER. AT 1445, IT WAS CONFIRMED THAT THE ENCLOSURES AROUND THE ISOLATION VALVES FOR THE REACTOR CONTAINMENT BUILDING (RB) SUMP RECIRCULATION LINES TO THE DECAY HEAT PUMP SUCTION DID NOT MATCH THE DESCRIPTION IN THE FINAL SAFETY ANALYSIS REPORT (FSAR). THE FSAR SECTION THAT DESCRIBES THESE ENCLOSURES WAS REVISED IN JUNE OF LAST YEAR. THE ENCLOSURE DESIGN HAS NOT BEEN MODIFIED. A SEARCH OF DESIGN CRITERIA BY THE ORIGINAL ARCHITECT/ENGINEERS HAS TO THIS POINT NOT PROVED THE FSAR CORRECT OR ERRONEOUS. A SEARCH OF

HISTORICAL DOCUMENTS WILL CONTINUE UNTIL FLORIDA POWER CORPORATION CAN RESOLVE THE ISSUE AND DETERMINE IF, IN FACT, THE PLANT WAS OUTSIDE THE DESIGN BASIS OR IF THERE WAS AN ERROR MADE IN REVISING THE FSAR. A SUPPLEMENT TO THIS REPORT PROVIDING RESOLUTION IS EXPECTED TO BE SUBMITTED ON JANUARY 25, 1991.

[54] DAVIS-BESSE 1 DOCKET 50-346 LER 89-007 REV 01
UPDATE ON ONE OF FOUR STRONG MOTION TRIAXIAL ACCELEROMETERS INOPERABLE FOR MORE THAN 30 DAYS.
EVENT DATE: 061489 REPORT DATE: 102590 NSSS: BW TYPE: PWR
VENDOR: TERRA TECHNOLOGY CORP.

(NSIC 219804) ON JUNE 14, 1989, DURING THE PERFORMANCE OF THE SEISMIC MONITORING SYSTEM CHANNEL CHECK, ST 5034.01, THE SYSTEM FAILED WHEN A +12VDC POWER SUPPLY FUSE BLEW. THE SOURCE OF THE PROBLEM WAS DETERMINED TO BE ZT2951, A STRONG MOTION TRIAXIAL ACCELEROMETER MOUNTED INSIDE CONTAINMENT AT THE 653 FT. ELEVATION. ZT2951 WAS ISOLATED, AND THE REMAINDER OF THE SEISMIC MONITORING SYSTEM RE-ENERGIZED WITH THREE OF THE FOUR TRIAXIAL ACCELEROMETERS AND THREE OF THREE PEAK ACCELEROMETERS OPERABLE. THIS REPORT COMPLETES THE TECHNICAL SPECIFICATION 3.3.3.3 ACTION STATEMENT 'A', WHICH REQUIRES THAT IF A SEISMIC INSTRUMENT HAS BEEN INOPERABLE FOR MORE THAN 30 DAYS, SUBMIT A SPECIAL REPORT PURSUANT TO SPECIFICATION 6.9.2 WITHIN THE NEXT 10 DAYS. DUE TO ALARA CONCERNS TROUBLESHOOTING AND MAINTENANCE OF THE COMPONENTS INSIDE CONTAINMENT WERE DELAYED UNTIL THE SIXTH REFUELING OUTAGE. THE EXAMINATION OF THE FAILED ZT2951 REVEALED A BURNED COMPONENT ON ITS VOLTAGE REGULATOR CIRCUIT BOARD. IT APPEARS THAT CONTACT OCCURRED BETWEEN THE COMPONENT AND A BOARD MOUNTING SCREW. OTHER SIMILAR BOARDS WERE INSPECTED. ALTHOUGH THEY ARE SIMILARLY CONFIGURED, THERE IS ADEQUATE CLEARANCE ON THESE BOARDS. THE LACK OF CLEARANCE APPARENTLY OCCURRED DURING ASSEMBLY OF THE CIRCUIT BOARD BY THE MANUFACTURE.

[55] DAVIS-BESSE 1 DOCKET 50-346 LER 90-014
INCOMPLETE DAILY HEAT BALANCE CALIBRATION TEST OF ONE OF FOUR POWER RANGE NUCLEAR INSTRUMENTS.
EVENT DATE: 092690 REPORT DATE: 103190 NSSS: BW TYPE: PWR

(NSIC 219904) ON 10/3/90, DURING THE REVIEW OF THE DAILY HEAT BALANCE CALIBRATION (DB-PF-03230) PERFORMED ON 9/26/90, A REACTOR ENGINEER NOTED THAT THE TEST WAS INCOMPLETE. ONLY THREE OF THE FOUR POWER RANGE NI'S WERE AVAILABLE AT THE TIME OF THE TEST BECAUSE NI-5 WAS BEING SIMULATED AND VARIED AS PART OF ANOTHER ROUTINE TEST, DB-MI-03058, REACTOR PROTECTION SYSTEM (RPS) CH2 QUARTERLY CALIBRATION. THE HEAT BALANCE CALIBRATION CHECK OF NI-5 WAS NOT COMPLETED BEFORE ITS TECH SPEC LATE TIME WAS EXCEEDED. THE CONTROL ROOM OPERATOR AND HIS SUPERVISOR SIGNED THE TEST AS COMPLETE BASED ON KNOWING THAT EVEN WITHOUT NI-5, THE MINIMUM CHANNELS OPERABLE REQUIREMENT OF THE TECH SPEC 3.3.1.1 WAS STILL BEING SATISFIED. HOWEVER, FAILING TO COMPLETE THE TECH SPEC TABLE 4.3-1, ITEM 2, SURVEILLANCE REQUIREMENT IS REPORTABLE AS AN LER UNDER 10CFR50.73(A)(2)(I)(B) AS A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS. THE RPS QUARTERLY CALIBRATION PROCEDURES, DB-MI-03057, 03058, 03059, AND 03060, WILL HAVE A NOTE ADDED TO ASSURE THE DAILY HEAT BALANCE CALIBRATION, DB-PF-03230, HAS BEEN PERFORMED JUST PRIOR TO STARTING. DB-PF-03230 WILL BE REVISED TO PROVIDE ENHANCED CLARITY IN THE ACCEPTANCE CRITERIA REQUIREMENTS.

[56] DIABLO CANYON 1 DOCKET 50-275 LER 84-045
VIOLATION OF CONTAINMENT FAN COOLER UNIT TECHNICAL SPECIFICATIONS DUE TO PERSONNEL ERROR.
EVENT DATE: 022084 REPORT DATE: 101290 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: DIABLO CANYON 2 (PWR)

(NSIC 219794) FEBRUARY 20, 1984, AND ON JULY 19, 1985, RESPECTIVELY, DIABLO CANYON POWER PLANT (DCPP) UNITS 1 AND 2 ENTERED MODE 4 (HOT SHUTDOWN) DURING PLANT HEATUP WITHOUT ADEQUATELY VERIFYING COOLING WATER FLOW TO EACH CONTAINMENT FAN COOLER UNIT (CFCU) REQUIRED BY TECHNICAL SPECIFICATIONS (TS) 4.6.2.3 AND 4.0.4. SURVEILLANCE TEST PROCEDURE (STP) M-51 IMPLEMENTS TS 4.6.2.3.A.2, WHICH REQUIRES VERIFICATION OF A 2000 GPM COOLING WATER FLOW RATE TO EACH CFCU COOLER.

THE DESIGN BASES OF THE CFCU ALSO REQUIRE 50 GPM TO THE MOTOR, FOR A TOTAL OF 2050 GPM TO EACH CFCU. STP M-51 HAS BEEN MEASURING THE COMBINED FLOW FROM THE COOLER AND MOTOR, THUS THE ACTUAL FLOW TO THE COOLERS HAS NOT BEEN VERIFIED. THIS TS VIOLATION WAS DISCOVERED ON SEPTEMBER 13, 1990, WHEN NECS ENGINEERING DETERMINED THAT STP M-51 DID NOT ADEQUATELY IMPLEMENT TS 4.6.2.3.A.2. THIS TS VIOLATION WAS CAUSED BY PERSONNEL ERROR. THE DESIGN BASES OF THE SYSTEM WERE NOT ADEQUATELY IMPLEMENTED DURING THE PREPARATION OF THE STP. TO PREVENT RECURRENCE, STP M-51 WAS REVISED TO ACCOUNT FOR MOTOR COOLER FLOW, AND ALL STPS IMPLEMENTING MULTIPLE FLOW REQUIREMENTS WILL BE REVIEWED TO ENSURE THAT THE STPS ARE IMPLEMEN..NG THE REQUIRED DESIGN BASES.

[57] DIABLO CANYON 2 DOCKET 50-323 LER 90-008
DEGRADATION OF RESIDUAL HEAT REMOVAL SYSTEM VALVES DUE TO IMPROPER WOODRUFF KEY MATERIAL.
EVENT DATE: 040390 REPORT DATE: 102690 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: DIABLO CANYON 1 (PWR)
VENDOR: FISHER CONTINENTAL

(NSIC 219861) THIS VOLUNTARY LER IS BEING SUBMITTED FOR INFORMATION PURPOSES ONLY AS DESCRIBED IN ITEM 19 OF SUPPLEMENT 1 TO NUREG-1022. ON APRIL 3, 1990, UNIT 2 RESIDUAL HEAT REMOVAL (RHR) HEAT EXCHANGER BYPASS VALVE RHR-2-HCV-670 WAS NOT CAPABLE OF BEING POSITIONED TO CONTROL FLOW DUE TO FAILURE OF THE WOODRUFF KEY CONNECTING THE VALVE SHAFT TO THE ACTUATOR LEVER ARM. SUBSEQUENT EVALUATION CONCLUDED THAT THE MINIMUM TECHNICAL SPECIFICATION REQUIRED FLOW FOR AN RHR TRAIN COULD NOT HAVE BEEN GUARANTEED HAD THE KEY BEEN INSTALLED AND SUBSEQUENTLY FAILED IN ANOTHER OF SEVERAL SIMILAR RHR VALVES. THE ROOT CAUSE WAS DETERMINED TO BE IMPROPER VENDOR-SUPPLIED KEY MATERIAL. THIS OCCURRED AS A RESULT OF THE VENDOR ORDERING KEY MATERIAL AS COMMERCIAL GRADE FROM A SUBSUPPLIER AND NOT ADEQUATELY VERIFYING THE MATERIAL. THE KEY WAS REPLACED. CORRECTIVE ACTIONS INCLUDE IDENTIFICATION AND INSPECTION OF OTHER POTENTIALLY AFFECTED VALVES AND REMOVING REDUNDANT DRAWINGS FROM THE DRAWING SYSTEM THAT DO NOT SPECIFY USE OF AN ALLOY STEEL KEY. A REQUEST WAS MADE TO THE VENDORS THAT THEY CONSIDER ISSUING A REPORT IN ACCORDANCE WITH 10 CFR 21 REGARDING THIS SUBJECT.

[58] DRESDEN 2 DOCKET 50-237 LER 90-009
TYPE B AND C PRIMARY CONTAINMENT LOCAL LEAK RATE TEST REQUIREMENTS EXCEEDED DUE TO LEAKING ISOLATION VALVES.
EVENT DATE: 092390 REPORT DATE: 101690 NSSS: GE TYPE: BWR
VENDOR: CRANE COMPANY

(NSIC 219812) ON SEPTEMBER 23, 1990, WITH UNIT 2 IN A REFUELING OUTAGE AND DURING PERFORMANCE OF MAIN STEAM ISOLATION VALVE (MSIV) LOCAL LEAK RATE TESTING (LLRT), THE A, C, AND D MAIN STEAM LINE VOLUMES WERE FOUND TO BE LEAKING IN EXCESS OF THE TECHNICAL SPECIFICATION LIMIT OF 11.5 SCFH. FURTHER DIAGNOSTIC TESTING INDICATED THAT MSIVS 2-203-1A, 1D, AND 2C WERE THE LEAKING VALVES. THE CAUSE OF THE LEAKAGE IS AT THIS TIME UNKNOWN. A SUPPLEMENT TO THIS REPORT WILL BE SUBMITTED OUTLINING THE CAUSE OF THE FAILURE, RETEST RESULTS, AND THE FINAL RESULTS FOR ALL TYPE B AND C LLRTS. ON SEPTEMBER 25, 1990, WHILE PERFORMING LLRT OF PRIMARY CONTAINMENT ISOLATION VALVES, OUTBOARD PRIMARY CONTAINMENT DRYWELL SPRAY VALVE 2-1501-27A LEAKED AN UNDETERMINED AMOUNT. THIS CAUSED THE AS-FOUND TOTAL TYPE B AND C LEAKAGE RATE TO BE IN EXCESS OF THE TECHNICAL SPECIFICATION LIMIT. AFTER FLUSHING OF THE VALVE SEAT, THE LEAK RATE WAS REDUCED TO A MINIMAL LEVEL. THE SAFETY SIGNIFICANCE FOR BOTH EVENTS IS MINIMAL SINCE IN EACH CASE THE IN LINE ISOLATION VALVES WERE NOT OBSERVED TO BE LEAKING. A PREVIOUS SIMILAR EVENT IS DOCUMENTED IN LER 89-009/050249.

[59] DRESDEN 2 DOCKET 50-237 LER 90-010
2B CORE SPRAY PUMP AUTOMATIC START DUE TO MANAGEMENT DEFICIENCY.
EVENT DATE: 100390 REPORT DATE: 102390 NSSS: GE TYPE: BWR

(NSIC 219813) ON 10/3/90, WITH UNIT 2 IN THE REFUEL MODE, A SPURIOUS AUTOMATIC START OF THE 2B CORE SPRAY PUMP OCCURRED. MAINTENANCE ACTIVITIES WERE UNDERWAY ON THE UNIT 2 DIESEL GENERATOR CUBICLE ON ELECTRICAL BUS 24-1. THE STATIONARY

AUXILIARY SWITCH, WHICH WAS BEING REPLACED, CONTAINS CONTACTS WHICH BYPASS THE DIESEL SEQUENCING TIMER AND ALLOWS THE 2B CORE SPRAY PUMP TO AUTOMATICALLY START UPON AN INITIATION SIGNAL WHEN THE NORMAL SOURCE OF AC POWER IS SUPPLYING THE BUS. IT IS HYPOTHESIZED THAT AN INADVERTENT GROUND OCCURRED WHILE INSTALLING THE NEW SWITCH. CORRECTIVE ACTION IMPLEMENTED BY THE WORKING DEPARTMENT REQUIRES MORE IN-DEPTH REVIEWS OF AFFECTED CIRCUITS AND INTERLOCKS BY MAINTENANCE AND OPERATING DEPARTMENT PERSONNEL PRIOR TO COMMENCEMENT OF WORK ON ENERGIZED CIRCUITRY. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL, BECAUSE OPERABILITY OF THE CORE SPRAY AND OTHER EMERGENCY CORE COOLING SYSTEMS WAS UNAFFECTED. A PREVIOUS EVENT INVOLVING AN UNPLANNED ENGINEERED SAFETY FEATURE ACTUATION DURING ELECTRICAL WORK WAS REPORTED BY LER 86-11/050249.

[60] FERMI 2 DOCKET 50-341 LER 90-001 REV 01
UPDATE ON BLOWN FUSE IN TESTABILITY CABINET H21-P083 CAUSED ENTRY INTO TECHNICAL SPECIFICATION 3.0.3.
EVENT DATE: 010890 REPORT DATE: 101990 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 219877) ON JANUARY 8, 1990, AT APPROXIMATELY 1410 HOURS, ANNUNCIATOR (ANN) 2D5 ALARMED FOR DIVISION II EMERGENCY CORE COOLING SYSTEM TESTABILITY LOGIC/POWER FAILURE. THE OPERATIONS PERSONNEL INVESTIGATED AND DISCOVERED THAT FUSE B21-F2B, WAS BLOWN. THE FUSE WAS REPLACED AT 1442 HOURS, BUT BLEW AGAIN APPROXIMATELY TEN MINUTES LATER. TECHNICAL SPECIFICATION 3.0.3 WAS ENTERED DUE TO VARIOUS INSTRUMENT TRIP UNITS AFFECTED AND THE INABILITY TO PLACE THEM IN THE REQUIRED TRIPPED CONDITIONS WITHOUT CAUSING ACTUATIONS/ISOLATIONS. AN UNUSUAL EVENT WAS DECLARED AT 1510 HOURS. A THIRD FUSE WAS INSTALLED DURING THE COURSE OF TROUBLESHOOTING WITH NO SUBSEQUENT PROBLEMS NOTED. CHANNEL FUNCTIONAL SURVEILLANCES WERE PERFORMED TO VERIFY THE OPERABILITY OF THE TRIP UNITS INVOLVED IN EVENT. ALL OF THE TRIP UNITS WERE FOUND TO BE OPERABLE AND THE UNUSUAL EVENT WAS TERMINATED AT 2100 HOURS. REACTOR POWER HAD BEEN REDUCED TO APPROXIMATELY 40 PERCENT. SUBSEQUENT INVESTIGATION REVEALED A FAILED CAPACITOR ON A TRIP UNIT WHICH SHORTED THE POWER SUPPLY CAUSING THE FUSES TO BLOW. FAILURE ANALYSIS OF THE SUSPECT CAPACITOR ON THE TRIP UNIT WAS COMPLETED. THE ANALYSIS DID NOT REVEAL A ROOT CAUSE. FERMI 2 ENGINEERING STAFF CONCLUDED THAT THE FAILURE OF THE CAPACITOR IS RANDOM IN NATURE.

[61] FERMI 2 DOCKET 50-341 LER 90-010
MSIV CLOSURE DUE TO OPERATOR ERROR.
EVENT DATE: 071890 REPORT DATE: 101590 NSSS: GE TYPE: BWR

(NSIC 219745) ON 7/18/90, DURING PERFORMANCE OF SURVEILLANCE 24.137.001, "MAIN STEAM LINE ISOLATION CHANNEL FUNCTIONAL TEST", THE OPERATOR CONDUCTING THE TEST INCORRECTLY DEPRESSED THE "CLOSE" PUSHBUTTON FOR "A" OUTBOARD MAIN STEAM LINE ISOLATION VALVE (MSIV) RATHER THAN THE "TEST" PUSHBUTTON, AS REQUIRED BY PROCEDURE. THIS ACTION CAUSED THE OUTBOARD "A" MSIV (B21-F028A) TO GO CLOSED. THE VALVE CLOSED NORMALLY AND ALL SYSTEMS RESPONDED AS EXPECTED. THERE WAS A BRIEF POWER INCREASE ABOVE 102% THERMAL POWER FOR APPROXIMATELY 32 SECONDS. THE OUTBOARD "A" MSIV WAS REOPENED. REACTOR POWER AND REACTOR PRESSURE RETURNED TO NORMAL. THE ROOT CAUSE OF THIS EVENT WAS AN ERROR BY A LICENSED REACTOR OPERATOR. AN EVENT CRITIQUE WAS CONDUCTED AND ISSUED AS REQUIRED READING. A REVISION WAS MADE TO THE PROCEDURE USED TO PERFORM THE SURVEILLANCE. AS A RESULT OF A REVIEW OF CONTROL ROOM CONTROLS, PROTECTIVE COVERS WILL BE PLACED OVER THE "CLOSE" PUSHBUTTONS FOR THE MSIVS. THIS LICENSEE EVENT REPORT IS SUBMITTED FOR INFORMATION ONLY. DETROIT EDISON CONSIDERS THIS INCIDENT SIGNIFICANT, THOUGH THE SPECIFIC REPORTING CRITERIA OF 10 CFR 50.73 ARE NOT APPLICABLE.

[62] FERMI 2 DOCKET 50-341 LER 90-009
LOSS OF SHUTDOWN COOLING AND SAFETY FEATURES ACTUATION DUE TO BREAKER TRIPPING.
EVENT DATE: 100290 REPORT DATE: 110190 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219903) ON 10/2/90, FERMI 2 WAS IN COLD SHUTDOWN. AT 1004 HOURS, A HALF SCRAM SIGNAL WAS RECEIVED WHEN POWER WAS LOST ON REACTOR PROTECTION SYSTEM BUS

"B". SEVERAL ENGINEERED SAFETY FEATURES WERE ACTUATED. ALL OF THE EXPECTED ACTUATIONS/ISOLATIONS WERE RECEIVED. SHUTDOWN COOLING WAS LOST AS A RESULT OF THE ISOLATIONS. SHUTDOWN COOLING WAS RESTORED IN 32 MINUTES. COOLANT TEMPERATURE INCREASED 13 DEGREES TO A MAXIMUM OF 160F DURING THIS EVENT. INVESTIGATION SHOWS THAT THE LOSS OF POWER WAS DUE TO A TRIP OF AN ELECTRICAL PROTECTION ASSEMBLY. INITIALLY, IT WAS SUSPECTED THAT THE TRIP WAS DUE TO AN UNDERVOLTAGE CONDITION, BUT TESTING RESULTS WERE INCONCLUSIVE. GENERAL ELECTRIC ISSUED A SERVICE INFORMATION LETTER ON SPURIOUS TRIPS IN THIS TYPE OF EQUIPMENT IN SEPTEMBER OF 1990, WHICH PROVIDES RECOMMENDATIONS FOR ADDRESSING THIS CONCERN. DETROIT EDISON IS EVALUATING THIS ISSUANCE AND WILL IMPLEMENT THE APPROPRIATE RECOMMENDATIONS FROM THAT DOCUMENT.

[63] FITZPATRICK DOCKET 50-333 LER 88-005 REV 01
UPDATE ON FAILURE OF VENTILATION BACKUP COOLING WATER SUPPLY CHECK VALVES.
EVENT DATE: 052588 REPORT DATE: 101690 NSSS: GE TYPE: BWR
VENDOR: VELAN VALVE CORP.

(NSIC 219797) ON FEBRUARY 27, 1987 AND MARCH 25, 1987, WHILE SHUTDOWN FOR REFUEL AND MAINTENANCE, SIX EMERGENCY SERVICE WATER (ESW) [BI] CHECK VALVES THAT SUPPLY VENTILATION SYSTEM BACKUP COOLING FOR NORMAL 4KV [EA] AND PART OF THE SAFETY-RELATED 600V AC [ED] SYSTEMS WERE FOUND INOPERABLE DUE TO AN ACCUMULATION OF SILT WHEN DISASSEMBLED FOR INSERVICE TESTS. AN INITIAL EVALUATION REVEALED THAT NO TECHNICAL SPECIFICATIONS OR FINAL SAFETY ANALYSIS REPORT (FSAR) BASIS FOR THE DESIGN OR SAFETY SIGNIFICANCE. LATER EVALUATION ESTABLISHED AN ORIGINAL DESIGN SAFETY BASIS. THE VALVES WERE CLEANED TO RESTORE OPERABILITY PRIOR TO PLANT STARTUP AT THE END OF THE REFUEL OUTAGE. PERIODIC FLUSHING WILL BE CONDUCTED TO MAINTAIN OPERABILITY. AN FSAR CHANGE HAS BEEN PREPARED FOR INCORPORATION INTO THE NEXT PERIODIC UPDATE TO REFLECT THE DESIGN BASIS OF THE VENTILATION SYSTEMS WHICH THE CHECK VALVES SUPPLY COOLING WATER. THERE HAVE BEEN NO SIMILAR LERS INVOLVING LACK OF DESIGN BASIS OR SAFETY CLASSIFICATION DOCUMENTATION. LER-90-012 DESCRIBES A SIMILAR EVENT CONCERNING A NUMBER OF CHECK VALVE PROBLEMS.

[64] FT. CALHOUN 1 DOCKET 50-285 LER 90-022
NONFUNCTIONAL FIRE BARRIER PENETRATIONS.
EVENT DATE: 090790 REPORT DATE: 100890 NSSS: CE TYPE: PWR

(NSIC 219712) ON SEPTEMBER 7, 1990, APPROXIMATELY 460 FIRE BARRIER PENETRATION SEALS, 60 FIRE DAMPERS AND 6 FIRE DOORS ASSOCIATED WITH 25 FIRE AREAS WERE DECLARED NONFUNCTIONAL DUE TO EITHER LACK OF DOCUMENTATION TO QUALIFY NON-VERIFIABLE PENETRATION CRITICAL PARAMETERS, OR PENETRATION "AS BUILT" CONFIGURATIONS WHICH DID NOT MATCH TYPICAL CONFIGURATIONS PREVIOUSLY QUALIFIED BY FIRE TESTS. THESE NONFUNCTIONAL PENETRATIONS WERE DISCOVERED THROUGH A SPECIAL DESIGN BASIS VERIFICATION WALKDOWN PROMPTED BY NRC INFORMATION NOTICE 88-04. AS REQUIRED BY TECHNICAL SPECIFICATIONS, THE APPROPRIATE COMPENSATORY MEASURES WERE IMPLEMENTED AND WILL REMAIN IN PLACE UNTIL THE AFFECTED FIRE BARRIER PENETRATIONS ARE RESTORED TO FUNCTIONAL STATUS THROUGH EVALUATION, REPAIR, OR REPLACEMENT. A SUPPLEMENT TO THIS LER WILL PROVIDE A CORRECTIVE ACTION PLAN FOR RESTORING AFFECTED FIRE BARRIER PENETRATIONS TO FUNCTIONAL STATUS. THIS REPORT IS SUBMITTED PURSUANT TO TECHNICAL SPECIFICATION 2.19(7) BECAUSE MOST OF THE NONFUNCTIONAL FIRE BARRIER PENETRATIONS WERE NOT RESTORED TO FUNCTIONAL STATUS WITHIN 7 DAYS. IT IS BEING SUBMITTED ALSO AS A VOLUNTARY LER DUE TO POTENTIAL REGULATORY AND INDUSTRY INTEREST.

[65] FT. CALHOUN 1 DOCKET 50-285 LER 90-020
POTENTIAL COMMON MODE FAILURE OF EMERGENCY DIESEL GENERATOR EXCITER CIRCUITS.
EVENT DATE: 091390 REPORT DATE: 101590 NSSS: CE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219847) ON JUNE 25, 1990, THE VOLTAGE REGULATOR ASSOCIATED WITH EMERGENCY DIESEL GENERATOR NO. 1 FAILED DURING FULL LOAD TESTING DUE TO A PARTIALLY FAILED TRANSISTOR IN THE STATIC EXCITER CIRCUIT. THIS WAS TENTATIVELY ATTRIBUTED TO HIGH TEMPERATURE IN THE CONTROL CABINET; THE CABINET DOORS FOR BOTH DIESEL

GENERATORS WERE REMOVED TO PROVIDE VENTILATION AND ASSURE OPERABILITY. ON SEPTEMBER 13, 1990, THE PLANT REVIEW COMMITTEE REVIEWED A FINALIZED ENGINEERING EVALUATION OF THE EXCITER FAILURE AND CONCLUDED THAT A REPORTABLE CONDITION EXISTED. THIS WAS REPORTED TO NRC AT 1402 HOURS CDT PURSUANT TO 10 CFR 50.72(B)(2)(III), DUE TO A POTENTIAL COMMON MODE FAILURE WHICH COULD HAVE PREVENTED THE DIESEL GENERATORS FROM PERFORMING THEIR SAFETY FUNCTION DURING A DESIGN BASIS ACCIDENT. THE ROOT CAUSE OF THIS CONDITION WAS IMPROPER DESIGN OF THE EXCITER CABINETS. A CONTRIBUTING FACTOR TO THE DURATION OF THIS CONDITION WAS THE FAILURE TO IDENTIFY IN PREVIOUS YEARS THE CABINET OVERHEATING AS THE PROBABLE CAUSE OF A HISTORY OF STATIC EXCITER COMPONENT FAILURES AT FORT CALHOUN STATION. CORRECTIVE ACTIONS INCLUDE INSTALLATION OF A CABINET VENTILATION SYSTEM TO PROVIDE ADEQUATE COOLING. UNTIL IMPLEMENTATION OF THIS MODIFICATION, THE CABINET DOORS WILL REMAIN REMOVED UNDER A TEMPORARY MODIFICATION. RECENTLY ENHANCED ENGINEERING RESOURCES SHOULD PRECLUDE RECURRENCE OF THIS TYPE CONDITION.

[66] FT. CALHOUN 1 DOCKET 50-285 LER 90-024
FAILURE TO CONDUCT HOURLY FIREWATCH.
EVENT DATE: 091590 REPORT DATE: 102690 NSSS: CE TYPE: PWR

(NSIC 219850) ON SEPTEMBER 27, 1990 AT 2358 HOURS WITH FORT CALHOUN STATION UNIT 1 OPERATING AT 100% POWER, THE SHIFT SECURITY SUPERVISOR DETERMINED THAT THE REQUIRED HOURLY FIREWATCH PATROL AT DOOR 989-8 WAS NOT BEING PERFORMED. THIS DOOR WAS ERRONEOUSLY COPIED AS DOOR 989-4 ONTO THE HOURLY FIREWATCH LOG ON SEPTEMBER 16, 1990 BY A DIFFERENT SHIFT SECURITY SUPERVISOR. FAILURE TO PERFORM THE FIREWATCH PATROL FROM SEPTEMBER 16 TO SEPTEMBER 27 WAS A VIOLATION OF TECHNICAL SPECIFICATION 2.19(7). UPON DISCOVERY THAT THE REQUIRED HOURLY FIREWATCH PATROL WAS NOT BEING PERFORMED, DOOR 989-8 WAS IMMEDIATELY RESTORED TO THE HOURLY FIREWATCH LOG. THIS EVENT RESULTED FROM PERSONNEL ERROR. APPROPRIATE DISCIPLINARY ACTION HAS BEEN ADMINISTERED. PLANT STAFF WILL IMPLEMENT A COMPUTER DRIVEN DATA BASE TO GENERATE THE HOURLY FIREWATCH LOG. THIS COMPUTERIZED SYSTEM SHOULD PRECLUDE MANUAL TRANSCRIPTION ERRORS.

[67] FT. CALHOUN 1 DOCKET 50-285 LER 90-023
SAFETY INJECTION PIPING AND RELIEF VALVES OUTSIDE DESIGN BASIS.
EVENT DATE: 092190 REPORT DATE: 102290 NSSS: CE TYPE: PWR

(NSIC 219848) ON SEPTEMBER 21, 1990, AT 1425 HOURS CDT, THE PLANT REVIEW COMMITTEE (PRC) DETERMINED THAT, FOR EACH OF THE FOUR TRAINS, THE SAFETY INJECTION (SI) PIPING BETWEEN THE SI TANK ISOLATION VALVES AND THE FIRST SI CHECK VALVE TO THE RCS WAS OUTSIDE THE PLANT DESIGN BASIS. THE SETPOINT OF THE RELIEF VALVE IN THIS SECTION OF PIPE HAS BEEN SIGNIFICANTLY HIGHER THAN THE DESIGN PRESSURE OF THE PIPING. IT WAS DETERMINED THAT THE PIPING IS OPERABLE SINCE THE MOST LIMITING COMPONENT WITHIN THE PRESSURE BOUNDARY HAS A PRESSURE RATING AT A CONSERVATIVE TEMPERATURE (550 DEGREES F) OF 395 PSIG, EQUAL TO THE SETPOINT OF THE RELIEF VALVE. DURING POWER OPERATION, THE SI TANK ISOLATION VALVES ARE OPEN, AND THE SI PIPING IS PROTECTED BY THE RELIEF VALVES ON THE SI TANKS. FURTHER, IT IS UNLIKELY THE PIPING INTEGRITY WOULD BE CHALLENGED SINCE THE PIPING COULD ONLY BE SUBJECTED TO A PRESSURE OF 395 PSIG WHEN THE SI TANK ISOLATION VALVES ARE CLOSED DURING TRANSITION TO OR FROM COLD SHUTDOWN. THIS CONDITION HAS EXISTED SINCE PLANT CONSTRUCTION, AND RESULTED FROM DESIGN AND ANALYSIS DEFICIENCIES BY THE ORIGINAL PLANT ARCHITECT/ENGINEER. CORRECTIVE ACTIONS INCLUDE INCORPORATION OF SAFETY RELATED RELIEF VALVES INTO THE INSERVICE TESTING PROGRAM, HYDROSTATIC TESTING OF THE AFFECTED PIPING, AND UPDATING OF APPLICABLE DESIGN BASIS DOCUMENTS.

[68] FT. CALHOUN 1 DOCKET 50-285 LER 90-025
COOLING WATER SYSTEMS OUTSIDE DESIGN BASIS.
EVENT DATE: 092990 REPORT DATE: 102990 NSSS: CE TYPE: PWR

(NSIC 219851) IN THE COURSE OF RESOLVING DESIGN BASIS DOCUMENT OPEN ITEMS, CONDITIONS WERE IDENTIFIED INVOLVING THE COMPONENT COOLING WATER (CCW), RAW WATER (RW), AND CONTAINMENT SPRAY (CS) SYSTEMS WHICH PLACED THE FORT CALHOUN STATION OUTSIDE ITS DESIGN BASIS FOR POST-ACCIDENT CONTAINMENT COOLING AS DEFINED IN THE UPDATED SAFETY ANALYSIS REPORT AND THE BASIS FOR TECHNICAL SPECIFICATION 2.4.

THE CCW CONDITIONS INVOLVED THE POTENTIAL FOR DEGRADATION OF CONTAINMENT AIR COOLER PERFORMANCE AND/OR LOSS OF CCW SYSTEM OPERABILITY FOLLOWING A LOSS OF INSTRUMENT AIR. THE RW CONDITIONS INVOLVED THE INABILITY OF THE RW SYSTEM TO PROVIDE BACKUP COOLING TO CCW FOR THE CONTAINMENT AIR COOLERS. THE CS CONDITIONS INVOLVED THE POTENTIAL FOR THE LOSS OF OPERABILITY OF A SINGLE OPERATING CS PUMP FOLLOWING A SINGLE ACTIVE FAILURE OF AN EMERGENCY DIESEL GENERATOR. THESE CONDITIONS RESULTED FROM ORIGINAL PLANT DESIGN DEFICIENCIES. CORRECTIVE ACTIONS INCLUDE IMPLEMENTATION OF A TEMPORARY MODIFICATION TO PRECLUDE LOSS OF CCW FOLLOWING A LOSS OF INSTRUMENT AIR, AND A PERMANENT MODIFICATION TO ASSURE ADEQUATE POST-ACCIDENT CS FLOW WITHOUT THREATENING THE OPERABILITY OF A SINGLE OPERATING CS PUMP. APPLICABLE BASES TO TECHNICAL SPECIFICATION 2.4 HAVE BEEN REVISED. A PERMANENT RESOLUTION TO THE CCW/RW CONDITIONS IS STILL BEING EVALUATED; RESULTS OF THIS EVALUATION WILL BE PROVIDED IN A SUPPLEMENT TO THIS LER.

[69] GINNA DOCKET 50-244 LER 90-012
TURBINE TRIP RELAY ACTUATION DUE TO DROPPED FLASHLIGHT IN RELAY RACK (PERSONNEL ERROR), CAUSES A REACTOR TRIP.
EVENT DATE: 092690 REPORT DATE: 102690 NSSS: WE TYPE: PWR
VENDOR: ATWOOD & MORRILL CO., INC.
PARKER HANNIFIN CORP.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 219814) ON SEPTEMBER 26, 1990, AT 1100 EDST WITH THE REACTOR AT APPROXIMATELY 97% FULL POWER, A REACTOR TRIP OCCURRED FROM AN OPENING OF THE "A" REACTOR TRIP BREAKER, FOLLOWED IN APPROXIMATELY SEVEN (7) SECONDS BY A LOW PRESSURIZER PRESSURE REACTOR TRIP SIGNAL AND THE OPENING OF THE "B" REACTOR TRIP BREAKER. THE "A" REACTOR TRIP BREAKER OPENING WAS CAUSED BY THE INADVERTENT DROPPING OF A FLASHLIGHT ON TWO OF THREE TURBINE AUTOSTOP TRIP RELAYS. THE LOW PRESSURIZER PRESSURE REACTOR TRIP WAS CAUSED BY THE REACTOR COOLANT SYSTEM COOLDOWN DUE TO THE REACTOR BEING TRIPPED WITH THE TURBINE STILL ON THE LINE. IMMEDIATE CORRECTIVE ACTION WAS TO STABILIZE THE PLANT IN HOT SHUTDOWN. CORRECTIVE ACTION TO PREVENT RECURRENCE WILL BE BASED UPON THE RECOMMENDATIONS OF A HUMAN PERFORMANCE ENHANCEMENT SYSTEM (HPES) EVALUATION OF THE DROPPED FLASHLIGHT EVENT. CORRECTIVE ACTION FOR SUBSEQUENT HARDWARE MALFUNCTIONS WILL ALSO BE TAKEN.

[70] GRAND GULF 1 DOCKET 50-416 LER 90-017 REV 01
UPDATE ON REACTOR SCRAM DUE TO LOSS OF BALANCE OF PLANT BUSSES.
EVENT DATE: 091690 REPORT DATE: 101190 NSSS: GE TYPE: BWR

(NSIC 219932) A DIVISION I LOAD SHEDDING AND SEQUENCING SYSTEM MALFUNCTION CAUSED A BALANCE OF PLANT (BOP) LOAD SHED ON SEPTEMBER 16, 1990. THE LOSS OF MAJOR PLANT EQUIPMENT, WHICH RECEIVED POWER FROM THE SHEDDED BOP BUSSES, RESULTED IN A REACTOR SCRAM, DUE TO MAIN TURBINE CONTROL VALVE FAST CLOSURE. SUBSEQUENT TO THE SCRAM, REACTOR WATER LEVEL DECREASED TO -41.6 INCHES WHERE AN AUTOMATIC HIGH PRESSURE CORE SPRAY SYSTEM ACTUATION OCCURRED. DURING RESTORATION OF MAIN STEAM ISOLATION VALVES, AS A PART OF SCRAM SUBSEQUENT ACTIONS, A SECOND REACTOR SCRAM OCCURRED DUE TO LOW REACTOR WATER LEVEL. THE LOAD SHED IS ATTRIBUTED TO A DEFECTIVE LIGHT BULB BEING PLACED IN THE LOAD SHED PANEL. THE SHORTED LIGHT BULB CAUSED AN OVERCURRENT WHICH SUBSEQUENTLY CAUSED DEGRADATION OF A COMPUTER CHIP WHICH INITIATED THE LOAD SHED. THE CARDS WHICH CONTAINED DEGRADED COMPUTER CHIPS, DUE TO THE OVERCURRENT, WERE REPLACED. THE DIVISION I LOAD SHED PANEL WAS TESTED SATISFACTORILY AND OPERABILITY WAS VERIFIED. ALL SAFETY SYSTEMS FUNCTIONED AS DESIGNED. THE MINIMUM WATER LEVEL REACHED WAS -54.1 INCHES WHICH WAS APPROXIMATELY 112 INCHES ABOVE THE TOP OF ACTIVE FUEL.

[71] HATCH 1 DOCKET 50-321 LER 90-017
DIESEL GENERATOR OUTPUT BREAKERS CLOSE IN GREATER THAN 12 SECONDS.
EVENT DATE: 082990 REPORT DATE: 100190 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 2 (BWR)

(NSIC 219727) ON 8/29/90 AT APPROX. 0330 CDT, UNITS 1 AND 2 WERE IN THE RUN MODE

AT APPROX. POWER LEVELS OF 2436 CMWT AND 2436 CMWT, RESPECTIVELY. AT THAT TIME, IT WAS DETERMINED DIESEL GENERATOR (D/G, EIIS CODE AK) 2R43-S001A WOULD NOT ENERGIZE ITS EMERGENCY 4160-VOLT BUS WITHIN 12 SECONDS OF RECEIPT OF A START SIGNAL AS EXPECTED. SUBSEQUENT TESTING OF THE 4 REMAINING D/GS PERFORMED BETWEEN 8/30/90 AND 9/3/90 REVEALED THEY WOULD NOT ENERGIZE THEIR RESPECTIVE 4160-VOLT EMERGENCY BUSSES WITHIN 12 SECONDS EITHER. THIS TESTING WAS BEING PERFORMED IN ANTICIPATION OF RECEIPT OF A TECH SPECS AMENDMENT REQUIRING PERIODIC VERIFICATION OF THIS TIME. BUS ENERGIZATION TIMES OF UP TO 24 SECONDS WERE ASSESSED BY PLANT HATCH'S ARCHITECT/ENGINEER AND GENERAL ELECTRIC. THE 24 SECOND ENERGIZATION TIME HAD NO SIGNIFICANT IMPACT ON THE RESULTS OF THE LICENSING BASIS ANALYSES; THEREFORE, THE OPERABILITY OF THE D/GS HAD NOT BEEN IMPACTED BY THE TEST RESULTS. THIS REPORT IS BEING SUBMITTED VOLUNTARILY BECAUSE THIS EVENT MAY BE OF INTEREST TO PLANTS WHOSE D/G TECH SPECS DO NOT CURRENTLY REQUIRE THEM TO TEST FOR EMERGENCY BUS ENERGIZATION TIME. THE CAUSE OF THIS EVENT WAS LESS THAN ADEQUATE DESIGN DOCUMENTATION. A TIME DELAY RELAY IN THE PERMISSIVE LOGIC FOR D/G OUTPUT BREAKER CLOSURE WAS NOT REQUIRED TO BE SET AT ITS MINIMUM VALVE.

[72] HATCH 1 DOCKET 50-321 LER 90-018
PERSONNEL ERROR RESULTS IN INADEQUATE PROCEDURE AND MISSED TECHNICAL SPECIFICATION SURVEILLANCE.
EVENT DATE: 090790 REPORT DATE: 100190 NSSS: GE TYPE: BWR

(NSIC 219728) ON 9/7/90, AT APPROXIMATELY 1115 CDT, UNIT 1 WAS IN THE RUN MODE AT APPROX. 2436 CMWT (APPROX. 100% OF RATED THERMAL POWER) WHEN NON-LICENSED PERSONNEL DETERMINED PROCEDURE 34SV-SUV-019-1S, "SURVEILLANCE CHECKS," DID NOT ADEQUATELY IMPLEMENT THE REQUIREMENTS OF UNIT 1 TECH SPECS TABLES 3.2-11 AND 4.2-11, ITEMS 12 AND 15. SPECIFICALLY, THE PROCEDURE DID NOT INCLUDE AN INSTRUMENT CHECK FOR THE POST LOCA RADIATION AND THE DRYWELL HIGH RANGE RADIATION MONITORING SYSTEMS' (EIIS CODE IP) RECORDERS. THE PROCEDURE DID, HOWEVER, INCLUDE AN INSTRUMENT CHECK OF THE INDICATORS WHICH PROVIDE DIRECT INPUT TO THE RECORDERS. UPON DISCOVERY OF THE EVENT, AN INSTRUMENT CHECK OF THE RECORDERS WAS SATISFACTORILY PERFORMED. PROCEDURE 34SV-SUV-019-1S WAS TEMPORARILY REVISED TO INCLUDE THE RECORDER INSTRUMENT CHECKS. THE DEFICIENCY WAS NOTED DURING AN ONGOING VALIDATION OF THE COMMITMENT MATRIX TRACKING SYSTEM. THE CAUSE OF THE EVENT IS PERSONNEL ERROR ON THE PART OF NON-LICENSED PERSONNEL. A PROCEDURE WRITER INADVERTENTLY DELETED THE INSTRUMENT CHECKS FROM THE PROCEDURE IN A REVISION MADE EFFECTED ON 12/9/88. ALSO, DURING A TECHNICAL REVIEW OF THE PROPOSED REVISION, THE REVIEWER FAILED TO IDENTIFY THE ERROR. CORRECTIVE ACTIONS INCLUDE PERMANENTLY REVISING THE PROCEDURE, COUNSELING APPROPRIATE PERSONNEL, AND REVIEWING THE PROCEDURE FOR SIMILAR PROBLEMS.

[73] HATCH 1 DOCKET 50-321 LER 90-019
MAIN STEAM LINE RADIATION MONITOR SETTINGS EXCEED TECHNICAL SPECIFICATION SETPOINT.
EVENT DATE: 092590 REPORT DATE: 102390 NSSS: GE TYPE: BWR
VENDOR: THERMAL INSTRUMENTS COMPANY

(NSIC 219859) ON 9/25/90, AT APPROXIMATELY 0955 CDT, UNIT 1 WAS IN THE RUN MODE AT APPROXIMATELY 2436 CMWT (APPROXIMATELY 100 PERCENT OF RATED THERMAL POWER). AT THAT TIME, THE UNIT 1 SHIFT SUPERVISOR WAS NOTIFIED BY NONLICENSED CHEMISTRY PERSONNEL THAT THE HI-HI TRIP SETPOINTS FOR THE MAIN STEAM LINE RADIATION MONITORS (MSLRMS, EIIS CODE IL) 1D11-K603A, B, C, AND D WERE NOT LESS THAN OR EQUAL TO 3 TIMES THE NORMAL MAIN STEAM LINE BACKGROUND RADIATION LEVELS AT RATED THERMAL POWER AS REQUIRED BY UNIT 1 TECHNICAL SPECIFICATIONS TABLE 3.1-1, ITEM 9; TABLE 3.2-1, ITEM 4; AND TABLE 3.2-8, ITEM 5. THE MONITORS WERE DECLARED INOPERABLE AND THE APPROPRIATE LIMITING CONDITION FOR OPERATION (LCO) WAS ENTERED. THE SETPOINTS WERE READJUSTED AS REQUIRED AND, SUBSEQUENTLY, AT 1235 CDT OF THE SAME DAY, THE LCO WAS TERMINATED. CAUSES OF THE EVENT INCLUDE AN AMBIGUOUS TECHNICAL SPECIFICATION, A LESS THAN ADEQUATE PROCEDURE, AND A MALFUNCTION OF A HYDROGEN FLOW MONITOR/ELEMENT. CORRECTIVE ACTIONS INCLUDE ISSUING A CLARIFICATION OF THE TECHNICAL SPECIFICATIONS, REVISING A PROCEDURE, AND REPAIRING THE FLOW MONITOR/ELEMENT. THE APPROPRIATE PERSONNEL HAVE BEEN COUNSELED AS TO THE REQUIREMENT FOR STRICT PROCEDURAL COMPLIANCE AND THE

IMPORTANCE OF TIMELY DISSEMINATION OF INFORMATION REGARDING PLANT OPERATING CONDITIONS.

[74] HATCH 1 DOCKET 50-321 LER 90-020
MAIN TURBINE HIGH VIBRATION RESULTS IN AUTOMATIC REACTOR SCRAM.
EVENT DATE: 100690 REPORT DATE: 102690 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.
LEEDS & NORTHRUP CO.

(NSIC 219860) ON 10/6/90, AT APPROXIMATELY 0337 CDT, UNIT 1 WAS IN THE RUN MODE AT APPROXIMATELY 536 CMWT (APPROXIMATELY 22 PERCENT OF RATED THERMAL POWER) AND DECREASING AS PART OF A SCHEDULED SHUTDOWN TO SUPPORT MAINTENANCE ON A RECIRCULATION PUMP SEAL. AT THAT TIME, A MAIN TURBINE TRIP OCCURRED ON HIGH VIBRATION AND A FULL REACTOR PROTECTION SYSTEM ACTUATION AND RECIRCULATION PUMP TRIP WERE INITIATED ON TURBINE STOP VALVE (TSV) CLOSURE. A SECOND RPS ACTUATION AND A PRIMARY CONTAINMENT ISOLATION SYSTEM GROUP 2 ISOLATION OCCURRED PER DESIGN WHEN REACTOR VESSEL WATER LEVEL DECREASED TO REACTOR LOW WATER LEVEL 3. REACTOR FEEDWATER PUMPS RESTORED AND MAINTAINED REACTOR VESSEL WATER LEVEL AND THE TURBINE BYPASS VALVES AUTOMATICALLY CONTROLLED REACTOR PRESSURE. CAUSES OF THE EVENT INCLUDE A VALVE MALFUNCTION, A LESS THAN OPTIMAL DESIGN OF TSV CLOSURE SCRAM BYPASS PRESSURE SWITCHES, AND A FAILED CENTRAL PROCESSING UNIT (CPU) BOARD. CORRECTIVE ACTIONS INCLUDE CHANGING A VALVE MOTOR OPERATOR TORQUE SWITCH SETTING, REPLACING A CPU BOARD, AND EVALUATING REPLACEMENT OF THE PRESSURE SWITCHES.

[75] HATCH 2 DOCKET 50-366 LER 90-007
PERSONNEL ERROR RESULTS IN INADEQUATE PROCEDURE AND MISSED TECHNICAL SPECIFICATION SURVEILLANCE.
EVENT DATE: 091890 REPORT DATE: 101290 NSSS: GE TYPE: BWR

(NSIC 219910) ON 9/18/90, WITH UNIT 2 IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2436 CMWT (APPROXIMATELY 100% OF RATED THERMAL POWER), IT WAS DETERMINED RESPONSE TIME TESTING WAS NOT BEING PERFORMED TO THE EXTENT REQUIRED BY UNIT 2 TECHNICAL SPECIFICATION SECTIONS 4.3.2.3 AND 4.3.3.3. SPECIFICALLY, PROCEDURE 42SV-SUV-033-2S, "TIME RESPONSE TEST COMPARISON WITH TECHNICAL SPECIFICATIONS," CONTAINED THE INCORRECT ACCEPTANCE CRITERION FOR SOME ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME TESTS AS LISTED IN UNIT 2 TECHNICAL SPECIFICATION TABLE 3.3.2-3. FURTHERMORE, THE PROCEDURE DID NOT REQUIRE THE ADDITION OF EMERGENCY BUS ENERGIZATION TIMES TO INSTRUMENTATION, VALVE, CORE SPRAY AND LOW PRESSURE COOLANT INJECTION RESPONSE TIMES TO OBTAIN SYSTEM RESPONSE TIMES REQUIRED BY TECHNICAL SPECIFICATION TABLE 3.3.2-3 AND SECTION 4.3.3.3. ADDITIONALLY, THERE WAS NO PROCEDURE TO OBTAIN THE TIME FOR THE DIESEL GENERATORS TO ENERGIZE THEIR RESPECTIVE EMERGENCY BUSSES; CONSEQUENTLY, THIS TIME WAS UNAVAILABLE TO BE INCLUDED IN THE ABOVE RESPONSE TIMES AS REQUIRED BY SECTIONS 4.3.2.3 AND 4.3.3.3. ALL ACTUAL RESPONSE TIMES HAVE BEEN VERIFIED ACCEPTABLE. THE CAUSE OF THIS EVENT IS COGNITIVE PERSONNEL ERROR. THE WRITER OF PROCEDURE 42SV-SUV-033-2S FAILED TO ENSURE THE REQUIREMENTS OF UNIT 2 TECHNICAL SPECIFICATION SECTIONS 4.3.2.3 AND 4.3.3.3 WERE INCORPORATED PROPERLY INTO THE PROCEDURE.

[76] HATCH 2 DOCKET 50-366 LER 90-008
COMPONENT FAILURE CAUSES UNPLANNED ESF ACTUATIONS.
EVENT DATE: 092890 REPORT DATE: 101890 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 1 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219911) ON 9/28/90 AT APPROXIMATELY 2025 CDT, UNIT 2 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2436 CMWT (APPROXIMATELY 100% OF RATED THERMAL POWER) AND UNIT 1 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2436 CMWT (APPROXIMATELY 100% OF RATED THERMAL POWER). AT THAT TIME, THE "A" TRAINS OF BOTH UNITS' STANDBY GAS TREATMENT (SBGT) SYSTEMS AUTOMATICALLY STARTED, BOTH UNITS' REFUELING FLOOR AND REACTOR BUILDING VENTILATION SYSTEMS (SECONDARY CONTAINMENTS) ISOLATED, AND THE UNIT 2 HYDROGEN AND OXYGEN ANALYZER AND FISSION

PRODUCTS MONITORING SYSTEMS ISOLATED. ADDITIONALLY, A GROUP 2 PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) ISOLATION SIGNAL TO SOME OTHER UNIT 2 INBOARD, GROUP 2 PRIMARY CONTAINMENT ISOLATION VALVES (PCIVS) WAS GENERATED. ALL SYSTEMS FUNCTIONED PER DESIGN AND UNIT OPERATION WAS UNAFFECTED. THE CAUSE OF THIS EVENT WAS COMPONENT FAILURE. THE INSULATION ON THE COIL IN RELAY 2C61-K60 FAILED AND EXPOSED THE WINDINGS. THE WINDINGS SHORTED CAUSING A CURRENT SURGE WHICH BLEW FUSE 2C61-F19. THIS FUSE IS IN THE POWER SUPPLY TO THE INITIATION/ISOLATION LOGIC FOR THE ABOVE SYSTEMS. WHEN THE FUSE FAILED, THESE LOGIC SYSTEMS LOST POWER AND THEIR ASSOCIATED SYSTEMS INITIATED/ISOLATED PER DESIGN. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED REPLACING THE FAILED RELAY AND THE BLOWN FUSE.

[77] HOPE CREEK 1 DOCKET 50-354 LER 90-016 REV 01
UPDATE ON DISCOVERY OF INADEQUATE DIESEL FUEL OIL ANALYSIS METHODS BY CONTRACT LABORATORY RESULTS IN ENTRY TO TECHNICAL SPECIFICATION.
EVENT DATE: 082490 REPORT DATE: 103190 NSSS: GE TYPE: BWR

(NSIC 219886) ON 8/23/90 AT 1630, THE SYSTEM ENGINEER RESPONSIBLE FOR THE HOPE CREEK EMERGENCY DIESEL GENERATORS (EDG) IDENTIFIED TO STATION MANAGEMENT AND THE SENIOR NUCLEAR SHIFT SUPERVISOR (SNSS, SRO LICENSED) THAT A CONTRACT LABORATORY HAD NOT CONDUCTED FUEL OIL ANALYSIS IAW TECHNICAL SPECIFICATIONS. WHEN INFORMED OF THE PROBLEM, THE SNSS INVOKED A 24 HOUR DELAY OF IMPLEMENTING TECH SPEC ACTION REQUIREMENTS (AS PERMITTED BY TECH SPEC 4.0.3) TO ALLOW FOR PROPER COMPLETION OF TESTING, RATHER THAN DECLARE ALL EDGS INOPERABLE. AN ADDITIONAL 48 HOUR WAIVER OF COMPLIANCE WAS REQUESTED FROM NRC REGION I DUE TO TIME CONSTRAINTS INVOLVED IN ARRANGING FOR SHIPMENT AND TESTING OF THE FUEL OIL SAMPLES BY A DIFFERENT CONTRACT LAB. THE WAIVER WAS NOT RECEIVED BY 1630 ON 8/24/90, AS SUCH, ALL EDGS WERE DECLARED INOPERABLE, AND TECH SPEC 3.0.3 WAS ENTERED. AT 1730, THE REQUESTED REGIONAL WAIVER WAS GRANTED AND TECH SPEC 3.0.3 WAS EXITED AT THIS TIME. SUBSEQUENT TESTING OF THE FUEL OIL PER ASTM-D2274-70 WAS COMPLETED AT 0345 ON 8/25/90, AND CONFIRMED THAT THE FUEL OIL WAS OF PROPER QUALITY. AN IN-DEPTH INVESTIGATION BY THE STATION QUALITY ASSURANCE DEPARTMENT DETERMINED THAT A VARIETY OF FACTORS CONTRIBUTED TO THIS OCCURRENCE, THE PRIMARY AMONG THESE FACTORS BEING A PROGRAMMATIC DEFICIENCY IN CONTROL OF THE FUEL OIL SAMPLING PROGRAM.

[78] HOPE CREEK 1 DOCKET 50-354 LER 90-017
CONTROL ROOM VENTILATION TRAIN TRIPS DUE TO LACK OF CALIBRATION OF A DIFFERENTIAL PRESSURE SWITCH.
EVENT DATE: 091290 REPORT DATE: 101290 NSSS: GE TYPE: BWR
VENDOR: BAILEY METER COMPANY

(NSIC 219887) ON 9/12/90 AT 1954, CONTROL ROOM OPERATORS NOTICED AN AUDIBLE SURGING OF THE "A" CONTROL ROOM CHILLER. BEFORE THE OPERATORS COULD RESPOND TO INVESTIGATE THE CAUSE OF THE SURGING, THE CHILLER TRIPPED, RENDERING THE "A" CONTROL ROOM VENTILATION (CRV) SYSTEM INOPERABLE. WITH THE "B" CRV TRAIN INOPERABLE FOR SCHEDULED MAINTENANCE, THE SENIOR NUCLEAR SHIFT SUPERVISOR (SNSS, SRO LICENSED) DIRECTED ENTRY INTO TECH SPEC 3.0.3 AT 2000. IMMEDIATE INVESTIGATION DETERMINED THAT THE "A" CHILLER HAD TRIPPED DUE TO A TRIP OF THE "C" SAFETY AUXILIARIES COOLING SYSTEM (SACS) PUMP DURING THE CALIBRATION OF THE "C" SACS PUMP DIFFERENTIAL PRESSURE TRANSMITTER (PDT). CONTROL ROOM PERSONNEL STARTED THE REDUNDANT "A" SACS PUMP, RESTARTED THE "A" CONTROL ROOM CHILLER, AND RETURNED THE "C" CRV SYSTEM TO A NORMAL OPERATIONAL CONFIGURATION. TECH SPEC 3.0.3 WAS EXITED AT 2001. EXTENSIVE TROUBLESHOOTING FOLLOWING THE EVENT DETERMINED THAT THE SETPOINT FOR THE DIFFERENTIAL PRESSURE SWITCH (PDSL) ASSOCIATED WITH THE "C" SACS PUMP PDT HAD DRIFTED HIGH, RESULTING IN AN INADVERTENT TRIP OF THE "C" SACS PUMP WHEN THE PDT WAS RETURNED TO SERVICE. THE PRIMARY CAUSE OF THIS OCCURRENCE WAS THE LACK OF A RECURRING TASK TO PERIODICALLY CALIBRATE THE PDSL. CORRECTIVE ACTIONS CONSISTED OF RECALIBRATING THE PDSL AND REVIEWING OTHER SACS INSTRUMENTATION FOR PROPERLY SCHEDULED CALIBRATION.

[79]   HOPE CREEK 1   DOCKET 50-354   LER 90-018
THROUGH WALL LEAK DUE TO SAFETY AUXILIARIES COOLING SYSTEM PUMP CASING LEAK RESULTS IN NRC NOTIFICATION DUE TO EQUIPMENT FABRICATION DEFICIENCY.
EVENT DATE: 092690   REPORT DATE: 101790   NSSS: GE   TYPE: BWR
VENDOR: INGERSOLL-RAND CO.

(NSIC 219905) ON 9/26/90 AT 0911, THE SENIOR NUCLEAR SHIFT SUPERVISOR (SNSS, SRO LICENSED) DECLARED THE "A" SAFETY AUXILIARIES COOLING SYSTEM (SACS) PUMP INOPERABLE IN PREPARATION FOR CONDUCTING NON-DESTRUCTIVE EXAMINATIONS ON AN APPARENT THROUGH-WALL LEAK ON THE LOWER PUMP CASING. TECH SPECS REQUIRE THAT AN INOPERABLE SACS PUMP BE RETURNED TO SERVICE WITHIN 72 HOURS, OR THAT THE PLANT BE IN HOT SHUTDOWN WITHIN THE NEXT 12 HOURS. RESULTS OF THE NDE DETERMINED THAT A ONE-INCH LINEAR INDICATION EXISTED IN THE PUMP CASING. SINCE THE STRUCTURAL INTEGRITY OF AN ASME III CLASS 3 COMPONENT WAS INVOLVED, IN ACCORDANCE WITH A PREVIOUS AGREEMENT BETWEEN PSE&G AND NRC REGION I, A NOTIFICATION OF THE DISCOVERED CONDITION WAS MADE AT 1040. THE APPARENT CAUSE OF THE DISCOVERED CONDITION WAS A CASTING DEFECT IN THE LOWER PUMP CASING DURING INTIAL FABRICATION. SUBSEQUENT INVESTIGATION DETERMINED THAT ASME III CODE REPAIRS COULD NOT BE ACCOMPLISHED ON THE PUMP CASING WITHIN TECH SPEC TIME CONSTRAINTS, AS SUCH, IT WAS NECESSARY TO REPLACE THE PUMP CASING. DUE TO THE AMOUNT OF TIME REQUIRED TO REPLACE AND RETEST THE PUMP, A 24 HOUR WAIVER OF COMPLIANCE WITH TECH SPEC PROVISIONS WAS REQUESTED ON 9/28/90. THE WAIVER WAS GRANTED BY NRC REGION I, EFFECTIVE AT 0911, 9/29/90, AND EXPIRING AT 0911, 9/30/90.

[80]   HOPE CREEK 1   DOCKET 50-354   LER 90-019
LATE INSERVICE INSPECTION SURVEILLANCE DUE TO SCHEDULING ERRORS.
EVENT DATE: 092790   REPORT DATE: 102490   NSSS: GE   TYPE: BWR

(NSIC 219906) ON 9/27/90 AT 1425, AN INSERVICE INSPECTION (ISI) SUPERVISOR INFORMED THE SENIOR NUCLEAR SHIFT SUPERVISOR (SNSS, SRO LICENSED) THAT A SEMI-ANNUALLY TYPE "B" LOCAL LEAK RATE TEST (LLRT) ON THE DRYWELL PERSONNEL AIRLOCK WAS DISCOVERED TO BE ONE DAY OVERDUE. TECHNICAL SPECIFICATIONS REQUIRE THAT TYPE "B" LLRT BE PERFORMED ON THE DRYWELL AIRLOCK AT LEAST ONCE PER 6 MONTHS (184 DAYS), WITH NO GRACE PERIOD AS NORMALLY ALLOWED BY SPECIFICATION 4.0.2. FOLLOWUP INVESTIGATION DETERMINED THE PRIMARY CAUSE OF THIS INCIDENT TO BE INADEQUATE SCHEDULING OF THE COMPLETION OF THE SUBJECT SURVEILLANCE BY ISI DEPARTMENT PERSONNEL. CORRECTIVE ACTIONS INCLUDE REVIEWING THIS OCCURRENCE WITH ALL ISI PERSONNEL, ADMINISTRATIVELY SCHEDULING THE SURVEILLANCE FOR COMPLETION 1 MONTH PRIOR TO THE ACTUAL DUE DATE, AND REVIEWING TECHNICAL SPECIFICATIONS TO ENSURE NO OTHER REQUIRED SURVEILLANCES EXIST WHICH DO NO ALLOW USE OF THE GRACE PERIOD IN SPECIFICATION 4.0.2.

[81]   HOPE CREEK 1   DOCKET 50-354   LER 90-020
PRE-PLANNED ENTRY INTO TECHNICAL SPECIFICATION 3.0.3 TO REPLACE SUSPECT ROSEMOUNT TRANSMITTER IN ACCORDANCE WITH NRC BULLETIN 90-01.
EVENT DATE: 100390   REPORT DATE: 110190   NSSS: GE   TYPE: BWR
VENDOR: ROSEMOUNT ENGINEERING COMPANY

(NSIC 219907) ON 10/03/90 AT 2031, THE SENIOR NUCLEAR SHIFT SUPERVISOR (SNSS, SRO LICENSED) DIRECTED ENTRY INTO TECH SPEC 3.0.3 TO REPLACE A SUSPECT PRESSURE TRANSMITTER IN ONE OF THE REACTOR VESSEL INSTRUMENT RACKS. THE TRANSMITTER WAS DISCOVERED TO BE EXHIBITING SIGNS OF FAILURE AS DESCRIBED IN NRC BULLETIN 90-01 DURING AN ENHANCED SURVEILLANCE. REPLACEMENT OF TRANSMITTER REQUIRED PLACING REACTOR VESSEL LEVEL AND PRESSURE INSTRUMENTS ON A COMMON SENSING LINE IN AN INOPERABLE CONDITION TO PRECLUDE INADVERTENT ESF ACTUATIONS. INCLUDED IN THESE INSTRUMENTS WERE LEVEL AND PRESSURE TRANSMITTERS WHICH PROVIDE INPUTS TO THE HIGH PRESSURE COOLANT INJECTION (HPCI) AND CORE SPRAY (CS) SYSTEM LOGICS. TWO ENTRIES INTO SPECIFICATION 3.0.3 ARE REQUIRED TO CONDUCT THIS EVOLUTION; THE FIRST OCCURRED AT 2031 AND WAS EXITED AT 2050, THE SECOND OCCURRED AT 2216 AND WAS EXITED AT 2240. SUBSEQUENT TO REPLACEMENT, IT WAS DETERMINED THAT A POTENTIAL ENVIRONMENTAL QUALIFICATION (EQ) CONCERN EXISTED WITH THE NEW TRANSMITTER, AS SUCH, ANOTHER ENTRY INTO SPECIFICATION 3.0.3 OCCURRED ON 10/4/90 AT 1806 TO AGAIN REPLACE THE TRANSMITTER. SPECIFICATION 3.0.3 WAS EXITED AT 1822, ENTERED AGAIN AT 2140 TO RETURN THE REFERENCE LEG TO OPERABILITY, AND WAS EXITED AT 2153 WHEN

TOTAL EQUIPMENT RESTORATION WAS COMPLETED. ROOT CAUSE OF THE EQ CONCERN WAS A DEFECTIVE TRANSMITTER.

[82] INDIAN POINT 2 DOCKET 50-247 LER 90-008
PLANT VENT GASEOUS ACTIVITY MONITOR ESF ACTUATION DUE TO ELECTRICAL SPIKE.
EVENT DATE: 091890 REPORT DATE: 101890 NSSS: WE TYPE: PWR

(NSIC 219816) AT THE CONCLUSION OF A CONTAINMENT PRESSURE RELIEVING OPERATION ON SEPTEMBER 18, 1990, WITH THE PLANT AT 96% POWER, THE PLANT VENT GASEOUS ACTIVITY MONITOR (R-14) EXPERIENCED A SPURIOUS ELECTRICAL SPIKE, WHICH IN TURN INITIATED CONTAINMENT VENTILATION ISOLATION AND PARTIALLY ACTUATED THE WELD CHANNEL AND CONTAINMENT PENETRATION PRESSURIZATION SYSTEM. A REVIEW OF OTHER RADIATION MONITORING INSTRUMENTATION CONFIRMED THERE HAD BEEN NO ACTUAL INCREASE IN GASEOUS ACTIVITY. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED BY THIS EVENT.

[83] INDIAN POINT 2 DOCKET 50-247 LER 90-010
ESF ACTUATION DUE TO ELECTRICAL SPIKE ON RADIATION MONITORS.
EVENT DATE: 092390 REPORT DATE: 102390 NSSS: WE TYPE: PWR

(NSIC 219818) DURING THE PERFORMANCE OF A PRESSURE RELIEF OF CONTAINMENT ON SEPTEMBER 23, 1990, WITH THE PLANT AT 97.5% POWER, THE CONTAINMENT RADIOGAS MONITOR (R-12) AND THE PLANT VENT GASEOUS ACTIVITY MONITOR (R-14) SIMULTANEOUSLY EXPERIENCED A SPURIOUS ELECTRICAL SPIKE, WHICH IN TURN INITIATED CONTAINMENT VENTILATION ISOLATION AND PARTIALLY ACTUATED THE WELD CHANNEL AND CONTAINMENT PENETRATION PRESSURIZATION SYSTEM. AFTER DETERMINING THERE HAD BEEN NO ACTUAL INCREASE IN GASEOUS ACTIVITY, BOTH RADIATION MONITORS WERE RESET AND PRESSURE RELIEF WAS REINSTITUTED. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED BY THIS EVENT.

[84] INDIAN POINT 2 DOCKET 50-247 LER 90-009
TOXIC GAS MONITOR ALARM RESULTING IN CENTRAL CONTROL ROOM VENTILATION SYSTEM ISOLATION.
EVENT DATE: 092690 REPORT DATE: 102690 NSSS: WE TYPE: PWR
VENDOR: WISCONSIN BRIDGE & IRON

(NSIC 219817) ON SEPTEMBER 26, 1990, AT ABOUT 0443 HOURS, WITH REACTOR POWER AT 97%, THE CHLORINE TOXIC GAS MONITOR CHANNEL 1 ALARMED, RESULTING IN THE TRANSFER OF THE CENTRAL CONTROL ROOM (CCR) VENTILATION SYSTEM FROM THE NORMAL MODE TO THE INCIDENT MODE. AS DESIGNED, THE DETECTION OF CHLORINE BY EITHER CHANNELS 1 OR 2 OF THE TOXIC GAS MONITORS WILL GENERATE AN ALARM IN THE CCR AND ISOLATE THE CCR VENTILATION SYSTEM. THE TOXIC GAS MONITORING SYSTEM IS CLASSIFIED AS AN ENGINEERED SAFETY FEATURE (ESF). NO TECHNICAL SPECIFICATION OR NRC LIMITS WERE EXCEEDED.

[85] KEWAUNEE DOCKET 50-305 LER 90-009
BOTH DIESEL GENERATORS INOPERABLE FOR 1 HOUR AND 40 MINUTES AS A RESULT OF A RETAINING RING INSTALLATION DUE TO POSSIBLE PERSONNEL ERROR.
EVENT DATE: 091790 REPORT DATE: 101790 NSSS: WE TYPE: PWR
VENDOR: ELECTRO - MOTIVE DIV. OF GM

(NSIC 219854) AT 1035 ON 9/19/90, WITH THE PLANT AT 100% POWER, IT WAS DETERMINED THAT BOTH DIESEL GENERATORS (DG) HAD BEEN OUT OF SERVICE FROM 1443 TO 1623 ON 9/17. THE EVENT WAS DISCOVERED WHEN IT WAS DETERMINED THAT THE 1A DG HAD BEEN INOPERABLE SINCE 9/14 AND THAT THE 1B DG HAD BEEN REMOVED FROM SERVICE DURING THIS TIME PERIOD FOR SURVEILLANCE TESTING. THE 1A DG WAS CONSIDERED INOPERABLE BECAUSE THE FUEL INJECTION ROCKER FOR THE NUMBER 10 CYLINDER BECAME DISENGAGED FROM ITS FUEL INJECTOR. THIS COULD HAVE PREVENTED THE 1A DG FROM REACHING ITS DESIGN CAPACITY. THEREFORE THE 1A DG WAS CONSERVATIVELY DECLARED INOPERABLE. THE LOST PROBABLE CAUSE OF THIS EVENT IS AN IMPROPERLY INSTALLED RETAINING SPRING. THE RETAINING SPRING IS LOCATED IN THE DG FUEL INJECTION SYSTEM. DEMONSTRATION OF DG OPERABILITY REQUIRES STARTING A DG BUT NOT LOADING IT. THEREFORE, THE INOPERABILITY OF THE 1A DG WAS NOT DISCOVERED PRIOR TO REMOVING

THE 1B DG FROM SERVICE ON 9/17. IMMEDIATE CORRECTIVE ACTIONS INCLUDED REPAIRING THE 1A DIESEL GENERATOR AND VERIFYING THAT THE REMAINING RETAINING SPRINGS ON THE 1A AND 1B DIESEL GENERATORS WERE PROPERLY INSTALLED. IN THE LONG TERM, THIS REPORT WILL BE INCLUDED IN THE NEXT TRAINING SESSION ON SIGNIFICANT INDUSTRY EVENTS FOR THE PLANT MAINTENANCE MECHANICS.

[86] LIMERICK 1 DOCKET 50-352 LER 90-013 REV 01
UPDATE ON THE AFFECTS OF UNDER-RATED DC FUSES AND THE FAILURE TO MAINTAIN ADEQUATE ELECTRICAL ISOLATION BETWEEN CLASS 1E AND NON-CLASS 1E COMPONENTS.
EVENT DATE: 061190 REPORT DATE: 110190 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 219880) ON JUNE 11, 1990, A REVIEW OF THE DC ELECTRICAL DISTRIBUTION SYSTEM IDENTIFIED THAT THE UNIT 1 AND UNIT 2, DIVISION 1 AND DIVISION 2, DC DISTRIBUTION SYSTEMS HAD INADEQUATE ISOLATION CAPABILITY BETWEEN CLASS 1E AND NON-CLASS 1E COMPONENTS AND ALSO HAD UNDER-RATED DC FUSES. UNITS 1 AND 2 DIVISIONS 1 AND 2 DC DISTRIBUTION SYSTEMS WERE DECLARED INOPERABLE UNTIL ELECTRICAL DISCONNECTS WERE OPENED TO ENSURE PROPER ELECTRICAL ISOLATION BETWEEN THE ASSOCIATED CLASS 1E AND NON-CLASS 1E COMPONENTS. A MODIFICATION WAS IMPLEMENTED THAT PROVIDES ADDITIONAL ISOLATION PROTECTION AND ADEQUATELY RATED FUSES. FURTHER INVESTIGATION ON JUNE 13, 1990 IDENTIFIED THAT FIRE PROTECTION SAFE SHUTDOWN (SSD) METHODS 'B' (UNIT 1) OR 'C' (UNIT 2) COULD BE AFFECTED DUE TO POSTULATED FIRE INDUCED HIGH IMPEDANCE FAULTS FAILING TO ISOLATE OVERLOAD CURRENT CONDITIONS DUE TO UNDER-RATED DC FUSES. IMMEDIATE CORRECTIVE ACTIONS WERE TAKEN TO ESTABLISH HOURLY FIRE WATCHES IN THE AFFECTED UNIT 2 FIRE AREAS UNTIL JUNE 26, 1990 WHEN A MODIFICATION WAS COMPLETED. THE AFFECTED UNIT 1 FIRE AREA WAS NOT FIRE WATCHED SINCE UNIT 1 WAS IN COLD SHUTDOWN AT THE TIME. PROXIMATE CAUSES OF THESE CONDITIONS ARE ERRORS MADE DURING THE ORIGINAL DESIGN WHEN WE CONCLUDED THAT THE POSITIVE LEG AND NEGATIVE LEG FUSES CONSTITUTED A TECHNICALLY CORRECT DOUBLE FUSING DESIGN.

[87] LIMERICK 1 DOCKET 50-352 LER 90-019
SPECIAL REPORT FOR DIESEL GENERATOR SURVEILLANCE TEST FAILURE.
EVENT DATE: 091590 REPORT DATE: 101790 NSSS: GE TYPE: BWR
VENDOR: BASLER ELECTRIC COMPANY

(NSIC 219881) ON SEPTEMBER 15, 1990, WITH UNIT 1 IN A REFUELING OUTAGE, PLANT PERSONNEL WERE PERFORMING SURVEILLANCE TEST (ST) PROCEDURE ST-1-092-113-1, "D13 DIESEL GENERATOR 4 KV SFGD LOSS OF POWER LSF/SAA AND OUTAGE TESTING" ON THE UNIT 1 D13 EMERGENCY DIESEL GENERATOR (EDG). A DIVISION 3 SAFEGUARD BUS OVERVOLTAGE CONDITION OCCURRED DURING ENERGIZATION OF THE SAFEGUARD BUS BY THE D13 EDG. THE EDG OUTPUT BREAKER WAS MANUALLY TRIPPED FROM THE MAIN CONTROL ROOM AND THE D13 EDG WAS DECLARED INOPERABLE. ADDITIONALLY, VARIOUS LOADS ON THE DIVISION 3 SAFEGUARD BUS INCURRED BLOWN FUSES AND MINOR BREAKER MALFUNCTIONS DURING THIS EVENT. THE EQUIPMENT POWERED BY THE DIVISION 3 SAFEGUARDS BUS WAS DECLARED INOPERABLE. FOLLOWING IMMEDIATE CORRECTIVE ACTIONS, THE EQUIPMENT POWERED BY THE DIVISION 3 SAFEGUARD BUS WAS DECLARED OPERABLE ON SEPTEMBER 18, 1990, AND THE D13 EDG WAS DECLARED OPERABLE ON SEPTEMBER 30, 1990. THE TERMINATED TEST WAS CLASSIFIED AS A VALID TEST FAILURE IN ACCORDANCE WITH THE GUIDANCE IN REGULATORY GUIDE 1.108. THE CAUSE OF THE EVENT WAS A MALFUNCTION IN THE NUMBER ONE EDG RECTIFIER BANK CONTAINED IN THE VOLTAGE REGULATION CIRCUIT. A FAILURE ANALYSIS IS BEING PERFORMED TO DETERMINE THE ROOT CAUSE OF THE RECTIFIER FAILURE. THE RESULTS OF THE ANALYSIS AND ANY PLANNED CORRECTIVE ACTIONS WILL BE PROVIDED IN A SUPPLEMENT TO THIS REPORT.

[88] LIMERICK 1 DOCKET 50-352 LER 90-020
MANUAL ISOLATION OF THE MAIN CONTROL ROOM DUE TO A HIGH TOXIC CHEMICAL CONCENTRATION SIGNAL.
EVENT DATE: 091890 REPORT DATE: 101690 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 219882) ON SEPTEMBER 18, 1990, AT 1120 HOURS THE HIGH TOXIC CHEMICAL CONCENTRATION ALARM WAS RECEIVED IN THE MAIN CONTROL ROOM (MCR). MCR PERSONNEL IMMEDIATELY IMPLEMENTED SPECIAL EVENT PROCEDURE SE-2, "TOXIC GAS," AND DONNED

SELF-CONTAINED BREATHING APPARATUS (SCBA). AT 1125 HOURS MCR PERSONNEL MANUALLY INITIATED A MCR VENTILATION SYSTEM CHLORINE ISOLATION, AN ENGINEERED SAFETY FEATURE (ESF). THE 'B' TRAIN OF THE CONTROL ROOM EMERGENCY FRESH AIR SUPPLY (CREFAS) SYSTEM, ALSO AN ESF, INITIATED AS DESIGNED AND PROVIDED TOTAL RECIRCULATION OF THE MCR AIR WITHOUT ANY INTAKE FROM THE OUTSIDE ATMOSPHERE. THE 'A' MCR TOXIC GAS ANALYZER READ A VINYL CHLORIDE CONCENTRATION OF 10.21 PPM (ALARM SETPOINT IS 10 PPM) WHICH IS WELL BELOW THE HAZARDOUS CONCENTRATION LIMIT. CHEMISTRY PERSONNEL OBTAINED AND ANALYZED A SAMPLE OF THE MCR AIR. THE SAMPLE SHOWED NO DETECTABLE VINYL CHLORIDE SO MCR PERSONNEL REMOVED SCBA. AFTER THE AIR SAMPLE FROM THE OUTSIDE AIR INTAKE PLENUM SHOWED NO DETECTABLE VINYL CHLORIDE, MCR OPERATORS RESET THE CHLORINE ISOLATION AT 1320 HOURS. THE CAUSE OF THE EVENT IS UNKNOWN.

[89] LIMERICK 1 DOCKET 50-352 LER 90-021
INOPERABILITY OF THE SEISMIC MONITORING SYSTEM DUE TO A FAULTY INVERTER BOARD IN THE MAGNETIC TAPE PLAYBACK SYSTEM.
EVENT DATE: 101190 REPORT DATE: 102290 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 219883) ON SEPTEMBER 11, 1990 AT 0841 HOURS, THE SEISMIC MONITORING SYSTEM (SMS) WAS DECLARED INOPERABLE IN PREPARATION FOR PERFORMANCE OF A SURVEILLANCE TEST (ST) PROCEDURE. THIS PARTICULAR ST PROCEDURE REQUIRES APPROXIMATELY 30 DAYS TO COMPLETE. WHILE THE ST PROCEDURE IS IN PROGRESS, THE PORTIONS OF THE SMS THAT MEASURE AND RECORD SEISMIC EVENT DATA ARE UNAVAILABLE. HOWEVER, THE COMPONENTS OF THE SMS THAT ALERT MAIN CONTROL ROOM (MCR) PERSONNEL OF AN OPERATING BASIS EARTHQUAKE ARE AVAILABLE. ON SEPTEMBER 21, 1990, WHILE PERFORMING THE ST PROCEDURE, INSTRUMENTATION AND CONTROLS (I&C) PERSONNEL DISCOVERED THAT THE INVERTER BOARD IN THE MAGNETIC TAPE PLAYBACK SYSTEM FAILED TO CALIBRATE TO THE REQUIRED SPECIFICATIONS. I&C PERSONNEL THEN INITIATED A PURCHASE REQUISITION FOR THE REPLACEMENT OF THE INVERTER BOARD. ON OCTOBER 11, 1990, AT 0841 HOURS, THE REPLACEMENT INVERTER BOARD HAD NOT BEEN RECEIVED AND THE SMS BECAME INOPERABLE FOR MORE THAN 30 DAYS, THEREFORE REQUIRING THE SUBMISSION OF A SPECIAL REPORT IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS SECTION 3.3.7.2 . THERE HAVE BEEN NO ADVERSE CONSEQUENCES ASSOCIATED WITH THIS EVENT. NO SEISMIC EVENTS HAVE OCCURRED TO DATE SINCE THE SMS WAS DECLARED INOPERABLE. THE CAUSE OF THE MALFUNCTION WAS A FAULTY INVERTER BOARD.

[90] LIMERICK 2 DOCKET 50-353 LER 90-016
FAILURE TO MEET TECHNICAL SPECIFICATION 3.7.6.4 SINCE THE HALON SYSTEM HAD BEEN INOPERABLE AND THE TS ACTION WAS NOT TAKEN IN THE APPROPRIATE TIME PERIOD.
EVENT DATE: 081190 REPORT DATE: 091090 NSSS: GE TYPE: BWR

(NSIC 219884) ON 7/5/90, UNIT 2 SURVEILLANCE TEST (ST) PROCEDURE ST-7-022-353-2, "HALON SYSTEM INVENTORY," WAS PERFORMED FOR THE AUXILIARY EQUIPMENT ROOM (AER) HALON FIRE SUPPRESSION SYSTEM. THIS ST IDENTIFIED ONE WEIGHT DEFICIENT BOTTLE IN THE HALON SYSTEM MAIN BANK. THE MAIN BANK WAS DECLARED INOPERABLE AND THE "MAIN/RESERVE" SWITCH (HS-22-283A) WAS PLACED IN THE "RESERVE" POSITION SWITCHING THE HALON SYSTEM TO ITS REDUNDANT RESERVE BANK AND ENSURING SYSTEM OPERABILITY. AN EQUIPMENT STATUS CONTROL TAG WAS THEN PLACED ON HS-22-283A TO INDICATE THE SYSTEM WAS IN AN OFF-NORMAL SYSTEM ALIGNMENT. THE DEFICIENT BOTTLE WAS REMOVED ON 7/24/90, TO BE REFILLED AND HS-22-283A WAS STILL IN THE "AS LEFT" POSITION OF "RESERVE." HOWEVER, ON 8/13/90, WHILE PREPARING TO REPLACE THE REFILLED MAIN BANK BOTTLE, STATION PERSONNEL DISCOVERED THAT HS-22-283A WAS IN THE "MAIN" POSITION WITH THE EQUIPMENT STATUS CONTROL TAG REMOVED. HS-22-283A WAS IMMEDIATELY PUT IN THE "RESERVE" POSITION BY THE FIRE PROTECTION SYSTEM ENGINEER AND HALON SYSTEM OPERABILITY WAS RESTORED. THERE WERE NO ADVERSE CONSEQUENCES AND NO RADIOACTIVE MATERIAL WAS RELEASED TO THE ENVIRONMENT AS A RESULT OF THIS EVENT. DURING THIS TIME THE UNIT 2 HALON SYSTEM WAS ALIGNED WITH THE MAIN BANK HALON SUPPLY, CONTAINING SIX FULLY CHARGED BOTTLES, NO FIRES OCCURRED IN THE UNIT 2 AER.

[91] LIMERICK 2 DOCKET 50-353 LER 90-017
AN INADVERTENT ACTUATION OF THE PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM DUE TO PERSONNEL ERROR.
EVENT DATE: 091690 REPORT DATE: 101690 NSSS: GE TYPE: BWR

(NSIC 219825) ON 9/16/90, AN INADVERTENT AUTOMATIC ACTUATION OF THE PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM (PCRVICS), AN ESF OCCURRED. THIS ESF ACTUATION RESULTED IN THE CLOSURE OF THE UNIT 2 HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM STEAM SUPPLY LINE OUTBOARD PRIMARY CONTAINMENT ISOLATION VALVE, HV-055-2F003 DUE TO RE-ENERGIZING THE MOTOR OPERATOR FOR THIS VALVE PRIOR TO RESETTING THE ISOLATION LOGIC FROM THE MAIN CONTROL ROOM (MCR). CAUSE OF THIS EVENT WAS THE FAILURE OF THE UNIT 2 REACTOR OPERATOR (RO) TO BE AWARE OF SYSTEM CONDITIONS PRIOR TO RESETTING THE PCRVICS ISOLATION LOGIC. AN ADDITIONAL CAUSAL FACTOR IS INADEQUATE COMMUNICATION BETWEEN THE UNIT 2 RO AND THE MCR SHIFT SUPERVISOR. BOTH OF THESE OPERATORS WERE COUNSELED. THERE WERE NO ADVERSE CONSEQUENCES AND NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT. IN THE EVENT OF AN ACCIDENT REQUIRING THE USE OF THE HPCI SYSTEM DURING THE TIME PERIOD THAT IT WAS ISOLATED, OPERATORS COULD HAVE UNISOLATED HV-055-2F003 AND THEN MANUALLY INITIATED THE HPCI SYSTEM. IN ADDITION, THE AUTOMATIC DEPRESSURIZATION SYSTEM, THE LOW PRESSURE EMERGENCY CORE COOLING SYSTEMS, AND THE REACTOR CORE ISOLATION COOLING SYSTEM WERE OPERABLE TO RESPOND TO AN ACCIDENT CONDITION IN THE EVENT THAT THE HPCI SYSTEM COULD NOT HAVE BEEN MANUALLY INITIATED.

[92] MAINE YANKEE DOCKET 50-309 LER 90-006
EMERGENCY FEEDWATER TRIP SYSTEM.
EVENT DATE: 091390 REPORT DATE: 091590 NSSS: CE TYPE: PWR

(NSIC 219892) ON 9/13/90, A DETERMINATION WAS MADE THAT A FEED TRAIN LINEUP UTILIZED DURING HOT STANDBY OPERATIONS MAY NOT CONFORM WITH TECH SPECS. THE FEED TRAIN TRIP SYSTEM IS REQUIRED TO BE OPERABLE WHENEVER THE REACTOR COOLANT SYSTEM BORON CONCENTRATION IS LESS THAN THAT REQUIRED FOR HOT SHUTDOWN (HOT STANDBY AND POWER OPERATIONS). DURING HOT STANDBY OPERATIONS, PLANT PROCEDURES PERMIT A LINEUP OF THE EMERGENCY FEEDWATER (EFW) PUMPS THROUGH A PORTION OF THE MAIN FEED SYSTEM. THIS LINEUP PERMITS PREHEATING OF THE EFW FLOW TO MINIMIZE THERMAL STRESSES ON FEEDWATER PIPING AND STEAM GENERATOR FEED RINGS (REFER TO THE ATTACHED FIGURE). THE LINEUP RESULTED IN THE FEED TRAIN TRIP SYSTEM BEING SINGLE FAILURE VULNERABLE BELOW 2% REACTOR POWER. A REVIEW OF PLANT DOCUMENTATION REVEALED THAT THE LINEUP HAD BEEN ANALYZED, AND WAS DETERMINED TO BE SAFE. HOWEVER, IT WAS NOT CLEAR WHETHER THE TECH SPEC PERMITTED SUCH A LINEUP. IT WAS DECIDED THAT A CLARIFICATION WOULD BE MADE TO TECH SPECS REGARDING SYSTEM LINEUP BELOW 2% POWER. IMMEDIATELY AFTER DISCOVERING THE POSSIBLE DISCREPANCY, PROCEDURES WERE REVISED TO PROHIBIT ALIGNING EFW FLOW THROUGH THE MAIN FEEDWATER SYSTEM DURING HOT STANDBY OPERATIONS.

[93] MCGUIRE 1 DOCKET 50-369 LER 90-010 REV 01
UPDATE ON ANNULUS VENTILATION AND CONTROL ROOM VENTILATION SYSTEM FILTER TRAIN HEATERS WERE INADEQUATELY SIZED BECAUSE OF A DESIGN DEFICIENCY.
EVENT DATE: 042790 REPORT DATE: 061390 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 219914) IN MARCH OF 1990, CATAWBA NUCLEAR STATION PERSONNEL WERE REVIEWING THE EFFECTS OF POTENTIALLY DEGRADED GRID VOLTAGE. IT WAS DISCOVERED THAT THE ANNULUS VENTILATION (VE) AND CONTROL ROOM VENTILATION (VC) SYSTEM HEATERS WOULD NOT OPERATE AS DESIGNED UNDER ALL POSTULATED OPERATING CONDITIONS. AFTER REVIEWING THE MCGUIRE VE AND VC SYSTEM HEATERS, IT WAS DETERMINED THEY WOULD NOT OPERATE AS DESIGNED UNDER ALL POSTULATED OPERATING CONDITIONS. UNIT 1 WAS IN MODE 5 (COLD SHUTDOWN) PREPARING TO ENTER MODE 4 (HOT SHUTDOWN). UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT 100% POWER. AN OPERABILITY EVALUATION WAS INITIATED ON UNIT 1 AND UNIT 2 VE AND VC SYSTEMS. UNIT 2 VE SYSTEM AND UNIT 1 AND 2 VC SYSTEMS WERE DETERMINED TO BE CONDITIONALLY OPERABLE. HOWEVER, UNIT 1 VE SYSTEMS WAS DETERMINED TO BE INOPERABLE. AN EMERGENCY TECH SPEC (TS) CHANGE REQUEST WAS SUBMITTED TO THE NUCLEAR REGULATORY COMMISSION (NRC) ON 5/9/90. THE NRC APPROVED THE EMERGENCY TS CHANGE ON 5/11/90. PRESENTLY, THE CHANGE TO UNIT 2 IS

ADMINISTRATIVE IN NATURE ONLY BECAUSE IT SHARES A COMMON TS DOCUMENT WITH UNIT 1. THE CHANGE WILL ONLY APPLY UNTIL 7/16/91. AFTER THAT TIME, THE NRC WILL REQUIRE THE UNIT 1 AND A UNIT 2 TS TO READ THE SAME. THIS EVENT IS ASSIGNED A CAUSE OF DESIGN DEFICIENCY BECAUSE OF UNANTICIPATED INTERACTION OF SYSTEMS DUE TO DESIGN OVERSIGHT.

[94] MCGUIRE 1 DOCKET 50-369 LER 90-025 REV 01
UPDATE ON SHUTDOWN BECAUSE OF UNIDENTIFIED REACTOR COOLANT SYSTEM LEAKAGE GREATER THAN TECHNICAL SPECIFICATION LIMITS CAUSED BY EQUIPMENT FAILURE.
EVENT DATE: 082790 REPORT DATE: 092690 NSSS: WE TYPE: PWR
VENDOR: BORG-WARNER CORP.

(NSIC 219915) ON AUGUST 27, 1990, AT 0554 HOURS, UNIT 1 BEGAN REDUCING LOAD TO COMPLY WITH THE UNIDENTIFIED LEAKAGE TECHNICAL SPECIFICATION. LEAKAGE CALCULATIONS INDICATED THAT UNIDENTIFIED LEAKAGE WAS GREATER THAN 1 GALLON PER MINUTE. AN UNUSUAL EVENT WAS DECLARED AT THIS TIME. UNIT 1 ENTERED MODE 3 (HOT STANDBY), AT 1013 HOURS. ENTRIES WERE MADE INTO CONTAINMENT AND PRESSURIZER PORV HEADER HI POINT VENT VALVE, 1NC-252, WAS FOUND TO HAVE A PACKING LEAK OF APPROXIMATELY 8 OUNCES PER HOUR. THE PACKING WAS ADJUSTED AND THE LEAK STOPPED. IN ADDITION, PRESSURIZER RELIEF ISOLATION VALVE, 1NC-33, WAS FOUND TO HAVE A PACKING LEAK AND THE VALVE STEM LEAKOFF LINE WAS SEPARATED FROM THE DRAIN PIPING. THE LEAKOFF LINE WAS REPLACED AND VALVE 1NC-33 WAS BACKSEATED TO STOP ITS LEAKAGE. THE PACKING ON VALVE 1NC-33 WAS TORQUED TO ITS MAXIMUM ALLOWED VALUE. THE POWER TO THE ASSOCIATED PRESSURIZER POWER OPERATED RELIEF VALVE SOLENOID WAS DEENERGIZED BECAUSE VALVE 1NC-33 IS TECHNICALLY INOPERABLE WHILE BACKSEATED. VALVE 1NC-33 WILL BE REPACKED DURING THE NEXT APPROPRIATE OUTAGE. THIS EVENT IS ASSIGNED CAUSE OF EQUIPMENT FAILURE. UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER AT THE TIME THE LEAKAGE WAS DISCOVERED.

[95] MILLSTONE 1 DOCKET 50-245 LER 90-010 REV 01
UPDATE ON EQUIPMENT ENVIRONMENTAL QUALIFICATION BARRIERS VIOLATED.
EVENT DATE: 061390 REPORT DATE: 100590 NSSS: GE TYPE: BWR

(NSIC 219815) ON 12/23/89, WITH THE PLANT AT 100% POWER (530F AND 1030 PSIG), THE TURBINE DECK TO HEATING AND VENTILATION (H&V) ROOM DOUBLE DOORS AND HEATING VENTILATION SUPPLY (HVS) 6A/B DOOR TO THE SWITCHGEAR AREA WERE OPENED FOR A DURATION OF APPROXIMATELY 16 HOURS. THIS WAS DONE IN AN ATTEMPT TO CORRECT A LOW TEMPERATURE CONDITION IN THE SWITCHGEAR AREA BY DRAWING WARM AIR FROM THE TURBINE DECK THROUGH THE VENTILATION SYSTEM TO THE SWITCHGEAR AREA. WITH THE SWITCHGEAR AREA AND H&V ROOM A MILD ENVIRONMENT, AND THE TURBINE DECK A POTENTIAL HARSH ENVIRONMENT, EEQ BARRIERS WERE VIOLATED. THIS RESULTED IN CHANGING THE ENVIRONMENTS IN THE H&V ROOM AND SWITCHGEAR ROOM TO POTENTIAL HARSH ENVIRONMENTS. A REPORTABILITY EVALUATION WAS INITIATED ON 5/14/90, TO DETERMINE IF A REPORTABLE CONDITION EXISTED. ON 6/13/90, RESULTS OF THE REPORTABILITY EVALUATION CONCLUDED BLOCKING OPEN THE DOUBLE DOORS BETWEEN THE H&V ROOM AND THE TURBINE DECK AND THE DOOR HVS 6A/B REPRESENTED A DEGRADATION BETWEEN POTENTIAL HARSH AND MILD ENVIRONMENTS. ON 7/20/90 WHILE REVIEWING ROUTINE PLANT EVALUATIONS TO DETERMINE IF ADDITIONAL EEQ HARSH-MILD STRUCTURAL BARRIERS ARE OPENED DURING THESE ACTIVITIES, IT WAS DISCOVERED THAT CERTAIN ACTIVITIES, SUCH AS BIMONTHLY LOADING OF RESINS FROM THE 14'6" ELEVATION OF THE TURBINE BUILDING TO THE 34'6" ELEVATION OF THE TURBINE BUILDING, INVOLVED OPENING AN ACCESS BETWEEN A HARSH AND MILD ENVIRONMENT.

[96] MILLSTONE 1 DOCKET 50-245 LER 90-014
LOW PRESSURE COOLANT INJECTION HEAT EXCHANGER FLOW RATES.
EVENT DATE: 090790 REPORT DATE: 100990 NSSS: GE TYPE: BWR
VENDOR: PERFEX, INC.

(NSIC 219694) ON 9/7/90, AT 1845 HOURS, WITH THE PLANT AT 100% POWER (530F AND 1030 PSIG), AN INCONSISTENCY BETWEEN PROCEDURAL AND DESIGN PARAMETERS ASSOCIATED WITH THE LOW PRESSURE COOLANT INJECTION (LPCI) HEAT EXCHANGER FLOW RATES WAS IDENTIFIED. THE INCONSISTENCY WAS ASSOCIATED WITH THE MAXIMUM LPCI FLOW PERMITTED THROUGH THE HEAT EXCHANGER TO PRECLUDE FAILURE DUE TO EROSION AND

FLOW-INDUCED VIBRATION, AND THE HEAT EXCHANGER FLOW RATES REQUIRED BY THE EMERGENCY OPERATING PROCEDURES (EOP'S). AFTER REVIEW OF THE PROCEDURES, THE DESIGN BASIS, AND DISCUSSIONS WITH THE HEAT EXCHANGER MANUFACTURER, IT WAS DETERMINED THAT OPERABILITY OF THE CONTAINMENT COOLING SYSTEM COULD NOT BE ASSURED DUE TO POTENTIAL MECHANICAL LIMITATIONS OF THE HEAT EXCHANGER. BOTH CONTAINMENT COOLING SUBSYSTEMS WERE DECLARED INOPERABLE AND A PLANT SHUTDOWN TO COLD SHUTDOWN WAS IMMEDIATELY INITIATED AS REQUIRED BY TECH SPECS. COLD SHUTDOWN WAS ACHIEVED ON 9/8/90 AT 1705 HOURS. NO SAFETY SYSTEMS WERE REQUIRED TO FUNCTION AS A RESULT OF THIS EVENT AND NO SAFETY CONSEQUENCES RESULTED FROM THIS EVENT.

[97] MILLSTONE 1 DOCKET 50-245 LER 90-015
REACTOR SCRAM ON LOW WATER LEVEL.
EVENT DATE: 091490 REPORT DATE: 101290 NSSS: GE TYPE: BWR

(NSIC 219695) ON 9/14/90, WITH THE PLANT AT 100% POWER (530F AND 1030 PSIG), A FULL REACTOR SCRAM OCCURRED ON LOW REACTOR WATER LEVEL (+8 INCHES) AFTER THE FEEDWATER REGULATING VALVES BEGAN TO CLOSE. THE FEEDWATER REGULATING VALVES WERE RESPONDING TO A HIGH INDICATED REACTOR WATER LEVEL SIGNAL FROM THE 'A' CHANNEL OF THE FEEDWATER CONTROL SYSTEM, WHICH WAS CONTROLLING FEEDWATER FLOW. AT THE TIME OF THE SCRAM, A TECHNICIAN WAS PERFORMING A CALIBRATION ON A PRESSURE SWITCH WHICH SENSES PRESSURE FROM AN INSTRUMENT LINE COMMON TO THE REFERENCE LEG OF THE 'A' FEEDWATER CONTROL SYSTEM. ALL SAFETY SYSTEMS FUNCTIONED AS REQUIRED AND NO SAFETY CONSEQUENCES RESULTED FROM THIS EVENT.

[98] MILLSTONE 2 DOCKET 50-336 LER 90-015
INADVERTENT ESAS ACTUATIONS DUE TO OPERATOR ERROR AND VOLTAGE TRANSIENTS.
EVENT DATE: 091990 REPORT DATE: 101990 NSSS: CE TYPE: PWR
VENDOR: CONSOLIDATED CONTROLS CORP.

(NSIC 219902) WHILE IN MODE 5 ON 9/19/90, AT 0244, THE UNIT EXPERIENCED AN INADVERTENT ISOLATION OF CONTAINMENT PURGE VALVES 2-AC-4, 2-AC-5, 2-AC-6, AND 2-AC-7. OPERATORS VERIFIED THAT THE ENGINEERED SAFEGUARDS ACTUATION SYSTEM (ESAS) FUNCTIONED PROPERLY, THAT THE ACTUATION WAS INADVERTENT AND REOPENED THE VALVES. THIS ACTUATION RESULTED FROM OPERATOR ERROR DURING THE OPERATION OF A WRONG CIRCUIT BREAKER. THERE WERE NO SAFETY IMPLICATIONS. ON 9/20/90, WHILE IN MODE 5, AT 1103, THE UNIT EXPERIENCED AN INADVERTENT ACTUATION OF THE FACILITY 2 SAFETY INJECTION ACTUATION SYSTEM (SIAS), CONTAINMENT ISOLATION ACTUATION SYSTEM (CIAS), AND ENCLOSURE BUILDING FILTRATION ACTUATION SYSTEM (EBFAS). WHEN THE ACTUATION OCCURRED, CHARGING PUMPS P-18B AND C STARTED. REACTOR OPERATORS MANUALLY OPENED THE CHARGING HEADER ISOLATION VALVE AND ONE OF THE TWO LOOP CHARGING SUPPLY ISOLATION VALVES TO PROVIDE A FLOW PATH AND PREVENT CHALLENGING THE PUMPS RELIEF VLAVES. THE ESTIMATED WATER ADDITION TO THE REACTOR COOLANT SYSTEM (RCS) WAS LESS THAN 50 GALLONS. OPERATORS IMPLEMENTED AOP 2571, VERIFIED THAT THE ACTUATION WAS INADVERTENT AND RETURNED THE ACTUATED EQUIPMENT TO THE LINEUP THAT EXISTED PRIOR TO THE EVENT. THE ACTUATION WAS CAUSED BY VOLTAGE TRANSIENTS CREATED WHEN THE CHANNEL "A" PRESSURIZER PRESSURE INHIBIT SWITCH WAS OPERATED WHILE CHANNEL "D" PRESSURIZER PRESSURE WAS BEING CALIBRATED.

[99] MILLSTONE 3 DOCKET 50-423 LER 89-018 REV 01
UPDATE ON INOPERABLE WASTE NEUTRALIZATION SUMP EFFLUENT RADIATION MONITOR DUE TO PERSONNEL ERROR.
EVENT DATE: 080289 REPORT DATE: 100190 NSSS: WE TYPE: PWR

(NSIC 219689) ON 8/2/89 AT 1430, IN MODE 1 AT 100% POWER, SHIFT PERSONNEL DECLARED THE CONDENSATE DEMINERALIZER WASTE NEUTRALIZATION SUMP EFFLUENT RADIATION MONITOR, 3CND-07 INOPERABLE DUE TO INADEQUATE TESTING OF THE HIGH ACTIVITY AUTOMATIC TERMINATION FEATURE, SINCE PLANT STARTUP. THE COMPENSATORY ACTIONS REQUIRED BY THE TECH SPECS HAD ALREADY BEEN INCORPORATED INTO EXISTING PROCEDURES AND WERE BEING PERFORMED. HOWEVER, TECH SPECS ALSO REQUIRES AN EXPLANATION BE SUBMITTED IN THE NEXT SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT INDICATING WHY THE "INOPERABILITY" WAS NOT CORRECTED IN A TIMELY MANNER (30 DAYS). THIS ACTION WAS NOT PERFORMED. THE ROOT CAUSE OF THE EVENT WAS

PERSONNEL ERROR. PLANT PERSONNEL FAILED TO INCORPORATE ALL OF THE AUTOMATIC RELEASE TERMINATION CAPABILITY OF 3CND-07 INTO THE APPLICABLE SURVEILLANCE PROCEDURE. WITH THE RADIATION MONITOR INOPERABLE AN EXPLANATION WAS NEVER PROVIDED IN THE SEMIANNUAL EFFLUENT RELEASE REPORT. NO IMMEDIATE CORRECTIVE ACTION WAS REQUIRED. THE SURVEILLANCE PROCEDURE WAS CHANGED TO INCORPORATE THE INTERLOCKS FOR ALL RELEASE PATHS AND WERE SATISFACTORILY TESTED. AN EXPLANATION WILL BE PROVIDED IN THE NEXT SEMIANNUAL RELEASE REPORT. NO OTHER DEFICIENCIES WERE IDENTIFIED IN OTHER RADIATION MONITORING SURVEILLANCES. LESSONS LEARNED WILL BE INCLUDED IN THE TRAINING FOR APPLICABLE PLANT PERSONNEL.

[100] MONTICELLO DOCKET 50-263 LER 90-007 REV 01
UPDATE ON ELECTRIC FAULT IN REACTOR RECIRCULATION PUMP DISCHARGE VALVE MOTOR WHICH COULD DISABLE LOW PRESSURE COOLANT INJECTION SYSTEM.
EVENT DATE: 070590 REPORT DATE: 110190 NSSS: GE TYPE: BWR
VENDOR: LIMITORQUE CORP.
RELIANCE ELECTRIC COMPANY

(NSIC 219830) ON JULY 5, 1990 WITH THE REACTOR IN COLD SHUTDOWN, THE #12 REACTOR RECIRCULATION PUMP DISCHARGE VALVE MOTOR FAILED WHILE CLOSING THE RECIRCULATION PUMP DISCHARGE VALVE. POSITION INDICATION LIGHTS WENT DARK APPROXIMATELY 5 SECONDS AFTER A CONTROL ROOM OPERATOR GAVE THE VALVE A CLOSE SIGNAL. THE VALVE WAS FOUND TO BE ABOUT 95 PERCENT CLOSED. THE REACTOR WAS IN COLD SHUTDOWN. THIS EVENT IS REPORTABLE BECAUSE THE #12 RECIRCULATION PUMP DISCHARGE VALVE IS REQUIRED TO CLOSE UPON LOW PRESSURE COOLANT INJECTION SYSTEM INITIATION WHEN RECIRCULATION SYSTEM "B" LOOP IS SELECTED FOR INJECTION. THE ROOT CAUSE OF THIS EVENT WAS FAILURE TO COMPLETELY EVALUATE THE POSSIBLE LATENT EFFECTS OVERLOADS OF THE OPERATOR IN 1984 AND 1986 MAY HAVE HAD ON THE OPERATOR. CUMULATIVE HEAT DEGRADATION OF THE MOTOR OR THE MOTOR FUSES DUE TO THE OVERLOAD EVENTS LED TO SINGLE PHASE OPERATION OF THE MOTOR IN 1990, AND ULTIMATELY TO FAILURE OF THE MOTOR. CORROSION OF THE MAGNESIUM ROTOR MATERIAL WAS NOT A CONTRIBUTING FACTOR TO THE FAILURE. THE MOTOR WAS REPLACED. THE POWER CIRCUIT COMPONENTS WERE INSPECTED AND/OR TESTED. THE VALVE WAS DECLARED OPERABLE PRIOR TO PLANT STARTUP.

[101] MONTICELLO DOCKET 50-263 LER 90-014
NONCONSERVATIVE CALIBRATION OF CORE FLOW MEASUREMENT SYSTEM DUE TO PROCEDURE INADEQUACY.
EVENT DATE: 090790 REPORT DATE: 100990 NSSS: GE TYPE: BWR

(NSIC 219699) A REVIEW OF A RECENT GENERAL ELECTRIC SERVICE INFORMATION LETTER CONCLUDED THAT THE CORE FLOW MEASUREMENT SYSTEM HAD NOT BEEN PROPERLY CALIBRATED. CALIBRATING THE CORE FLOW MEASUREMENT SYSTEM USING THE METHOD RECOMMENDED IN THE SERVICE INFORMATION LETTER RESULTS IN DECREASING THE INDICATED CORE FLOW BY APPROXIMATELY 3.5%. HAVING AN INDICATED FLOW 3.5% HIGHER THAN IT SHOULD BE RESULTS IN THE RATED DRIVE FLOW ALSO BEING 3.5% TOO HIGH. THIS IN TURN RESULTS IN NONCONSERVATIVE AVERAGE POWER RANGE MONITOR FLOW BIASED ROD BLOCK AND SCRAM SETPOINTS OF AS MUCH AS 2% AT RATED CORE FLOW. ALSO, INDICATED CORE FLOW 3.5% HIGHER THAN ACTUAL CORE FLOW RESULTS IN A 1% NONCONSERVATISM IN THE CALCULATION OF THE REACTOR CORE THERMAL LIMITS. THE ROOT CAUSE IS AN INADEQUATE CORE FLOW MEASUREMENT SYSTEM INSTRUCTION MANUAL. THE IMMEDIATE CORRECTIVE ACTIONS WERE TO INCREASE THE RATED DRIVE FLOW SIGNAL USED IN THE FLOW BIAS SETPOINT CIRCUITRY BY 3.5% AND TO REDUCE THE ACTION LIMIT ON THE THERMAL LIMITS BY 1%. THE LONG TERM CORRECTIVE ACTIONS WILL BE TO CALIBRATE THE CORE FLOW MEASUREMENT SYSTEM ONCE PER CYCLE, REVIEW FOR ACCURACY OTHER MEASURED INPUTS TO THE THERMAL LIMITS CALCULATIONS, AND AS PART OF THE ON-GOING DESIGN BASIS DOCUMENTATION PROGRAM. FURTHER REVIEW WILL BE DONE ON SIMILAR SETPOINTS.

[102] MONTICELLO DOCKET 50-263 LER 90-015 REV 01
UPDATE ON FAILURE TO PERFORM REQUIRED STROKE TIMING OF PRIMARY CONTAINMENT ISOLATION VALVE FOLLOWING VALVE MAINTENANCE.
EVENT DATE: 091690 REPORT DATE: 110190 NSSS: GE TYPE: BWR

(NSIC 219831) MINOR MAINTENANCE AND SUBSEQUENT POST MAINTENANCE TESTING WERE PERFORMED ON A PRIMARY CONTAINMENT AUTOMATIC ISOLATION VALVE. IT WAS RETURNED TO

SERVICE FOLLOWING THIS MAINTENANCE. LATER, IT WAS DISCOVERED THAT REQUIRED STROKE TIMING OF THE VALVE WAS NOT PERFORMED AS A PART OF THE POST MAINTENANCE TESTING. THIS WAS A VIOLATION OF TECHNICAL SPECIFICATIONS WHICH REQUIRE PRIMARY CONTAINMENT AUTOMATIC ISOLATION VALVES TO BE STROKE TIMED PRIOR TO PLACING THEM BACK IN SERVICE. THE FAILURE TO STROKE TIME VALVE MO-2373 FOLLOWING MAINTENANCE AS REQUIRED BY TECHNICAL SPECIFICATIONS WAS ATTRIBUTED TO COGNITIVE PERSONNEL ERROR. THE CORRECTIVE ACTION TAKEN WAS TO STROKE TIME THE VALVE. ADDITIONAL ACTIONS TO PREVENT RECURRENCE ARE TO IMPLEMENT IMPROVED ADMINISTRATIVE CONTROLS AND TO REMIND PLANT PERSONNEL OF THE STROKE TIMING REQUIREMENT.

[103] NINE MILE POINT 2 DOCKET 50-410 LER 90-014
SERVICE WATER PUMP INSERVICE TEST PROCEDURE NOT IN COMPLIANCE WITH ASME SECTION XI REQUIREMENTS.
EVENT DATE: 092490 REPORT DATE: 102490 NSSS: GE TYPE: BWR

(NSIC 219927) ON SEPTEMBER 24, 1990, IT WAS DETERMINED THAT THE NINE MILE POINT UNIT 2 (NMP2) TEN YEAR INSERVICE INSPECTION PROGRAM PLAN WAS NOT BEING IMPLEMENTED IN ACCORDANCE WITH ASME CODE, SECTION XI GUIDELINES AS REQUIRED BY THE PLANT TECHNICAL SPECIFICATION. THE METHODS DEVELOPED TO DETERMINE SERVICE WATER PUMP OPERABILITY ACCEPTANCE, AND SPECIFICALLY, THE USE OF PUMP PERFORMANCE CURVES, WERE NOT CONSISTENT WITH ASME SECTION XI REQUIREMENTS. NMP2 WAS IN A REFUELING OUTAGE AT THE TIME THIS CONDITION WAS DETERMINED REPORTABLE. THE CAUSE FOR THIS CONDITION WAS A PROCEDURAL DEFICIENCY. THE ROOT CAUSE WAS DETERMINED TO BE A MISINTERPRETATION OF AN ASME SECTION XI REQUIREMENT. INITIAL CORRECTIVE ACTION INCLUDED REVISING THE OPERATING SURVEILLANCE PROCEDURE TO ELIMINATE USE OF PERFORMANCE CURVES. AN ADDITIONAL ACTION INCLUDES PERFORMING A 100 PERCENT REVIEW OF ASSOCIATED INSERVICE TESTING PROGRAM IMPLEMENTING PROCEDURES.

[104] NORTH ANNA 2 DOCKET 50-339 LER 89-006 REV 01
UPDATE ON PRESSURIZER CODE SAFETY VALVES OUT OF TOLERANCE.
EVENT DATE: 031589 REPORT DATE: 100590 NSSS: WE TYPE: PWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV

(NSIC 219687) AT 1600 HOURS ON 3/15/89, WITH UNIT 2 IN MODE 6 (REFUELING) THE "A" PRESSURIZER CODE SAFETY VALVE, 2-RC-SV-2551A, "AS FOUND" SET PRESSURE WAS FOUND TO BE OUT OF TOLERANCE. THE "AS FOUND" SET PRESSURE WAS NOT WITHIN THE SET PRESSURE OF 2485 PSIG +/- 1% ALLOWED BY TECH SPEC 3.4.3. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B). ON 3/8/89, ALL THREE PRESSURIZER CODE SAFETY VALVES WERE SENT TO WYLE LABS FOR THE PERFORMANCE OF THE "PRESSURIZER CODE SAFETY VALVE SETPOINT VERIFICATION" PERIODIC TEST (2-PT-50). EACH VALVE WAS FUNCTIONALLY TESTED FOR THE "AS FOUND" SET PRESSURE AND LEAK TIGHTNESS. THE "AS FOUND" SET PRESSURE FOR THE "A" SAFETY VALVE WAS FOUND TO BE BELOW THE MINIMUM SET PRESSURE ALLOWED BY TECH SPEC 3.4.3. ALSO, ALL SAFETY VALVES LEAKED FOLLOWING "AS FOUND" TESTING. AS A CORRECTIVE ACTION, THE SAFETY VALVES WERE REPAIRED AND READJUSTED AT WYLE LABS TO WITHIN THE CORRECT SETPOINT TOLERANCE ALLOWED BY TECH SPEC 3.4.3. FOLLOWING REPAIR AND READJUSTMENT NONE OF THE SAFETY VALVES EXHIBITED LEAKAGE. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE THE "A" PRESSURIZER CODE SAFETY VALVE WOULD HAVE PERFORMED ITS SAFETY FUNCTION IN THE EVENT OF AN OVERPRESSURE CONDITION.

[105] NORTH ANNA 2 DOCKET 50-339 LER 90-004
GREATER THAN ONE PERCENT OF INSERVICE TUBES DEFECTIVE ON STEAM GENERATORS "A" AND "C".
EVENT DATE: 100290 REPORT DATE: 102590 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 219875) ON OCTOBER 2, 1990 WITH UNIT 2 IN MODE 6 (REFUELING), IT WAS DETERMINED BY INSERVICE INSPECTION THAT "A" STEAM GENERATOR (S/G) HAD 33 TUBES DEFECTIVE OUT OF THE 3277 TUBES IT HAD IN SERVICE. ON OCTOBER 5, 1990 WITH THE SAME PLANT CONDITIONS, IT WAS ALSO DETERMINED THAT "C" S/G HAD 40 TUBES DEFECTIVE OUT OF THE 3249 TUBES IT HAD IN SERVICE. THUS, GREATER THAN ONE PERCENT OF TUBES INSPECTED ON EACH OF THE S/G'S, "A" AND "C", WERE FOUND TO BE SUFFICIENTLY DEFECTIVE TO REQUIRE PLUGGING. AS THE RESULT, BOTH S/G'S ARE CLASSIFIED AS

CATEGORY C-3. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(V)(C) AS REQUIRED BY TECHNICAL SPECIFICATION (TS) 4.4.5.5.C. A FOUR HOUR REPORT WAS MADE PURSUANT TO 10CFR50.72(B)(2)(I). NO REACTOR CORE SAFETY CONCERNS EXIST SINCE EACH S/G IS BOUNDED BY THE TUBE PLUGGING LIMIT OF EIGHT (8) PERCENT ALLOWED BY THE SAFETY ANALYSIS. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED AT ANY TIME DURING THIS EVENT.

[106] NORTH ANNA 2 DOCKET 50-339 LER 90-005
PRESSURIZER AND MAIN STEAM SAFETY VALVE SETPOINTS OUT OF TOLERANCE DUE TO SETPOINT DRIFT.
EVENT DATE: 100590 REPORT DATE: 101790 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE & GAGE CO.
DRESSER INDUSTRIAL VALVE & INST DIV

(NSIC 219876) AT 0950 HOURS ON OCTOBER 5, 1990, WITH UNIT 2 IN MODE 6 (REFUELING), THE "AS FOUND" SET PRESSURES FOR THE 3 PRESSURIZER SAFETY VALVES AND 10 OF 15 MAIN STEAM SAFETY VALVES (MSSVS) WERE FOUND TO BE OUTSIDE THE SETPOINT TOLERANCES ALLOWED BY TECHNICAL SPECIFICATIONS 3.4.3 AND 3.7.1.1, RESPECTIVELY. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B) FOR CONDITIONS PROHIBITED BY TECHNICAL SPECIFICATIONS 3.4.2, 3.4.3.1 AND 3.7.1.1. THE SAFETY VALVES WERE SENT TO WYLE LABS FOR TESTING TO ENSURE CONFORMANCE TO TECHNICAL SPECIFICATIONS 3.4.2, 3.4.3.1 AND 3.7.1.1, RESPECTIVELY. THE "AS FOUND" SET PRESSURES FOR THE 3 PRESSURIZER SAFETY VALVES AND 10 MSSVS WERE FOUND TO BE OUTSIDE THE TOLERANCE OF TECHNICAL SPECIFICATIONS 3.4.2, 3.4.3.1 AND 3.7.1.1. HOWEVER, THE EXPECTED PEAK PRESSURE WAS FOUND TO BE LESS THAN THE DESIGN BASIS PRESSURE. THE SAFETY VALVES WERE REPAIRED AND READJUSTED, AS NECESSARY TO BE WITHIN THE CORRECT SETPOINT TOLERANCE ALLOWED BY TECHNICAL SPECIFICATIONS. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE THE SAFETY VALVES WOULD HAVE PERFORMED THEIR SAFETY FUNCTION IN THE EVENT OF AN OVERPRESSURE CONDITION. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED AT ANY TIME DURING THIS EVENT.

[107] OCONEE 2 DOCKET 50-270 LER 90-001
MANUAL REACTOR TRIP WHILE SUBCRITICAL AFTER 2A FEEDWATER PUMP TRIP DUE TO EQUIPMENT MALFUNCTION.
EVENT DATE: 091390 REPORT DATE: 101590 NSSS: BW TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219701) ON 9/13/90, AT 0244 HOURS, WITH UNIT 2 SUBCRITICAL AND GROUP 1 CONTROL RODS POSITIONED AT 50% WITHDRAWN, THE REACTOR WAS MANUALLY TRIPPED IN ACCORDANCE WITH THE OPERATIONS MANAGEMENT PROCEDURE WHEN THE ONLY OPERATING MAIN FEEDWATER PUMP TURBINE (FWPT), 2A FWPT, TRIPPED. UNIT 2 WAS IN THE PROCESS OF BEING SHUT DOWN IN PREPARATION FOR A REFUELING OUTAGE. EMERGENCY FEEDWATER (EFDW) WAS MANUALLY INITIATED AFTER THE 2A FWPT TRIPPED BECAUSE ITS AUTOMATIC ACTUATION HAD BEEN PREVIOUSLY BYPASSED BY THE SHUTDOWN PROCEDURE. EFDW PERFORMED AS EXPECTED AND COOLDOWN WAS MAINTAINED WITHIN LIMITS. THE REACTOR COOLANT SYSTEM (RCS) WAS AT APPROXIMATELY 490F AND 2150 PSIG PRIOR TO THE EVENT. EFDW WAS SECURED 47 MINUTES LATER WITH THE CONDENSATE BOOSTER PUMP FEEDING THE STEAM GENERATORS AND THE RCS WAS AT APPROXIMATELY 465F AND 1950 PSIG. THE ROOT CAUSE IS EQUIPMENT MALFUNCTION.

[108] OYSTER CREEK DOCKET 50-219 LER 90-013
TECHNICAL SPECIFICATION VIOLATION DUE TO MISSED FIRE WATCH CAUSED BY PERSONNEL ERROR.
EVENT DATE: 092690 REPORT DATE: 102490 NSSS: GE TYPE: BWR

(NSIC 219811) ON SEPTEMBER 26, 1990 AT APPROXIMATELY 1500 HOURS, A PREVENTIVE MAINTENANCE ACTIVITY ASSOCIATED WITH THE C BATTERY ROOM VENTILATION WAS COMPLETED AND THE HOURLY FIRE WATCH FOR THE C BATTERY ROOM WAS SECURED BY THE ON DUTY SHIFT SUPERVISOR. AT 1700 HOURS, AFTER SHIFT TURNOVER, THE NEW ON DUTY GROUP SHIFT SUPERVISOR REVIEWED THE FIRE WATCH PATROL LIST AND REALIZED THAT THERE WAS STILL A REQUIREMENT FOR THE C BATTERY ROOM FIRE WATCH, DUE TO A MALFUNCTIONING DOOR. THE FIRE WATCH WAS IMMEDIATELY REINSTATED, HOWEVER ONE OF THE HOURLY FIRE

WATCHES WAS MISSED. THE CAUSE OF THIS OCCURRENCE IS ATTRIBUTED TO PERSONNEL ERROR. A CONTRIBUTING FACTOR TO THIS OCCURRENCE IS THE PRIORITY ASSIGNED TO THE MAINTENANCE ACTIVITY ASSOCIATED WITH REPAIRING THE MALFUNCTIONING DOOR. THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL BECAUSE ALL FIRE DETECTORS IN THE C BATTERY ROOM, AS WELL AS ADJACENT AREAS WERE OPERABLE. THE MISSED FIRE WATCH WAS QUICKLY RECOGNIZED AND CORRECTED. FOR CORRECTIVE ACTION, MANAGEMENT WILL ENSURE APPROPRIATE PERSONNEL ARE AWARE OF THE NEED TO GIVE A HIGH PRIORITY TO ACTIVITIES ASSOCIATED WITH THE RESTORATION OF FIRE PROTECTION EQUIPMENT. ALSO, THIS LICENSEE EVENT REPORT WILL BE ASSIGNED AS REQUIRED READING FOR ALL LICENSED PERSONNEL IN THE OPERATIONS DEPARTMENT.

[109] PALISADES DOCKET 50-255 LER 90-016
PRIMARY COOLANT SYSTEM VENT VALVE SIZING ERROR.
EVENT DATE: 091190 REPORT DATE: 101190 NSSS: CE TYPE: PWR

(NSIC 219825) ON SEPTEMBER 11, 1990 THE PLANT WAS OPERATING AT APPROXIMATELY 80% POWER. TECHNICAL SPECIFICATION 3.1.8 REQUIRES THAT WHENEVER THE TEMPERATURE OF ONE OR MORE OF THE PRIMARY COOLANT SYSTEM COLD LEGS IS LESS THAN 430 DEGREES F, BOTH POWER OPERATOR RELIEF VALVES (PORV'S) BE OPERABLE. IF ONE OR BOTH PORV'S ARE INOPERABLE THEN THE TECHNICAL SPECIFICATIONS PROVIDES ACTION STATEMENTS TO EVENTUALLY DEPRESSURIZE AND EITHER VENT THE PCS THROUGH A VENT GREATER THAN OR EQUAL TO 1.3 SQUARE INCHES IN SIZE, OR OPEN BOTH PORV VALVES AND BOTH PORV BLOCK VALVES. THIS REQUIREMENT ENSURES THAT THE 10CFR50 APPENDIX G PRESSURE LIMITS WILL NOT BE EXCEEDED. DURING THE 1989 FALL MAINTENANCE OUTAGE THE PORV'S AND BLOCK VALVES WERE REPLACED, AND VENT VALVES WERE ADDED UNDER FACILITY CHANGE (FC) 791. AN ENGINEERING ANALYSIS PERFORMED AS PART OF THE FACILITY CHANGE SHOWED THE VENT VALVES TO HAVE THE REQUIRED VENT PATH OPENING SIZE OF GREATER THAN 1.3 SQUARE INCHES. THIS WAS INITIALLY CONFIRMED BY THE VALVE VENDOR. THE LATER CONFIRMATORY CALCULATIONS BY THE VALVE VENDOR SHOWED THAT THE VALVES DID NOT PROVIDE A VENT PATH OPENING OF 1.3 SQUARE INCHES. THEREFORE THE OPERATING PROCEDURES IN PLACE ALLOWED A PLANT CONFIGURATION THAT WAS NOT IN CONFORMANCE WITH THE TECHNICAL SPECIFICATIONS LIMIT.

[110] PALISADES DOCKET 50-255 LER 90-017
LOW TEMPERATURE OVER-PRESSURIZATION PROTECTION INOPERABLE AS POWER OPERATED RELIEF VALVE ACTUATION CIRCUITRY INAPPROPRIATELY DISABLED.
EVENT DATE: 092690 REPORT DATE: 102690 NSSS: CE TYPE: PWR

(NSIC 219826) ON SEPTEMBER 26, 1990 AT 0830 HOURS THE PLANT WAS IN COLD SHUTDOWN AND ON SHUTDOWN COOLING. TECHNICAL SPECIFICATION 3.1.8, "OVERPRESSURE PROTECTION SYSTEMS", REQUIRES THAT WHENEVER THE TEMPERATURE OF ONE OR MORE OF THE PRIMARY COOLANT SYSTEM (PCS) COLD LEGS IS LESS THAN 430 DEGREES F, BOTH POWER OPERATED RELIEF VALVES (PORVS) BE OPERABLE. DURING A DAILY CONTROL ROOM TOUR, THE INSTRUMENT AND CONTROL (I&C) SUPERVISOR NOTICED THAT AN ALARM WAS ANNUNCIATED ON THE MAIN CONTROL BOARDS AS "NO PCS PROTECTION" AND QUESTIONED THE OPERABILITY OF THE PORVS WHICH WERE BEING USED FOR OVERPRESSURE PROTECTION. DISCUSSIONS WITH THE PROJECT ENGINEER AND THE I&C SUPERVISOR FAILED TO PROVIDE A CLEAR ANSWER AS TO THE OPERABILITY OF THE SYSTEM. DURING THIS REVIEW, THE PORVS AND ASSOCIATED BLOCK VALVES WERE OPENED TO PROVIDE A POSITIVE VENT PATH. TO VERIFY THE ACTUAL SYSTEM RESPONSE, PORTIONS OF MI-27E TEST PROCEDURE WERE PERFORMED TO TEST THE OPERABILITY OF THE OVERPRESSURE PROTECTION SYSTEM IN THE SHUTDOWN COOLING MODE. IT WAS DETERMINED THAT THE PORVS WOULD NOT OPEN UNDER AN ACTUAL HIGH PRESSURE CONDITION, WITH A FAILURE OF THE TEMPERATURE INPUT. THIS EVENT IS REPORTABLE AS A CONDITION PROHIBITED BY PLANT TECHNICAL SPECIFICATIONS IN THAT THE REQUIREMENT TO VERIFY THE PCS VENT PATH EVERY 12 HOURS WHEN THE VENT IS BEING USED FOR OVERPRESSURE PROTECTION WAS NOT MET.

[111] PALO VERDE 2 DOCKET 50-529 LER 90-009
VOLUNTARY REPORT ON FIRE BARRIER INSPECTION.
EVENT DATE: 091190 REPORT DATE: 102490 NSSS: CE TYPE: PWR

(NSIC 219957) PALO VERDE UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER WHEN THE UNIT 2 EIGHTEEN MONTH INSPECTION OF FIRE AREA BOUNDARIES (WALLS,

FLOORS, AND CEILINGS) AND ASSOCIATED SEALED PENETRATIONS (EXCEPT FIRE DOORS) WAS COMPLETED ON SEPTEMBER 11, 1990. APPROXIMATELY 256 INSTANCES WERE IDENTIFIED OUT OF APPROXIMATELY 2000 ATTRIBUTES EXAMINED WHERE EITHER AN INSPECTION ACCEPTANCE CRITERION WAS NOT MET OR THE ADEQUACY OF THE INSTALLED CONFIGURATION WAS QUESTIONABLE. BECAUSE OF THE EXTENT OF OUR INSPECTION ACTIVITY AND THE TYPES OF DISCREPANCIES IDENTIFIED, APS IS SUBMITTING THIS INFORMATIONAL LER. IF, THROUGH POST-INSPECTION ANALYSIS, A DISCREPANCY IS IDENTIFIED THAT MEETS THE REPORTING REQUIREMENTS OF 10 CFR 50.73, A SUPPLEMENT TO THIS REPORT WILL BE SUBMITTED. FIRE BARRIER IMPAIRMENT COMPENSATORY ACTIONS WERE ESTABLISHED IN ACCORDANCE WITH PLANT PROCEDURES. A SCHEDULE TO DISPOSITION/CORRECT THE IDENTIFIED DISCREPANCIES IS BEING DEVELOPED. ADDITIONALLY, EVALUATION OF THE EFFECTS OF THE DISCREPANCIES ON PLANT SAFETY WILL BE PERFORMED AS PART OF THE DISPOSITION OF EACH DISCREPANCY. SIMILAR FIRE BARRIER/PENETRATION INSPECTIONS ARE SCHEDULED FOR UNITS 1 AND 3.

[112] PEACH BOTTOM 2 DOCKET 50-277 LER 90-028
MISSED SURVEILLANCE ON A DRYWELL-SUPPRESSION CHAMBER VACUUM BREAKER DUE TO AN IMPROPER USE OF A PARTIAL PROCEDURE.
EVENT DATE: 080190 REPORT DATE: 101890 NSSS: GE TYPE: BWR

(NSIC 219842) WHILE REVIEWING A COMPLETED PROCEDURE ON 9/19/90, IT WAS DISCOVERED THAT A DRYWELL-SUPPRESSION CHAMBER VACUUM BREAKER HAD NOT BEEN EXERCISED AS REQUIRED BY TECHNICAL SPECIFICATIONS. IN PREPARATION FOR PERFORMING THE VALVE EXERCISE TESTING ON 7/11/90, THE SHIFT SUPERVISOR DIRECTED THAT A PARTIAL TEST BE PERFORMED. THE "C" VACUUM BREAKER WAS EXCLUDED FROM BEING EXERCISED BECAUSE IT WAS THOUGHT TO BE INOPERABLE. THE CAUSE OF THE EVENT WAS AN IMPROPER USE OF A PARTIAL PROCEDURE. THERE WERE NO SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT. IT WAS PROVEN DURING THE FOLLOWING SCHEDULED SURVEILLANCE PERFORMANCE THAT THE VACUUM BREAKER WAS CAPABLE OF PERFORMING ITS INTENDED SAFETY FUNCTION. AN ENTRY WAS MADE IN THE LCO LOG STATING THAT THE "C" VACUUM BREAKER SHOULD BE EXERCISED MONTHLY IN ADDITION TO BYPASS AREA TESTING WHICH WAS BEING PERFORMED BECAUSE THE VACUUM BREAKER WOULD NOT INDICATE FULLY SEATED. THE INDIVIDUAL INVOLVED HAS BEEN INFORMED OF THIS EVENT. THE PERTINENT INFORMATION CONTAINED IN THIS REPORT WILL BE INCLUDED IN REQUIRED READING FOR APPROPRIATE OPERATIONS PERSONNEL. NO PREVIOUS SIMILAR LERS WERE IDENTIFIED.

[113] PEACH BOTTOM 2 DOCKET 50-277 LER 90-021
PLANT SHUTDOWN REQUIRED DUE TO INOPERABLE REACTOR WATER LEVEL INSTRUMENTATION.
EVENT DATE: 083090 REPORT DATE: 100190 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 219707) ON 8/30/90 AT 1750 HOURS ALL UNIT 2 REACTOR LEVEL INSTRUMENTATION ASSOCIATED WITH THE 2B REFERENCE LEG CONDENSING CHAMBER WAS DECLARED INOPERABLE. PLANT SHUTDOWN WAS INITIATED AT 1812 HOURS TO COMPLY WITH TECHNICAL SPECIFICATION 3.0.C. SHUTDOWN OF UNIT 2 WAS COMPLETED ON 8/30/90 AT 2151 HOURS. THE CAUSE OF THIS EVENT IS BELIEVED TO BE A LOSS OF INVENTORY IN THE 2B CONDENSING CHAMBER REFERENCE LEG COUPLED WITH A LOSS OF MAKEUP CAPABILITY TO THE REFERENCE LEG. THIS RESULTED IN ERRONEOUS REACTOR WATER LEVEL READINGS. PROCEDURES HAVE BEEN ENHANCED WITH TIGHTER ACCEPTANCE CRITERIA. MONITORING OF LEVEL INSTRUMENTATION WILL BE INCREASED. THE DESIGN AND CONFIGURATION OF THE CONDENSING CHAMBERS IS BEING EVALUATED. NO PREVIOUS SIMILAR LERS HAVE BEEN IDENTIFIED.

[114] PEACH BOTTOM 2 DOCKET 50-277 LER 90-023
CONDITION OUTSIDE DESIGN BASIS CONCERNING WALL PENETRATION SEALS NOT FLOODTIGHT DUE TO PROGRAMMATIC WEAKNESS.
EVENT DATE: 091090 REPORT DATE: 101090 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 219837) ON 9/10/90, ENGINEERING PERSONNEL DISCOVERED AN UNQUALIFIED EXTERNAL TEMPORARY PENETRATION SEAL ASSEMBLY IN THE SOUTH WALL OF THE UNIT 2 HIGH PRESSURE SERVICE WATER (HPSW)/EMERGENCY SERVICE WATER (ESW) PUMP MOTOR AREA COMPARTMENT. THIS CONDITION COULD COMPROMISE THE WATERTIGHT INTEGRITY OF THIS AREA IN THE EVENT OF A LOCAL EXTERNAL FLOOD. ON 9/14/90, IT WAS DISCOVERED THAT AN UNQUALIFIED TEMPORARY SEAL IN A STANDBY GAS TREATMENT (SBGT) SYSTEM CONDUIT

COULD RESULT IN SBGT INOPERABILITY IF AN INTERNAL FLOOD OCCURRED IN THE HIGH PRESSURE COOLANT INJECTION SYSTEM COMPARTMENT. AS A RESULT OF THESE TWO DEFICIENCIES ADDITIONAL REVIEW ON 9/24/90 REVEALED AN UNQUALIFIED SEAL IN THE EAST WALL OF THE UNIT 3 HPSW/ESW PUMP MOTOR AREA COMPARTMENT. THESE THREE DEFICIENCIES WERE OUTSIDE THE DESIGN BASIS OF THE PLANT. THE CAUSE OF THE EVENT IS DUE TO PROGRAMMATIC WEAKNESSES INVOLVING INSTALLATION OF PENETRATION SEALS DURING MODIFICATIONS. THERE WERE NO ACTUAL SAFETY CONSEQUENCES. THE THREE SEAL ASSEMBLIES WERE REPLACED WITH QUALIFIED SEALS. BOTH TRAINS OF SBGT WERE OPERABLE BY 9/15/90. AN INSTALLATION PROCEDURE WAS PROVIDED WITH INTERIM GUIDANCE TO REQUIRE AN INDEPENDENT REVIEW AND 10 CFR 50.59 REVIEW BE PERFORMED PRIOR TO FLOOD SEAL WORK. A REVIEW OF EXTERNAL FLOOD BARRIERS WILL BE PERFORMED.

[115] PEACH BOTTOM 2 DOCKET 50-277 LER 90-024
FAILURE TO PERFORM TECHNICAL SPECIFICATION SURVEILLANCES DUE TO A DEFICIENT PROCEDURE AND LESS THAN ADEQUATE TECHNICAL REVIEW.
EVENT DATE: 091090 REPORT DATE: 101090 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 219838) ON 9/10/90 AT 1300 HOURS, IT WAS DISCOVERED THAT DAILY INSTRUMENT CHANNEL CHECKS OF THE DRYWELL HIGH PRESSURE AND CORE SPRAY SYSTEM SPARGER DIFFERENTIAL PRESSURE INSTRUMENTATION WERE NOT INCLUDED IN SURVEILLANCE TEST PROCEDURE ST 9.1.A. THIS OMISSION RESULTED IN THE FAILURE TO PERFORM THE DAILY CHANNEL CHECKS FOR THESE INSTRUMENTS DURING SHUTDOWN AND REFUELING CONDITIONS AS REQUIRED BY TECHNICAL SPECIFICATION 3.2.B. THE CHANNEL CHECKS WERE OMITTED IN 1988 DURING AN EXTENSIVE REVISION TO ST 9.1.A. THE CAUSE OF THE FAILURE TO PERFORM THE DAILY INSTRUMENT CHANNEL CHECKS WAS THE DEFICIENCY OF ST-9.1.A WHICH WAS THE RESULT OF AN INAPPROPRIATE INTERPRETATION OF TECHNICAL SPECIFICATIONS. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. ST 9.1.A WILL BE REVISED TO INCLUDE DAILY INSTRUMENT CHANNEL CHECKS OF THE DRYWELL HIGH PRESSURE AND CORE SPRAY SYSTEM SPARGER DIFFERENTIAL PRESSURE INSTRUMENTATION. ST 9.1.A WILL BE TECHNICALLY REVIEWED FURTHER TO ENSURE THAT OTHER TECH SPEC SURVEILLANCES ARE SPECIFIED TO BE PERFORMED AS REQUIRED. FOUR PREVIOUS SIMILAR EVENTS WERE IDENTIFIED.

[116] PEACH BOTTOM 2 DOCKET 50-277 LER 90-025
HIGH PRESSURE/EMERGENCY SERVICE WATER VENTILATION OUTSIDE DESIGN BASIS DUE TO DEFICIENT DESIGN.
EVENT DATE: 091390 REPORT DATE: 101090 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 219839) ON 9/13/90 DURING A REVIEW RELATED TO AN INTERNAL PECO SAFETY SYSTEM FUNCTIONAL INSPECTION ON THE HIGH PRESSURE SERVICE WATER (HPSW) SYSTEM, ENGINEERING PERSONNEL DISCOVERED A CONDITION THAT COULD PREVENT THE UNIT 2 AND 3 HPSW/EMERGENCY SERVICE WATER (ESW) VENTILATION SYSTEMS FROM AUTOMATICALLY OPERATING DURING DESIGN BASIS EVENTS. IT WAS DISCOVERED THAT THE TEMPERATURE CONTROL SYSTEM COULD FAIL TO OPERATE DUE TO UNAVAILABILITY OF THE INSTRUMENT AIR SUPPLY TO THE CONTROL SYSTEM DURING DESIGN BASIS EVENTS INVOLVING LOSS OF INSTRUMENT AIR. THE FAILURE OF THE VENTILATION SYSTEM TO AUTOMATICALLY START COULD POTENTIALLY RESULT IN HPSW AND ESW PUMP MOTORS TO OVERHEAT AND FAIL. THE CAUSE OF THE EVENT WAS A DEFICIENT DESIGN OF THE HPSW/ESW VENTILATION CONTROL SYSTEM. A CONTRIBUTING CAUSE WAS A DESIGN REVIEW OVERSIGHT ASSOCIATED WITH NRC GENERIC LETTER 88-14. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. LATER ON 9/13/90, ONE SUPPLY AND ONE EXHAUST FAN PER UNIT WERE PLACED IN THE RUN POSITION THEREBY ENSURING OPERABILITY DURING DESIGN BASIS CONDITIONS. THE VENTILATION CONTROLS WILL BE MODIFIED. A REVIEW PERFORMED ON OTHER POSSIBLE SIMILAR DEFICIENCIES REVEALED NO SIMILAR PROBLEMS. THERE WERE NO PREVIOUS SIMILAR EVENTS IDENTIFIED.

[117] PEACH BOTTOM 2 DOCKET 50-277 LER 90-026
HIGH PRESSURE COOLANT INJECTION SYSTEM DECLARED INOPERABLE DUE TO LOW EMERGENCY SERVICE WATER FLOW THROUGH ROOM COOLERS.
EVENT DATE: 091390 REPORT DATE: 101590 NSSS: GE TYPE: BWR

(NSIC 219840) ON 9/13/90 AT 0235 HOURS DURING THE PERFORMANCE OF SURVEILLANCE TEST (ST) 21.5-2 "UNIT 2 EMERGENCY SERVICE WATER (ESW) ROOM COOLER FLOW TEST", THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM WAS DECLARED INOPERABLE DUE TO LOW FLOW THROUGH THE HPCI ROOM COOLER. THE CAUSE OF THE EVENT IS DUE TO THE BUILDUP OF LOOSE CORROSION PRODUCTS AND SILT IN THE COOLER DISCHARGE LINE WHICH RESTRICTED THE FLOW. THE HPCI SYSTEM WAS RETURNED TO AN OPERABLE STATUS ON 9/13/90 AT 1400 HOURS. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. AN EVALUATION OF THE POSSIBILITY OF REDUCING THE SW PRESSURE TO THE ESW RING HEADER, IN ORDER TO MINIMIZE INFLUX OF CORROSION PRODUCTS AND SILT, IS BEING PERFORMED. ONE PREVIOUS SIMILAR LER HAS BEEN IDENTIFIED.

[118] PEACH BOTTOM 2 DOCKET 50-277 LER 90-027
REACTOR WATER CLEANUP ISOLATION AND STANDBY GAS TREATMENT SYSTEM ACTUATION DUE TO LIGHTNING STRIKE.
EVENT DATE: 091690 REPORT DATE: 101690 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 219841) ON 9/16/90, AT 1805 HOURS, ACTUATIONS OCCURRED ON BOTH THE UNIT 2 AND UNIT 3 PRIMARY CONTAINMENT ISOLATION SYSTEMS (PCIS) DUE TO A LIGHTNING STRIKE WHICH CAUSED THE OPENING OF THE UNIT 3 STARTUP FEED BREAKER. UNIT 2 AND UNIT 3 EMERGENCY BUSSES ASSOCIATED WITH THE UNIT 3 STARTUP FEED TRANSFERRED TO THE ALTERNATE SUPPLY FOLLOWING THE LOSS OF THE FEED AS DESIGNED. THE FAST TRANSFER RESULTED IN ISOLATION OF THE REACTOR WATER CLEANUP (RWCU) SUCTION OUTBOARD ISOLATION VALVES ON BOTH UNITS. ON UNIT 2, THE "B" REACTOR PROTECTION SYSTEM MOTOR-GENERATOR SET ALSO TRIPPED RESULTING IN 1/2 OF A GROUP III ISOLATION, WHICH CAUSED THE "B" TRAIN OF THE STANDBY GAS TREATMENT ISOLATION SYSTEM TO START ALONG WITH THE LINEUP OF THE APPROPRIATE OUTBOARD ISOLATION VALVES. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. RWCU WAS OUT OF SERVICE FOR A TOTAL OF APPROXIMATELY 15 MINUTES, WHICH DID NOT PRESENT A REACTOR WATER CHEMISTRY CONCERN. FOLLOWING THE EVENT THE UNIT 3 STARTUP FEED BREAKER WAS CLOSED. THE EMERGENCY BUSSES WERE RETURNED TO THEIR NORMAL FEEDS. THE ISOLATIONS WERE RESET, THE RWCU SYSTEMS WERE PLACED BACK INTO SERVICE, AND THE VENTILATION SYSTEMS AND THE SBGT SYSTEM WERE NORMALIZED. TWO PREVIOUS SIMILAR LERS HAVE BEEN IDENTIFIED.

[119] PEACH BOTTOM 3 DOCKET 50-278 LER 90-011
HIGH PRESSURE COOLANT INJECTION SYSTEM INOPERABLE DUE TO DIRTY RELAY CONTACT IN AUXILIARY OIL PUMP MOTOR STARTER.
EVENT DATE: 091090 REPORT DATE: 101090 NSSS: GE TYPE: BWR
VENDOR: CUTLER-HAMMER

(NSIC 21984?) ON 9/10/90, AT 1900 HOURS WITH UNIT 3 OPERATING AT APPROXIMATELY 80% LICENSED REACTOR POWER, THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM AUXILIARY OIL PUMP (AOP) FAILED TO START AS REQUIRED DURING THE PERFORMANCE OF ST 6.5.1, "AUXILIARY OIL PUMP SURVEILLANCE". FAILURE OF THE AOP TO START PREVENTS THE HPCI TURBINE STOP AND CONTROL VALVES FROM OPENING, THUS PREVENTING HPCI FROM OPERATING AS DESIGNED. THE CAUSE OF THE EVENT WAS A BOUND RELAY IN THE AOP LOCAL MOTOR STARTER DUE TO DIRT IN THE RELAY MECHANISM. A CONTRIBUTING CAUSE WAS THE LACK OF PREVENTATIVE MAINTENANCE (PM) ON THE LOCAL MOTOR STARTER. THE RELAYS IN THE LOCAL MOTOR STARTER PANEL WERE CLEANED. HPCI WAS DECLARED OPERABLE AT 1750 HOURS ON 9/11/90 FOLLOWING THE SATISFACTORY COMPLETION OF ST 6.5.1. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. NO PREVIOUS SIMILAR LER'S WERE IDENTIFIED.

[120] PERRY 1 DOCKET 50-440 LER 90-025
LOCAL LEAK RATE TESTS RESULT IN EXCEEDING ALLOWABLE PRIMARY CONTAINMENT LEAKAGE RATES FOR MAIN STEAM LINES A, B, C, AND D.
EVENT DATE: 091690 REPORT DATE: 101690 NSSS: GE TYPE: BWR

(NSIC 219935) DURING THE PERIOD SEPTEMBER 16-17, 1990, FOLLOWING SHUTDOWN AND COOLDOWN OF THE PERRY NUCLEAR POWER PLANT, UNIT 1 FOR THE SECOND REFUELING OUTAGE, LOCAL LEAK RATE TESTING (LLRT) OF THE MAIN STEAM ISOLATION VALVES (MSIV) WAS CONDUCTED. ALL FOUR OF THE MAIN STEAM LINE PENETRATIONS EXHIBITED LEAKAGE IN

EXCESS OF THE TECHNICAL SPECIFICATION 3.6.1.2(C) LIMIT OF 25 SCFH WHEN TESTED AT PA (11.31 PSIG). SEVERAL OF THE MSIV'S HAD NOT BEEN FAST CLOSED FOLLOWING THE REACTOR SHUTDOWN. THE MSL'S WERE TESTED AGAIN FOLLOWING THE OPENING AND FAST CLOSING OF EACH MSIV AND WERE STILL FOUND TO BE LEAKING IN EXCESS OF TECHNICAL SPECIFICATION REQUIREMENTS. THE CAUSE OF THESE FAILURES IS UNDER INVESTIGATION. AS A RESULT OF THESE RECENT FAILURES, THE MSIV'S ARE BEING REWORKED AND MODIFICATIONS TO ENHANCE PROPER SEATING OF THE MSIV'S ARE BEING PURSUED. THESE MODIFICATIONS INCLUDE POPPET ANTI-ROTATION DEVICES, NOSE CONE IMPROVEMENTS AND STEM/COVER MODIFICATIONS TO MINIMIZE VIBRATION OF THE POPPET WHEN THE MSIV IS OPEN. ALL OF THE MSL'S WILL BE TESTED AND VERIFIED TO BE OPERABLE PRIOR TO STARTUP AND A SUPPLEMENTAL REPORT WILL BE ISSUED DETAILING THE WORK NECESSARY TO RETURN THEM TO SERVICE. THIS SUPPLEMENTAL REPORT WILL IDENTIFY THE MODIFICATIONS TO BE IMPLEMENTED DURING THE NEXT REFUELING OUTAGE.

[121] PERRY 1 DOCKET 50-440 LER 90-026
FAILED LOCAL LEAK RATE TEST RESULTS IN EXCEEDING ALLOWABLE SECONDARY CONTAINMENT BYPASS LEAKAGE.
EVENT DATE: 091990 REPORT DATE: 101990 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 219936) ON SEPTEMBER 19, 1990 AT APPROXIMATELY 1115 THE TOTAL COMBINED SECONDARY CONTAINMENT BYPASS LEAKAGE RATE DEFINED BY TECHNICAL SPECIFICATION 3.6.1.2.D WAS DETERMINED TO HAVE BEEN EXCEEDED. THE MAJOR CONTRIBUTORS TO THE TOTAL LEAKAGE WERE THE POST ACCIDENT SAMPLING SYSTEM (PASS) INSTRUMENT SAMPLE LINE CONTAINMENT ISOLATION VALVES, 1P87-F049 AND F055. ONE OF THESE VALVES WAS SUBSEQUENTLY REBUILT AND THE OTHER REPLACED. THEY WERE SATISFACTORILY LEAK TESTED ON OCTOBER 7, 1990. THE CAUSE OF THIS EVENT HAS NOT BEEN DETERMINED. A ROOT CAUSE ANALYSIS IS BEING PERFORMED ON THE REPLACED VALVE. A SUPPLEMENTAL REPORT WILL BE ISSUED TO DISCUSS THE RESULTS OF THIS ANALYSIS AND TO IDENTIFY CORRECTIVE ACTIONS WHICH WILL BE TAKEN. THIS SUPPLEMENTAL REPORT WILL ALSO IDENTIFY MAJOR CONTRIBUTORS TO SECONDARY CONTAINMENT BYPASS LEAKAGE. PENDING COMPLETION OF THE ROOT CAUSE ANALYSIS, CORRECTIVE ACTION WILL BE TAKEN TO LIMIT THE CYCLING OF THESE VALVES FOR POSITION INDICATION TESTING AND PASS TRAINING TO ONCE PER QUARTER. FLOW WILL ONLY BE ALLOWED THROUGH THE VALVES SEMI-ANNUALLY FOR PASS SAMPLING AND WHEN REQUIRED AS A BACKUP TO THE NORMAL REACTOR WATER SAMPLING PANEL.

[122] PERRY 1 DOCKET 50-440 LER 90-027
PERSONNEL ERROR RESULTS IN AIR ROLL OF DIVISION 1 DIESEL GENERATOR AND LICENSE VIOLATION.
EVENT DATE: 092590 REPORT DATE: 101990 NSSS: GE TYPE: BWR

(NSIC 219849) ON 9/25/90, AT APPROX. 0700, AN AIR ROLL TEST WAS PERFORMED ON THE DIVISION 1 DIESEL GENERATOR WHILE THE DIVISION 2 DIESEL GENERATOR WAS INOPERABLE. THIS TESTING RESULTED IN THE INOPERABILITY OF THE DIVISION 1 DIESEL GENERATOR AND A FACILITY OPERATING LICENSE VIOLATION. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR, FAILURE TO FOLLOW PROCEDURE. THE OPERATORS WHO PERFORMED THE TESTING DID NOT TAKE HEED OF THE NOTE PRECEDING THE FIRST STEP OF A SECTION WHICH PROHIBITED THEM FROM PERFORMING THAT SECTION OF THE PROCEDURE. ALTHOUGH THE OPERATORS WERE AWARE OF THE DIVISION 2 DIESEL GENERATOR BEING INOPERABLE, THE CONTROL ROOM OPERATOR WHO WAS DIRECTING OPERATOR ACTIVITIES AT THE DIVISION 1 DIESEL GENERATOR DID NOT DIRECTLY CONSULT THE PROCEDURE AT THE TIME THE OPERATORS AT THE DIVISION 1 DIESEL GENERATOR REQUESTED THE PERFORMANCE OF SWITCH MANIPULATIONS TO PROCEED WITH THE AIR ROLL TEST. THE CONTROL ROOM OPERATOR PERFORMED THE SWITCH MANIPULATIONS ALSO WITHOUT READING THE NOTE AND AS A RESULT, THE FACILITY OPERATING LICENSE VIOLATION OCCURRED. TO PREVENT RECURRENCE, THE OPERATIONS PERSONNEL WHO PERFORMED THE AIR ROLL ON THE DIVISION 1 DIESEL GENERATOR WERE COUNSELED CONCERNING THE EVENT AND CONCERNING THE IMPORTANCE OF PROCEDURAL COMPLIANCE. THESE TOPICS WERE ALSO DISCUSSED BETWEEN OPERATIONS SHIFT SUPERVISORS AND THEIR OPERATING CREWS.

[123] PILGRIM 1 DOCKET 50-293 LER 90-015
UNPLANNED PARTIAL ISOLATIONS OF THE HYDROGEN AND OXYGEN ANALYZER SYSTEM AND THE REACTOR COOLANT PRESSURE BOUNDARY LEAK DETECTION SYSTEM DURING JUMPER INSTALLATION.
EVENT DATE: 091390 REPORT DATE: 101390 NSSS: GE TYPE: BWR

(NSIC 219888) ON 9/13/90 AT 1640 HOURS, UNPLANNED ACTUATIONS OF PORTIONS OF THE PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM (PCIS) CAUSED PARTIAL ISOLATIONS OF THE HYDROGEN AND OXYGEN ANALYZER SYSTEM AND THE REACTOR COOLANT PRESSURE BOUNDARY LEAK DETECTION SYSTEM. THE ISOLATIONS OCCURRED WHILE INSTALLING A GROUNDING JUMPER REQUIRED FOR THE PLANNED REPLACEMENT OF RELAY 16A-K29 IN PANEL C-941. WHILE ATTEMPTING TO ATTACH THE GROUNDING JUMPER TO TERMINAL NUMBER 14 OF RELAY 16A-K17X IN PANEL C-941, TERMINALS NUMBER 10 AND NUMBER 12 WERE ACCIDENTALLY CONTACTED WHICH CAUSED FUSES PPFU1 AND FU-RADA TO BLOW. THE ACCIDENTAL CONTACT WITH TERMINALS NUMBER 10 AND NUMBER 12 DURING JUMPER INSTALLATION WAS A RESULT OF AN INADEQUATE WORK PLAN. THE LOCATION OF THE RELAY AND ARRANGEMENT OF PANEL C-941 MADE THE JUMPER INSTALLATION DIFFICULT AND CONTRIBUTED TO THE EVENT. THE EVENT OCCURRED WHILE SHUTDOWN WITH THE REACTOR MODE SELECTOR SWITCH IN THE REFUEL POSITION FOR INSTRUMENT CHECKS. REACTOR POWER LEVEL WAS 0%, THE REACTOR VESSEL (RV) PRESSURE WAS 0 PSIG AND THE RV WATER TEMPERATURE WAS APPROXIMATELY 102F. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(IV). THE BLOWN FUSES WERE REPLACED AND THE VALVE LINEUPS WERE RETURNED TO NORMAL. A TASK FORCE HAS BEEN FORMED TO DETERMINE THE VARIOUS PROBLEMS THAT COULD OCCUR WHEN INSTALLING JUMPERS.

[124] PILGRIM 1 DOCKET 50-293 LER 90-016
AUTOMATIC CLOSING OF THE GROUP 1 ISOLATION VALVES WHILE SHUTDOWN DUE TO HIGH REACTOR WATER LEVEL.
EVENT DATE: 091790 REPORT DATE: 101690 NSSS: GE TYPE: BWR

(NSIC 219852) ON SEPTEMBER 17, 1990 AT 1923 HOURS, AN AUTOMATIC ACTUATION OF THE MAIN STEAM SYSTEM/GROUP 1 (ONE) PORTION OF THE PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM (PCIS) OCCURRED WHILE SHUTDOWN DUE TO A HIGH REACTOR VESSEL (RV) WATER LEVEL. THE ACTUATION RESULTED IN THE AUTOMATIC CLOSING OF THE MAIN STEAM ISOLATION VALVES THAT WERE IN THE OPEN POSITION. THE CAUSE OF THE ACTUATION WAS HIGH RV WATER LEVEL. THE HIGH RV WATER LEVEL OCCURRED DUE TO REACTOR WATER SWELL WHEN THE SHUTDOWN COOLING (SDC) MODE OF THE RESIDUAL HEAT REMOVAL SYSTEM (RHRS) WAS BEING SECURED WITH TWO PUMPS IN SERVICE. THE RV LEVEL WAS HIGHER THAN NORMAL PRIOR TO THIS EVOLUTION BECAUSE, IN PREPARATION FOR REACTOR STARTUP THE RV WAS BEING HEATED I.E. VESSEL HEAD TEMPERATURE WAS BEING INCREASED . THE CAUSE OF THE HIGH WATER LEVEL WAS THAT THE RHRS OPERATING PROCEDURE DID NOT ADDRESS TWO PUMP RHRS OPERATION. CORRECTIVE ACTION INCLUDES REVISING THE RHRS OPERATING PROCEDURE TO ADDRESS SECURING SDC WITH TWO PUMPS IN OPERATION AND INCLUDE AN ACCEPTABLE RV WATER LEVEL BAND IN ORDER TO COMPENSATE FOR A RV WATER LEVEL INCREASE WHEN THE RHRS IS SECURED FROM SDC SERVICE THE EVENT OCCURRED WHILE IN COLD SHUTDOWN WITH THE REACTOR MODE SELECTOR SWITCH IN THE REFUEL POSITION FOR INSTRUMENTATION CHECKS. THE REACTOR POWER LEVEL WAS ZERO PERCENT WITH THE CONTROL RODS IN THE INSERTED POSITION.

[125] PRAIRIE ISLAND 1 DOCKET 50-282 LER 90-014
AUTO-START OF ONE TRAIN OF CONTROL ROOM SPECIAL VENTILATION DUE TO INADEQUATE WORK INSTRUCTIONS.
EVENT DATE: 091290 REPORT DATE: 101890 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)

(NSIC 219846) ON SEPTEMBER 18, 1990 UNIT 1 WAS AT 100% POWER AND UNIT 2 WAS AT COLD SHUTDOWN FOR REFUELING. MAINTENANCE WAS PLANNED FOR CIRCUIT BREAKER 226. THE SHIFT SUPERVISOR HAD APPROVED COMMENCEMENT OF THE WORK AFTER VERIFYING THE TRAIN A CHLORINE MONITORS OPERABLE AND AFTER REVIEWING THE ELECTRICAL LOAD LIST. HE WARNED THE LEAD OPERATOR THAT WHEN CIRCUIT BREAKER 226 WAS TURNED OFF, ONE OF THE TWO TRAIN B CHLORINE MONITORS WOULD BE DE-ENERGIZED, AND HE COULD EXPECT A CHLORINE MONITOR ALARM BUT NO OTHER ACTUATION. WHEN CIRCUIT BREAKER 226 WAS TURNED OFF, AT 1500, THE CHLORINE MONITOR ALARM WAS RECEIVED, NO. 122 CONTROL ROOM CLEANUP FAN STARTED AUTOMATICALLY AND OUTSIDE AIR TO THE CONTROL ROOM WAS

ISOLATED. INVESTIGATION SHOWED THAT THE WORK PACKAGE CONTAINED INFORMATION ALERTING THE OPERATOR TO THE FACT THAT ONE TRAIN B CHLORINE DETECTOR WOULD BE DE-ENERGIZED WHEN CIRCUIT BREAKER 226 WAS TURNED OFF. FURTHER INVESTIGATION REVEALED THAT, IN FACT, BOTH TRAIN B CHLORINE DETECTORS ARE POWERED BY CIRCUIT BREAKER 226 AND WERE BOTH DE-ENERGIZED WHEN CIRCUIT BREAKER 226 WAS TURNED OFF. THIS INFORMATION WAS MISSING FROM THE WORK PACKAGE BECAUSE THE ELECTRICAL LOAD LIST, WHICH WAS USED TO PREPARE THE WORK PACKAGE, WAS IN ERROR. THIS WAS A NON-ESF ACTUATION OF ESF EQUIPMENT.

[126] PRAIRIE ISLAND 2 DOCKET 50-306 LER 90-008
AUTO-START OF NO. 22 COMPONENT COOLING WATER PUMP WHILE SWITCHING RESIDUAL HEAT REMOVAL PUMPS.
EVENT DATE: 092390 REPORT DATE: 102390 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 1 (PWR)

(NSIC 219856) ON SEPTEMBER 23, 1990 UNIT 2 WAS IN COLD SHUTDOWN FOR REFUELING. DECAY HEAT WAS BEING REMOVED BY USE OF ONE TRAIN OF THE RESIDUAL HEAT REMOVAL SYSTEM. CONTROL ROOM OPERATORS WERE IN THE PROCESS OF SWITCHING TO THE OTHER RESIDUAL HEAT REMOVAL TRAIN FOR DECAY HEAT REMOVAL. BOTH COMPONENT COOLING WATER PUMPS WERE RUNNING TO SUPPORT THE SWITCHOVER. AFTER THE SWITCHOVER WAS COMPLETED, THE OPERATOR TRIED TO STOP NO. 22 COMPONENT COOLING WATER PUMP. THE OPERATOR HELD THE CONTROL SWITCH FOR NO. 22 COMPONENT COOLING WATER PUMP IN THE STOP POSITION UNTIL DISCHARGE PRESSURE HAD STABILIZED IN ACCORDANCE WITH GUIDANCE ISSUED AFTER THE LAST EVENT, UNIT 1 LER 1-90-009. WHEN THE CONTROL SWITCH WAS RELEASED AT 1010, NO. 22 COMPONENT COOLING WATER PUMP RESTARTED AUTOMATICALLY. THIS WAS A NON-ESF ACTUATION OF ESF EQUIPMENT. SYSTEM AND COMPONENT TESTING WAS DONE WHICH SHOWED THAT HEAT REMOVAL CAPACITY WAS ADEQUATE IN VARIOUS EQUIPMENT CONFIGURATIONS, BUT IN CERTAIN CONFIGURATIONS, AN AUTOSTART OF A COMPONENT COOLING WATER PUMP MAY RESULT DUE TO DISCHARGE PRESSURE FLUCTUATIONS DROPPING BELOW THE ACTUATION SETPOINT AND NOT INCREASING ABOVE THE RESET SETPOINT. PROCEDURES WILL BE REVISED TO PREVENT EQUIPMENT CONFIGURATIONS THAT CAN RESULT IN RECURRENCE OF THIS EVENT.

[127] QUAD CITIES 1 DOCKET 50-254 LER 90-021
RESIDUAL HEAT REMOVAL VALVE 1001-50 FAILED TO OPEN DUE TO THERMAL BINDING.
EVENT DATE: 081290 REPORT DATE: 102?90 NSSS: GE TYPE: BWR
VENDOR: LIMITORQUE CORP.
RELIANCE ELECTRIC COMPANY

(NSIC 219823) ON AUGUST 12, 1990, AT 0215 HOURS, UNIT ONE WAS IN THE SHUTDOWN MODE AT 0 PERCENT OF RATED CORE THERMAL POWER. WHILE PERFORMING QOP 1000-5, SHUTDOWN COOLING STARTUP AND OPERATION, MOTOR OPERATED VALVE (MOV) 1-1001-50 FAILED TO OPEN. FURTHER ATTEMPTS TO OPEN THE VALVE ALSO FAILED. AT 0400 HOURS, THE VALVE WAS MANUALLY OPENED TO ESTABLISH SHUTDOWN COOLING AS REQUIRED BY THE PROCEDURE. AT 1035 HOURS, IT WAS DETERMINED THAT THE MOTOR HAD FAILED. ON AUGUST 15, 1990, THE MOTOR WAS REPLACED AND THE VALVE SUCCESSFULLY TESTED. THE CAUSE OF THIS EVENT IS BELIEVED TO BE DUE TO THERMAL BINDING OF THE VALVE WITH A CONTRIBUTING CAUSE BEING HYDRAULIC LOCK OF THE VALVE OPERATOR. CORRECTIVE ACTIONS WILL INCLUDE EVALUATING AND CORRECTING ANY VALVE OPERATOR PROBLEMS, ENHANCING THE MOV PROGRAM AND OPERATING DEPARTMENT PROCEDURES, AND DISCUSSING THIS EVENT WITH OPERATING PERSONNEL. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR 50.73(A)(2)(I)(A).

[128] QUAD CITIES 1 DOCKET 50-254 LER 90-019
MISSED TECHNICAL SPECIFICATION SURVEILLANCES ON THE MAIN STEAM LINE RADIATION MONITORS DUE TO OPERATOR MISJUDGEMENT.
EVENT DATE: 081490 REPORT DATE: 102690 NSSS: GE TYPE: BWR

(NSIC 219821) ON AUGUST 14, 1990, AT 1245 HOURS, UNIT ONE WAS IN THE REFUEL MODE AT 0 PERCENT OF RATED CORE THERMAL POWER. WHILE REVIEWING OPERATING LOGS, THE SHIFT CONTROL ROOM ENGINEER (SCRE) DISCOVERED THAT TWO ONCE PER SHIFT CHECKS HAD BEEN MISSED FOR THE MAIN STEAM LINE (MSL) RADIATION MONITORS UNDER PROCEDURE QOS 005-S1, OPERATIONS DEPARTMENT WEEKLY SUMMARY OF DAILY SURVEILLANCES. ON

SEPTEMBER 9, 1990, AFTER FURTHER REVIEW, THIS EVENT WAS DETERMINED TO BE REPORTABLE UNDER 10 CFR 50.73. THE CAUSE OF THE MISSED SURVEILLANCES IS BEING ATTRIBUTED TO OPERATOR MISJUDGMENT. A CONTRIBUTING CAUSE IS THAT THE REVIEWS PERFORMED TO VERIFY COMPLETION ALSO FAILED TO IDENTIFY THE MISSED SURVEILLANCES. CORRECTIVE ACTION TAKEN WAS TO DISCUSS THIS EVENT WITH OPERATORS. FURTHER ACTION WILL INCLUDE A TRAINING EVALUATION AND PROCEDURE ENHANCEMENT. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73 (A)(2)(I)(B).

[129] QUAD CITIES 1 DOCKET 50-254 LER 90-020
CONTINUOUS FIRE WATCHES BEING PERFORMED ON A TWENTY MINUTE ROVING BASIS DUE TO A MISINTERPRETATION OF CONTINUOUS FIRE WATCH.
EVENT DATE: 092090 REPORT DATE: 101990 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 219822) AT 1048 HOURS ON SEPTEMBER 20, 1990, UNIT ONE WAS IN THE RUN MODE AT 100 PERCENT OF RATED CORE THERMAL POWER. AT THIS TIME, IT WAS DETERMINED THAT A MISINTERPRETATION OF THE MEANING OF CONTINUOUS FIRE WATCH HAD OCCURRED. IT WAS ALSO DETERMINED THAT THE HATCHWAYS FOR THE UNIT ONE CABLE TUNNEL HAD BEEN OPENED WITHOUT ESTABLISHING ANY COMPENSATORY MEASURES. A FIRE WATCH WAS IN EFFECT FOR SEVERAL OTHER AREAS IN THE PLANT AS A ROVING TWENTY MINUTE FIRE WATCH WHICH ACTUALLY REQUIRED A CONTINUOUS FIRE WATCH. THE CAUSE OF THIS EVENT IS PERSONNEL ERROR. THE REQUIREMENTS OF A CONTINUOUS FIRE WATCH WERE PERCEIVED TO BE SATISFIED BY A ROVING TWENTY MINUTE INSPECTION. THE HATCHWAYS TO THE CABLE TUNNEL WERE NOT KNOWN BY THE OPERATING DEPARTMENT TO BE FIRE BARRIERS. CONTINUOUS FIRE WATCHES WERE IMMEDIATELY ESTABLISHED FOR THE NECESSARY AREAS. ALL FIRE BARRIERS WILL BE INSPECTED TO ENSURE THAT THEY ARE DESIGNATED AS SUCH. QAP 1170-14 HAS BEEN CHANGED TO DESCRIBE A CONTINUOUS FIRE WATCH AND THE EXISTING FIRE PROTECTION PROGRAM WILL BE ENHANCED TO AVOID ANY FUTURE CONFUSION. THIS REPORT IS BEING SUBMITTED PER 10CFR50.73(A)(2)(II)(B).

[130] QUAD CITIES 1 DOCKET 50-254 LER 90-022
PIPING SYSTEM OUTSIDE FSAR COMPLIANCE CAUSED BY COMPUTER USER INPUT ERROR.
EVENT DATE: 092790 REPORT DATE: 102690 NSSS: GE TYPE: BWR

(NSIC 219824) ON SEPTEMBER 27, 1990 AT 0850 HOURS, UNIT ONE WAS IN THE RUN MODE AT 96 PERCENT OF RATED CORE THERMAL POWER. AT THIS TIME, THE STATION WAS INFORMED THAT A DESIGN INPUT ERROR WAS DISCOVERED BY AN ARCHITECT ENGINEER (AE) ASSIGNED TO ANALYZED THE RESIDUAL HEAT REMOVAL (RHR) TORUS ATTACHED PIPING. THE RHR LINES AFFECTED WERE 1-1013A-16", 1-1014A-14" AND 1-1017A-6". THE ERROR RESULTED IN A PORTION OF THE PIPING MODEL BEING ANALYZED FOR A DESIGN TEMPERATURE OF 90 DEGREES FAHRENHEIT (F) RATHER THAN 146F. THE RESULT IS THAT THIS PIPING MODEL DID NOT MEET THE FINAL SAFETY ANALYSIS REPORT (FSAR) CRITERIA FOR THE RHR SYSTEM. HOWEVER, OPERABILITY CRITERIA FOR THE PIPING, PIPING SUPPORTS, AND TORUS PENETRATIONS WAS VERIFIED. THE APPARENT CAUSE OF THIS EVENT WAS COMPUTER USER INPUT ERROR. A REVIEW OF THE OTHER TORUS ATTACHED PIPING ANALYZED IN THE MARK I PROGRAM REVEALED TEN OTHER SIMILAR INPUT ERRORS, HOWEVER NONE OF THESE EVENTS ARE SIGNIFICANT WITH RESPECT TO FSAR CRITERIA. CORRECTIVE ACTION FOR THIS EVENT IS TO RETURN THE THREE RHR LINES TO FSAR COMPLIANCE, COMPLETE TRAINING FOR USERS OF THE COMPUTER PROGRAM, AND REVISE THE USERS MANUAL. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73(A (2)(II) B .

[131] QUAD CITIES 2 DOCKET 50-265 LER 90-009
HIGH PRESSURE COOLANT INJECTION DECLARED INOPERABLE DUE TO FAILURE OF THE GLAND SEAL HOTWELL PUMP CAUSED BY LEVEL SWITCH FAILURE.
EVENT DATE: 091590 REPORT DATE: 101590 NSSS: GE TYPE: BWR
VENDOR: MERCOID CORP.

(NSIC 219832) ON SEPTEMBER 15, 1990 AT 2225 HOURS, UNIT TWO WAS IN THE RUN MODE AT 88 PERCENT OF RATED CORE THERMAL POWER. AT THIS TIME, THE HIGH PRESSURE COOLANT INJECTION (HPCI) GLAND SEAL HOTWELL PUMP WAS OBSERVED TO BE CYCLING OFF AND ON. THE HPCI SUBSYSTEM WAS DECLARED INOPERABLE AND A 24 HOUR LIMITING CONDITION FOR OPERATION (LCO) WAS ENTERED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.5.C.3. RON SEPTEMBER 16, 1990 AT 0153 HOURS, AN YESRGENCY

NOTIFICATION SYSTEM (ENS) PHONE NOTIFICATION WAS COMPLETED AS REQUIRED BY 10 CFR 50.72(B)(2)(III). AT 0330 HOURS, ELECTRICAL MAINTENANCE (EM) PERSONNEL FOUND THAT THE HIGH LEVEL SWITCH FOR THE HPCI GLAND SEAL HOTWELL PUMP HAD FAILED. THE SWITCH WAS REPAIRED AND THE SYSTEM WAS SUCCESSFULLY TESTED. CORRECTIVE ACTION WILL INCLUDE INSPECTING THE HPCI LEVEL SWITCHES AND PERFORMING AN EVALUATION OF THE INSPECTION RESULTS. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(V)(D).

[132] RIVERBEND 1 DOCKET 50-458 LER 86-045 REV 02
UPDATE ON REACTOR SCRAM DUE TO HIGH STEAM SEAL EVAPORATOR TANK LEVEL.
EVENT DATE: 071386 REPORT DATE: 101190 NSSS: GE TYPE: BWR
VENDOR: FISHER CONTROLS CO.

(NSIC 219677) AT 0310 HOURS ON 7/13/86 WITH THE REACTOR AT 55% POWER THE MAIN TURBINE TRIPPED DUE TO LOSS OF CONDENSER VACUUM, RESULTING IN A REACTOR SCRAM. REACTOR POWER HAD BEEN REDUCED FROM 85% STARTING AT 0305 HOURS WHEN IT WAS NOTED THAT VACUUM WAS STARTING TO DECAY. VACUUM WAS LOST BECAUSE OF THE FAILURE OF THE TURBINE STEAM SEAL SYSTEM TO SUPPLY ADEQUATE SEALING STEAM DUE TO A MALFUNCTION IN THE STEAM SEAL EVAPORATOR CONTROL SYSTEM. A LEVEL INDICATION TRANSIENT CAUSED THE ISOLATION OF CONDENSATE FLOW TO THE EVAPORATOR. WHEN THE EVAPORATOR SHELL SIDE LEVEL DROPPED BELOW THE HEATING TUBES, THE SUPPLY OF SEALING STEAM TO THE TURBINE WAS LOST. WHEN CONDENSATE FLOW WAS REINTRODUCED TO THE EVAPORATOR THE CHANGE IN LEVEL CAUSED A SIGNIFICANT CHANGE IN FLOW TO THE TUBE SIDE DRAIN TANK CAUSING THE TANK LEVEL TO SURGE. THIS LEVEL SURGE RESULTED IN A HIGH TUBE SIDE DRAIN TANK LEVEL INDICATION CAUSING AN ISOLATION OF THE MAIN STEAM SUPPLY TO THE EVAPORATOR. THE MAIN STEAM SUPPLY VALVE WAS RE-OPENED WITH A RESULTANT DRAIN TANK LEVEL SURGE AND ANOTHER ISOLATION. ON THE NEXT ATTEMPT TO RE-OPEN THIS VALVE IT TRIPPED ON THERMAL OVERLOAD RESULTING IN A COMPLETE LOSS OF THE SYSTEM. THIS EVENT DID NOT THREATEN THE HEALTH AND SAFETY OF THE PUBLIC.

[133] RIVERBEND 1 DOCKET 50-458 LER 90-027
MISSED FIRE WATCH DUE TO PERSONNEL ILLNESS.
EVENT DATE: 091690 REPORT DATE: 101690 NSSS: GE TYPE: BWR

(NSIC 219949) AT APPROXIMATELY 1540 HOURS ON 9/16/90, THE ELECTRICAL MAINTENANCE FOREMAN DISCOVERED THAT NO ONE HAD SIGNED THE FIRE WATCH LOG FOR THE 0900 ROVING FIRE WATCH PATROL IN THE CONTROL BUILDING (*NA*), RADWASTE BUILDING (*NE*), TURBINE BUILDING (*NM*), NORMAL SWITCHGEAR AND SERVICES BUILDINGS (*MF*). THE FOREMAN QUESTIONED PERSONNEL INVOLVED AND DETERMINED THAT THE FIRE WATCH HAD NOT BEEN PERFORMED. THE PERSON ASSIGNED TO THE MISSED FIRE WATCH BECAME VERY ILL BETWEEN 0700 AND 0800 AND WAS TOLD SHE SHOULD GO HOME. HER FOREMAN CONTACTED THE LEVEL I TECHNICIAN TO REPLACE HER ON SHIFT. THE LEVEL I TECHNICIAN PROCEEDED TO GET THE CONTROL BUILDING KEYS AND STARTED THE FIRE WATCH RUN AT 0850, BUT DID NOT COMPLETE THE FIRE WATCH IN TIME. IN ADDITION, THE LEVEL I TECHNICIAN FAILED TO IMMEDIATELY INFORM THE FOREMAN AFTER THE FIRE WATCH WAS MISSED. TO PREVENT RECURRENCE, EACH PERSON ASSIGNED TO A FIRE WATCH ON SHIFT AND THE LEVEL I TECHNICIAN WILL BE REQUIRED TO CARRY A COMPLETE SET OF KEYS TO MAKE ALL ROUNDS. RETRAINING OF ALL FIRE PROTECTION PERSONNEL WILL BE CONDUCTED DUE TO THIS INCIDENT.

[134] RIVERBEND 1 DOCKET 50-458 LER 90-030
VIOLATION OF A HIGH RADIATION AREA DUE TO PERSONNEL ERROR.
EVENT DATE: 101090 REPORT DATE: 110190 NSSS: GE TYPE: BWR

(NSIC 219950) ON OCTOBER 10, 1990 AT APPROXIMATELY 2213 HOURS WITH THE PLANT IN OPERATIONAL CONDITION 5 (REFUELING), TWO WORKERS ENTERED A HIGH RADIATION AREA (HRA) WITHOUT AN ALARMING DOSIMETER OR A RADIATION PROTECTION (RP) TECHNICIAN ESCORTING THEM. SINCE THE REQUIREMENTS OF TECHNICAL SPECIFICATION 6.12.1 WERE NOT MET, THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B) AS OPERATION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR ON THE PART OF A CONTRACT RP TECHNICIAN; HE FAILED TO RECOGNIZE THE HRA. CORRECTIVE ACTIONS INCLUDED LIMITING THE NUMBER OF RP TECHNICIANS AUTHORIZED TO SET ENTRY REQUIREMENTS INTO AN AREA FOR WHICH A CONTROL POINT IS

RESPONSIBLE. IN ADDITION, REVIEW OF THIS EVENT AND THE PROCEDURE REQUIREMENTS FOR HRA ENTRY HAVE BEEN COMPLETED FOR ALL GSU AND CONTRACT RP TECHNICIANS SUPPORTING THE THIRD REFUELING OUTAGE. THE MAXIMUM EXPOSURE RECEIVED BY THE INDIVIDUALS INVOLVED WAS 5 MREM, WELL BELOW THE REGULATORY OR STATION ADMINISTRATIVE LIMITS. THIS EVENT DID NOT ADVERSELY AFFECT THE HEALTH AND SAFETY OF THE PUBLIC OR THE SAFETY OF THE PLANT.

[135] ROBINSON 2 DOCKET 50-261 LER 90-012
POTENTIAL OF INADEQUATE NET POSITIVE SUCTION HEAD FOR SAFETY INJECTION PUMPS.
EVENT DATE: 092890 REPORT DATE: 102590 NSSS: WE TYPE: PWR

(NSIC 219829) ON SEPTEMBER 25, 1990, WITH H. B. ROBINSON UNIT NO. 2 IN COLD SHUTDOWN FOR A SCHEDULED REFUELING OUTAGE, A SPECIAL TEST WAS PERFORMED TO DETERMINE THE CAPABILITY OF THE ONE SAFETY INJECTION PUMP INJECTING INTO THREE COLD LEGS OF THE REACTOR COOLANT SYSTEM, WHICH COULD BE AN EXPECTED ALIGNMENT DURING CERTAIN DESIGN BASIS LOSS OF COOLANT ACCIDENTS. THE RESULTS OF THE TEST INDICATED THE POTENTIAL FOR INADEQUATE NET POSITIVE SUCTION HEAD (NPSH) AVAILABLE FOR EITHER SAFETY INJECTION PUMP. BASED ON THIS TEST, BOTH SAFETY INJECTION PUMPS WERE CONSERVATIVELY DECLARED INOPERABLE. IT SHOULD BE NOTED THAT THE PLANT WAS IN A MODE WHERE THE PUMPS WERE NOT REQUIRED TO BE OPERABLE, AND THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC. THE CAUSE FOR THE POTENTIAL FOR PUMP RUNOUT HAS NOT BEEN DETERMINED DUE TO THE UNAVAILABILITY OF INFORMATION NECESSARY TO ADEQUATELY ASSESS THE SIGNIFICANCE OF THE CONCERN. RESOLUTION OF THIS ISSUE IS BEING ACTIVELY PURSUED, AND RESULTS WILL BE REPORTED IN A SUPPLEMENT TO THIS LICENSEE EVENT REPORT.

[136] SALEM 1 DOCKET 50-272 LER 90-032
1R13A AND B ALARM SETPOINTS WRONG DUE TO PERSONNEL ERROR.
EVENT DATE: 091990 REPORT DATE: 101990 NSSS: WE TYPE: PWR
VENDOR: LFE CORP.

(NSIC 219834) ON 9/19/90, THE SYSTEM ENGINEER FOR THE RADIATION MONITORING SYSTEM (RMS) DISCOVERED THAT THE ALARM SETPOINT FOR THE 1R13A AND 1R13B (NOS. 11 AND 12 CFCU SERVICE WATER MONITORS) RMS CHANNELS WAS NOT CORRECT. THE SETPOINT WAS SET AT 2600 CPM INSTEAD OF 700 CPM. A 1984/85 DESIGN CHANGE REPLACED THE 1R13A & B DETECTORS WITH 1/2" CRYSTALS (PREVIOUSLY THEY WERE 1" CRYSTALS). PER THE VENDOR MANUALS, THE 1R13A & B DETECTOR SENSITIVITY IS ROUGHLY ONE HALF THAT OF THE OTHER 1R13 DETECTORS. RECENT PROCEDURE MODIFICATIONS INCORRECTLY RESET THE 1R13A & B ALARM SETPOINTS TO THE SAME SETPOINT AS THE 1R13C, D & E DETECTORS. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO PERSONNEL ERROR. IN 1984/85, WHEN THE ODCM WAS REVISED, IT DID NOT INCLUDE THE CORRECT SETPOINT FOR THE 1R13A & B CHANNELS. THE LATEST REVISION TO THE ODCM (REV. 6 ISSUED IN MARCH 1990) STILL HAS THE SAME 1R13 SETPOINT. THE APPROPRIATE PROCEDURES WERE REVISED TO THE SETPOINT IDENTIFIED IN THE ODCM. SUBSEQUENT REVIEW OF THIS EVENT REVEALED THAT THE CHANNEL CALIBRATION HAS NOT BEEN CORRECTLY PERFORMED SINCE THE INSTALLATION OF THE NEW DETECTORS. APPROPRIATE PROCEDURES HAVE BEEN REVISED TO ADDRESS THE CORRECT ALARM SETPOINT. AN ADMINISTRATIVE PROCEDURE, DETAILING THE REQUIRED ACTIONS NECESSARY TO MAINTAIN THE ODCM, WILL BE PREPARED. THE CURRENT ADMINISTRATIVE CONTROLS ASSOCIATED WITH CONFIGURATION CONTROL DOCUMENTATION WILL BE REVIEWED.

[137] SALEM 1 DOCKET 50-272 LER 90-033
TWO CHANNELS MADE INOPERABLE DUE TO COMMON EQUIPMENT FAILURE.
EVENT DATE: 092390 REPORT DATE: 102390 NSSS: WE TYPE: PWR
VENDOR: CONDE MILKING MACHINE COMPANY

(NSIC 219835) ON 9/23/90, A CONTROL ROOM OPERATOR OBSERVED THAT THE 1R11A/1R12A/1R12B CONTAINMENT RADIATION MONITORING SYSTEM (RMS) PUMP OPERABILITY INDICATION WAS NOT ILLUMINATED. SINCE IT WAS QUESTIONABLE AS TO PUMP OPERABILITY, THE 1R11A, 1R12A AND 1R12B RMS CHANNELS WERE DECLARED INOPERABLE AND TECHNICAL SPECIFICATION ACTION STATEMENTS 3.4.6.1 AND 3.9.9 WERE ENTERED AS PER TECHNICAL SPECIFICATION TABLE 3.3-6 ACTION 20 (I.E., 1R11A INOPERABLE) AND ACTION 22 (I.E., 1R12A INOPERABLE). THE 1R12B CHANNEL IS NOT TAKEN CREDIT FOR IN THE ACCIDENT ANALYSES. THE 1R11A AND 1R12A CHANNELS PROVIDE AUTOMATIC ISOLATION OF

THE CONTAINMENT PURGE/PRESSURE - VACUUM RELIEF SYSTEM UPON DETECTION OF HIGH ACTIVITY. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO AN EQUIPMENT/DESIGN CONCERN. CONDENSATION HAD CAUSED CORROSION (RUST) WHICH SEIZED THE MOVING VANES OF THE POSITIVE DISPLACEMENT SAMPLING PUMP. UPON REPLACEMENT OF THE SAMPLE PUMP, THE 1R11A AND 1R12A MONITORS WERE DECLARED OPERABLE AND THE APPROPRIATE ACTION STATEMENT WERE EXITED. AN INSPECTION PROGRAM WILL BE INITIATED TO IDENTIFY THE MEAN TIME TO SAMPLE PUMP FAILURE. THE DESIGN OF THE CONTAINMENT 1R11A/1R12A/1R12B RMS WILL BE REVIEWED. APPROPRIATE DESIGN CHANGE(S) TO ADDRESS CONDENSATION CONCERNS WILL BE MADE AS APPLICABLE. THE PUMP WILL BE REPLACED WITH A MORE RELIABLE PUMP.

[138] SALEM 1 DOCKET 50-272 LER 90-034
CONTAINMENT VENTILATION ISOLATION DUE TO 1R11A RMS CHANNEL SPIKES.
EVENT DATE: 100590 REPORT DATE: 110590 NSSS: WE TYPE: PWR
VENDOR: LFE CORP.

(NSIC 219836) ON 10/5/90, A CONTAINMENT PURGE/PRESSURE-VACUUM RELIEF SYSTEM (CP/P-VRS) ISOLATION SIGNAL ACTUATED AS A RESULT OF A CHANNEL SPIKE ON THE RADIATION MONITORING SYSTEM (RMS) CONTAINMENT PARTICULATE RADIATION MONITOR, 1R11A. SUBSEQUENTLY, TECH. SPEC. 3.4.6.1 ACTION STATEMENT WAS ENTERED. THE ISOLATION VALVES WERE CLOSED AT THE TIME OF THIS SIGNAL; THEY DID NOT CHANGE POSITION. ON 10/6/90, UPON COMPLETION OF A SUCCESSFUL CHANNEL CALIBRATION, THE CHANNEL WAS DECLARED OPERABLE AND TECHNICAL SPECIFICATION 3.4.6.1 ACTION STATEMENT WAS EXITED. ON 11/1/90, ANOTHER 1R11A CHANNEL SPIKE OCCURRED RESULTING IN ACTUATION OF A CP/P-VRS SIGNAL. A CONTAINMENT PRESSURE RELIEF WAS IN PROGRESS AT THE TIME OF THIS EVENT. THE ASSOCIATED VALVES CLOSED IN RESPONSE TO THE ISOLATION SIGNAL. TECH. SPEC. 3.4.6.1 ACTION STATEMENT WAS AGAIN ENTERED. THE ROOT CAUSE OF THE TWO (2) 1R11A CHANNEL SPIKES HAS NOT YET BEEN IDENTIFIED. INVESTIGATIVE TROUBLESHOOTING IS CONTINUING. A HISTORICAL REVIEW OF PRIOR 1R11A CHANNEL FAILURES WAS CONDUCTED. THE CHANNEL HAS BEEN SUBJECT TO SPIKES (APPROXIMATELY ONCE OR TWICE A YEAR). THE CORRECTIVE ACTIONS FOR THE PRIOR EVENTS INCLUDED CONTACT CLEANING AND CONNECTOR ADJUSTMENT. NO FAILED COMPONENT WAS IDENTIFIED FOR SEVERAL OF THE PRIOR EVENTS. A PREVIOUS RMS HISTORICAL REVIEW WAS CONDUCTED (REFERENCE LER 311/90-031-00).

[139] SALEM 2 DOCKET 50-311 LER 90-037
2C DIESEL GENERATOR START AND START OF SAFEGUARDS EQUIPMENT DUE TO 2C SEC EQUIPMENT FAILURE.
EVENT DATE: 092290 REPORT DATE: 101790 NSSS: WE TYPE: PWR
VENDOR: POTTER & BRUMFIELD

(NSIC 219893) ON 9/22/90 AT 0245 HOURS, APPROXIMATELY ONE (1) MINUTE AFTER RESETTING THE 2C SEC (UPON SUCCESSFUL COMPLETION OF A 2C DIESEL GENERATOR (D/G) ONE (1) HOUR SURVEILLANCE RUN), A 2C SEC MODE I ACCIDENT SAFEGUARDS ACTUATION OCCURRED. FOLLOWING THE 2C SEC MODE I ACTUATION, THE SEC WAS SUCCESSFULLY RESET AND WAS THEREFORE CONSIDERED OPERABLE. THE SEC IS REQUIRED TO BE RESET AS PART OF THE PROCEDURE FOR RESTORING A DIESEL GENERATOR (D/G) TO NORMAL ALIGNMENT UPON COMPLETION OF A ONE (1) HOUR SURVEILLANCE RUN. WHEN THE D/G SURVEILLANCE IS INITIATED, THE D/G INFEED BREAKER CHANGES STATE TO ALLOW LOADING OF THE D/G. DUE TO THE CHANGE OF STATE OF THE D/G INFEED BREAKER, THE SEC "AUTO TEST FEATURE" (WHICH CONTINUOUSLY TESTS SEC CIRCUIT CONTINUITY) WILL CAUSE ACTUATION OF THE "SEC TROUBLE ALARM". THE 2C SEC MODE I ACCIDENT SAFEGUARDS INITIATION WAS VERIFIED BY OBSERVATION OF THE ACTUATION OF THE 2C SEC "INPUT LAMP NO. 1" AND THE PRINTOUT OF THE CONTROL ROOM AUXILIARY ALARM TYPEWRITER. SPECIFICALLY, 2C D/G STARTED ALONG WITH SAFEGUARDS EQUIPMENT ASSOCIATED WITH THE 2C VITAL BUS. THE ROOT CAUSE OF THE 2C SEC MODE I ACCIDENT LOADING AND 2C D/G START IS EQUIPMENT FAILURE. INVESTIGATION REVEALED THAT THE XK1 INPUT RELAY (POTTER-BRUMFIELD RELAY MODEL KUP-14-A15) HAD FAILED. THE RELAY FAILURE WOULD NOT HAVE PREVENTED THE 2C SEC FROM PERFORMING ITS DESIGN FUNCTION(S).

[140] SAN ONOFRE 1 DOCKET 50-206 LER 90-007 REV 01
UPDATE ON REACTOR TRIP ON A SPURIOUS LOW REACTOR COOLANT SYSTEM FLOW SIGNAL.
EVENT DATE: 043090 REPORT DATE: 102390 NSSS: WE TYPE: PWR
VENDOR: FOXBORO CO., THE
ROME CABLE CORP

(NSIC 219807) AT 2202 ON 4/30/90, WITH UNIT 1 AT 91% POWER, A REACTOR TRIP OCCURRED DUE TO ACTUATION OF THE REACTOR PROTECTION SYSTEM (RPS) ON A SPURIOUS LOW REACTOR COOLANT SYSTEM (RCS) FLOW SIGNAL IN LOOP B. ALL SYSTEMS RESPONDED NORMALLY TO THE TRIP AND THE OPERATORS STABILIZED THE PLANT IN MODE 3. THE RPS OPERATED IN ACCORDANCE WITH DESIGN WITH NO MALFUNCTIONS NOTED; THEREFORE THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT. THE ROOT CAUSE OF THIS EVENT IS BELIEVED TO BE THE EXISTENCE OF VOIDS IN THE LOOP B RCS FLOW TRANSMITTER COIL INSULATION, WHICH CAUSED A GROUND IN THE COIL. IT HAS BEEN DETERMINED THAT THE VOIDS WERE DUE TO AN ISOLATED MANUFACTURING DEFECT. SCE BELIEVES THAT THE GROUND IN THE FLOW TRANSMITTER COIL CAUSED THE INSTRUMENT LOOP TO BE SENSITIVE TO THE PRESENCE OF MAGNETIC FIELDS ADJACENT TO THE INSTRUMENT LOOP CABLES. THE OCCURRENCE OF A MAGNETIC FIELD, WHICH COULD BE CAUSED BY THE NORMAL OPERATION OF PLANT EQUIPMENT, NEAR THE FLOW TRANSMITTER SIGNAL CABLE (OR A STEADY MAGNETIC FIELD IF THE GROUND WAS INTERMITTENT) COULD HAVE PRODUCED A NOISE PULSE LARGE ENOUGH AND OF SUFFICIENT DURATION TO CAUSE AN INDUCED CURRENT IN THE FLOW INSTRUMENTATION CIRCUIT SUFFICIENT TO ACTUATE THE LOW FLOW TRIP SIGNAL, THEREBY CAUSING THE REACTOR TRIP. CORRECTIVE ACTIONS INCLUDED REPLACEMENT OF THE DEFECTIVE TRANSMITTER AND VERIFYING THAT ALL RCS LOOP FLOW INSTRUMENTS WERE OPERATING PROPERLY.

[141] SAN ONOFRE 1 DOCKET 50-206 LER 90-018
VALVES NOT INSERVICE TESTED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 4.7, "INSERVICE INSPECTION REQUIREMENTS".
EVENT DATE: 092090 REPORT DATE: 102290 NSSS: WE TYPE: PWR

(NSIC 219808) ON 09/20/90 AND 09/25/90, WITH UNIT 1 DEFUELED, ENGINEERING EVALUATIONS DETERMINED THAT THE REQUIREMENTS OF TECHNICAL SPECIFICATION 4.7, "INSERVICE INSPECTION REQUIREMENTS," MAY NOT HAVE BEEN MET. THE EVALUATIONS DETERMINED THAT TEN MAIN STEAM [SB] ASME CLASS 2 VALVES (FOUR MOISTURE SEPARATOR REHEATER (MSR) AIR OPERATED WARM-UP [FCV] VALVES, FOUR MOTOR OPERATED MSR BLOCK VALVES [ISV], AND TWO AIR OPERATED MAIN CONDENSER STEAM DUMP VALVES [PCV]) WHICH ARE REQUIRED TO BE CLOSED PER GDC-57, "CLOSED SYSTEM ISOLATION VALVES," HAD BEEN DELETED FROM THE INSERVICE TESTING (IST) PROGRAM. A PRELIMINARY INVESTIGATION REVEALED THAT THE TEN MS VALVES HAD BEEN DELETED FROM THE PROGRAM IN 1978. A CLEAR BASIS FOR DELETION OF THESE VALVES HAS NOT YET BEEN ESTABLISHED. THEY HAVE NOW BEEN INCLUDED IN THE IST PROGRAM AND THEIR OPERABILITY WILL BE VERIFIED BEFORE UNIT 1 IS RETURNED TO SERVICE. SIMILAR DISCREPANCIES WERE PREVIOUSLY REPORTED IN LERS 89-013, 90-004, AND 90-009 (DOCKET 50-206). AS A RESULT OF THESE EVENTS AND OTHER SCE REVIEWS, CORRECTIVE ACTIONS WERE UNDERTAKEN TO UPDATE THE SONGS IST PROGRAM. REVIEWS ARE BEING CONTINUED IN ORDER TO ENSURE THAT THE SONGS IST PROGRAM FOR ALL THREE UNITS ACCURATELY REFLECTS PRESENT DESIGN, REGULATORY GUIDANCE, AND CODE REQUIREMENTS.

[142] SAN ONOFRE 3 DOCKET 50-362 LER 90-002 REV 01
UPDATE ON REACTOR TRIP WITH PRESSURIZER SAFETY VALVE OPENING RESULTING FROM SPURIOUS MAIN STEAM ISOLATION SYSTEM ACTUATION CAUSED BY DEGRADED ACTUATION PUSHBUTTON.
EVENT DATE: 022390 REPORT DATE: 103190 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 2 (PWR)
VENDOR: BAILEY INSTRUMENT CO., INC.
HONEYWELL CORP.
NEWARK

(NSIC 219908) AT 2257 ON FEBRUARY 23, 1990, WITH UNIT 3 AT 100% POWER, DURING THE PERFORMANCE OF MAIN STEAM ISOLATION SYSTEM (MSIS) SUBGROUP RELAY TESTING, A SPURIOUS MSIS ACTUATION OCCURRED. THIS MSIS ACTUATION INITIATED CLOSURE OF THE MAIN STEAM ISOLATION VALVES (MSIVS) AND RELATED IN A REACTOR TRIP ON HIGH PRESSURIZER PRESSURE. THE PLANT TRANSIENT INCLUDED: 1) LIFTING OF THE MAIN

STEAM SAFETY VALVES (MSSVS); (2) INITIATION OF THE EMERGENCY FEEDWATER ACTUATION SYSTEM (EFAS); AND 3) THE BRIEF OPENING OF A PRESSURIZER SAFETY VALVE (PSV). HOWEVER, DURING THE TRANSIENT, RCS PRESSURE NEVER EXCEEDED THE MINIMUM PSV SET POINT VALUE ALLOWED BY TECHNICAL SPECIFICATION (TS) 3.4.2. OPERATORS STABILIZED THE PLANT IN MODE 3. FOLLOWING THIS EVENT, IT WAS DETERMINED THAT THE LIFT SET POINTS FOR THE UNITS 2 AND 3 PSVS WERE OUTSIDE THE TS REQUIREMENTS. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE: 1) ALL REACTOR PROTECTION SYSTEM, MSIS, AND EFAS COMPONENTS WERE VERIFIED TO ACTUATE IN ACCORDANCE WITH DESIGN; AND 2) THE PLANT RESPONSE WITH THE AS-FOUND PSV LIFT SET POINTS WAS DETERMINED TO BE ACCEPTABLE. CONTACTS ON THE MSIS MANUAL ACTUATION PUSHBUTTON (PB) WERE FOUND TO BE DEGRADED SUCH THAT ONE TRIP LEG OF MSIS DID NOT RESET DURING THE SUBGROUP RELAY TESTING.

[143] SAN ONOFRE 3 DOCKET 50-362 LER 90-010
MISALIGNMENT OF CONTAINMENT EMERGENCY SUMP ISOLATION VALVE 3HV-9302.
EVENT DATE: 092890 REPORT DATE: 102990 NSSS: CE TYPE: PWR

(NSIC 219909) ON 9/28/90, AT 0900, WITH UNIT 3 AT 100% POWER, A NORMALLY CLOSED TRAIN "B" CONTAINMENT EMERGENCY SUMP (CES) SUCTION ISOLATION VALVE (3HV-9303) WAS DISCOVERED TO BE OPEN; IT WAS PROMPTLY CLOSED. 3HV-9302 IS REQUIRED TO BE IN THE CLOSED POSITION DURING THE INJECTION PHASE OF ECCS/CS AND OPENS ON A RECIRCULATION ACTUATION SIGNAL (RAS) TO PROVIDE A FLOW PATH FROM THE CES DURING THE RECIRCULATION PHASE. WITH THIS VALVE OPEN, THE FUNCTION OF THE TRAIN "B" EMERGENCY CORE COOLING (ECCS) AND CONTAINMENT SPRAY (CS) SYSTEMS WOULD HAVE BEEN IMPAIRED DURING POSTULATED ACCIDENTS IN WHICH THE CONTAINMENT PRESSURE EXCEEDS THAT AVAILABLE FROM THE REFUELING WATER STORAGE TANK. THE VALVE WAS DETERMINED TO HAVE BEEN INADVERTENTLY OPENED ON 9/24/90 AND REMAINED OPEN FOR 95 HOURS. THIS CONDITION WAS CONTRARY TO TSS 3.5.2, "ECCS SUBSYSTEMS," AND 3.6.2, "CONTAINMENT SPRAY SYSTEMS," SINCE THE 72-HOUR ACTION REQUIREMENT WAS EXCEEDED. AS A RESULT OF ONGOING EVALUATIONS OF THE SIGNIFICANCE OF THE OPEN SUMP VALVE, IT WAS SUBSEQUENTLY DETERMINED AND REPORTED TO THE NRC ON 10/5/90, THAT THE MISALIGNMENT ALSO RESULTED IN DEGRADED CONTAINMENT INTEGRITY. THE EVALUATION OF THIS CONDITION ALSO REVEALED A REMOTE POTENTIAL FOR A SINGLE FAILURE SUSCEPTIBILITY WHICH COULD RESULT IN SIMULTANEOUS OPENING OF BOTH THE INSIDE AND OUTSIDE CONTAINMENT VALVES. A MODIFICATION WAS IMPLEMENTED TO PRECLUDE THIS SINGLE FAILURE SUSCEPTIBILITY.

[144] SEABROOK 1 DOCKET 50-443 LER 90-015 REV 01
UPDATE ON TURBINE TRIP WITH REACTOR TRIP DUE TO GROUND FAULT RELAY ACTUATION.
EVENT DATE: 062090 REPORT DATE: 102290 NSSS: WE TYPE: PWR

(NSIC 219937) ON 6/20/90, AT 4:39 P.M., WHILE IN MODE 1 AT 30% REACTOR POWER AND INCREASING, A TURBINE-GENERATOR TRIP WITH REACTOR TRIP OCCURRED. THE TRIP WAS INITIATED BY THE ACTUATION OF A MAIN GENERATOR GROUND FAULT RELAY DESIGNED TO PROTECT THE LAST 5% OF GENERATOR WINDINGS FROM A GROUND FAULT. AN EMERGENCY FEEDWATER (EFW) ACTUATION ALSO OCCURRED DUE TO LOW-LOW STEAM GENERATOR NARROW RANGE LEVEL. STEAM GENERATOR "A" EFW ISOLATED DUE TO HIGH EFW FLOW ONE MINUTE INTO THE EVENT. THE TURBINE GENERATOR TRIPPED DUE TO THE GROUND FAULT RELAY ACTUATING. ALL THE APPLICABLE TRIPS AND INTERLOCKS ASSOCIATED WITH A TURBINE GENERATOR, REACTOR TRIP AND FEEDWATER ISOLATION FUNCTIONED AS DESIGNED. PARAMETERS ASSOCIATED WITH THE REACTOR WERE REVIEWED BY THE ON-SHIFT REACTOR ENGINEER AND DETERMINED TO BE NORMAL. THE EFW ISOLATION OCCURRED WHEN THE TURBINE DRIVEN PUMP CAME UP TO FULL SPEED, THUS CREATING A SITUATION WHERE BOTH EFW PUMPS WERE SUPPLYING FEEDWATER SIMULTANEOUSLY. THE ROOT CAUSE FOR THE ACTUATION OF THE GROUND FAULT RELAY HAS BEEN DETERMINED TO BE INCORRECT RELAY SETTINGS PROVIDED BY THE VENDOR. A MINOR MODIFICATION (MMOD) WILL BE ISSUED TO REVISE THE RELAY SETTINGS AND BYPASS THE TRIP FUNCTION OF THE RELAY. ADDITIONAL CORRECTIVE ACTIONS INCLUDE REVISING THE EFW ISOLATION SETPOINT AS WELL AS REVIEWING AND UPDATING EMERGENCY OPERATING PROCEDURES.

[145] SEABROOK 1 DOCKET 50-443 LER 90-023
UNSECURED HIGH RADIATION AREA RESULTS IN TECH SPEC NONCOMPLIANCE.
EVENT DATE: 092790 REPORT DATE: 102690 NSSS: WE TYPE: PWR

(NSIC 219938) ON 9/27/90 AT 4:07 P.M. EDT, WHILE IN MODE 1, IT WAS DISCOVERED THAT A DOOR TO A LOCKED HIGH RADIATION AREA, THE "DEMIN ALLEY", LOCATED IN THE PRIMARY AUXILIARY BUIDING, WAS UNLOCKED AND HAD BEEN UNLOCKED FOR APPROXIMATELY FIVE HOURS. THIS EVENT OCCURRED CONTRARY TO SEABROOK STATION TECH SPEC 6.11.2. ON 9/27/90 AT 11:08 A.M. A HEALTH PHYSICS FOREMAN ENTERED THE DEMIN ALLEY THROUGH DOOR P203. UPON CLOSURE, HOWEVER, THE DOOR LATCH DID NOT COMPLETELY ENGAGE AND THE DOOR WAS CONSEQUENTLY LEFT UNLOCKED. ADDITIONALLY, THE FOREMAN INVOLVED FAILED TO CHECK THE DOOR TO ENSURE THAT IT WAS PROPERLY LOCKED PRIOR TO EXITING THE AREA. THE ROOT CAUSE HAS BEEN ATTRIBUTED TO PERSONNEL ERROR. THE HEALTH PHYSICS FOREMAN WAS COUNSELLED AND RECEIVED DISCIPLINARY ACTION. ADDITIONAL CORRECTIVE ACTIONS INCLUDE: (1) THE INSTALLATION OF A PADLOCK MECHANISM ON CURRENT LOCKED HIGH RADIATION DOORS, EXCEPT FOR THE CONTAINMENT PERSONNEL HATCH CONTROLS AND AREAS INSIDE CONTAINMENT; (2) REMOVAL OF KEY CARD ACCESS AUTHORIZATION TO LOCKED HIGH RADIATION AREAS FOR ALL PERSONNEL; (3) NEW PROCEDURAL REQUIREMENTS WHICH INCLUDES A DOUBLE VERIFICATION SIGN-OFF THAT THE LOCKED HIGH RADIATION AREA DOOR IS INDEED LOCKED; AND (4) A REQUIREMENT THAT A PERSON WITH THE WORK PARTY REMAIN OUTSIDE THE LOCKED HIGH RADIATION AREA DOOR TO ENSURE PROPER ACCESS CONTROLS.

[146] SEQUOYAH 1 DOCKET 50-327 LER 90-018 REV 01
UPDATE ON REQUIRED SURVEILLANCE INSPECTION NOT PERFORMED ON FOUR FIRE DOOR DAMPERS AS A RESULT OF AN INADEQUATE PROCEDURE.
EVENT DATE: 083190 REPORT DATE: 102990 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 219868) THE LER IS BEING REVISED TO ADD REFERENCE TO ONE PREVIOUS OCCURRENCE OF A SIMILAR EVENT INADVERTENTLY OMITTED FROM THE INITIAL LER. ON AUGUST 31, 1990, WITH UNITS 1 AND 2 IN MODE 1, IT WAS DISCOVERED THAT SURVEILLANCE INSPECTIONS ON FOUR FIRE DAMPERS LOCATED IN FIRE DOORS IN THE AUXILIARY BUILDING HAD NOT BEEN PERFORMED WITHIN THE REQUIRED TIME INTERVAL. ALTHOUGH THE SUBJECT FIRE DOORS ARE INCLUDED IN THE APPROPRIATE SURVEILLANCE INSTRUCTION (SI), NO ACCEPTANCE CRITERIA ARE GIVEN FOR THE FIRE DAMPERS. THEREFORE, THERE IS NO RECORD OF THE FIRE DAMPERS HAVING BEEN INSPECTED. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO AN INADEQUATE PROCEDURE. UPON DISCOVERY OF THE PROBLEM LIMITING CONDITION FOR OPERATION (LCO) 4.0.3 WAS ENTERED. A WORK REQUEST WAS PREPARED AND COMPLETED THE SAME DAY FOR INSPECTION OF EACH OF THE FIRE DAMPERS. ALL FOUR FIRE DAMPERS WERE VERIFIED TO BE FUNCTIONAL, AND LCO 4.0.3 WAS EXITED. THIS PROBLEM WAS VERIFIED TO BE LIMITED TO ONLY THE FOUR SUBJECT DAMPERS. THE SI WILL BE REVISED TO INCLUDE ACCEPTANCE CRITERIA FOR THE FIRE DAMPERS PRIOR TO THE NEXT SCHEDULED PERFORMANCE, BUT NO LATER THAN DECEMBER 1, 1991.

[147] SEQUOYAH 1 DOCKET 50-327 LER 90-021
REACTOR TRIP FROM LOW STEAM GENERATOR LEVEL AS A RESULT OF A FEEDWATER TRANSIENT INITIATED BY THE FAILURE OF A VITAL INVERTER DURING A TRANSFER FROM THE MAINTENANCE OF THE NORMAL POWER SUPPLY.
EVENT DATE: 091490 REPORT DATE: 101590 NSSS: WE TYPE: PWR
VENDOR: SOLID STATE CONTROLS, INC.

(NSIC 219736) ON SEPTEMBER 14, 1990, WITH UNIT 1 IN MODE 1, A REACTOR TRIP OCCURRED AT 1613 EASTERN DAYLIGHT TIME. THE TRIP WAS GENERATED FROM A LOW-LOW STEAM GENERATOR WATER LEVEL SIGNAL IN LOOP 2. THE LOW LEVEL WAS THE RESULT OF A FEEDWATER TRANSIENT INITIATED BY THE FAILURE OF A VITAL INVERTER. THE INVERTER FAILURE OCCURRED AFTER THE COMPLETION OF MAINTENANCE ACTIVITIES ON THE INVERTER AND DURING THE TRANSFER OF THE INVERTER FROM ITS MAINTENANCE POWER SUPPLY TO ITS NORMAL POWER SUPPLY. DURING THE TRANSFER, THE INVERTER OUTPUT VOLTAGE DROPPED TO ZERO BECAUSE OF THE RANDOM FAILURE OF THE INVERTER'S SILICON-CONTROLLED RECTIFIERS. THIS DEENERGIZED THE 1-II VITAL INSTRUMENT POWER BOARD. THE LOSS OF POWER RESULTED IN THE MAIN FEEDWATER REGULATOR VALVES CLOSING AND THE MAIN FEEDWATER PUMPS DROPPING TO MINIMUM SPEED. THIS REDUCED FEEDWATER FLOW TO ALL

FOUR STEAM GENERATORS. PLANT SYSTEMS RESPONDED PROPERLY AND THE SHUTDOWN POSED NO DANGER TO PLANT EMPLOYEES OR THE GENERAL PUBLIC. THE UNIT WAS STABILIZED IN ACCORDANCE WITH PLANT PROCEDURES. THE VITAL INVERTER WAS REPAIRED AND RETURNED TO SERVICE ON SEPTEMBER 15, 1990.

[148] SEQUOYAH 1 DOCKET 50-327 LER 90-022
REACTOR TRIP AS A RESULT OF A TURBINE TRIP CAUSED FROM CORRODED AND SHORTED TERMINALS ON THE SPARE (A PHASE) MAIN TRANSFORMER'S GAS RELAY.
EVENT DATE: 091990 REPORT DATE: 101990 NSSS: WE TYPE: PWR
VENDOR: ASEA ELECTRIC, INC.

(NSIC 219869) ON SEPTEMBER 19, 1990, WITH UNIT 1 OPERATING AT APPROXIMATELY 60 PERCENT REACTOR POWER, 2235 POUNDS PER SQUARE INCH GAUGE (PSIG), AND 564 DEGREES FAHRENHEIT (F), A TURBINE TRIP FOLLOWED BY A REACTOR TRIP OCCURRED AT 0357 EASTERN DAYLIGHT TIME. THE TURBINE TRIPPED AS A RESULT OF "A" PHASE MAIN TRANSFORMER DIFFERENTIAL RELAY (SUDDEN PRESSURE) OPERATION CAUSED FROM CORRODED AND SHORTED TERMINALS ON THE TRANSFORMER GAS RELAY. BECAUSE THE REACTOR POWER WAS GREATER THAN THE REACTOR TRIP INTERLOCK FOR AUTOMATIC BLOCK OF REACTOR TRIP ON TURBINE TRIP PERMISSIVE (P-9), A REACTOR TRIP OCCURRED AS A RESULT OF THE TURBINE TRIP. OPERATORS RESPONDED TO THE TRIP USING EMERGENCY OPERATING PROCEDURE 1-E-0, "REACTOR TRIP OR SAFETY INJECTION," AND STABILIZED THE REACTOR AT HOT STANDBY CONDITIONS (MODE 3) AT 547 DEGREES F AND 2235 PSIG. ALL REACTOR PROTECTION SYSTEMS OPERATED AS DESIGNED, AND NO ANOMALIES OCCURRED. THE SUDDEN PRESSURE RELAY WAS REPLACED, AND THE EQUIPMENT WAS RETURNED TO SERVICE.

[149] SEQUOYAH 2 DOCKET 50-328 LER 90-012
GAS ACCUMULATION IN CENTRIFUGAL CHARGING PUMP SUCTION PIPING AS RESULT OF GAS STRIPPING IN MINIFLOW LINE ORIFICES.
EVENT DATE: 082290 REPORT DATE: 102290 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 1 (PWR)

(NSIC 219870) THIS REPORT IS BEING SUBMITTED AS A VOLUNTARY LER. ON AUGUST 22, 1990, WITH BOTH UNITS IN MODE 1, UNIT 2 WAS DISCOVERED TO BE IN A POTENTIALLY UNANALYZED CONDITION WHEN A QUANTITY OF GAS WAS FOUND IN THE CENTRIFUGAL CHARGING PUMP (CCP) SUCTION HEADER THAT EXCEEDED THE MAXIMUM ALLOWED AS VOLUME. IT WAS SUBSEQUENTLY DETERMINED THAT HYDROGEN GAS HAD BEEN COMING OUT OF SOLUTION ON BOTH UNITS AND ACCUMULATING IN THE SUCTION PIPING AS A PROBABLE RESULT OF GAS STRIPPING BY THE CCP MINIFLOW ORIFICES. ON SEPTEMBER 6, 1990, UNIT 1 WAS ALSO DISCOVERED TO HAVE MORE THAN THE MAXIMUM ALLOWED GAS VOLUME IN PIPING THAT COULD BE ALIGNED TO THE CCP SUCTION. THE SUCTION PIPING IS BEING MONITORED AND PERIODICALLY VENTED. POSITIVE DISPLACEMENT CHARGING PUMPS ARE BEING OPERATED AS THE PREFERRED CHARGING SOURCE AT THIS TIME. RELATED NUCLEAR EXPERIENCE REVIEW ITEMS THAT WERE PREVIOUSLY CLOSED ARE BEING REVIEWED. LONG-TERM CORRECTIVE ACTIONS, WHICH ARE EXPECTED TO INVOLVE HARDWARE MODIFICATIONS, ARE STILL BEING DEVELOPED.

[150] SEQUOYAH 2 DOCKET 50-328 LER 90-013
LOSS OF RESIDUAL HEAT REMOVAL SHUTDOWN COOLING BECAUSE OF THE INADVERTENT DEENERGIZATION AND ACTUATION OF AUTOCLOSURE INTERLOCK CIRCUITRY.
EVENT DATE: 091190 REPORT DATE: 101190 NSSS: WE TYPE: PWR

(NSIC 219737) ON SEPTEMBER 11, 1990, AT 1333 EASTERN DAYLIGHT TIME WITH UNIT 2 IN MODE 5, RESIDUAL HEAT REMOVAL (RHR) SHUTDOWN COOLING WAS LOST WHEN SUCTION ISOLATION VALVE 2-FCV-74-1 WAS INADVERTENTLY CLOSED. 2-FCV-74-1 CLOSED WHEN POWER WAS REMOVED FROM AN RHR AUTOCLOSURE INTERLOCK (ACI) BISTABLE DURING THE EAGLE 21 MODIFICATION TO THE REACTOR PROTECTION SYSTEM (RPS). OPERATIONS RESPONDED IN ACCORDANCE WITH PLANT PROCEDURES, AND RHR COOLING WAS REINITIATED IN APPROXIMATELY FIVE MINUTES. RCS TEMPERATURE INCREASED 5 DEGREES FAHRENHEIT DURING THIS EVENT. THE INADVERTENT ACTUATION OF THE ACI CIRCUITRY IS ATTRIBUTED TO PERSONNEL OVERSIGHT DURING THE IMPACT EVALUATION PERFORMED IN SUPPORT OF THE EAGLE 21 MODIFICATION. THE EVENT HAS BEEN REVIEWED WITH THE PERSONNEL INVOLVED TO STRESS THE IMPORTANCE OF CORRECTLY IDENTIFYING THE COMPONENTS AFFECTED WHEN REMOVING RPS RACKS FROM SERVICE. MODIFICATIONS' MANAGEMENT HAS REVIEWED THIS

EVENT WITH MODIFICATIONS' ENGINEERS TO EMPHASIZE THAT DETAILED INFORMATION SUPPORTING A WORKPLAN MUST BE PRESENTED IN A CLEAR, CONCISE MANNER. THE SUCTION ISOLATION VALVES WILL BE TAGGED OPEN FOR THE REMAINDER OF THE REFUELING OUTAGE DURING EAGLE 21 MODIFICATION WORK WHEN RHR IS IN SERVICE. THE RHR ACI FUNCTION IS BEING REMOVED THIS OUTAGE IN ACCORDANCE WITH THE RECOMMENDATIONS OF GENERIC LETTER 88-17.

[151] SEQUOYAH 2 DOCKET 50-328 LER 90-014
CONTAINMENT VENTILATION ISOLATION EVENT RESULTING FROM AN INADVERTENT GROUND ON ONE OF THE HANDSWITCHES FOR THE RADIATION MONITOR ISOLATION VALVES.
EVENT DATE: 092190 REPORT DATE: 102290 NSSS: WE TYPE: PWR

(NSIC 219871) ON SEPTEMBER 21, 1990, AT APPROXIMATELY 0915 EASTERN DAYLIGHT TIME (EDT) WITH UNIT 2 IN MODE 6 FOR A REFUELING OUTAGE, A B TRAIN CONTAINMENT VENTILATION ISOLATION (CVI) OCCURRED ON UNIT 2. LIMITING CONDITION FOR OPERATIONS 3.3.3.1 WAS ENTERED AT 0915 EDT. THE EXACT CAUSE OF THE EVENT CANNOT BE DETERMINED. HOWEVER, THE CONCLUSION DRAWN FROM THE INVESTIGATIONS IS THAT THE ACTIVITY ASSOCIATED WITH MODIFICATION WORK ON THE UNIT 2 CONTROL ROOM PANELS CAUSED THE CVI. ELECTRICIANS WERE WORKING IN AND AROUND PANEL 2-M-6, WHICH HOUSES THE CVI CONTROL HANDSWITCHES. AN INADVERTENT GROUND OR SHORT INDUCED ON ONE OF THE SWITCH TERMINATIONS WOULD HAVE INITIATED THE CVI. AFTER THE CVI WAS RECOGNIZED, OPERATIONS' PERSONNEL APPROPRIATELY VERIFIED THAT THE CVI WAS INVALID, RETURNED THE B TRAIN RADIATION MONITOR ISOLATION VALVES TO NORMAL CONFIGURATION, RECOVERED FROM THE EVENT, AND EXITED LCO 3.3.3.1 AT 1035 EDT.

[152] SEQUOYAH 2 DOCKET 50-328 LER 90-015
FIRE WATCH FAILED TO FOLLOW PROCEDURES AND SURVEY AN AREA ON HIS ASSIGNED PATROL ROUTE WITH AN INOPERABLE FIRE BARRIER PENETRATION.
EVENT DATE: 092790 REPORT DATE: 102590 NSSS: WE TYPE: PWR

(NSIC 219872) ON SEPTEMBER 27, 1990, AT 1430 EASTERN DAYLIGHT TIME (EDT) WITH UNIT 2 IN MODE 6 FOR A REFUELING OUTAGE, IT WAS DISCOVERED THAT ONE OF THE FIRE WATCH PERSONNEL FAILED TO FOLLOW HIS ASSIGNED PATROL ROUTE AND SURVEY AN AREA WITH AN INOPERABLE FIRE BARRIER PENETRATION. BREACH PERMIT 10308 WAS ISSUED SEPTEMBER 22, 1990, FOR THE UNIT 2 PRESSURIZER HEATER TRANSFORMER ROOM ON ELEVATION 759 TO PERMIT CABLE PULLING ACTIVITIES. IN COMPLIANCE WITH THE LIMITING CONDITION FOR OPERATION (LCO) 3.7.12 ACTION STATEMENT, A ROVING FIRE WATCH WAS ESTABLISHED TO PERIODICALLY SURVEY THE ROOM FOR DETECTION AND PREVENTION OF FIRE. ONE OF FIVE FIRE WATCH PERSONNEL ASSIGNED THE PATROL ROUTE SIGNED THE LOG SHEET WITHOUT INSPECTING THE UNIT 2 PRESSURIZER HEATER TRANSFORMER ROOM. THE FIRE WATCH SIGN-OFF LOG WAS ORIGINALLY LOCATED IN THE UNIT 2 PRESSURIZER HEATER TRANSFORMER ROOM. HOWEVER, THE LOG HAD BEEN MOVED AND WAS LOCATED ON ELEVATION 734. THE CAUSE OF THIS EVENT IS THE FIRE WATCH PERSON FAILED TO FOLLOW PROCEDURES. THE CORRECTIVE ACTION WAS TO RELOCATE THE SIGN-OFF LOG OUTSIDE THE PRESSURIZER HEATER TRANSFORMER ROOM ON ELEVATION 759 AND APPROPRIATELY DISCIPLINE THE FIRE WATCH EMPLOYEE.

[153] SHEARON HARRIS 1 DOCKET 50-400 LER 90-020
TECHNICAL SPECIFICATION VIOLATION DUE TO MISSED RADWASTE SAMPLING REQUIREMENTS.
EVENT DATE: 091390 REPORT DATE: 101590 NSSS: WE TYPE: PWR
VENDOR: GEMS, INC.

(NSIC 219926) ON 9/12/90, PLANT WAS OPERATING IN MODE 1, AT 100% REACTOR POWER. SECONDARY WASTE SAMPLE TANK (SWST) WAS IN THE CONTINUOUS RELEASE MODE OF OPERATION. THE SWST RADIATION MONITOR (REM-21WS-3542) WAS IN NORMAL CONTINUOUS OPERATION AS REQUIRED BY TECH SPEC (TS) 3.3.3.10 TABLE 3.3-12 ITEM 1.A.3. AN INSTRUMENT & CONTROLS (I&C) TECHNICIAN (TECH) WORKING IN THE AREA OBSERVED THAT SWST RADIATION MONITOR WAS INDICATING FLOW THROUGH THE SAMPLE SYSTEM BUT THE SAMPLE PUMP WAS SWITCHED TO THE OFF POSITION. THERE WAS A DEFICIENCY TAG ON THE MONITOR FOR LOSS OF SAMPLE FLOW AND THE I&C TECH ASSUMED THAT THE MONITOR HAD BEEN DECLARED INOPERABLE. ON 9/13/90, THE I&C TECH DISCUSSED WHAT HE HAD OBSERVED WITH A RADIATION MONITORING SYSTEM (RMS) TECH. INVESTIGATING THE SITUATION, RMS TECH FOUND THE SAMPLE PUMP TURNED OFF BUT THE SAMPLE FLOW

INSTRUMENTATION WAS READING 3 GPM WHICH PREVENTED ANY LOSS OF FLOW TROUBLE ALARMS FROM ANNUNCIATING. HE CONTACTED THE MAIN CONTROL ROOM AND MONITOR REM-21WS-3542 WAS DECLARED INOPERABLE AT 08:30. 9/5/90, WAS THE LAST KNOWN TIME THE PUMP WAS STARTED AND THEREFORE, IT IS CONCLUDED THAT THE MONITOR WAS MALFUNCTIONING FROM 9/5/90, TO 9/13/90. COMPENSATORY ACTIONS AS REQUIRED BY TS TABLE 3.3-12 WERE INITIATED. RESIN CAUSED FLOW SWITCH TO STICK OPEN.

[154] SOUTH TEXAS 1 DOCKET 50-498 LER 90-021
UNPLANNED ENGINEERED SAFETY FEATURES ACTUATIONS AND REACTOR SHUTDOWN DUE TO AN INVERTER FAILURE.
EVENT DATE: 090990 REPORT DATE: 100990 NSSS: WE TYPE: PWR
VENDOR: ELGAR, CORP.

(NSIC 219954) ON SEPTEMBER 9, 1990, UNIT 1 WAS IN MODE 1 AT 100% POWER. AT 0822 HOURS, THE INVERTER WHICH FEEDS THE CLASS 1E AC VITAL DISTRIBUTION PANEL DP002 FAILED. THIS CAUSED ENGINEERED SAFETY FEATURES ACTUATIONS OF THE CONTROL ROOM, REACTOR CONTAINMENT BUILDING AND FUEL HANDLING BUILDING HVAC SYSTEMS DUE TO A LOSS OF POWER TO THEIR RESPECTIVE RADIATION MONITORS. TROUBLESHOOTING ACTIVITIES WERE EXTENSIVE AND COULD NOT BE COMPLETED WITHIN THE TWENTY-FOUR HOUR ACTION STATEMENT OF TECHNICAL SPECIFICATION 3.8.3.1. AT 0822 HOURS ON SEPTEMBER 10, 1990, A SHUTDOWN OF UNIT 1 COMMENCED. THE CAUSE OF THIS EVENT WAS FAILURE OF A POWER FILTER CAPACITOR WHICH INTERRUPTED POWER TO THE INVERTER CONTROLLER CARD AND BLEW TWO MAIN POWER FUSES. CORRECTIVE ACTIONS INCLUDE REPLACEMENT OF THE DC TO DC CONVERTER BOARD WHICH WILL BE RETURNED TO THE MANUFACTURER FOR ANALYSIS, TRENDING AND ANALYSIS OF MEASUREMENTS TAKEN FROM THE DC TO DC CONVERTER BOARDS OF SIMILAR INVERTERS, AND REVISION OF THE MAINTENANCE MANUAL.

[155] SOUTH TEXAS 1 DOCKET 50-498 LER 90-023
MANUAL REACTOR TRIP DUE TO FULL CLOSURE OF A FEEDWATER ISOLATION VALVE DURING PARTIAL STROKE TESTING.
EVENT DATE: 092990 REPORT DATE: 102990 NSSS: WE TYPE: PWR

(NSIC 219955) ON SEPTEMBER 29, 1990, UNIT 1 WAS IN MODE 1 AT 100% POWER. AT 0232 HOURS, FEEDWATER ISOLATION VALVE 1A FULLY CLOSED DURING AN "ALTERNATE" PARTIAL STROKE SURVEILLANCE TEST. THE RESULTANT LOSS OF FEEDWATER FLOW CAUSED A DECREASE IN STEAM GENERATOR LEVEL AND THE REACTOR WAS MANUALLY TRIPPED. DURING RECOVERY, A STEAM GENERATOR POWER OPERATED RELIEF VALVE (PORV) WAS MANUALLY OPENED TO APPROXIMATELY 30% WHILE INDICATING APPROXIMATELY 5% OPEN. AT 0345 HOURS, AN AUXILIARY FEEDWATER (AFW) ACTUATION OCCURRED ON LOW-LOW STEAM GENERATOR (SG) LEVEL. THE SG PORV WAS CLOSED AND THE PLANT WAS STABILIZED. THE FEEDWATER ISOLATION VALVE CLOSURE WAS CAUSED BY A TECHNICIAN INADVERTENTLY CONTACTING THE WRONG TERMINAL WITH A TEST JUMPER. THE CAUSE OF THE AFW ACTUATION DURING RECOVERY WAS FAILURE OF A REACTOR OPERATOR TO CONFIRM THE POSITION OF THE SG PORV BY MONITORING SG LEVEL AND PRESSURE INDICATIONS WHEN THE VALVE POSITION INDICATOR DISPLAYED SOME UNCERTAINTY. CORRECTIVE ACTIONS INCLUDE: ADDITION OF SPECIAL CONNECTORS ON THE TERMINALS IDENTIFIED IN THE "ALTERNATE" PARTIAL STROKE TEST PROCEDURE; THE USE OF SPECIAL INSULATED JUMPERS DURING PERFORMANCE OF THE STROKE TEST; PERFORMANCE OF THE PARTIAL STROKE TEST ON A MONTHLY BASIS VICE WEEKLY.

[156] SOUTH TEXAS 2 DOCKET 50-499 LER 90-013
A REACTOR TRIP CAUSED BY MANIPULATION OF THE INCORRECT REACTOR TRIP BREAKER TEST PUSHBUTTON.
EVENT DATE: 091790 REPORT DATE: 101790 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 219948) ON 9/17/90, UNIT 2 WAS IN 100% POWER, AT 0330 HOURS, A REACTOR TRIP OCCURRED DURING PERFORMANCE OF THE TRAIN S REACTOR TRIP BREAKER TRIP ACTUATING DEVICE OPERATIONAL TEST. A MAIN FEEDWATER ISOLATION OCCURRED ON LOW REACTOR COOLANT SYSTEM AVERAGE TEMPERATURE AND THE AUXILIARY FEEDWATER SYSTEM ACTUATED ON LOW STEAM GENERATOR LEVEL. CONTROL ROOM PERSONNEL RESPONDED IN ACCORDANCE WITH PROCEDURES AND STABLIZED THE PLANT. ALL SYSTEMS RESPONDED AS EXPECTED. THE CAUSE OF THE EVENT WAS FAILURE OF A NON-LICENSED OPERATOR TO SELF VERIFY THAT HE WAS IN POSITION TO OPEN THE CORRECT REACTOR TRIP BREAKER PANEL PRIOR TO

MANIPULATION OF THE AUTO SHUNT TRIP TEST PUSHBUTTON. THE NON-LICENSED OPERATOR WAS COUNSELED AND RECEIVED DISCIPLINARY ACTION, IN ADDITION A TRAINING MODULE EMPHASIZING THE IMPORTANCE OF ATTENTION TO DETAIL AND SELF-VERIFICATION WAS DEVELOPED AND PRESENTED TO EMPLOYEES ENGAGED IN OPERATIONS AND MAINTENANCE OF THE PLANT.

[157] SOUTH TEXAS 2 DOCKET 50-499 LER 90-014
INADVERTENT ENGINEERED SAFETY FEATURES ACTUATION DUE TO IMPROPER USE OF TEST EQUIPMENT.
EVENT DATE: 092690 REPORT DATE: 102490 NSSS: WE TYPE: PWR

(NSIC 219956) ON SEPTEMBER 26, 1990, UNIT 2 WAS IN MODE 1 AT 100% POWER. AT 1319 HOURS A CONTAINMENT VENTILATION ISOLATION (CVI) ACTUATION OCCURRED AS A RESULT OF IMPROPER USE OF TEST EQUIPMENT DURING A MONTHLY SURVEILLANCE ON THE SPENT FUEL POOL RADIATION MONITORS. THE HIGH RADIATION DETECTION RELAYS FOR CVI, FUEL HANDLING BUILDING (FHB) HVAC AND CONTROL ROOM (CR) HVAC ARE CONTAINED IN A SINGLE CONTROL ROOM PANEL. DURING THE SURVEILLANCE TEST MAINTENANCE TECHNICIAN INADVERTENTLY CONNECTED A TEST LEAD TO THE CURRENT SENSING JACK ON A DIGITAL MULTIMETER. A SECOND TECHNICIAN ATTEMPTED TO MEASURE THE VOLTAGES ACROSS THE COIL OF A FHB HVAC ACTUATION RELAY WITH THE MULTIMETER, EFFECTIVELY SHORTING THE POWER SUPPLY AND DEENERGIZING THE ACTUATION RELAYS FOR CVI. THE CAUSE OF THIS EVENT WAS FAILURE OF THE TECHNICIAN TO VERIFY THAT THE TEST EQUIPMENT WAS CONNECTED PROPERLY. CORRECTIVE ACTIONS INCLUDE THE DEVELOPMENT OF A TRAINING MODULE EMPHASIZING THE IMPORTANCE OF ATTENTION TO DETAIL AND SELF-VERIFICATION WHICH WAS PRESENTED TO EMPLOYEES ENGAGED IN MAINTENANCE AND OPERATION OF THE PLANT.

[158] ST. LUCIE 1 DOCKET 50-335 LER 90-009
ENGINEERED SAFETY FEATURE VALVE CLOSURE DUE TO A FAILED RELAY.
EVENT DATE: 092490 REPORT DATE: 102490 NSSS: CE TYPE: PWR
VENDOR: CONSOLIDATED CONTROLS CORP.

(NSIC 219901) ON 9/24/90, WITH UNIT 1 IN MODE 1 AT 100% POWER, SEVERAL NON-SAFETY RELATED SECONDARY PLANT HIGH TEMPERATURE ALARMS WERE RECEIVED IN A SHORT PERIOD OF TIME. INVESTIGATION REVEALED THAT THE INTAKE COOLING WATER ISOLATION VALVE, MV-21-2, WHICH SUPPLIES THE 'B' HEADER OF THE NON-ESSENTIAL TURBINE COOLING WATER SYSTEM WAS IN THE CLOSED POSITION WHEN IT SHOULD HAVE REMAINED OPEN. TROUBLESHOOTING OF THE CAUSE OF THE ISOLATION VALVE GOING CLOSED SHOWED THAT A COMPONENT ACTUATION RELAY IN THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) ACTUATION CIRCUITRY HAD FAILED. THIS RELAY ACTUATED THE SAFETY INJECTION ACTUATION SIGNAL (SIAS) FOR MV-21-2. WHEN IT FAILED IN ITS DEENERGIZED STATE MV-21-2 CLOSED TO ITS SIAS POSITION. THE VALVE WAS REOPENED AND THE RELAY REPLACED. THE UNIT REMAINED AT 100% POWER. THE ROOT CAUSE OF THE EVENT WAS A FAILED ESFAS RELAY. THE RELAY WAS REPLACED. IN CONSULTATION WITH THE VENDOR, IT HAS BEEN DETERMINED THAT THE FAILURE OF THIS TYPE OF RELAY IS A VERY RARE OCCURRENCE. THEREFORE, THE ONLY FUTURE CORRECTIVE ACTIONS PLANNED ARE TO CONTINUE TO TRACK THESE TYPE OF RELAYS TO DETERMINE IF THEIR FAILURE RATE IS INCREASING. THIS LER IS BEING SUBMITTED VOLUNTARILY. IT IS NOT REPORTABLE UNDER 10CFR50.73(A)(2)(IV) SINCE NO ESFAS ACTUATION LOGIC WAS COMPLETED AND NO ESFAS CHANNEL ACTUATED.

[159] SURRY 1 DOCKET 50-280 LER 90-010
LATE TYPE "B" AND TYPE "C" TESTING DUE TO 10CFR50 APPENDIX J INTERPRETATION.
EVENT DATE: 091390 REPORT DATE: 101290 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 219844) ON SEPTEMBER 13, 1990 WITH UNIT 1 OPERATING AT 70% POWER AND UNIT 2 OPERATING AT 100% POWER, IT WAS DETERMINED THAT THE METHODOLOGY IN USE FOR SPECIFYING 10CFR50 APPENDIX J, SECTION III.D TEST INTERVALS DIFFERED FROM NRC REQUIREMENTS IN TWO RESPECTS. FIRST, TYPE C VALVE TESTING DATES WERE NOT TRACKED ON A COMPONENT BASIS ONLY IN AGGREGATE. THE DATE OF THE LAST INDIVIDUAL COMPONENT TEST WAS RECORDED AS THE COMPLETION DATE FOR THE AGGREGATE TEST. SECOND, A +/- 25% EXTENSION ALLOWANCE HAD BEEN APPLIED TO CONTAINMENT AIR LOCK

TYPE B TESTING FREQUENCY. A REVIEW OF COMPLETED TEST DATA REVEALED THAT THE FIRST METHODOLOGY HAD CAUSED UNIT 1 TYPE C TESTING INTERVALS TO BE EXCEEDED FOR APPROXIMATELY TWO MONTHS IN EARLY 1990. ADDITIONALLY, SEVERAL INSTANCES PREDATING 1986 WERE IDENTIFIED WHERE UNIT 1 AND UNIT 2 CONTAINMENT AIR LOCK TYPE B TESTING WAS APPARENTLY NOT PERFORMED WITHIN THE ASSUMED +/- 25% GRACE PERIOD. THESE OCCURRENCES CONSTITUTED A VIOLATION OF TECHNICAL SPECIFICATION (TECH SPEC) 4.4.D WHICH REQUIRES TESTING TO BE PERFORMED AS SPECIFIED BY 10CFR50 APPENDIX J, SECTION III.D.

[160] SURRY 1 DOCKET 50-280 LER 90-013
FAILURE TO OBTAIN WASTE GAS DECAY TANK SAMPLE WITHIN TECHNICAL SPECIFICATION SAMPLING INTERVAL.
EVENT DATE: 092890 REPORT DATE: 102690 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 219845) ON SEPTEMBER 28, 1990, WITH UNIT 1 AT 57% POWER AND UNIT 2 AT 100% POWER, THE TECHNICAL SPECIFICATION SAMPLING INTERVAL FOR THE "A" WASTE GAS DECAY TANK WAS EXCEEDED. THIS EVENT IS REPORTABLE IN ACCORDANCE WITH 10CFR50.73(A)(2)(I)(B) SINCE IT IS A CONDITION NOT ALLOWED BY TECHNICAL SPECIFICATIONS. THE EVENT WAS CAUSED BY UNEXPECTED DELAYS ENCOUNTERED IN RADIOLOGICAL CONTROL ACTIVITIES. A SAMPLE WAS TAKEN, ANALYZED, AND VERIFIED TO BE WITHIN ACCEPTABLE LIMITS. MODIFIED HYDROGEN AND OXYGEN ANALYZERS ARE EXPECTED TO BE IN SERVICE BY DECEMBER 31, 1990, ELIMINATING THE REQUIREMENT FOR ROUTINE GRAB SAMPLING.

[161] SUSQUEHANNA 1 DOCKET 50-387 LER 90-020
"AS FOUND" MAIN STEAM LINE PENETRATION LEAKAGE RATE EXCEEDS TECHNICAL SPECIFICATION LIMITS.
EVENT DATE: 092090 REPORT DATE: 101990 NSSS: GE TYPE: BWR
VENDOR: ATWOOD & MORRILL CO., INC.

(NSIC 219919) AT 1230 HOURS ON SEPTEMBER 20, 1990 WITH UNIT 1 IN ITS FIFTH REFUELING AND INSPECTION OUTAGE, AN EVALUATION OF DATA FROM THE SCHEDULED MAIN STEAM LINE (MSL) PENETRATION LOCAL LEAK RATE TESTS (LLRTS) DETERMINED THAT THE "AS FOUND" LEAKAGE WAS IN EXCESS OF THE LIMIT OF TECHNICAL SPECIFICATION 3.6.1.2(C) FOR THE TOTAL MSL CONTAINMENT PENETRATION LEAKAGE OF 46.0 STANDARD CUBIC FEET PER HOUR (SCFH). THE "AS FOUND" MINIMUM PATH LEAKAGE RATE WAS 110.9 SCFH. THE C MSL CONTAINMENT PENETRATION ACCOUNTED FOR 99.6 SCFH OF THE TOTAL PENETRATION LEAKAGE. IN ACCORDANCE WITH 10CFR50.72(B)(2)(I) AN ENS 4 HOUR NON-EMERGENCY CALL WAS MADE AT 1355 HOURS. THE C MSL INBOARD AND OUTBOARD MAIN STEAM ISOLATION VALVES WERE REWORKED AND A POST MAINTENANCE LLRT WAS PERFORMED WITH SATISFACTORY RESULTS. THE INSPECTION ONLY REVEALED SLIGHT AREAS OF LIGHT OXIDATION AND MINOR SURFACE SCRATCHING, NEITHER OF WHICH ARE CONSIDERED CONTRIBUTORS TO THE CAUSE OF THE HIGH "AS FOUND" LEAKAGE RATE. TOTAL MSL CONTAINMENT PENETRATION LEAKAGE WAS REDUCED TO 20.9 SCFH. THERE WAS NO SAFETY SIGNIFICANCE OR RISK TO THE HEALTH AND SAFETY OF THE PUBLIC DUE TO THIS EVENT.

[162] SUSQUEHANNA 2 DOCKET 50-388 LER 90-009
HPCI & RCIC STEAM SUPPLY PRESSURE LOW RESPONSE TIME TESTING NOT PERFORMED WITHIN REQUIRED TIME INTERVAL.
EVENT DATE: 091990 REPORT DATE: 101990 NSSS: GE TYPE: BWR

(NSIC 219920) ON SEPTEMBER 19, 1990 WITH UNIT 2 IN CONDITION 1 AT 100% POWER, IT WAS DISCOVERED THAT TIME RESPONSE TESTING FOR HPCI AND RCIC STEAM SUPPLY LOW PRESSURE INSTRUMENTATION WAS NOT COMPLETED WITHIN THE REQUIRED 18 MONTH SURVEILLANCE INTERVAL SPECIFIED BY TECHNICAL SPECIFICATIONS. THE CAUSE OF THE EVENT WAS DUE TO AN INCORRECT DETERMINATION OF THE REQUIRED TESTING FREQUENCY OF EACH CHANNEL OF THE AFFECTED INSTRUMENTATION DURING A REVIEW IN 1989. THIS RESULTED IN PROCEDURAL REVISIONS WHICH ALLOWED TESTING AT A LESSER THAN PER ALLOWABLE FREQUENCY. THIS EVENT WAS DETERMINED REPORTABLE PER 10CFR50.73(A)(2)(I)(B) IN THAT NOT COMPLETING SURVEILLANCES WITHIN THE REQUIRED FREQUENCY REPRESENTED A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS. THERE WERE NO SAFETY CONSEQUENCES OR COMPROMISE TO THE PUBLIC

HEALTH OR SAFETY AS A RESULT OF THIS EVENT. AS AN IMMEDIATE CORRECTIVE ACTION, THE AFFECTED HPCI AND RCIC INSTRUMENTS WERE RESPONSE TESTED WITH SATISFACTORY RESULTS. APPLICABLE UNIT 1 AND 2 RESPONSE TIME TESTING SURVEILLANCE PROCEDURES WERE REVIEWED BY I&C ENGINEERING PERSONNEL FOR CORRECT TESTING FREQUENCIES. THE UNIT 1 AND 2 HPCI AND RCIC STEAM SUPPLY LOW PRESSURE INSTRUMENT SURVEILLANCES WERE THE ONLY PROCEDURES FOUND TO BE IMPACTED AND THESE WERE REVISED. THIS LER IS NOT EXPECTED TO BE UPDATED.

[163] SUSQUEHANNA 2 DOCKET 50-388 LER 90-010
CONDENSATE TRANSFER PUMP DISCHARGE LOW PRESSURE ALARM LOGIC SURVEILLANCE NOT COMPLETED WITHIN THE REQUIRED MONTHLY TIME INTERVAL.
EVENT DATE: 100290 REPORT DATE: 110190 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 1 (BWR)

(NSIC 219921) ON OCTOBER 2, 1990 WITH UNIT 1 IN CONDITION 5 AND DEFUELED AND UNIT 2 IN CONDITION 1 AT 100% POWER, IT WAS DISCOVERED THAT THE MONTHLY CHANNEL FUNCTIONAL TEST OF THE CONDENSATE TRANSFER DISCHARGE LOW PRESSURE ALARM LOGIC HAD NOT BEEN TESTED WITHIN THE REQUIRED SURVEILLANCE INTERVAL. THE TEST HAD INADVERTENTLY BEEN CATEGORIZED AS NOT BEING APPLICABLE FOR EXISTING PLANT CONDITIONS. THE TEST WAS COMMON TO BOTH UNITS AND IN ACTUALITY IT WAS APPLICABLE TO UNIT 2. THIS EVENT WAS DETERMINED REPORTABLE PER 10CFR50.73(A)(2)(I)(B) IN THAT NOT COMPLETING A SURVEILLANCE WITHIN THE REQUIRED FREQUENCY REPRESENTED A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS. THERE WERE NO SAFETY CONSEQUENCES OR COMPROMISE TO THE PUBLIC HEALTH OR SAFETY AS A RESULT OF THIS EVENT. THE SURVEILLANCE WAS IMMEDIATELY PERFORMED UPON DISCOVERY AND THE RESULTS WERE SATISFACTORY. A REVIEW WAS CONDUCTED WHICH ENSURED NO OTHER COMMON I&C PROCEDURES WERE INAPPROPRIATELY CATEGORIZED DUE TO PLANT CONDITION CHANGES ON UNIT 1. PROCEDURAL CHANGES WILL BE ACCOMPLISHED TO REQUIRE A SECOND LEVEL OF REVIEW BEFORE SURVEILLANCES CAN BE CATEGORIZED AS NOT REQUIRED DUE TO EXISTING PLANT CONDITIONS. THIS LER IS NOT EXPECTED TO BE UPDATED.

[164] TROJAN DOCKET 50-344 LER 90-039
CENTRIFUGAL CHARGING PUMP SEAL LEAKAGE EXCEEDS DESIGN LIMITS AND RESULTS IN BOTH TRAINS OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM BEING CONSIDERED INOPERABLE.
EVENT DATE: 092390 REPORT DATE: 102390 NSSS: WE TYPE: PWR
VENDOR: CRANE PACKING CO.

(NSIC 219878) ON SEPTEMBER 23, 1990, THE TROJAN NUCLEAR PLANT WAS OPERATING AT 100 PERCENT RATED THERMAL POWER. AT 2038, DURING THE PERFORMANCE OF WEEKLY OPERATING ROUTINES, THE OPERATING CENTRIFUGAL CHARGING PUMP WAS SWITCHED FROM THE "B" PUMP TO THE "A" PUMP. WHEN THE "A" PUMP WAS STARTED, IT DEVELOPED A SMALL LEAK PAST THE OUTBOARD SHAFT SEAL. AT 2101 THE "B" PUMP WAS RE-STARTED AND THE "A" PUMP WAS SHUT DOWN. DURING THE COAST DOWN OF THE "A" PUMP, THE RATE OF LEAKAGE INCREASED AND WAS ESTIMATED TO BE GREATER THAN ONE-HALF GALLON PER MINUTE. AT 2117, THE "A" CENTRIFUGAL CHARGING PUMP WAS ISOLATED, AND THE LEAK STOPPED. THE FINAL SAFETY ANALYSIS REPORT THYROID DOSES TO CONTROL ROOM PERSONNEL WERE CALCULATED ASSUMING A LEAKAGE RATE FOR RECIRCULATION WATER OUTSIDE CONTAINMENT OF 1772 CUBIC CENTIMETERS PER HOUR. SINCE THE LEAKAGE EXCEEDED THIS VALUE, THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM'S ABILITY TO MAINTAIN THYROID DOSES TO THE PLANT OPERATORS WITHIN ANALYZED LIMITS COULD HAVE BEEN EXCEEDED DURING A DESIGN BASIS ACCIDENT. THE SEAL FAILURE APPEARS TO BE ASSOCIATED WITH A BUILDUP OF BORIC ACID CRYSTALS ON THE SEAL COMPONENTS. IT IS SUSPECTED THAT THE BORIC ACID CRYSTALS INTERFERED WITH THE SEALING SURFACES. THE SEALS WERE REPLACED.

[165] TROJAN DOCKET 50-344 LER 90-040
INTERMITTENT FAILURE OF CONTAINMENT MONITORING SYSTEM PROCESS AND EFFLUENT RADIATION MONITOR IODINE DETECTOR CAUSES INITIATION OF A CONTAINMENT VENTILATION ISOLATION SIGNAL.
EVENT DATE: 092990 REPORT DATE: 102990 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 219879) ON SEPTEMBER 29, 1990, THE TROJAN NUCLEAR PLANT WAS IN MODE 3 (HOT STANDBY). AT 0941, THE COUNT RATE ON THE IODINE CHANNEL OF THE CONTAINMENT MONITORING SYSTEM PROCESS AND EFFLUENT RADIATION MONITOR (PERM-1B) RAPIDLY INCREASED FROM APPROXIMATELY 2,000 TO OVER 40,000 COUNTS PER MINUTE (CPM) RESULTING IN INITIATION OF A CONTAINMENT VENTILATION ISOLATION SIGNAL. NO RELEASES WERE IN PROGRESS AT THE TIME AND NO COMPONENT ACTUATION OCCURRED. TROUBLESHOOTING OF PERM-1B WAS PERFORMED; NO APPARENT PROBLEMS WERE FOUND. THE IODINE ABSORBER CARTRIDGE WAS ANALYZED. THE ANALYSIS INDICATED THAT THE LEVEL OF IODINE BUILT UP IN THE CARTRIDGE SHOULD HAVE RESULTED IN A COUNT RATE NEAR 40,000 CPM. A CONTAINMENT ATMOSPHERE GRAB SAMPLE SHOWED NO ABNORMAL LEVELS OF IODINE WHICH WOULD HAVE CAUSED THE RAPID INCREASE IN ACTIVITY INDICATED BY PERM-1B. THE MONITOR WAS RECALIBRATED, TESTED AND RETURNED TO SERVICE WITH A NEW IODINE ABSORBER CARTRIDGE. AT 0144, ON SEPTEMBER 30, CONTAINMENT VENTING WAS STARTED. THE COUNT RATE ON PERM-1B AT THE TIME THE PRESSURE REDUCTION WAS STARTED WAS APPROXIMATELY 400 COUNTS PER MINUTE. AT 0248, THE COUNT RATE INCREASED TO APPROXIMATELY 900 COUNTS PER MINUTE, CAUSING A CONTAINMENT VENTILATION ISOLATION. THESE EVENTS WERE CAUSED BY A RANDOM INTERMITTENT FAILURE OF THE PERM-1B DETECTOR.

[166] TURKEY POINT 3 DOCKET 50-250 LER 88-026 REV 04
UPDATE ON OUTSIDE FINAL SAFETY ANALYSIS REPORT DESIGN BASIS WITH REGARD TO HURRICANE FLOOD PROTECTION.
EVENT DATE: 110788 REPORT DATE: 101790 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)

(NSIC 219795) ON NOVEMBER 7, 1988, AT APPROXIMATELY 1900, WITH UNIT 3 IN COLD SHUTDOWN AND UNIT 4 DEFUELED, IT WAS DETERMINED THAT UNITS 3 AND 4 WERE OUTSIDE THEIR DESIGN BASIS WITH REGARD TO HURRICANE FLOOD PROTECTION. THE FINAL SAFETY ANALYSIS REPORT STATES, "THE UNIT IS DESIGNED FOR A HURRICANE TIDE TO AN ELEVATION OF +20 FEET WITH WAVE RUN-UP TO AN ELEVATION OF 22.5 FEET ON THE EAST SIDE OF THE UNIT." THE FOLLOWING CONDITIONS WERE IDENTIFIED BY A QUALITY ASSURANCE AUDITOR: 1) THE DIESEL OIL TRANSFER PUMPS ARE MOUNTED AT ELEVATION 19.0 FEET WITHOUT FLOOD PROTECTION. 2) A SECTION OF A FLOOD PROTECTION WALL BETWEEN THE EMERGENCY DIESEL GENERATOR BUILDING AND THE UNIT 3 SWITCHGEAR ENCLOSURE HAS BEEN TEMPORARILY REMOVED AS A PORTION OF A PLANT MODIFICATION. 3) THE STOP LOGS ON THE EAST FACE OF THE AUXILIARY BUILDING PROVIDE PROTECTION ONLY TO ELEVATION 20 FEET. SUBSEQUENT TO THESE CONCERNS, SEVERAL STOP LOGS ON THE NORTH, SOUTH, AND WEST SIDES OF THE PLANT WERE IDENTIFIED AS BEING DEFICIENT. ITEMS 1 AND 2 ABOVE WERE CAUSED BY DESIGN ERROR. ITEMS 3 AND 4 WERE CAUSED BY INADEQUATE PLANT DRAWINGS; NO PLANT DRAWINGS EXIST WHICH CLEARLY IDENTIFIES THE STOP LOG'S DESIGN CRITERIA. APPROPRIATE COMPENSATORY MEASURES HAVE BEEN TAKEN IN THE EVENT OF A HURRICANE WARNING, PRIOR TO COMPLETION OF MODIFICATIONS TO CORRECT THE ABOVE CONCERNS.

[167] TURKEY POINT 3 DOCKET 50-250 LER 90-018
MISSED FIRE PROTECTION SURVEILLANCE ON THE 4160V SWITCHGEAR ROOM LOUVER SPRAYS DUE TO PERSONNEL ERRORS.
EVENT DATE: 091790 REPORT DATE: 101790 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)

(NSIC 219819) ON 9/17/90, WITH UNITS 3 & 4 IN MODE 1 (POWER OPERATION) AT 100% POWER, THE QUALITY ASSURANCE GROUP (QA), RELEASED A REPORT INDICATING THAT UNITS 3 AND 4. 4160V SWITCHGEAR ROOM LOUVER SPRAYS HAD NOT BEEN INSPECTED IN ACCORDANCE WITH TECH SPEC (TS) 4.15.3.A.2. THIS TS REQUIRES A VISUAL INSPECTION OF THE LOUVER SPRAYS EVERY 18 MONTHS. THE LOUVER SPRAYS WERE LAST INSPECTED ON 1/18/85 IN ACCORDANCE WITH MAINTENANCE PROCEDURE (MP) 15537.2, "FIRE PROTECTION EQUIPMENT-ANNUAL MAINTENANCE." PRIOR TO THE 1985 SURVEILLANCE, ASSOCIATED LOUVERS HAD BEEN REMOVED AND REPLACED WITH 3-HOUR RATED FIRE BARRIERS. INSTALLATION OF BARRIERS RENDERED THE FUNCTION OF LOUVER SPRAYS OBSOLETE. ON 3/29/85, PROCEDURE MP 15537.2 WAS UPDATED BY THE FIRE PROTECTION GROUP TO DELETE THOSE STEPS THAT REQUIRED LOUVER SPRAYS INSPECTION. ON 8/20/85, FPL SUBMITTED PROPOSED LICENSE AMENDMENT (PLA)-025 TO THE NRC REQUESTING THAT THE OPERABILITY TS FOR THE 4160V SWITCHGEAR ROOM LOUVER SPRAYS BE DELETED. ON 3/6/87 NOTIFICATION WAS RECEIVED FROM THE NRC INDICATING THAT NO ACTION WOULD BE TAKEN

ON THE PLA REQUEST TO DELETE THE FP TS REQUIREMENT. HOWEVER, PRIOR TO SUBMITTING THE PLA TO THE NRC, SURVEILLANCE STEPS FOR THE LOUVER SPRAYS WERE DELETED IN PROCEDURE MP 15537.2. THE LOUVER SPRAYS WERE INSPECTED AND DETERMINED OPERABLE ON 9/27/90.

[168] TURKEY POINT 3 DOCKET 50-250 LER 90-019
CRITICAL HEAT TRACING CIRCUITS INOPERABLE DUE TO INADEQUATE WORK CONTROLS.
EVENT DATE: 100390 REPORT DATE: 101990 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)

(NSIC 219820) AT APPROXIMATELY 0843, ON OCTOBER 3, 1990, WITH UNITS 3 AND 4 IN MODE 1 AT 100 PERCENT POWER, AN ALARM WAS RECEIVED ON CONTROL ROOM ANNUCIATOR X-7/6, "HEAT TRACING TROUBLE." A NUCLEAR OPERATOR WAS DISPATCHED TO THE AUXILIARY BUILDING TO INVESTIGATE THE CAUSE FOR THE ALARM. AT 0855 AND AT 0920, RESPECTIVELY, CONTROL ROOM PERSONNEL WERE NOTIFIED THAT CRITICAL HEAT TRACING (CHT) CIRCUIT #6, UNIT 4 BORIC ACID TRANSFER PUMPS (BATPS) DISCHARGE TO THE BORIC ACID FILTER, AND CHT CIRCUIT #69, 3A BATP STRIP HEATER, WERE READING BELOW 145 DEGREES F. TECHNICAL SPECIFICATION (TS) 3.6 REQUIRES AT LEAST ONE CHANNEL OF CHT FOR THE FLOW PATH FROM THE BORIC ACID STORAGE TANKS (BASTS) TO BE OPERABLE DURING POWER OPERATION. WITH CHT CIRCUITS #6 AND #69 READING BELOW 145 DEGREES F, BOTH CHANNELS OF EACH CIRCUIT WERE CONSIDERED TO BE INOPERABLE. UNITS 3 AND 4 ENTERED TS 3.0.1 AT 0920 AND 0855, RESPECTIVELY. TEMPORARY LAGGING ON CHT CIRCUITS #6 AND #69 BECAME WET DURING AREA DECONTAMINATION OPERATIONS. THE COOLING EFFECT OF THE WET LAGGING DID NOT PERMIT THE CHT CIRCUITS TO MAINTAIN BORIC ACID PIPING ABOVE 145 DEGREES F. THE LAGGING BECAME WET DUE TO INADEQUATE WORK CONTROLS. CHT CIRCUIT #6 AND #69 TEMPERATURES WERE STABILIZED ABOVE 145 DEGREES F. UNITS 3 AND 4 EXITED TS 3.0.1 AT 1015 AND 1240, RESPECTIVELY. THIS EVENT HAS BEEN DISCUSSED WITH THE DECON SHIFT SUPERVISOR.

[169] VERMONT YANKEE DOCKET 50-271 LER 90-012 REV 01
UPDATE ON 1990 APPENDIX "J" TYPE "B" AND "C" FAILURE DUE TO SEAT LEAKAGE.
EVENT DATE: 090390 REPORT DATE: 100390 NSSS: GE TYPE: BWR
VENDOR: ALLIS CHALMERS
ANCHOR/DARLING VALVE CO.

(NSIC 219833) ON 9/3/90 AND 9/6/90 WHILE PERFORMING TYPE C LEAK RATE TESTING WITH THE PLANT SHUT-DOWN FOR THE 1990 REFUEL OUTAGE FEEDWATER CHECK VALVE FDW-96A (EIIS=SJ) AND PRIMARY CONTAINMENT ATMOSPHERIC CONTROL VALVE PCAC-6B (EIIS=BB) WERE FOUND TO HAVE SEAT LEAKAGE ABOVE THAT PERMITTED BY TECHNICAL SPECIFICATION 3.7.A.4. ON 9/3/90 THE SUM TOTAL LEAKAGE FOR TYPE B (PENETRATIONS) AND TYPE C (VALVES) EXCEEDED THAT ALLOWED BY 10 CFR 50 APPENDIX J. THE ASSIGNED PATHWAY LEAKAGE EXCEEDED THAT ALLOWED BY APPENDIX J AS A RESULT OF THE LEAKAGE THROUGH CHECK VALVE FDW-96A. APPENDIX J LIMITS THE TOTAL B AND C PENETRATION LEAKAGE TO 0.60 LA. VERMONT YANKEE HAS PERFORMED MAINTENANCE ON THE VALVES THAT WERE FOUND TO BE LEAKING TO DETERMINE THE CAUSE OF THE FAILURE. THE VALVES WERE REPAIRED AND RETESTED TO VERIFY THAT SEAT LEAKAGE IS WITHIN ALLOWABLE LIMITS PRIOR TO PLANT STARTUP FOLLOWING THE 1990 REFUELING OUTAGE.

[170] VERMONT YANKEE DOCKET 50-271 LER 90-013
RELIEF VALVE ACCUMULATOR FAILED DUE TO CHECK VALVE LEAKAGE.
EVENT DATE: 090590 REPORT DATE: 100590 NSSS: GE TYPE: BWR
VENDOR: NUPRO COMPANY

(NSIC 219703) ON 9/5/90, WITH THE PLANT SHUTDOWN FOR THE 1990 REFUELING OUTAGE, THE "C" MAIN STEAM (EIIS=SB) RELIEF VALVE ACCUMULATOR ASSEMBLY (REFER TO ATTACHED SKETCH #1) WAS FOUND TO HAVE CHECK VALVE SEAT LEAKAGE THAT EXCEEDED THE ACCEPTANCE CRITERIA SPECIFIED IN THE SURVEILLANCE PROCEDURE. THE OTHER THREE MAIN STEAM RELIEF VALVE ACCUMULATOR ASSEMBLIES PASSED THE INITIAL LEAK TESTS. THE ACCUMULATOR ASSEMBLY EXCEEDED THE ACCEPTANCE CRITERIA DUE TO A CHECK VALVE NOT PROPERLY SEATING. THE CHECK VALVE WAS DISASSEMBLED, THE INTERNALS WERE CLEANED AND A NEW RESILIENT SEAT WAS INSTALLED. ON 9/20/90, THE ASSEMBLY WAS SUCCESSFULLY RETESTED. THE ROOT CAUSE FOR THE LEAKAGE IS DIRT/CORROSION PRODUCTS ON THE CHECK VALVE SEAT THAT PRECLUDED PROPER SEATING OF THE VALVE. THE

DIRT/CORROSION PRODUCT IS FROM THE CONTAINMENT ATMOSPHERE SYSTEM (EIIS=LE) CARBON STEEL PIPING, THAT EXISTS FROM WHEN THE SYSTEM WAS SUPPLIED BY AIR. THE SYSTEM IS CURRENTLY SUPPLIED WITH CLEAN, DRY NITROGEN. THE CARBON STEEL PIPING HAS BEEN REPLACED WITH STAINLESS STEEL PIPING WITH FILTERS INSTALLED ON THE INLET TO EACH OF THE ACCUMULATOR ASSEMBLIES. THE NEW SUPPLY LINES WERE INSTALLED DURING THE 1990 REFUELING OUTAGE.

[171] VOGTLE 1 DOCKET 50-424 LER 90-019
PROGRAM INADEQUACY RESULTS IN NOT REPERFORMING A LEAK RATE TEST AFTER A VALVE INSPECTION.
EVENT DATE: 031490 REPORT DATE: 103190 NSSS: WE TYPE: PWR

(NSIC 219933) ON 9/19/90, THE PLANT ENGINEER ASSIGNED TO MONITOR LEAK RATE TESTING DURING THE ONGOING UNIT 2 REFUELING OUTAGE FOUND A CONDITION WHICH WOULD ALLOW INSERVICE INSPECTIONS (ISIS) OF SOME CONTAINMENT ISOLATION VALVES TO BE PERFORMED FOLLOWING THE COMPLETION OF ASME SECTION XI LOCAL LEAK RATE TESTING (LLRT) FOR VALVES. THE ENGINEER QUESTIONED IF THE INSPECTIONS INVALIDATED THE TEST RESULTS BECAUSE THE VALVE BONNETS AND SOME OF THE INTERNALS WERE BEING REMOVED. AN EXAMPLE WAS FOUND TO HAVE OCCURRED ON 3/14/90 DURING A UNIT 1 REFUELING OUTAGE WHEN VALVE 1-1206-U6-016, A CONTAINMENT ISOLATION CHECK VALVE, WAS RETURNED TO SERVICE FOLLOWING AN ISI INSPECTION WITHOUT BEING FUNCTIONALLY (LLRT) TESTED. FOLLOWING A NUMBER OF CONVERSATIONS WITH NRC PERSONNEL IT WAS DETERMINED ON 10-2-90 THAT A RETEST WOULD BE REQUIRED. A LLRT ON VALVE 1-1206-U6-016 WAS COMPLETED ON 10-4-90 WITH SATISFACTORY RESULTS. THE CAUSE OF THIS EVENT WAS A PROCEDURAL DEFICIENCY WHICH ALLOWED REMOVAL OF VALVE COMPONENTS, SUBSEQUENT TO THE COMPLETION OF AN LLRT, WITHOUT A RETEST. THE APPROPRIATE PROCEDURE HAS BEEN REVISED.

[172] VOGTLE 2 DOCKET 50-425 LER 90-013
REMOVING POWER TO PLANT VENT MONITOR LEADS TO TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 092690 REPORT DATE: 102390 NSSS: WE TYPE: PWR

(NSIC 219934) ON 9-26-90 AT 2300 CDT, MOTOR CONTROL CENTER (MCC) 2ABC WAS REMOVED FROM SERVICE FOR PREVENTIVE MAINTENANCE (PM). THIS MCC FEEDS POWER PANEL 2NYC2, WHICH IN TURN POWERS THE AIR PUMP FOR PLANT VENT RADIATION MONITOR 2RE-12444. ONCE THIS AIR FLOW WAS STOPPED, 2RE-12444 WAS NO LONGER CAPABLE OF MEASURING THE RADIATION LEVELS PRESENT IN THE AIR ESCAPING THROUGH THE PLANT VENT. ANOTHER PLANT VENT MONITOR, 2RE-12442, WAS IN SERVICE AT THE TIME WITH THE EXCEPTION OF CHANNEL A, THE PARTICULATE MONITOR. THEREFORE, THE REQUIREMENTS OF TECH SPEC (TS) SECTION 3.3.3.10 FOR PARTICULATE MONITORING WERE NO LONGER BEING MET BECAUSE NEITHER 2RE-12444 NOR 2RE-12442 WAS PERFORMING THE PARTICULATE MONITORING FUNCTION. ON 9/27/90 AT 1800 CDT, PERSONNEL PERFORMING THE DAILY CHECK OF 2RE-12444 FOUND THAT NO AIR FLOW WAS REGISTERING ACROSS THE MONITOR. AUXILIARY EQUIPMENT WAS INSTALLED TO INITIATE CONTINUOUS SAMPLING AS REQUIRED BY THE TS ACTION STATEMENT. THE CAUSE OF THIS EVENT WAS A DEFICIENCY IN THE ELECTRICAL DRAWING WHICH WAS REVIEWED WHILE PLANNING THE PM. THE DRAWING DID NOT SHOW THAT THE 2NYC2 POWER PANEL WOULD BE AFFECTED. A REQUEST FOR A DRAWING CHARGE TO CORRECT THE DEFICIENCY HAS BEEN INITIATED.

[173] WATERFORD 3 DOCKET 50-382 LER 90-003 REV 01
UPDATE ON REACTOR TRIP DUE TO GRID DISTURBANCE.
EVENT DATE: 032990 REPORT DATE: 101190 NSSS: CE TYPE: PWR

(NSIC 219916) AT 0730 HOURS ON MARCH 29, 1990, WITH WATERFORD STEAM ELECTRIC STATION UNIT 3 AT 99.9% POWER, A SEVERE TRANSIENT ON THE 230 KV POWER TRANSMISSION GRID AT TAFT, LOUISIANA, RESULTED IN A REACTOR TRIP. THE TRANSIENT WAS INITIATED WHEN AN OCCIDENTAL CHEMICAL COMPANY EMPLOYEE CAUSED A FAULT AT THE 230 KV SUBSTATION OWNED BY THE CHEMICAL COMPANY. THE REACTOR COOLANT PUMPS SLOWED TO LESS THAN 96.5 % OF NORMAL SPEED AS VOLTAGE DROPPED. THIS GENERATED A LOW MULTIPLIER IN THE CORE PROTECTION CALCULATORS AND A REACTOR TRIP OCCURRED DUE TO AN ANTICIPATORY DEPARTURE FROM NUCLEATE BOILING RATIO TRIP SIGNAL. THIS EVENT IS REPORTABLE AS AN AUTOMATIC REACTOR PROTECTION SYSTEM ACTUATION. THIS SUPPLEMENT IS SUBMITTED FOR POTENTIAL GENERIC INTEREST WITH RESPECT TO OFF-SITE

GRID CONDITIONS. AS A RESULT OF THIS SYSTEM TRANSIENT, A DEGRADED VOLTAGE CONDITION OCCURRED ON THE 4.16 KV 'B' SAFETY BUS WHEN A 'B' BUS TIE BREAKER OPENED AUTOMATICALLY WHICH DE-ENERGIZED THE 'B' SAFETY BUS. EMERGENCY DIESEL GENERATOR 'B' STARTED AND RE-ENERGIZED THE 'B' SAFETY BUS AS DESIGNED. THE ROOT CAUSE OF THIS EVENT WAS A GRID VOLTAGE DROP ON THE LOUISIANA POWER AND LIGHT COMPANY 230 KV SYSTEM CAUSED BY A FAULT AT OCCIDENTAL CHEMICAL COMPANY. BECAUSE PLANT PROTECTIVE FEATURES FUNCTIONED AS DESIGNED, THIS EVENT DID NOT THREATEN THE HEALTH OR SAFETY OF THE GENERAL PUBLIC OR PLANT PERSONNEL.

[174] WATERFORD 3 DOCKET 50-382 LER 90-014
INADVERTENT CONTROL ROOM OUTSIDE AIR ISOLATION DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 092290 REPORT DATE: 102290 NSSS: CE TYPE: PWR

(NSIC 219917) ON SEPTEMBER 22, 1990, WATERFORD STEAM ELECTRIC STATION UNIT 3 EXPERIENCED AN UNPLANNED ACTUATION OF THE ENGINEERED SAFETY FEATURE (ESF) PORTION OF THE CONTROL ROOM VENTILATION SYSTEM. THE ACTUATION WAS INITIATED BY A HIGH ALARM FROM ONE OF THE FOUR NORMAL CONTROL ROOM OUTSIDE AIR INTAKE (CROAI) RADIATION MONITORS, RESULTING IN A CONTROL ROOM ISOLATION AND AN AUTOMATIC START OF THE ASSOCIATED CONTROL ROOM EMERGENCY FILTRATION UNIT. ALL OTHER CROAI RADIATION MONITORS WERE INDICATING NORMAL RADIATION LEVELS AND AIR SAMPLES TAKEN IN THE AREA OF THE ALARMING RADIATION MONITOR SHOWED NO DETECTABLE ACTIVITY. THIS EVENT IS REPORTABLE AS AN UNPLANNED ESF ACTUATION. THE ROOT CAUSE OF THIS ACTUATION WAS AN INADEQUATE PROCEDURE WHICH CAUSED THE CROAI HIGH ALARM SETPOINT TO BE INADVERTENTLY SET AT A LOW VALUE. NORMAL BACKGROUND RADIATION FLUCTUATION EXCEEDED THE ALARM SETPOINT AND CAUSED THIS ACTUATION. THE CONTROL ROOM EMERGENCY FILTRATION SYSTEM FUNCTIONED AS DESIGNED AND THERE WAS NO ACTUAL RELEASE OF RADIOACTIVE MATERIAL; THEREFORE, THIS EVENT DID NOT RESULT IN AN INCREASED RISK TO THE HEALTH AND SAFETY OF THE PUBLIC OR PLANT PERSONNEL. THE ALARM SETPOINT HAS BEEN CORRECTED AND THE ALARM SETPOINT PROCEDURE WILL BE REVISED.

[175] WATERFORD 3 DOCKET 50-382 LER 90-015
INADVERTENT CONTROL ROOM VENTILATION ACTUATION DUE TO EQUIPMENT MALFUNCTION.
EVENT DATE: 092990 REPORT DATE: 102990 NSSS: CE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 219918) ON SEPTEMBER 29, 1990, WATERFORD STEAM ELECTRIC STATION UNIT 3 EXPERIENCED AN UNPLANNED ACTUATION OF THE ENGINEERED SAFETY FEATURE (ESF) PORTION OF THE CONTROL ROOM VENTILATION SYSTEM. THE ACTUATION WAS INITIATED BY A HIGH ALARM FROM THE CONTROL ROOM OUTSIDE AIR INTAKE (CROAI) RADIATION MONITOR, RESULTING IN A CONTROL ROOM ISOLATION AND AN AUTOMATIC START OF THE ASSOCIATED CONTROL ROOM EMERGENCY FILTRATION UNIT. ALL OTHER CROAI RADIATION MONITORS WERE INDICATING NORMAL RADIATION LEVELS AND AIR SAMPLES TAKEN IN THE AREA OF THE ALARMING RADIATION MONITOR SHOWED NO DETECTABLE ACTIVITY. THIS EVENT IS REPORTABLE AS AN UNPLANNED ESF ACTUATION. THE ROOT CAUSE OF THE ESF ACTUATION IS EQUIPMENT MALFUNCTION OF THE HIGH VOLTAGE POWER SUPPLY TO THE CROAI RADIATION MONITOR. THE HIGH VOLTAGE POWER SUPPLY WAS REPLACED AND THE CROAI RADIATION MONITOR RETURNED TO SERVICE. THE CONTROL ROOM EMERGENCY FILTRATION SYSTEM FUNCTIONED AS DESIGNED AND THERE WAS NO ACTUAL RELEASE OF RADIOACTIVE MATERIAL; THEREFORE, THIS EVENT DID NOT RESULT IN AN INCREASED RISK TO THE HEALTH AND SAFETY OF THE PUBLIC OR PLANT PERSONNEL.

[176] WPPSS 2 DOCKET 50-397 LER 90-019
PRESSURE SUPPRESSION PRESSURE LIMIT CURVE IN THE EOPS DID NOT AGREE WITH THE DESIGN CALCULATION DUE TO INADEQUATE COMMUNICATION/PROCEDURAL REVIEW AND VERIFICATION.
EVENT DATE: 091290 REPORT DATE: 101290 NSSS: GE TYPE: BWR

(NSIC 219922) ON SEPTEMBER 12, 1990 AT 1108 HOURS IT WAS DETERMINED THAT A DISCREPANCY WITH THE PRESSURE SUPPRESSION LIMIT (PSPL) CURVE IN THE EMERGENCY OPERATING PROCEDURES (EOPS) WAS REPORTABLE AS A CONDITION ALONE THAT COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF STRUCTURES OR SYSTEMS THAT ARE NEEDED TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT. THIS DISCREPANCY WAS

DISCOVERED BY AN NRC INSPECTOR DURING A RECENT NRC TEAM INSPECTION OF THE EOPS. THE PSPL CURVE IN THE EOP PROCEDURE DID NOT AGREE WITH THE DESIGN CALCULATION THAT FORMED THE BASIS FOR THE CURVE. THE PROCEDURAL CURVE SPECIFIED A WETWELL PRESSURE LIMIT WHICH WAS AS MUCH AS 2.0 PSI TOO HIGH BETWEEN 19.2 AND 37.0 FEET OF SUPPRESSION POOL WATER LEVEL. THIS PRESSURE IS A FUNCTION OF PRIMARY CONTAINMENT WATER LEVEL AND THE LIMIT IS USED DURING EMERGENCY SITUATIONS TO ENSURE THAT THE PRESSURE SUPPRESSION FUNCTION OF THE CONTAINMENT IS MAINTAINED WHILE THE RPV IS AT PRESSURE. THE PROCEDURAL DISCREPANCY COULD HAVE RESULTED IN INAPPROPRIATE OPERATOR ACTION IF A SITUATION HAS OCCURRED WHICH REQUIRED THAT THE GRAPH BE USED.

[177] WPPSS 2 DOCKET 50-397 LER 90-020
OVERHEATING OF CONTROL CABINET INTERNAL COMPONENTS COULD CAUSE DIESEL GENERATOR FAILURE DUE TO INADEQUATE DESIGN AND PROGRAMMATIC CONTROLS.
EVENT DATE: 091790 REPORT DATE: 101790 NSSS: GE TYPE: BWR
VENDOR: STEWART & STEVENSON SERVICES, INC.

(NSIC 219923) ON 9/17/90 IT WAS DETERMINED BY PLANT ENGINEERS THAT A SINGLE CONDITION, OVERHEATING IN THE DG EXCITATION CONTROL CABINETS, COULD PREVENT THE DIVISION 1 AND 2 DIESEL GENERATORS (DG1 AND DG2 FROM FULFILLING THEIR SAFETY FUNCTION. DURING WORST CASE DESIGN CONDITIONS, THE DG ELECTRICAL EQUIPMENT ROOM WOULD REACH ITS DESIGN TEMPERATURE OF 104F. IT WAS DETERMINED BY PLANT TESTS THAT OPERATION OF THE DGS, COINCIDENT WITH A 104F ROOM TEMPERATURE WOULD CAUSE THE TEMPERATURE IN THE DG EXCITATION CONTROL CABINET TO STABILIZE AT 136F. THE STATIC EXCITOR VOLTAGE REGULATOR (SEVR) MANUFACTURER'S RECOMMENDED CONTINUOUS OPERATION (OPERATION EXCEEDING 24 HOURS) TEMPERATURE IS 122F. THUS, FOR THESE CONDITIONS, OPERABILITY OF THE SEVR IS NOT ASSURED AND, SINCE THE SEVR CONTROLS THE VOLTAGE AND FIELD OF THE GENERATOR, THE OPERABILITY OF THE DG ALSO CANNOT BE GUARANTEED. IN 11/89, WHILE RE-EVALUATING THE DG ROOM HEATUP UNDER DESIGN BASIS ACCIDENT CONDITIONS, IT WAS DETERMINED BY SUPPLY SYSTEM ENGINEERS THAT THERE WAS INSUFFICIENT DATA TO DETERMINE THE HEATUP CHARACTERISTICS OF THE DG EXCITATION CONTROL CABINETS. TEMPERATURE TESTS WERE THEN INITIATED TO OBTAIN THE NECESSARY DATA TO COMPLETE THE RE-EVALUATION. CAUSES OF THIS EVENT ARE A MANUFACTURING ERROR AND AN ERROR IN THE SUPPLY SYSTEM START-UP PROGRAM.

[178] WPPSS 2 DOCKET 50-397 LER 90-021
MANUAL REACTOR SCRAM CAUSED BY TURBINE CONTROL OIL LEAK DUE TO PIPE NIPPLE FAILURE.
EVENT DATE: 092590 REPORT DATE: 101790 NSSS: GE TYPE: BWR

(NSIC 219924) AT 0557 HOURS ON SEPTEMBER 25, 1990 PLANT OPERATORS MANUALLY SCRAMMED THE REACTOR AFTER EXPERIENCING MAIN TURBINE HYDRAULIC CONTROL OIL PRESSURE PROBLEMS IN THE DIGITAL ELECTRO-HYDRAULIC (DEH) SYSTEM. THE DEH SYSTEM PROVIDES AUTOMATIC AND MANUAL CONTROL OF THE MAIN TURBINE GENERATOR BY POSITIONING THE TURBINE STEAM ADMISSION AND BYPASS VALVES. THE POSITIONING OF THESE VALVES ALSO AFFECTS REACTOR PRESSURE AND POWER. THE CAUSE OF THIS EVENT WAS TRACED TO A BROKEN PIPE NIPPLE IN THE AUTO STOP OIL HEADER PORTION OF THE TURBINE LUBE OIL (TO) SYSTEM. THIS RESULTED IN A DECREASE IN PRESSURE IN THE DEH SYSTEM. THE PRELIMINARY ROOT CAUSE OF THE EVENT WAS TRACED TO THE IMPROPER INSTALLATION OF A STABILIZER STRAP THAT MAY HAVE CONTRIBUTED TO THE FAILURE. THE INSTALLATION COMBINED WITH THE UNANTICIPATED VIBRATION IN THE AREA OF THE PIPE NIPPLE LED TO FATIGUE FAILURE. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO PLACE THE PLANT IN A COLD SHUTDOWN CONDITION. ALL SYSTEMS AND COMPONENTS PERFORMED AS DESIGNED. PLANT MAINTENANCE PERSONNEL REPLACED THE BROKEN NIPPLE AND REINSTALLED THE RELIEF VALVE AND THE STABILIZER STRAP. THE EVENT POSED NO THREAT TO THE HEALTH AND SAFETY OF EITHER THE PUBLIC OR PLANT PERSONNEL.

[179] WPPSS 2 DOCKET 50-397 LER 90-022
ENGINEERED SAFETY FEATURE ACTUATION OF CONTAINMENT INSTRUMENT AIR CAUSED BY DEPLETING THE NITROGEN CRYOGENIC TANK.
EVENT DATE: 093090 REPORT DATE: 102690 NSSS: GE TYPE: BWR

(NSIC 219925) AT 1600 HOURS ON SEPTEMBER 30, 1990, WHILE PLANT OPERATORS WERE

INERTING THE PRIMARY CONTAINMENT DURING A REACTOR STARTUP, A PRESSURE DECREASE OCCURRED IN THE CONTAINMENT INSTRUMENT AIR (CIA) SYSTEM. THE PRESSURE LOSS OCCURRED WHEN NITROGEN CRYOGENIC TANK CN-TK-1 (THE NORMAL SUPPLY USED FOR CONTAINMENT INERTING AND FOR CIA) RAS INADVERTENTLY DEPLETED OF LIQUID NITROGEN. THIS PRESSURE DECREASE CAUSED THE SAFETY RELATED PART OF THE CIA SYSTEM TO BE ISOLATED AND AUTOMATICALLY PLACED THE BACKUP BOTTLED NITROGEN SOURCE INTO SERVICE. THIS ACTION IS CONSIDERED AN ENGINEERED SAFETY FEATURE ACTUATION. FURTHER EVALUATION SHOWED THAT THE PRESSURE MAINTAINED BY THE BOTTLED NITROGEN SOURCE IN DIVISION II DID NOT MEET DESIGN REQUIREMENTS BECAUSE OF A MISADJUSTED PRESSURE REGULATOR. THE ROOT CAUSE OF THIS EVENT WAS LESS THAN ADEQUATE PROCEDURES THAT DID NOT CONTAIN ADEQUATE PRECAUTIONS FOR CONTAINMENT INERTING WITH LOW TANK LEVELS. THIS ALLOWED THE LEVEL IN THE TANK TO DROP TO THE POINT WHERE THERE WAS A LOSS OF PRESSURE IN THE CIA SYSTEM. THE ROOT CAUSE OF THE LOW NITROGEN PRESSURE IN DIVISION II WAS PROCEDURES NOT BEING FOLLOWED BY PLANT OPERATORS WITH A CONTRIBUTING CAUSE BEING LACK OF CLARITY IN THESE SAME PROCEDURES.

[180] YANKEE ROWE DOCKET 50-029 LER 90-007
INCORE INSTRUMENTATION SPIRE FAILURE RESULTS IN PRESSURE BOUNDARY LEAKAGE.
EVENT DATE: 092790 REPORT DATE: 102990 NSSS: WE TYPE: PWR

(NSIC 219806) ON SEPTEMBER 27, 1990 AND OCTOBER 13, 1990, FOLLOWING THE CORE 20/21 REFUELING OUTAGE AND CORRECTIVE MAINTENANCE, PRESSURE BOUNDARY LEAKAGE WAS DISCOVERED DURING PLANT HEAT-UP WITH BOTH INSTANCES BEING RELATED TO THE SAME CAUSE. IN BOTH CASES, UNUSUAL EVENTS (UE) WERE DECLARED SINCE TECHNICAL SPECIFICATION (TS) 3.4.5.2 DOES NOT ALLOW PRESSURE BOUNDARY LEAKAGE. NOTIFICATIONS WERE MADE TO THE COMMONWEALTH OF MASSACHUSETTS AND THE STATE OF VERMONT AND TO THE NRC. THE UES WERE TERMINATED FOLLOWING ORDERLY PLANT COOLDOWNS AND ENTRY INTO MODE 5, COLD SHUTDOWN. THE LEAKAGE IS SUSPECTED TO BE OCCURRING AT A PRESSURE BOUNDARY SEAL IN AN INCORE INSTRUMENTATION SPIRE LOCATED BELOW THE REACTOR HEAD. INITIAL CORRECTIVE ACTION INSTALLED MECHANICAL PLUGS IN TUBING AT THE PRIMARY SEAL LOCATION TO STOP SUSPECTED INSTRUMENTATION TUBE LEAKAGE. WHILE THIS ATTEMPT WAS UNSUCCESSFUL, ADDITIONAL CORRECTIVE ACTION IS IN PROGRESS TO INSTALL A SEAL CAP TO ENCAPSULATE ANY LEAKAGE WHILE PROVIDING AN EXTENSION OF THE PRIMARY SYSTEM PRESSURE BOUNDARY. THERE WAS NO ADVERSE EFFECT TO THE PUBLIC HEALTH AND SAFETY AS A RESULT OF EITHER EVENT.

[181] ZION 1 DOCKET 50-295 LER 90-020
INADVERTENT AUTOSTART OF 1A AUXILIARY FEEDWATER PUMP.
EVENT DATE: 091590 REPORT DATE: 101590 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 219889) ON 9/15/90 AT 1226, WHILE PERFORMING PERIODIC TEST (PT) 5B, "REACTOR PROTECTION LOGIC REACTOR AT NORMAL OPERATION CONDITIONS", SECTION 4 OF THE TEST FAILED. SECTION 4 TESTS THE TRAIN A 2/4 REACTOR COOLANT PUMP (RCP) BUS UNDERVOLTAGE REACTOR TRIP LOGIC BY DE-ENERGIZING LOGIC RELAYS TO SIMULATE RCP BUS UNDERVOLTAGE. TROUBLESHOOTING EFFORTS IDENTIFIED THAT THE BUS 144 UNDERVOLTAGE LOGIC RELAY, 27-2/XA, WAS DEFECTIVE. DURING THE REPLACEMENT OF THE RELAY, 1A AUXILIARY FEEDWATER (AFW) PUMP STARTED AUTOMATICALLY. DE-TERMINATION OF THE RELAY CONTACTS OPENED THE NORMALLY ENERGIZED 2/4 LOGIC CIRCUIT WHICH AUTOSTARTED THE AFW PUMP. FOLLOWING RELAY REPLACEMENT, THE "RCP BUS UV RX TRIP" ANNUNCIATOR DID NOT ACTUATE AS REQUIRED DURING TESTING DUE TO A WIRING ERROR ON RELAY 27-2/XA. THE WIRING DISCREPANCY WAS CORRECTED AND SECTION 4 OF THE TEST COMPLETED SATISFACTORILY. THERE WAS MINIMAL EFFECT ON PLANT SAFETY SINCE TRAIN B OF REACTOR PROTECTION WAS ALWAYS AVAILABLE.

[182] ZION 1 DOCKET 50-295 LER 90-021
GAS SAMPLE NOT ANALYZED DUE TO PERSONNEL ERROR.
EVENT DATE: 092290 REPORT DATE: 102290 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 2 (PWR)

(NSIC 219890) RADIATION MONITORS 1 AND 2 RT-PR12 (REACTOR VESSEL LEAK DETECTION RADIATION MONITORS) WERE OUT OF SERVICE, REQUIRING SHIFTLY AIR SAMPLES TO BE

TAKEN AND ANALYZED. ON 9/22/90 ON THE 3-11 PM SHIFT, THE SAMPLES WERE TAKEN, BUT INADVERTENTLY WERE NOT ANALYZED. THIS IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B). THE CAUSE OF THIS EVENT WAS PROCEDURAL DEFICIENCY. THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT, AS REDUNDANT MONITORING EQUIPMENT WAS OPERABLE. CORRECTIVE ACTIONS INCLUDE TAILGATE SESSIONS WITH BOTH RADIATION AND CHEMISTRY TECHNICIANS, AND A PROCEDURE CHANGE TO REQUIRE SIGNATURES WHEN ANY SAMPLE IS TRANSFERRED BETWEEN RADIATION PROTECTION DEPARTMENT AND CHEMISTRY DEPARTMENT.

[183] ZION 2 DOCKET 50-304 LER 90-011
2W MAIN TRANSFORMER FAILURE DUE TO UNKNOWN CAUSE.
EVENT DATE: 092290 REPORT DATE: 102290 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 219891) ON 9/22/90, AT APPROXIMATELY 2151, UNIT 2 WAS AT 39% AND RAMPING UP IN POWER WHEN THE 2W MAIN POWER TRANSFORMER EXPERIENCED AN INTERNAL FAULT. A MAJOR OIL FIRE RESULTED FROM THE 2W TRANSFORMER EXPLOSION. THE FIRE WAS SUCCESSFULLY CONTAINED BY THE DELUGE SYSTEM, AND WAS EXTINGUISHED BY THE FIRE BRIJADE. NEITHER THE 2E MAIN POWER, STATION AUXILIARY, OR UNIT AUXILIARY TRANSFORMERS WERE DAMAGED DURING THE EVENT. THE NSSS AND SECONDARY SYSTEMS SHUTDOWN RESPONSE WAS NORMAL. THE 2W MAIN POWER TRANSFORMER EXPERIENCED HEAVY MECHANICAL DAMAGE DUE TO THE PRESSURE SPIKE RESULTING FROM THE FAULT. ADDITIONALLY, THE ISOLATED PHASE BUS DUCT WAS SIGNIFICANTLY DAMAGED BY MECHANICAL DISPLACEMENT FROM THE TRANSFORMER EXPLOSION AND BUS DUCT FLASHOVERS THAT OCCURRED AFTER THE INITIAL FAULT. NO DAMAGE WAS FOUND TO THE MAIN GENERATOR. THE ROOT CAUSE IS STILL UNDER INVESTIGATION AND WILL REQUIRE DISASSEMBLY OF THE 2W TRANSFORMER. SUSPECTED PRIMARY CAUSAL FACTORS INCLUDE STATIC ELECTRICIFICATION, WINDING-TO-WINDING FAILURE, AND NITROGEN BREAKOUT FROM THE TRANSFORMER OIL. THE TRANSFORMER IS CURRENTLY BEING REPLACED WITH A SPARE UNIT, AND THE BUS DUCTS AND ASSOCIATED EQUIPMENT ARE BEING REPAIRED. ADDITIONAL PRECAUTIONARY MEASURES BEING PURSUED INCLUDE TRANSFORMER COOLER BANK SEQUENCING, ADDITION OF BUS DUCT LIGHTNING ARRESTERS, AND ADDITION OF A SEPARATE BUS DUCT GROUND CABLE.

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NRC FORM 335 U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** SEE INSTRUCTIONS ON THE REVERSE	1. REPORT NUMBER NUREG/CR-2000, Vol. 9, No. 12 ORNL/NSIC-200
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13. ABSTRACT (200 words or less)

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

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS		15. AVAILABILITY STATEMENT
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