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Licensee Event Report (LER) Compilation

For month of October 1990

Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

Available from

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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

Gary T. Mays
Nuclear Operations Analysis Center
Oak Ridge National Laboratory
P. O. Box 2009, Oak Ridge, TN 37831-8065
Telephone: 615/574-0391, FTS Number 624-0391

Questions regarding LER searches should be directed to

W. P. Poore (same address as above)
Telephone: 615/574-0325, FTS Number 624-0325

[1] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 90-006
PERSONNEL ERROR RESULTS IN INCORRECTLY EVALUATING THE ENVIRONMENTAL QUALIFICATION REQUIREMENTS OF FOUR MOTOR OPERATED VALVES WHICH WOULD BECOME SUBMERGED DURING AN ACCIDENT CONDITION (LOCA).
EVENT DATE: 072090 REPORT DATE: 081990 NSSS: BW TYPE: PWR

(NSIC 219349) ON JULY 20, 1990, WHILE PROCESSING CALCULATIONS RELATED TO A PROPOSED LEVEL INCREASE IN THE BORATED WATER STORAGE TANK, IT WAS DISCOVERED THAT FOUR MOTOR OPERATED VALVES AND THEIR ACTUATORS WERE LOCATED BELOW THE CURRENT ACCIDENT FLOOD ELEVATION IN THE REACTOR BUILDING (RB) WITHOUT HAVING BEEN PREVIOUSLY QUALIFIED FOR SUBMERGENCE. THREE OF THE FOUR VALVES RECEIVE AN ENGINEERED SAFEGUARDS SIGNAL AND PROVIDE RB ISOLATION DURING AN ACCIDENT CONDITION. THE THREE RB ISOLATION VALVES HAVE A SUFFICIENTLY SHORT OPERATING TIME TO ALLOW CLOSURE PRIOR TO BECOMING SUBMERGED WHILE THE FOURTH VALVE IS INITIALLY OPEN AND MAY REMAIN OPEN THROUGHOUT THE ACCIDENT DURATION. REMOTE POSITION INDICATION MAY BECOME UNAVAILABLE FOLLOWING VALVE SUBMERGENCE, HOWEVER, OTHER INDIRECT PARAMETER INDICATION IS AVAILABLE TO VERIFY VALVE POSITION AND SAFETY FUNCTION FOLLOWING SUBMERGENCE. THE ROOT CAUSE OF THIS CONDITION IS PERSONNEL ERROR DURING THE DEVELOPMENT OF ENVIRONMENTAL QUALIFICATION REQUIREMENTS WITH RESPECT TO EQUIPMENT OPERATING CONDITIONS. UPON DISCOVERY OF THIS CONDITION, VALVE OPERABILITY WAS ASSESSED, CONTROL ROOM OPERATORS WERE INFORMED OF THE CONDITION, AND A PLANT CHANGE ACTION WAS INITIATED TO REPLACE THE AFFECTED COMPONENTS WITH EQUIPMENT QUALIFIED FOR SUBMERGENCE.

[2] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 90-007
INSTALLATION OF LEAD SHIELDING ON THE 'B' DECAY HEAT PUMP JACKET HEAT EXCHANGER SERVICE WATER SUPPLY LINE RESULTS IN A POSTULATED FAILURE OF THE LINE DURING A SEISMIC EVENT.
EVENT DATE: 073190 REPORT DATE: 083090 NSSS: BW TYPE: PWR

(NSIC 219422) ON JULY 31, 1990, WHILE CONDUCTING SYSTEM WALKDOWNS TO OBTAIN PHYSICAL DIMENSIONS OF EXISTING PIPING FOR STRUCTURAL ANALYSIS INVOLVING THE INSTALLATION OF A FLUSH LINE, ANO DISCOVERED THAT A 1/2 INCH SERVICE WATER SUPPLY LINE FOR 'B' DECAY HEAT PUMP (P34B) JACKET HEAT EXCHANGER WAS NOT SEISMICALLY QUALIFIED DUE TO EXCESSIVE LOADING BY LEAD SHIELDING. THE LEAD SHIELDING WAS PARTIALLY SUPPORTED BY THE SERVICE WATER LINE AND WAS INITIALLY INSTALLED TO REDUCE RADIATION EXPOSURE DURING PREVIOUS SHUTDOWN MAINTENANCE ACTIVITIES. THE AFFECTED LINE WAS INSPECTED FOLLOWING SHIELDING REMOVAL AND NO DAMAGE WAS IDENTIFIED NOR WAS LEAKAGE DETECTED. LEAD SHIELDING INSTALLATION REQUEST FORMS DURING INITIAL INSTALLATION DID NOT REQUIRE ENGINEERING EVALUATING NOR DID THEY CONSIDER THAT SHIELDING INSTALLATION AND REMOVAL WAS DEPENDENT ON EQUIPMENT AVAILABILITY REQUIREMENTS IN REGARDS TO PLANT STATUS. CONSIDERING THOSE FACTORS, THE ROOT CAUSE OF THIS CONDITION IS INADEQUATE PROCEDURAL GUIDANCE FOR CONTROLLING EVALUATION AND INSTALLATION OF LEAD SHIELDING. UPON DISCOVERY OF THIS CONDITION, SYSTEM WALKDOWNS WERE COMPLETED TO IDENTIFY AND CORRECT SIMILAR CONDITIONS. ADDITIONALLY, A PROCEDURAL REVISION WAS INITIATED TO PROVIDE ENHANCED GUIDANCE FOR THE INSTALLATION OF LEAD SHIELDING.

[3] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 90-008
CONTROL ROOM WALL NOT SEISMICALLY QUALIFIED DUE TO PERSONNEL ERROR INVOLVING INSTALLATION ACCEPTANCE OF A BLOCKOUT CONSTRUCTION.
EVENT DATE: 080990 REPORT DATE: 091090 NSSS: BW TYPE: PWR

(NSIC 219472) ON 7/26/90, WHILE PERFORMING A FOLLOWUP INSPECTION OF FIRE BARRIER PENETRATION SEALS, ANO PERSONNEL IDENTIFIED AIR FLOW AROUND A CONDUIT PENETRATING A BLOCKOUT IN THE SOUTH WALL OF THE ANO-1 CONTROL ROOM WHICH HAD BEEN FILLED WITH CONCRETE BLOCK. IT WAS DETERMINED THAT THE BLOCK FILLED WALL WAS NOT SEISMICALLY QUALIFIED DUE TO THE ABSENCE OF STRUCTURAL REINFORCEMENT. THE POTENTIAL EFFECT OF A SEISMIC EVENT WAS EVALUATED AND IT WAS CONCLUDED THAT A VITAL ELECTRICAL DISTRIBUTION PANEL, MOUNTED ON THE BLOCKOUT, COULD BECOME INOPERABLE. FAILURE OF THE PANEL WOULD RESULT IN TRAIN 'A' OF ENGINEERED SAFEGUARDS (ES) AND EMERGENCY FEEDWATER SYSTEM BECOMING INOPERABLE. STRUCTURAL SUPPORT WAS ADDED TO SEISMICALLY SUPPORT THE BLOCK FILLED WALL. A REVIEW OF EXISTING DRAWINGS REVEALED THAT THE WALL WAS ANALYZED TO BE SEISMICALLY QUALIFIED BUT WAS NOT

CONSTRUCTED AS ANALYZED; THEREFORE, THE ROOT CAUSE OF THIS EVENT WAS AN ERROR ON THE PART OF THE LEAD INSTALLATION FIELD ENGINEER IN ACCEPTING THE DEFICIENT INSTALLATION DURING THE CONSTRUCTION OF THE BLOCKOUT FILLED WALL. WHILE REPAIRS WERE IN PROGRESS, THE CONTROL ROOM OPERATORS WERE PROVIDED INTERIM GUIDANCE REGARDING SEISMIC EVENT RESPONSE. ADDITIONALLY, A REVIEW OF ALL SEISMIC MASONARY WALLS IS IN PROGRESS TO DETERMINE IF ANY SIMILAR CONDITIONS EXIST AT ANO.

[4] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 90-009
INADVERTENT ACTUATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM CAUSED BY A FAULTY RADIATION MONITOR CABLE.
EVENT DATE: 081990 REPORT DATE: 091890 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 2 (PWR)

(NSIC 219541) ON AUGUST 19, 1990, AT APPROXIMATELY 1348, AN INADVERTENT ACTUATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) OCCURRED WHEN THE ELECTRICAL CABLE FOR THE ANO-2 CONTROL ROOM VENTILATION SYSTEM RADIATION MONITOR WAS INADVERTENTLY DISTURBED. DISTURBING THE CABLE RESULTED IN A FULL SCALE DEFLECTION OF THE RADIATION MONITOR WHICH INITIATED AN AUTOMATIC CREVS ACTUATION. THE CAUSE OF THIS EVENT WAS DETERMINED TO BE A FAULTY CABLE/CONNECTOR ASSEMBLY FOR THE RADIATION MONITOR WHICH RESULTED IN GROUNDING OF CABLE CONDUCTOR(S) TO THE SHIELD. THE FAULTY CABLE/CONNECTOR ASSEMBLY RESULTED FROM DEVIATION FROM DESIGN DOCUMENTS DURING INITIAL SYSTEM INSTALLATION. AFTER DISASSEMBLY AND REASSEMBLY OF THE CABLE CONNECTOR, THE ERRATIC RADIATION MONITOR READINGS COULD NOT BE REPRODUCED. INTERIM CORRECTIVE ACTION TO PREVENT RECURRENCE WAS TO PLACE A PLACARD IN THE VICINITY OF THE RADIATION MONITOR REQUIRING THE SHIFT SUPERVISOR TO BE NOTIFIED PRIOR TO ACCESSING THE AREA SO THAT APPROPRIATE ACTIONS ARE INITIATED TO PREVENT AN INADVERTENT CREVS ACTUATION. AN EVALUATION WAS INITIATED TO DETERMINE THE APPROPRIATE CORRECTIVE ACTIONS FOR THE CABLE/CONNECTOR ASSEMBLY. ADDITIONALLY, A REVIEW OF PROCESSES AND CONTROLS OVER FIELD INSTRUCTIONS IS BEING PERFORMED TO ENSURE THAT THEY ARE ADEQUATE TO PREVENT SIMILAR OCCURRENCES.

[5] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 89-025 REV 02
UPDATE ON PERSONNEL ERROR RESULTED IN NOT IDENTIFYING FIRE BARRIERS RENDERING FIRE BARRIER PENETRATIONS INOPERABLE DUE TO FAILURE TO PERFORM SURVEILLANCE REQUIREMENTS WITHIN APPROPRIATE TIME INTERVAL.
EVENT DATE: 122189 REPORT DATE: 090490 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 1 (PWR)

(NSIC 219479) ON 12/21/89, IT WAS IDENTIFIED THAT A PORTION OF A WALL LOCATED IN THE AUX. BLDG. BETWEEN THE 354 AND 360 FOOT ELEVATIONS HAD NOT BEEN PREVIOUSLY IDENTIFIED AS A TECH SPEC FIRE BARRIER. TWO PIPING PENETRATIONS LOCATED IN THE BARRIER HAD NOT BEEN SURVEILED AS REQUIRED BY TECH SPECS. A VISUAL INSPECTION OF ONE SIDE OF THE PENETRATIONS WAS PERFORMED WITH NO DISCREPANCIES IDENTIFIED. IT IS REASONABLE TO BELIEVE SINCE NO DISCREPANCIES WERE IDENTIFIED THAT THE PENETRATION FIRE BARRIERS HAD PREVIOUSLY BEEN FUNCTIONAL. ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. DURING THE INITIAL REVIEW OF PLANT AREAS THE DESIGN CONFIGURATION ON DIFFERENT ELEVATIONS WAS NOT CONSIDERED. A REVIEW OF THE DRAWINGS FOR ANO-1 AND ANO-2 HAS BEEN PERFORMED TO ENSURE ANY OTHER BARRIERS THAT EXIST ON DIFFERENT PLANT ELEVATIONS HAVE BEEN PROPERLY ACCOUNTED FOR AS TECH SPEC BARRIERS. SEVERAL BARRIERS HAVE BEEN IDENTIFIED WHICH ARE LOCATED ON DIFFERENT PLANT ELEVATIONS AND A WALKDOWN OF THESE BARRIERS HAS BEEN COMPLETED. A FIRE WATCH WAS POSTED WHEN NECESSARY AS REQUIRED BY TECH SPECS. A VISUAL INSPECTION OF THE FIRE BARRIER PENETRATIONS HAS BEEN PERFORMED AND THE REQUIRED PHYSICAL CHANGES WILL BE MADE.

[6] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 90-018
INADEQUATE INSERVICE INSPECTION PROGRAM RESULTED IN THE FAILURE TO PERFORM THE REQUIRED ASME CODE SECTION XI INSPECTIONS FOR SEVERAL CLASS 3 COMPONENT SUPPORTS.
EVENT DATE: 073190 REPORT DATE: 083090 NSSS: CE TYPE: PWR

(NSIC 219436) ON JULY 31, 1990, AS PART OF THE ARKANSAS NUCLEAR ONE (ANO) BUSINESS PLAN (ITEM D.5.N), A REVIEW OF THE FIRST 10-YEAR INTERVAL FOR INSERVICE INSPECTION (ISI) WAS PERFORMED. AS A RESULT OF THE REVIEW APPROXIMATELY NINETY

CLASS 3 COMPONENT SUPPORTS WERE IDENTIFIED TO NOT HAVE BEEN EXAMINED AS REQUIRED BY ASME CODE, SECTION XI DURING THE FIRST INSPECTION PERIOD AND ABOUT SEVEN WERE NOT INSPECTED IN THE SECOND PERIOD. DURING THE THIRD PERIOD ONE HUNDRED PERCENT OF THE COMPONENTS WERE INSPECTED. THE ROOT CAUSE OF THE FAILURE TO EXAMINE THE CLASS 3 SUPPORTS WAS AN INADEQUATE MAINTENANCE OF THE ISI PROGRAM AND LACK OF ISI INVOLVEMENT BY ANO PERSONNEL. THE FIRST PERIOD INSPECTIONS WERE PERFORMED BY CONTRACT PERSONNEL WITHOUT APPROPRIATE INVOLVEMENT BY ANO PERSONNEL. THE MISSED EXAMINATIONS IN THE SECOND PERIOD WERE NOT PROPERLY SCHEDULED DUE TO THE ISI PROGRAM NOT BEING ADEQUATELY MAINTAINED. A COMPUTER DATA BASE OF THE ISI PROGRAMS AND EXAMINATIONS HAS BEEN ESTABLISHED TO PROVIDE A MORE EFFICIENT MEANS OF TRACKING AND SCHEDULING INSPECTIONS. EACH SUPPORT HAS BEEN INSPECTED WITH NO SIGNIFICANT SAFETY CONCERNS IDENTIFIED. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B).

[7] ARNOLD DOCKET 50-331 LER 90-010
CONTROL ROD DRIVE INSERT/WITHDRAW LINE LEAKAGE AND INDICATIONS IN ONE BUNDLE DUE TO HIGH CYCLE FATIGUE.
EVENT DATE: 051990 REPORT DATE: 082390 NSSS: GE TYPE: BWR
VENDOR: CARGOCAIRE ENGINEERING CORP.

(NSIC 219357) ON MAY 19, 1990, DURING A VIDEO PROBE INSPECTION OF THE SMALL (APPROXIMATELY TWO INCH) AIR GAP BETWEEN THE STEEL DRYWELL SHELL AND THE OUTER CONCRETE RADIOLOGICAL SHIELD, A CONTROL ROD DRIVE (CRD) WITHDRAW LINE WAS OBSERVED TO HAVE A SMALL (PINHOLE) LEAK. SUBSEQUENT INVESTIGATION USING ULTRASONIC TESTING AND METALLURGICAL ANALYSIS FOUND OTHER INDICATIONS (CRACKS) ON THE INSERT AND WITHDRAW LINES AT THE SOUTHWEST CRD BUNDLE PENETRATION INTO THE DRYWELL. THESE INDICATIONS ARE CIRCUMFERENTIAL, FROM THE OUTSIDE DIAMETER IN, AND AT THE TOE OF THE DRYWELL SHELL TO CRD PIPE FILLET WELD. THEY OCCUR BOTH INSIDE AND OUTSIDE THE DRYWELL, WITH THE MOST EXTENSIVE BEING OUTSIDE. A SUBSEQUENT SURVEY OF THE THREE OTHER CRD INSERT/WITHDRAW BUNDLE PENETRATIONS FOUND NO EVIDENCE OF SIMILAR INDICATIONS. AT THIS TIME, THE ROOT CAUSE OF THE INDICATIONS AT THE SOUTHWEST CRD PENETRATION REMAINS UNKNOWN. BASED ON METALLURGICAL EXAMINATIONS, AN UNKNOWN FORCING FUNCTION CAUSING HIGH CYCLE FATIGUE IS SUSPECTED. IT HAS NOT BEEN DETERMINED IF THE SOUTHWEST CRD BUNDLE WAS MORE SUSCEPTIBLE TO SUCH A FUNCTION OR IF THE FUNCTION IS UNIQUE TO THAT BUNDLE. INVESTIGATION IS CONTINUING WITH INSTRUMENTATION OF THE CRD LINES. THE SOUTHWEST CRD LINES ARE BEING REPLACED USING A NEW DRYWELL PENETRATION WELD DESIGN. THIS REPORT IS BEING FILED FOR INFORMATION ONLY.

[8] ARNOLD DOCKET 50-331 LER 90-009
INADEQUATE FIRE BARRIER SEAL.
EVENT DATE: 073090 REPORT DATE: 082990 NSSS: GE TYPE: BWR

(NSIC 219428) WITH THE PLANT SHUTDOWN IN A REFUELING OUTAGE, WORK WAS IN PROGRESS ON A DESIGN CHANGE PACKAGE INVOLVING POSITIVE PRESSURE MODIFICATIONS TO THE CONTROL BUILDING HEATING, VENTILATION, AND AIR CONDITIONING SYSTEM. DURING TESTING, PERSONNEL IDENTIFIED AIR LEAKAGE FORM THE CONTROL ROOM TO THE CABLE SPREADING ROOM AND CONDUIT CHASE AROUND THE FIRE BARRIER SEAL IN THE "SEISMIC GAP" BETWEEN THE BUILDINGS. INVESTIGATION DETERMINED THE BARRIER SEAL DID NOT MEET THE REQUIREMENTS OF 10 CFR 50 APPENDIX R. THE SEAL WAS REPLACED IN ACCORDANCE WITH 10 CFR 50 APPENDIX R. THIS EVENT IS BEING REPORTED PURSUANT TO 10 CFR 50.73 (A)(2)(I).

[9] BEAVER VALLEY 1 DOCKET 50-334 LER 90-012
PLANT SHUTDOWN DUE TO FAILURE OF THE NO. 2 BATTERY CHARGER.
EVENT DATE: 080890 REPORT DATE: 083190 NSSS: WE TYPE: PWR
VENDOR: LA MARCHE MFG CO.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 219430) ON 8/1/90 AT 1719 HOURS THE CONTROL ROOM RECEIVED THE #2 BATTERY CHARGER FAILURE ALARM. AN INVESTIGATION DETERMINED THAT THE "A" PHASE LEAD OF THE 480 VOLT INPUT BREAKER HAD FAILED, CAUSING THE BREAKER TO BE INOPERABLE. SURVEILLANCE TESTING OF THE #2 BATTERY WAS IMMEDIATELY INITIATED AS REQUIRED BY

TECHNICAL SPECIFICATIONS. AT 1805 HOURS, THE #2 BATTERY WAS DECLARED INOPERABLE DUE TO THE VOLTAGE BEING LESS THAN REQUIRED. AT 2005 HOURS A CONTROLLED PLANT SHUTDOWN WAS INITIATED PER TECHNICAL SPECIFICATIONS DUE TO AN INOPERABLE BATTERY. THE PLANT ENTERED MODE 3 AT 0159 HOURS ON 8/2/90. WHILE INVESTIGATING THE BREAKER PROBLEM, A VOLTAGE CONTROL PROBLEM WAS ALSO DISCOVERED WITH THE #2 BATTERY CHARGER. THE VOLTAGE CONTROL PROBLEM WAS DUE TO A FAILED COLLECTOR RESISTOR ON THE CHARGER OUTPUT. THE BATTERY CHARGER RESISTOR AND THE INPUT BREAKER WERE REPLACED. THE BATTERY WAS THEN PLACED ON CHARGE. THE #2 BATTERY WAS DECLARED OPERABLE AT 1330 HOURS AND THE BATTERY CHARGER WAS DECLARED OPERABLE AT 2100 HOURS ON 8/2/90. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. ALL TECHNICAL SPECIFICATION ACTION STATEMENTS WERE COMPLIED WITH. ADDITIONALLY, ONE TRAIN OF 120 VAC VITAL POWER WAS FULLY OPERABLE THROUGHOUT THIS EVENT. THE REDUNDANCY OF THE TWO AVAILABLE TRAINS IS DESCRIBED IN UFSAR SECTION 8.5.4, "120 VOLT A-C VITAL BUS SYSTEM."

[10] BEAVER VALLEY 1 DOCKET 50-334 LER 90-013
CONTROL ROOM EMERGENCY BREATHING AIR PRESSURIZATION SYSTEM ACTUATION.
EVENT DATE: 081590 REPORT DATE: 090790 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BEAVER VALLEY 2 (PWR)

(NSIC 219475) ON 8/15/90 AT 1521 HOURS, WITH THE UNIT IN POWER OPERATION AT 100% REACTOR POWER, CONTROL ROOM PERSONNEL RECEIVED ALARMS ASSOCIATED WITH THE RADIATION MONITORS. THESE ALARMS WERE: RADIATION MONITOR POWER SUPPLY FAILURE, RADIATION HIGH AND RADIATION HIGH-HIGH. THE CONTROL ROOM EMERGENCY BREATHING AIR PRESSURIZATION SYSTEM (CREBAPS), TRAIN A, ACTUATED, PRESSURIZING THE COMMON CONTROL ROOM ENVELOPE. THE TRAIN A UNIT 1 CONTROL ROOM RADIATION MONITOR, RM-RM-218A, WAS FOUND TO INDICATE A HIGH, HIGH-HIGH AND A FAILURE ALARM. THE MONITOR ALARM COULD NOT BE ACKNOWLEDGED. THE OPERATORS VERIFIED THE SIGNAL WAS ERRONEOUS BY COMPARISONS WITH THE REDUNDANT RADIATION MONITORS IN BOTH CONTROL ROOMS, AND ISOLATED CREBAPS AT 1523 HOURS. ISOLATING THE AIR SUPPLY PLACES BOTH UNITS 1 AND 2 INTO TECH SPEC 3.0.3. INSTRUMENT AND CONTROL PERSONNEL BEGAN TROUBLESHOOTING RM-RM-218A BY DISABLING THE MONITOR'S SIGNAL TO CREBAPS. THIS ALLOWED RESETTING OF THE INITIATION SIGNAL AND RESTORATION OF THE AIR SUPPLY AT 1600 HOURS, EXITING TS 3.0.3. RM-RM-218A WAS RETURNED TO SERVICE AT 1355 HOURS ON 8/23/90, FOLLOWING OF MAINTENANCE AND SURVEILLANCE TESTING ACTIVITIES. REDUNDANT CONTROL ROOM RADIATION MONITORS REMAINED OPERABLE THROUGHOUT THIS EVENT TO PROVIDE CONTROL ROOM ISOLATION AND PRESSURIZATION.

[11] BEAVER VALLEY 2 DOCKET 50-412 LER 90-004 REV 01
UPDATE ON INADVERTENT ESF ACTUATION DURING SAFEGUARDS PROTECTION SYSTEM TESTING.
EVENT DATE: 042390 REPORT DATE: 082390 NSSS: WE TYPE: PWR

(NSIC 219370) ON 4/23/90, WHILE THE PLANT WAS AT 87% POWER, TRAIN B SAFEGUARDS PROTECTION SYSTEM SAFETY INJECTION SYSTEM (SIS, GO TESTING WAS IN PROGRESS. THIS TEST VERIFIES THE ACTUATION OF SEVERAL SIS COMPONENTS IN RESPONSE TO A SAFETY INJECTION SIGNAL. IN ORDER TO CONDUCT THIS TEST DURING POWER OPERATION, THE ACTUATION OF CERTAIN AFFECTED SIS COMPONENTS ARE TEMPORARILY BLOCKED, PER PROCEDURE. DURING THIS TEST, AN OPERATOR INADVERTENTLY UNBLOCKED THE OPERATION OF CHARGING CONTAINMENT ISOLATION VALVE [2CHS-MOV310] BEFORE THE ACTUATION SIGNAL WAS RESET. THIS CAUSED THE VALVE TO CLOSE, WITH SUBSEQUENT LOSS OF NORMAL CHARGING FLOW TO THE REACTOR COOLANT SYSTEM. OPERATORS RESET THE SIGNAL AND RESTORED CHARGING. A HUMAN PERFORMANCE ENHANCEMENT SYSTEM ANALYSIS DETERMINED THAT THE OPERATOR INCORRECTLY PERFORMED SYSTEM REALIGNMENT AS REQUIRED BY TEST PROCEDURES. THIS WAS A RESULT OF THE OPERATOR FAILING TO FULLY ASSESS THE OVERALL SYSTEM CONFIGURATION. ALL OPERATIONS PERSONNEL WILL BE TRAINED ON THE PERSONNEL ERROR INVOLVED IN THIS EVENT. THE TESTING PROCEDURE WILL BE REVISED TO SIMPLIFY THE DOUBLE VERIFICATION PROCESS. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. THIS EVENT IS BOUNDED BY THE ANALYSIS OF BEAVER VALLEY UNIT 2 UFSAR SECTION 15.5.1, "INADVERTENT OPERATION OF EMERGENCY CORE COOLING SYSTEM DURING POWER OPERATION."

[12] BEAVER VALLEY 2 DOCKET 50-412 LER 90-005 REV 01
UPDATE ON INADVERTENT ESF ACTUATION DURING QUENCH SPRAY FLOW SWITCH CALIBRATION.
EVENT DATE: 042390 REPORT DATE: 082390 NSSS: WE TYPE: PWR
VENDOR: TARGET ROCK CORP.

(NSIC 219371) ON 4/24/90, OPERATORS DISCOVERED THAT THREE VALVES IN THE QUENCH SPRAY CHEMICAL ADDITION SYSTEM HAD SPURIOUSLY CLOSED. REVIEW OF COMPUTER LOGS FOUND THAT ONE HAD CLOSED EARLIER THAT DAY, WHILE THE OTHER TWO HAD CLOSED ON 4/23/90. INVESTIGATION FOUND THAT THE VALVES HAD CLOSED DUE TO AN ERROR IN FOUR NEWLY REVISED CALIBRATION PROCEDURES FOR THE QUENCH SPRAY FLOW SWITCHES. EACH PROCEDURE HAD TECHNICIANS TEMPORARILY DE-ENERGIZE THE LIMIT SWITCHES ON THE VALVE ASSOCIATED WITH THE FLOW SWITCH. WITH THE OPEN LIMIT SWITCH DE-ENERGIZED, THE VALVES AUTOMATICALLY CLOSED. TWO OF THE PROCEDURES WERE PERFORMED ON 4/23/90 AND ONE WAS PERFORMED ON 4/24/90. THE FOURTH TEST HAD NOT BEEN PERFORMED SINCE ITS REVISION. OPERATIONS OPENED THE AFFECTED VALVES. THE FLOW SWITCH CALIBRATION PROCEDURES HAVE BEEN REVISED. A ROOT CAUSE ANALYSIS OF THIS EVENT WAS PERFORMED AND DETERMINED THAT THE PROCEDURE HAD BEEN INADEQUATELY REVIEWED PRIOR TO BEING ISSUED. NEW GUIDELINES FOR PROCEDURE REVIEW/WALK DOWN HAVE BEEN ISSUED FOR USE DURING FUTURE REVISIONS OF INSTRUMENTATION CALIBRATION PROCEDURES. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. IN THE EVENT OF A QUENCH SPRAY ACTUATION, THE VALVES WERE OPERABLE AND WOULD HAVE BEEN AUTOMATICALLY POSITIONED CORRECTLY IN RESPONSE TO THEIR ESF SIGNALS.

[13] BIG ROCK POINT DOCKET 50-155 LER 90-003
DEFECT IN REACTOR COOLANT BOUNDARY IN REACTOR CLEAN-UP SYSTEM.
EVENT DATE: 072290 REPORT DATE: 082190 NSSS: GE TYPE: BWR

(NSIC 219336) ON JULY 18, 1990, PLANT SHUTDOWN WAS INITIATED DUE TO PROBLEMS WITH THE TURBINE GLAND SEAL SYSTEM. FOLLOWING SHUTDOWN, A SHIFT SUPERVISOR IDENTIFIED A "PINHOLE" LEAK NEAR THE WELD ON THE THREE INCH RETURN LINE FROM THE CLEAN-UP SYSTEM. AT THAT TIME THE PLANT WAS TAKEN TO COLD SHUTDOWN. ON JULY 22 NON-DESTRUCTIVE EXAMINATIONS REVEALED A POTENTIAL INTERGRANULAR DEFECT. EXAMINATIONS ON TWO ADDITIONAL SIMILAR WELDS DID NOT REVEAL ANY INDICATIONS. PRELIMINARY INVESTIGATION HAS DETERMINED THE PROBABLE CAUSE OF DEFECT IS RELATED TO THE CONFIGURATION OF THE THERMAL SLEEVE FILET WELD BEING CLOSE TO CLEAN-UP PIPE BUTT WELD. ON JULY 24, THE SECTION OF PIPE WITH THE DEFECT AND THERMAL SLEEVE WAS REMOVED AND NEW SECTION WAS WELDED INTO PLACE. TO PREVENT RECURRENCE THE THERMAL SLEEVE WAS RELOCATED FURTHER AWAY FROM THE BUTT WELD. ANOTHER THERMAL SLEEVE IN THE SYSTEM WAS ALSO CUT OUT AND RELOCATED TO MINIMIZE DEFECT DEVELOPMENT.

[14] BIG ROCK POINT DOCKET 50-155 LER 90-004
INOPERABLE DIESEL GENERATOR DUE TO OVERCURRENT LOGIC WIRING ERROR.
EVENT DATE: 072890 REPORT DATE: 082490 NSSS: GE TYPE: BWR

(NSIC 219337) ON JULY 27, 1990, DURING A NRC MAINTENANCE TEAM INSPECTION, A DISCREPANCY WAS IDENTIFIED IN THE CONTROL PANEL WIRING FOR THE EMERGENCY DIESEL GENERATOR (EDG). FURTHER INVESTIGATION DETERMINED THAT THE ERROR WAS IN THE OVERCURRENT TRIP LOGIC. PER THE FHSR, THE EDG OVERCURRENT RELAYS ARE REQUIRED TO PROVIDE COINCIDENT LOGIC TRIP UTILIZING TWO INDEPENDENT SENSORS. IN CONTRAST, ONE PHASE OF THE OVERCURRENT LOGIC WAS MISWIRED RESULTING IN SINGLE SENSOR LOGIC. ON JULY 28, 1990 THE IDENTIFIED ERROR WAS CORRECTED AND LOGIC CIRCUIT TESTING VERIFIED THE CIRCUIT WAS ACCEPTABLE. CAUSE OF THE ERROR WAS DETERMINED TO BE A PERSONNEL ERROR DURING INSTALLATION OF THE MODIFICATION IN 1977. TESTING PERFORMED IN 1977 WAS ALSO INADEQUATE AND FAILED TO IDENTIFY THE ERROR. THE PLANT WAS IN COLD SHUTDOWN WHEN THE ERROR WAS IDENTIFIED AND CORRECTIVE ACTIONS WERE COMPLETE PRIOR TO PLANT START-UP.

[15] BIG ROCK POINT DOCKET 50-155 LER 90-005
MOTOR OPERATED VALVE FAILURE RESULTING IN PLANT SHUTDOWN.
EVENT DATE: 080390 REPORT DATE: 083190 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 219394) ON AUGUST 3, 1990 AT 1006 HOURS, OPERATIONS PERSONNEL PERFORMED A SURVEILLANCE TEST ON A CORE SPRAY MOTOR-OPERATED VALVE. ON THE FIRST ATTEMPT, THE VALVE FAILED TO OPEN. AT 1440 HOURS IT WAS DECIDED TO INITIATE A PLANT SHUTDOWN SINCE TROUBLESHOOTING AND REPAIR EFFORTS WERE EXPECTED TO TAKE MORE THAN TWENTY-FOUR HOURS. AN UNUSUAL EVENT WAS DECLARED. AT 0145 HOURS ON AUGUST 4, ALL CONTROL RODS WERE INSERTED AND PLANT TEMPERATURE WAS LESS THAN 212F THUS THE UNUSUAL EVENT WAS TERMINATED. ON AUGUST 5 A SETPOINT CHANGE WAS COMPLETED TO INCREASE THE TORQUE SETTING OF THE VALVE AND AT 1415 HOURS THE VALVE WAS SUCCESSFULLY TESTED AND DECLARED OPERABLE. AT 1421 HOURS REACTOR START-UP COMMENCED. CAUSE OF THE FAILURE WAS ATTRIBUTED TO INCREASED TORQUE REQUIREMENT WHILE TESTING THE VALVE UNDER A HIGH DIFFERENTIAL PRESSURE CONDITION. EVALUATION OF THE FAILURE CONTINUES.

[16] BRAIDWOOD 1 DOCKET 50-456 LER 90-011
SHUTDOWN DUE TO FAILED DISCHARGE CHECK VALVE ON THE 1A ESSENTIAL SERVICE WATER PUMP.
EVENT DATE: 080290 REPORT DATE: 083090 NSSS: WE TYPE: PWR
VENDOR: ANDERSON, GREENWOOD & CO.
JAMES BURY CORP.

(NSIC 219529) AT 0845 ON 8/1/90 IT WAS IDENTIFIED THAT THE DISCHARGE CHECK VALVE OF THE 1A ESSENTIAL SERVICE WATER PUMP (SX) HAD FAILED PERMITTING THE 1A SX PUMP TO ROTATE IN THE REVERSE DIRECTION WHEN SHUTDOWN. AT 0905 THE 1A SX PUMP WAS DECLARED INOPERABLE AND THE DISCHARGE VALVE WAS ISOLATED. ISOLATING THE DISCHARGE VALVE DECREASED THE REVERSE ROTATION BUT THE VALVE FAILED TO FULLY ISOLATE THE FLOW. THE PUMP SUCTION ISOLATION VALVE WAS CLOSED BUT ALSO FAILED TO FULLY ISOLATE THE FLOW. SEVERAL ADDITIONAL UNIT 1 SX VALVES WERE ISOLATED AND THE REVERSE PUMP ROTATION STOPPED. IN AN ATTEMPT TO SEAT THE CHECK VALVE THE 1A SX PUMP WAS "BUMPED". AFTER THE BUMP THE DISCHARGE VALVES WERE UNISOLATED BUT THE PUMP BEGAN ROTATING IN THE REVERSE DIRECTION SO THE VALVES WERE ISOLATED. ON 8/2/90 IT WAS CONCLUDED THAT THE CHECK VALVE REPLACEMENT WOULD REQUIRE THE 1A SX PUMP TO BE INOPERABLE FOR A LONGER DURATION THAN THE 72 HOURS ALLOWED BY THE PLANT TECH SPECS. AT 1720 A UNIT SHUTDOWN WAS INITIATED. AT 0618 ON 8/4/90 THE SHUTDOWN WAS COMPLETED. AT 0904 ON 9/10/90 AFTER COMPLETION OF POST MAINTENANCE TESTING, THE 1A SX PUMP WAS DECLARED OPERABLE. THE CAUSE OF THE EVENT WAS COMPONENT FAILURE. THE VALVE WAS REPAIRED. SIMILAR VALVES WILL BE MONITORED. THE PUMP COMPONENTS WERE INSPECTED AND THE EFFECTS OF THE REVERSE ROTATION WERE CONCLUDED TO BE NEGLIGIBLE.

[17] BRAIDWOOD 1 DOCKET 50-456 LER 90-012
1B SAFETY INJECTION ACCUMULATOR, 1A, AND 1B RESIDUAL HEAT REMOVAL PUMPS DECLARED INOPERABLE DUE TO A SMALL WELD CRACK IN A 3/4 INCH TEST LINE.
EVENT DATE: 080390 REPORT DATE: 083190 NSSS: WE TYPE: PWR

(NSIC 219530) AT 1000 ON 8/3/90 A SMALL LEAK WAS DISCOVERED ON A 3/4" TEST LINE WHICH WAS CONNECTED TO THE 10 INCH RCS LOOP 2 COLD LEG SAFETY INJECTION (SI)/ACCUMULATOR LINE. THE SOCKET WHERE THE LINE WAS WELDED CONTAINED A 3/8 INCH FLOW RESTRICTING ORIFICE THAT WAS THE ASME CLASS BREAK BETWEEN THE RCS AND NON RCS SUB SYSTEM COMPONENTS. AT 1300 IT WAS CONCLUDED THAT THE LEAK COULD IMPAIR THE ABILITY OF BOTH RESIDUAL HEAT REMOVAL (RH) TRAINS FROM INJECTING INTO ONE OF THE FOUR RCS COLD LEGS AND THE 1B SI ACCUMULATOR FROM INJECTING INTO ITS ASSOCIATED COLD LEG. THE ACCUMULATOR AND BOTH TRAINS OF RH WERE CONSERVATIVELY DECLARED INOPERABLE PENDING DETAILED EVALUATION. THE TECH SPEC DID NOT PROVIDE FOR BOTH TRAINS OF RH TO BE INOPERABLE IN MODEL 3. LIMITING CONDITION FOR OPERATION (LCO) 3.0.3 WAS ENTERED. AT 1635 THE UNIT ENTERED MODE 4 - HOT SHUTDOWN, LCO 3.0.3 WAS EXITED. CAUSE OF THIS EVENT WAS A SMALL CRACK IN THE WELD. THE CRACK WAS 1/2 INCH IN LENGTH AND WAS LOCATED 3/32 INCH AWAY FROM THE EDGE OF THE SOCKET. THE PIPING HAS BEEN REMOVED AND FORWARDED OFF SITE FOR METALLURGICAL ANALYSIS. THE LINE WAS REPLACED, SIMILAR CONNECTIONS WERE INSPECTED ON UNIT 1 AND WILL BE INSPECTED ON UNIT 2. NO PREVIOUS OCCURRENCES.

[18] BRAIDWOOD 1 DOCKET 50-456 LER 90-013
FEEDWATER ISOLATION DUE TO SPURIOUS ACTUATION OF THE TRAIN B FEEDWATER ISOLATION CIRCUITRY.
EVENT DATE: 081190 REPORT DATE: 090790 NSSS: WE TYPE: PWR

(NSIC 219531) STEAM GENERATOR (SG) LEVELS WERE BEING CONTROLLED MANUALLY WITH FLOW THROUGH THE FEEDWATER (FW) TEMPERING LINE FLOW CONTROL VALVES, 1FW035A, B, C, AND D. AT 1430 ON 8/11/90 THE UNIT 1 NUCLEAR STATION OPERATOR (NSO) OBSERVED THAT VALVES 1FW035A, B, C, AND D HAD ISOLATED. NO ALARMS ANNUNCIATED FOR THE ISOLATION. THE STATUS LIGHT FOR THE B TRAIN FEEDWATER ISOLATION WAS ILLUMINATED. AFTER VERIFYING THAT A FEEDWATER ISOLATION CONDITION DID NOT EXIST, THE NSO RESET THE FEEDWATER ISOLATION AND RE-ESTABLISHED FLOW TO THE STEAM GENERATORS THROUGH THE 1FW035A, B, C, AND D. BRAIDWOOD UNIT 1 OPERATING SURVEILLANCE (BWOS) 3.1.1-21, "UNIT 1 SSPS, REACTOR TRIP BREAKER, AND REACTOR TRIP BYPASS BREAKER BI-MONTHLY (STAGGERED) SURVEILLANCE (TRAIN B)", WAS PERFORMED TO TEST TRAIN B SOLID STATE PROTECTION SYSTEM (SSPS). TRAIN B SSPS FUNCTIONED PROPERLY AND THE EVENT DID NOT REPEAT. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO A SPURIOUS ACTUATION OF THE FEEDWATER ISOLATION CIRCUITRY WITHIN THE B TRAIN OF SSPS. THE B TRAIN OF SSPS WAS TESTED IN AN ATTEMPT TO REPRODUCE THE EVENT. TRAIN B SSPS WAS FOUND TO BE FUNCTIONING PROPERLY AND HAS PERFORMED SATISFACTORILY SINCE THAT TIME. THE EVENT COULD NOT BE REPEATED. THERE HAVE BEEN NO PREVIOUS SIMILAR OCCURRENCES.

[19] BRAIDWOOD 1 DOCKET 50-456 LER 90-014
ENTRY INTO MODE 3 PRIOR TO PERFORMING THE 1B MAIN STEAMLINE ISOLATION VALVE OPERABILITY TEST DUE TO MANAGEMENT DEFICIENCY.
EVENT DATE: 081290 REPORT DATE: 091190 NSSS: WE TYPE: PWR

(NSIC 219532) A UNIT 1 HEATUP WAS IN PROGRESS. REPAIRS HAD BEEN PERFORMED ON THE 1B MAIN STEAMLINE ISOLATION VALVE (MSIV) WHICH HAD BEEN EXPERIENCING ERRATIC BEHAVIOR DURING THE 10% CLOSURE TESTING. ON 8/11/90 A PARTIAL STROKE TEST WAS PERFORMED. THE VALVE OSCILLATED INDICATING THAT THE REPAIR ATTEMPT HAD BEEN UNSUCCESSFUL. THE STATION CONTROL ROOM ENGINEER (SCRE) CONTACTED THE DUTY OPERATING ENGINEER (DOE) TO DISCUSS THE EFFECT OF THIS ON THE MODE CHANGE. IT WAS CONCLUDED THAT THE MSIV FULL STROKE TEST WOULD BE REQUIRED FOR OPERABILITY. THE SCRE FOCUSED HIS ATTENTION AWAY FROM THE MSIV WORK PACKAGE AND ONTO THE PERFORMANCE OF TEST PROCEDURE. IN ACCORDANCE WITH A PREREQUISITE THE SCRE MARKED SEVERAL STEPS NOT APPLICABLE BECAUSE THE PROCEDURE WAS NOT BEING PERFORMED FOR RESPONSE TIME TESTING. THE SCRE INFORMED THE SHIFT ENGINEER (SE) THAT THE PROCEDURE HAD BEEN COMPLETED. THE SE REMOVED THE ITEM FROM THE MODE CHANGE CHECKLIST. AT 0757 ON 8/12/90 UNIT 1 ENTERED MODE 3. AT 0900 A SHIFT SUPERVISOR IDENTIFIED THAT THE STEPS THAT WOULD HAVE TESTED THE STANDBY ACCUMULATOR HAD BEEN MARKED "NA". THE CAUSES OF THE EVENT WERE MANAGEMENT DEFICIENCY, PROCEDURAL DEFICIENCY, COMMUNICATION DEFICIENCY, AND INADEQUATE REVIEW. TRAINING WILL BE PROVIDED, NEW OPERABILITY DETERMINATION METHODS WILL BE IMPLEMENTED, AND THE PROCEDURE WILL BE REVISED.

[20] BRAIDWOOD 2 DOCKET 50-457 LER 90-011
CONTAINMENT VENTILATION ISOLATION SIGNAL DUE TO SPURIOUS SPIKING FROM 2RE-AR012 AS A RESULT OF COMPONENT FAILURE.
EVENT DATE: 080590 REPORT DATE: 083190 NSSS: WE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 219533) AT 0950 ON 8/5/90 THE CONTAINMENT - FUEL HANDLING INCIDENT AREA RADIATION MONITOR (AR), 2RT-AR012, MOMENTARILY WENT INTO HIGH ALARM AND INTERLOCK ACTUATION, DUE TO SUDDEN SPIKING. THIS INITIATED A TRAIN B CONTAINMENT VENTILATION ISOLATION SIGNAL. NO COMPONENTS REPOSITIONED AS ALL VALVES WERE SECURED IN THE CLOSED POSITION. AT 1231 A SECOND SPIKE OCCURRED ON THE 2RT-AR012 WHICH ALSO RESULTED IN A VENTILATION ISOLATION SIGNAL. IT WAS SUSPECTED THAT DETECTOR REPLACEMENT WAS REQUIRED. DUE TO THE PHYSICAL AND RADIOLOGICAL DIFFICULTIES IN ACCESSING THE DETECTOR THE REPLACEMENT REQUIRED EXTENSIVE JOB PLANNING. AT 1956 ON 8/6/90 PRIOR TO INITIATING THE DETECTOR REPLACEMENT A SPIKE AGAIN OCCURRED WHICH ALSO RESULTED IN A VENTILATION ISOLATION SIGNAL. ON 8/7/90 THE DETECTOR OF 2RT-AR012 WAS REPLACED. THE CAUSE OF THIS EVENT WAS COMPONENT

FAILURE. THE FAULTY DETECTOR WAS CREATING SPURIOUS HIGH RADIATION SPIKES WHICH RESULTED IN THE MONITOR INITIATING THE VENTILATION ISOLATION SIGNALS. THE DETECTORS ARE REPLACED ON AN 18 MONTH FREQUENCY. THE FAILED DETECTOR IN THIS EVENT HAD BEEN IN SERVICE SINCE APRIL OF 1990. THE DETECTOR HAS BEEN RETURNED TO THE VENDOR FOR FAILURE ANALYSIS PREVIOUS CORRECTIVE ACTIONS ARE NOT APPLICABLE TO THIS EVENT.

[21] BROWNS FERRY 1 DOCKET 50-259 LER 90-012
HIGH PRESSURE FIRE PROTECTION SYSTEM IN VIOLATION OF TECH SPECS BECAUSE FUNCTIONAL TEST NOT PERFORMED.
EVENT DATE: 080190 REPORT DATE: 083090 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 219490) ON AUGUST 1, 1990 DURING A REVIEW OF THE FIRE PROTECTION TECHNICAL SPECIFICATIONS (TS) IT WAS DISCOVERED THAT HIGH-PRESSURE FIRE PROTECTION SYSTEM (HPFP) FIRE PUMP START LOGIC IS NOT BEING FUNCTIONALLY TESTED TO VERIFY THAT THE FIRE PROTECTION SYSTEM PUMPS MAINTAIN 120 PSIG AFTER AN INITIAL PUMP STARTUP. THE ROOT CAUSE OF THIS EVENT HAS NOT BEEN DETERMINED. THE REQUIREMENT TO VERIFY SEQUENTIAL STARTING OF THE HPFP SYSTEM FIRE PUMPS AT A PRESSURE GREATER THAN OR EQUAL TO 120 PSIG WAS APPROVED DECEMBER 27, 1988 IN AN AMENDMENT TO TSS. CORRECTIVE ACTIONS FOR THIS EVENT WILL BE DETERMINED BY THE ROOT CAUSE EVALUATION OF THIS EVENT. TVA IS PRESENTLY EVALUATING THE SURVEILLANCE INSTRUCTIONS (SI) ISSUED SINCE THE PLANT WENT INTO AN EXTENDED OUTAGE TO VERIFY THE SIS ADEQUATELY IMPLEMENT TSS.

[22] BROWNS FERRY 1 DOCKET 50-259 LER 90-013
UNPLANNED ENGINEERING SAFETY FEATURES ACTUATION DUE TO A BLOWN FUSE CAUSED BY FAILED RELAY.
EVENT DATE: 080790 REPORT DATE: 090690 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 219406) ON AUGUST 7, 1990, AT 1212 HOURS, A BLOWN FUSE IN UNIT 1 PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) LOGIC PANEL RESULTED IN THE ACTUATION OF ENGINEERED SAFETY FEATURES. THE SYSTEM AFFECTED BY THIS EVENT WAS THE UNIT 1 REACTOR ZONE VENTILATION SYSTEM. INVESTIGATION OF THIS EVENT REVEALED THAT A RELAY IN THE UNIT 1 SECONDARY CONTAINMENT ISOLATION CIRCUITRY FAILED. THIS CAUSED A FAULTED CONDITION WHICH RESULTED IN BLOWING THE FUSE. THE ROOT CAUSE OF THIS EVENT WAS A RANDOM FAILURE OF THE RELAY. THE CAUSE OF THE FAILED RELAY WAS DUE TO A FAILED RELAY COIL. FOLLOWING THE BLOWING OF THE FUSE, THE FAILED RELAY WAS REPLACED AND THE PRIMARY CONTAINMENT ISOLATION LOGIC WAS RESET ON AUGUST 9, 1990 AT 0340 HOURS.

[23] BROWNS FERRY 1 DOCKET 50-259 LER 90-014
UNPLANNED ENGINEERED SAFETY FEATURES ACTUATION DUE TO A BLOWN FUSE.
EVENT DATE: 081090 REPORT DATE: 091090 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)

(NSIC 219491) ON AUGUST 10, 1990 AT 1250 HOURS, UNIT 1 REACTOR PROTECTION SYSTEM (RPS) BUS 1A WAS DEENERGIZED RESULTING IN THE ACTUATION OF A PRIMARY CONTAINMENT ISOLATION OF THE UNIT 1 REACTOR BUILDING VENTILATION SYSTEM. INVESTIGATION OF THIS EVENT REVEALED THAT A CONTROL POWER FUSE IN MG SET 1A HAD BLOWN. THIS CAUSED THE MG SET TO TRIP AND REMOVED NORMAL POWER SOURCE FROM THE RPS BUS. THE ROOT CAUSE OF THIS EVENT WAS A RANDOM FAILURE OF THE FUSE. THERE WERE NO ABNORMALITIES IDENTIFIED IN THE MG SET CONTROL CIRCUIT THAT WOULD CAUSE THE CONTROL POWER FUSE TO BLOW. FOLLOWING THE BLOWING OF THE FUSE, THE FUSE WAS REPLACED, AND RPS BUS 1A WAS REALIGNED TO ITS ALTERNATE POWER SUPPLY ON AUGUST 11, 1990 AT 2135 HOURS.

[24]   BRUNSWICK 1   DOCKET 50-325   LER 90-013
COMPUTER AIDED DESIGN SYSTEM DESIGN DEFICIENCIES.
EVENT DATE: 072590   REPORT DATE: 082490   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)

(NSIC 219335) ON 7/25/90, IT WAS DETERMINED THAT THE CAC/CAD SUBSYSTEM DID NOT MEET FSAR AND DESIGN REQUIREMENTS RELATIVE TO REDUNDANCY OF POWER SUPPLIES FOR THE VAPORIZER TRAINS AND DIVISION I AND II ELECTRICAL SEPARATION CRITERIA FOR THE INLET AND THE PURGE/EXHAUST SIDES OF THE CAD SUBSYSTEM. THIS DEFICIENCY HAS EXISTED SINCE CONSTRUCTION OF THE PLANT, AND IS THE RESULT OF CHANGING REGULATORY COMMITMENTS THAT WERE MADE RELATIVE TO THE SYSTEM NOT BEING PROPERLY INCORPORATED INTO THE DESIGN OF THE PLANT. REVIEWS SINCE THE DATE HAVE NOT FOUND THESE DISCREPANCIES. CORRECTIVE ACTIONS INCLUDE THE DEVELOPMENT OF ENGINEERING EVALUATIONS TO ELIMINATE THE IMMEDIATE CONCERNS, AND A REVIEW OF REVIEW OF THE SYSTEM TO DETERMINE LONG-TERM RESOLUTION. IN ADDITION, A REVIEW OF THE INVOLVED PANEL FOR ADDITIONAL SEPARATION CONCERNS WILL BE DONE AS WELL AS A LICENSING BASIS REVIEW BEING INCLUDED AS A PART OF THE SYSTEM DESIGN BASIS DOCUMENT DEVELOPMENT. EVALUATIONS HAVE DETERMINED THAT SINCE MANUAL OPERATION OF THE SYSTEM IS READILY ACHIEVABLE, AND THE SYSTEM IS NOT NEEDED UNTIL WELL INTO THE ACCIDENT SCENARIO, THE SYSTEM WOULD HAVE BEEN ABLE TO PROVIDE ITS SAFETY FUNCTION.

[25]   BRUNSWICK 2   DOCKET 50-324   LER 90-007
GROUP IV ISOLATION WHILE PERFORMING A REACTOR WATER CLEANUP PLANT MODIFICATION.
EVENT DATE: 072490   REPORT DATE: 082390   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 1 (BWR)

(NSIC 219354) THE UNIT 2 HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM RECEIVED THREE ISOLATIONS OF THE OUTBOARD VALVE WHILE PERFORMING A PLANT MODIFICATION ON THE REACTOR WATER CLEANUP SYSTEM. A LOOSE WIRE ON A HPCI TEMPERATURE MODULE RESULTED IN A GROUP 4 ISOLATION (E41-F003 ONLY). BEFORE AND AFTER THE EVENT, THE UNIT WAS OPERATING AT 100% POWER. ON EACH OCCASION HPCI WAS RESTORED TO SERVICE WITHIN A FEW MINUTES OF THE OUTBOARD VALVE CLOSING. THE INBOARD ISOLATION VALVE WAS NOT AFFECTED. HPCI RESPONDED AS DESIGNED TO THE EVENT. OTHER EMERGENCY CORE COOLING SYSTEMS REMAINED OPERABLE. SIMILAR EVENTS INVOLVING A TEMPERATURE MODULE AND GROUP ISOLATIONS HAVE BEEN REPORTED IN LERS 2-84-017, 1-86-008, 1-86-014, 2-87-007, 1-88-002, AND 1-89-005.

[26]   BRUNSWICK 2   DOCKET 50-324   LER 90-008
SCRAM RESULTING FROM TURBINE TRIP ON HIGH LEVEL DUE TO BLOWN FUSE IN FEEDWATER LOGIC.
EVENT DATE: 081690   REPORT DATE: 091790   NSSS: GE   TYPE: BWR
VENDOR: SHAWMUT COMPANY

(NSIC 219474) ON 8/16/90, UNIT 2 REACTOR WAS AT 100% POWER. RPS, HPCI, RCIC, ADS, RHR/LPCI, CS, SBGT, SLC, DG AND PLANT ELECTRICAL SYSTEM WERE OPERABLE AND IN STANDBY READINESS. REACTOR FEEDWATER LEVEL CONTROL SYSTEM WAS OPERATING IN AUTOMATIC - THREE ELEMENT CONTROL AND LEVEL WAS BEING MAINTAINED AT 185 INCHES. AT 0942 ON 8/16/90, THE REACTOR AUTOMATICALLY SHUTDOWN ON A "TSV FAST CLOSURE" RPS TRIP SIGNAL CAUSED BY A TURBINE TRIP ON REACTOR HIGH WATER LEVEL. DURING THIS EVENT, HPCI TURBINE STOP VALVE CYCLED CLOSED AND THEN OPEN, WATER INTRUSION INTO THE HPCI OIL WAS NOTED, RCIC BAROMETRIC CONDENSER VACUUM PUMP EXPERIENCED AN ELECTRICAL FAULT, AND A LOSS OF RECIRCULATION PUMPS RESULTED IN TEMPERATURE TRANSIENTS IN THE VESSEL. HPCI AND RCIC OPERABILITY WERE NOT AFFECTED AND RECIRCULATION PUMPS ARE NOW BEING POWERED FROM THE START UP AUXILIARY TRANSFORMER TO PREVENT THEIR LOSS DURING FUTURE REACTOR TRIPS. CAUSE OF THIS EVENT WAS FAILURE OF PRIMARY POWER FUSE C32-F5, WHICH SUPPLIED POWER TO THE STEAM FLOW INPUTS OF THE THREE ELEMENT FEEDWATER CONTROL LOGIC. LOSS OF THE STEAM FLOW INPUTS RESULTED IN A MAXIMUM DEMAND SIGNAL TO THE RFPS AND A RAPID INCREASE IN REACTOR LEVEL UP TO THE HIGH LEVEL TURBINE TRIP POINT WHICH, IN TURN, CAUSED A REACTOR SCRAM ON TSV POSITION. NO CAUSE FOR FUSE FAILURE HAS BEEN DETERMINED. SIMILAR EVENTS 2-88-018 AND 1-88-023.

[27] BRUNSWICK 2 DOCKET 50-324 LER 90-010
INADVERTENT DEENERGIZATION OF REACTOR PROTECTION SYSTEM BUS 2A.
EVENT DATE: 082290 REPORT DATE: 091790 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 1 (BWR)

(NSIC 219543) ON 8/22/90 AT 0438, WITH THE UNIT 2 REACTOR IN COLD SHUTDOWN REACTOR PROTECTION SYSTEM (RPS) BUS "A" WAS INADVERTENTLY DEENERGIZED DUE TO PERSONNEL ERROR WHEN THE BUS OUTPUT BREAKERS EPA-1 AND 2 WERE OPENED INSTEAD OF THE RESPECTIVE UNIT 1 BREAKERS. THE UNIT 1 BUS "A" WAS BEING REMOVED FROM SERVICE FOR SURVEILLANCE TESTING 1MST-RPS-21SA ON ELECTRICAL PROTECTION ASSEMBLY BREAKERS (EPA) 1 AND 2. THE TRIP RESULTED IN A LOGIC TRIP OF THE RPS "A" BUS AND EXPECTED ACTUATION AND ISOLATION OF THE STANDBY GAS TRAINS AND PRIMARY CONTAINMENT ISOLATION SYSTEMS RESPECTIVELY, INCLUDING ISOLATION OF SHUTDOWN COOLING. THE SAFETY SIGNIFICANCE WAS MINIMAL IN THAT POWER WAS RESTORED IN 2 MINUTES AND SHUTDOWN COOLING RETURNED TO SERVICE IN 22 MINUTES. THE HIGHEST REACTOR VESSEL TEMPERATURE REACHED 147 DEGREE F A 5 DEGREE F RISE FROM PRIOR TO THE EVENT. THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL SINCE ALL SYSTEMS FUNCTIONED AS DESIGNED AND THE SHORT DURATION OF THE POWER LOSS. THE INVOLVED OPERATORS HAVE BEEN APPROPRIATELY DISCIPLINED AND THE CLEARANCE PROCEDURE WILL BE REVISED TO CLARIFY DOUBLE/INDEPENDENT VERIFICATION REQUIREMENTS. TRAINING WILL BE CONDUCTED ON THE USE AND INTENT OF DOUBLE/INDEPENDENT VERIFICATIONS AND THE COMPONENT IDENTIFICATION LABEL WILL BE ENHANCED. A SIMILAR EVENT WAS REPORTED IN LER 2-86-021.

[28] BYRON 1 DOCKET 50-454 LER 90-010
AUXILIARY FEEDWATER PUMP AUTOMATIC START DUE TO MODULE FAILURE IN THE ANTICIPATED TRANSIENT WITHOUT SCRAM SYSTEM.
EVENT DATE: 081890 REPORT DATE: 091390 NSSS: WE TYPE: PWR
VENDOR: SCIENCE APPLICATIONS, INC.

(NSIC 219556) AT 0247 ON 8/8/90, WITH UNIT 1 IN MODE 1 AT 84% POWER, AN UNEXPECTED AUTOMATIC START OF THE 1A AUXILIARY FEEDWATER PUMP (AF) (BA) OCCURRED. THE CAUSE OF THE AUTO START WAS DUE TO THE K1AA RELAY IN THE ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) MITIGATION SYSTEM ENERGIZING WITHOUT THE NECESSARY 3 OF 4 LOW STEAM GENERATOR LEVEL SIGNALS. THE LICENSED REACTOR OPERATOR PLACED THE ATWS MITIGATING SYSTEM (AMS) IN BYPASS AND RETURNED THE PUMP TO STANDBY SERVICE. TROUBLESHOOTING DID NOT IDENTIFY AN IMMEDIATE FAILURE MECHANISM AND THE EVENT COULD NOT BE REPEATED. HOWEVER, AFTER CONTINUED OBSERVATION, THE K1AA ACTUATION RELAY STATUS LIGHT RETURNED TO A PARTIALLY ILLUMINATED CONDITION ON 8/22/90. THE DRIVER MODULE PANEL WAS REPLACED AND AMS WAS RETURNED TO NORMAL ON 8/31/90. THE MOST PROBABLE CAUSE OF THE DRIVE MODULE FAILURE IS THE GRADUAL DEGRADATION OF AN ELECTRONIC METAL-OXIDE SEMICONDUCTOR FIELD EFFECT TRANSISTOR DUE TO INDUCTIVE FLYBACK WHICH ALLOWED TRICKLE CURRENT FLOW UNTIL THERE WAS SUFFICIENT AMOUNT OF CURRENT TO ENERGIZE THE RELAY. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV) DUE TO THE AUTOMATIC ACTUATION OF AN ENGINEERED SAFETY FEATURE.

[29] BYRON 1 DOCKET 50-454 LER 90-011
HIGH NEGATIVE FLUX RATE REACTOR TRIP DUE TO LOSS OF CONTROL ROD DRIVE POWER SUPPLIES DURING LIGHTNING STRIKE.
EVENT DATE: 081990 REPORT DATE: 091390 NSSS: WE TYPE: PWR
VENDOR: LAMBDA ELECTRONICS

(NSIC 219557) AT 0425 ON 8/19/90, WITH SEVERE LIGHTNING ACTIVITY NEAR BYRON STATION, A UNIT 1 REACTOR TRIP OCCURRED FROM 78% POWER. A LIGHTNING STRIKE INDUCED A VOLTAGE SURGE THAT ACTIVATED NINE OUT OF TEN OVER-VOLTAGE PROTECTION DEVICES INSTALLED ON POWER SUPPLIES IN THE ROD DRIVE (RD)(AA) POWER CABINETS. THIS ACTIVATION RELEASED TWELVE OUT OF FIFTEEN ROD CONTROL CLUSTER ASSEMBLY GROUPS INTO THE CORE AND RESULTED IN A HIGH NEGATIVE FLUX RATE REACTOR TRIP. DUE TO SEVERAL COMMONWEALTH EDISON AND INDUSTRY WIDE LIGHTNING INDUCED REACTOR TRIPS, SEVERAL MODIFICATIONS HAVE PREVIOUSLY BEEN MADE TO BOTH THE CONTAINMENT LIGHTNING PROTECTION SYSTEM AND THE ROD DRIVE OVER VOLTAGE PROTECTORS. ADDITIONAL ENHANCEMENTS ARE BEING PURSUED AND WILL BE DOCUMENTED IN A SUPPLEMENTAL REPORT.

THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV) AS A RESULT OF THE AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM.

[30] BYRON 2 DOCKET 50-455 LER 90-003
P-14 FEEDWATER ISOLATION DUE TO INABILITY TO CONTROL LEVEL IN D-5 STEAM GENERATORS FOR LOW POWER LEVELS.
EVENT DATE: 071790 REPORT DATE: 081590 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 219390) AT 1216 ON 7/17/90, WITH UNIT 2 AT 6% POWER AND POWER ASCENSION IN PROGRESS, A P-14 FEEDWATER ISOLATION TURBINE TRIP OCCURRED WHEN THE 2A STEAM GENERATOR NARROW RANGE LEVEL REACHED THE HIGH-HIGH WATER LEVEL SETPOINT. STEAM GENERATOR LEVEL IMMEDIATELY DROPPED BELOW THE P-14 SETPOINT AND THE SIGNAL WAS RESET. THE FEEDWATER SYSTEM WAS REALIGNED AND POWER ASCENSION WAS RESUMED. A TURBINE TRIP DID NOT OCCUR AS THE TURBINE WAS NOT LATCHED. THE ROOT CAUSE OF THE HIGH-HIGH NARROW RANGE LEVEL ON THE 2A STEAM GENERATOR WAS THE INHERENT DIFFICULTY IN CONTROLLING LEVEL IN THE D-5 MODEL STEAM GENERATORS AT LOW POWER LEVELS. THE DIFFICULTY IN CONTROLLING LEVEL IS CAUSED BY THE SMALL SPAN (APPROXIMATELY 60% OF THE MORE STABLE D-4 MODEL) OF THE NARROW RANGE INDICATION. THE INDUSTRY RECOGNIZES THE UNIQUE BEHAVIOR OF THE D-5 STEAM GENERATORS, AND AS A RESULT LICENSED OPERATOR TRAINING ADDRESSES TECHNIQUES IN CONTROLLING D-5 STEAM GENERATOR LEVEL. TO ENHANCE THIS KNOWLEDGE, THIS EVENT WILL BE INCLUDED IN LICENSED OPERATOR REQUIRED READING.

[31] CALLAWAY 1 DOCKET 50-483 LER 90-009
TECHNICAL SPECIFICATION ENTRY DUE TO LOSS OF THE DIGITAL ROD POSITION INDICATION SYSTEM ON A FAILED CENTRAL CONTROL CARD.
EVENT DATE: 080290 REPORT DATE: 090490 NSSS: WE TYPE: PWR

(NSIC 219520) ON 8/2/90 AT 0917 CDT, TECH SPEC (T/S) 3.0.3 WAS UNINTENTIONALLY ENTERED WHEN THE DIGITAL ROD POSITION INDICATION (DRPI) SYSTEM CONTROL ROD INDICATION WAS DE-ENERGIZED DUE TO A FAILED DRPI CENTRAL CONTROL CARD. THE PLANT WAS IN MODE 1 - POWER OPERATIONS AT 100% REACTOR POWER. ON 8/1/90, SPURIOUS 'ROD DEVIATION' ALARMS OCCURRED DURING A CONTROL ROD MOVEMENT SURVEILLANCE (T/S 4.1.3.1.2). UTILITY INSTRUMENT AND CONTROL (I&C) PERSONNEL BEGAN TROUBLESHOOTING AT APPROXIMATELY 0900. TROUBLESHOOTING REVEALED THE DRPI +15 VDC POWER SUPPLY WAS ACTUALLY SUPPLYING +20 VDC WITH A RIPPLE OF 200 MILLIVOLTS RMS. ON 8/2/90, IN AN ATTEMPT TO OBTAIN DRPI -15 VDC POWER SUPPLY READINGS, DRPI CENTRAL CONTROL CARD A104 WAS REMOVED TO ALLOW THE INSTALLATION OF AN EXTENDER CARD. DUE TO INSULATING PAPER WHICH HAD FALLEN OFF THE DRPI 'DATA A' INPUT/OUTPUT (I/O) CARD IN ADJACENT SLOT A105, THE EXTENDER CARD COULD NOT BE INSTALLED. AT 0917, THE 'DATA A' I/O CARD WAS REMOVED TO REPAIR THE INSULATION. UPON REMOVAL, THE DRPI DISPLAY WENT BLANK FOR APPROXIMATELY TEN SECONDS. THE +15 VDC POWER SUPPLY OVER-VOLTAGE CONDITION, IN CONJUNCTION WITH THE REMOVAL OF THE A105 I/O CARD, RESULTED IN THE FAILURE OF DRPI CENTRAL CONTROL CARD A102 AND THE TEMPORARY LOSS OF DRPI. THE +15 VDC POWER SUPPLY HAS BEEN ISOLATED FROM THE CIRCUIT AND WILL BE REPAIRED IN THE NEXT REFUELING OUTAGE.

[32] CALVERT CLIFFS 1 DOCKET 50-317 LER 90-014 REV 01
UPDATE ON DELTA T CHANNELS OUT OF SPECIFICATION CAUSED BY LACK OF PROCEDURAL GUIDANCE RESULTS IN ENTERING TS LCO 3.0.3.
EVENT DATE: 041490 REPORT DATE: 082490 NSSS: CE TYPE: PWR

(NSIC 219351) ON APRIL 13, 1990 WITH UNIT 1 IN MODE 1 AT 30% AND AGAIN ON APRIL 19, 1990 WITH UNIT 1 IN MODE 1 AT 65%, CALVERT CLIFFS UNIT 1 ENTERED TECHNICAL SPECIFICATION (TS) LIMITING CONDITION OF OPERATION (LCO) 3.0.3. THIS LCO WAS ENTERED DUE TO THE POTENTIAL INOPERABILITY OF THREE OUT OF FOUR REACTOR PROTECTIVE SYSTEM (RPS) DELTA T POWER CHANNELS. THE DELTA T POWER CHANNELS WERE DECLARED INOPERABLE AFTER THEY WERE ADJUSTED TO MATCH SECONDARY CALORIMETRIC POWER. THE AMOUNT THE CHANNELS WERE ADJUSTED EXCEEDED THE AMOUNT ALLOWED IN THE CALVERT CLIFFS SETPOINT FILE. IN THE FIRST EVENT, THE ROOT CAUSE WAS AN INADEQUATE WRITTEN PROCEDURE. THE ROOT CAUSE OF THE SECOND EVENT WAS PERSONNEL ERROR RESULTING IN INSUFFICIENT DEPTH OF ASSESSMENT OF THE FIRST EVENT AND

INSUFFICIENT CONTINGENCY PLANNING. CORRECTIVE ACTIONS INCLUDED INCREASING THE ALLOWABLE TOLERANCE BETWEEN THE DELTA T POWER CHANNELS AND CALORIMETRIC POWER AND CHANGING OPERATIONS PROCEDURES TO PROVIDE BETTER GUIDANCE TO THE OPERATORS.

[33] CALVERT CLIFFS 1 DOCKET 50-317 LER 90-022
LOW TEMPERATURE OVERPRESSURE PROTECTION CALCULATION PROBLEMS.
EVENT DATE: 050490 REPORT DATE: 082290 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 2 (PWR)

(NSIC 219352) FOLLOWING THE DEVELOPMENT OF LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) CONTROLS, SEVERAL ERRORS AND NON-CONSERVATISMS WERE FOUND IN THE CALCULATIONS SUPPORTING THESE CONTROLS. THESE PROBLEMS INVOLVED THE POWER OPERATED RELIEF VALVE (PORV) RESPONSE TIME, TWO-PHASE FLOW THROUGH THE PORVS, MODELING OF THE REACTOR COOLANT PUMP START TRANSIENT, AND ASSUMED DECAY HEAT LOAD. THESE EVENTS ARE VOLUNTARILY REPORTED BECAUSE OF NUCLEAR REGULATORY COMMISSION (NRC) INTEREST IN CONJUNCTION WITH RELATED TECHNICAL SPECIFICATION (TS) SUBMITTALS. THESE DEFICIENCIES WERE CAUSED BY INSUFFICIENT INVESTIGATION AND DOCUMENTATION OF ASSUMPTIONS AND INITIAL CONDITIONS AND OVER-RELIANCE ON THE VALIDITY OF PREVIOUS CALCULATIONS AND SUBMITTALS. REVALIDATED LTOP TS CONTROLS FOR UNIT 1 HAVE BEEN SUBMITTED, APPROVED, AND IMPLEMENTED. THE REACTOR COOLANT SYSTEM FOR UNIT 2 WILL NOT BE PRESSURIZED UNTIL REVISED CONTROLS ARE REVIEWED AND IMPLEMENTED. SIMILAR CALCULATIONS ARE BEING REVIEWED TO DETERMINE IF ADDITIONAL ERRORS AND OMISSIONS AFFECTING PLANT OPERATION WERE MADE. EXPECTATIONS FOR PREPARATION, REVIEW AND DOCUMENTATION OF ANALYTICAL ASSUMPTIONS, INPUT DATA, AND CALCULATIONAL METHODS WILL BE REEVALUATED AND RESTATED.

[34] CALVERT CLIFFS 1 DOCKET 50-317 LER 90-023
ENGINEERED SAFETY FEATURES ACTUATIONS DUE TO FAILED FUSE AND POWER SUPPLY.
EVENT DATE: 080290 REPORT DATE: 090490 NSSS: CE TYPE: PWR
VENDOR: SHAWMUT COMPANY
VITRO ENGINEERING DIVISION

(NSIC 219449) ON AUGUST 2, 1990 UNIT 1 EXPERIENCED AN ACTUATION OF SUBSYSTEM ZA OF THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS). THE ACTUATION OCCURRED WHILE POWER WAS BEING RESTORED TO SUBSYSTEM ZA FOLLOWING MAINTENANCE AND WAS CAUSED BY THE OPERATION OF A FUSE IN THE VITAL AC DISTRIBUTION PANEL. NO CAUSE FOR THE OPERATION OF THE FUSE COULD BE IDENTIFIED AND THE FUSE HAS BEEN SENT TO A LABORATORY FOR FAILURE ANALYSIS. ON AUGUST 7, 1990 PRIOR TO THE RESTORATION OF ESFAS SUBSYSTEM ZA, UNIT 1 EXPERIENCED A PARTIAL ACTUATION OF SUBSYSTEM ZB OF THE ESFAS WHICH WAS CAUSED BY A DEGRADED 15 VDC POWER SUPPLY IN THE ACTUATION LOGIC CABINET. NO CAUSE FOR THE FAILURE OF THE POWER SUPPLY WAS IDENTIFIED AND THE POWER SUPPLY HAS BEEN RETURNED TO THE MANUFACTURER FOR FAILURE ANALYSIS. IN BOTH CASES, PLANT SYSTEMS AND COMPONENTS PERFORMED AS REQUIRED BY THE DESIGN, CONSISTENT WITH PLANT CONDITIONS AND SYSTEM LINEUPS. FOLLOWING THE ACTUATION ON AUGUST 7, CONTAINMENT INTEGRITY WAS ESTABLISHED IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS BECAUSE LOSS OF THE UNDERVOLTAGE START SIGNAL ASSOCIATED WITH ESFAS REQUIRED THAT THE UNIT 1 EMERGENCY DIESELS BE DECLARED INOPERABLE. THE RESULTS OF THE FAILURE ANALYSES ON THE FUSE AND THE POWER SUPPLY WILL BE REVIEWED TO DETERMINE IF ADDITIONAL CORRECTIVE ACTION IS REQUIRED.

[35] CATAWBA 1 DOCKET 50-413 LER 90-029
INADEQUATE TECHNICAL REVIEWS OF A COMPENSATORY ACTION RESULTING IN A TECHNICAL SPECIFICATION VIOLATION DUE TO INADEQUATE DIRECTIVE/POLICY.
EVENT DATE: 071890 REPORT DATE: 082990 NSSS: WE TYPE: PWR

(NSIC 219513) ON JULY 18, 1990, WITH UNIT 1 IN MODE 1, POWER OPERATION, A COMPENSATORY ACTION WAS APPROVED TO SECURE THE UPPER ANNULUS ACCESS DOOR OPEN TO PURGE FREON FROM AN INOPERABLE ANNULUS VENTILATION (VE) SYSTEM FILTER TRAIN A. THE ACCESS DOOR WAS OPENED AT APPROXIMATELY 1600 HOURS, AND REMAINED OPEN UNTIL JULY 19 AT APPROXIMATELY 0800 HOURS, WITH A SECURITY OFFICER AT THE DOOR IN COMPLIANCE WITH THE REQUIREMENT OF THE COMPENSATORY ACTION SHEET TO MAINTAIN VE FILTER TRAIN B OPERABILITY. THE COMPENSATORY ACTION REQUIREMENT WAS THAT THE SECURITY OFFICER CLOSE THE ACCESS DOOR "IMMEDIATELY UPON NOTIFICATION" FROM THE

CONTROL ROOM. TECHNICAL SPECIFICATIONS (T/S) REQUIRE THAT THE VE SYSTEM ACHIEVE AN ANNULUS NEGATIVE PRESSURE OF AT LEAST 0.5 INWG WITHIN ONE MINUTE AFTER A START SIGNAL. AN OPERABILITY EVALUATION BY DESIGN ENGINEERING CONCLUDED THAT THE COMPENSATORY ACTION DID NOT MEET T/S REQUIREMENTS FOR VE SYSTEM OPERABILITY. THIS INCIDENT IS ATTRIBUTED TO MANAGEMENT DEFICIENCY DUE TO AN INADEQUATE POLICY/DIRECTIVE. STATION DIRECTIVE GUIDANCE ON OPERABILITY EVALUATIONS DID NOT ENSURE AN ADEQUATE TECHNICAL REVIEW OF COMPENSATORY ACTIONS PRIOR TO THEIR USE. ALL SUBSEQUENT COMPENSATORY ACTIONS WILL BE INITIATED WITH AN APPROPRIATE 10CFR50.59 EVALUATION.

[36] CATAWBA 1 DOCKET 50-413 LER 90-028
TECHNICAL SPECIFICATION 3.0.3 ENTERED DUE TO TWO INOPERABLE TRAINS OF THE CONTROL ROOM VENTILATION SYSTEM.
EVENT DATE: 072390 REPORT DATE: 082190 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 219372) ON JULY 23, 1990 AT 1135 HOURS, WITH UNIT 1 IN MODE 1, POWER OPERATION, AND UNIT 2 IN MODE 6, REFUELING (WITH CORE ALTERATIONS IN PROGRESS), CONTROL ROOM OPERATORS (CROS) NOTICED A LIGHT HAZE OF SMOKE IN THE CONTROL ROOM (C/R). INSPECTION OF THE OPERATING TRAIN B CONTROL ROOM AREA VENTILATION (VC) AND CHILLED WATER (YC) SYSTEM EQUIPMENT REVEALED THAT ALL BUT ONE FAN BELT ON THE TRAIN B C/R AIR HANDLING UNIT (2CR-AHU-1) HAD BROKEN. THE BROKEN FAN BELTS WERE TWISTED ON THE DRIVE SHAFT AND HAD CREATED THE SMOKE THAT ENTERED THE C/R. AT 1210 HOURS, DUE TO BOTH TRAINS OF VC/YC INOPERABLE (TRAIN A WAS ALREADY INOPERABLE FOR PLANNED MAINTENANCE), UNIT 1 ENTERED TECHNICAL SPECIFICATION (T/S) 3.0.3 ACTION STATEMENT AND UNIT 2 CORE ALTERATIONS WERE SUSPENDED. AT 1455 HOURS, TRAIN B VC/YC WAS DECLARED OPERABLE AND T/S 3.0.3 WAS EXITED ON UNIT 1. THIS INCIDENT IS ATTRIBUTED TO AN EQUIPMENT FAILURE DUE TO THE FAILURE OF THE FAN BELTS ON 2CR-AHU-1. REPLACEMENT BELTS WERE INSTALLED AND THE EQUIPMENT WAS RETURNED TO SERVICE. THIS INCIDENT IS ALSO ATTRIBUTED TO AN INAPPROPRIATE ACTION IN THAT A WORK REQUEST HAD BEEN WRITTEN ON MAY 10, 1990 TO REPLACE THE FAN BELTS, BUT HAD FAILED TO BE SCHEDULED. DISCUSSIONS WERE HELD WITH RESPONSIBLE PERSONNEL TO PREVENT RECURRENCE.

[37] CATAWBA 2 DOCKET 50-414 LER 90-011
TECHNICAL SPECIFICATION VIOLATION DUE TO INOPERABLE CONTAINMENT PENETRATION DUE TO INAPPROPRIATE ACTIONS AND A DEFECTIVE PROCEDURE.
EVENT DATE: 071190 REPORT DATE: 082190 NSSS: WE TYPE: PWR

(NSIC 219374) ON JULY 11, 1990, WITH UNIT 2 IN NO MODE, DEFUELED, A CATAWBA SAFETY REVIEW GROUP (CSRG) STAFF MEMBER NOTED, DURING A REVIEW OF TAGOUTS, THAT THE TAGOUT RECORD SHEET FOR THE CLOSURE OF 2NC-61, NC TO PZR SMPL HX ISOL VALVE, HAD BEEN ACTIVE FOR SOME TIME. VALVE 2NC-61 WAS SPECIFIED TO BE LOCKED OPEN, AND CLOSING IT COULD ALLOW POST ACCIDENT PRESSURE INSIDE THE PRESSURIZER SAMPLE PENETRATION TO EXCEED DESIGN PRESSURE. FURTHER INVESTIGATION REVEALED THAT 2NC-61 HAD PREVIOUSLY BEEN CLOSED PER PROCEDURE. THIS VALVE WAS CLOSED, IN BOTH CASES, TO ISOLATE PRESSURIZER STEAM FROM LEAKING INTO AND DILUTING PRESSURIZER WATER SAMPLES BEING TAKEN BY CHEMISTRY. THE UNIT OPERATED IN VIOLATION OF THE TECHNICAL SPECIFICATION REQUIRING THIS PENETRATION TO BE OPERABLE FROM FEBRUARY 19, 1988 TO APRIL 27, 1988, FROM APRIL 29, 1988 TO MARCH 11, 1989, AND FROM JUNE 2, 1989 TO JUNE 11, 1990. THIS INCIDENT IS ATTRIBUTED TO INAPPROPRIATE ACTIONS AND A DEFECTIVE PROCEDURE, FOR NOT REALIZING THE SAFETY SIGNIFICANCE OF THE DESIGN POSITION OF 2NC-61. CORRECTIVE ACTIONS WILL INCLUDE TRAINING ON THE SIGNIFICANCE OF VALVE POSITIONS AS THEY RELATE TO RELIEF PATHS, PROCEDURE ENHANCEMENTS, AND THE DEVELOPMENT OF A LIST OF LOCKED VALVES WITH THE REASONS FOR THEIR BEING LOCKED.

[38] CATAWBA 2 DOCKET 50-414 LER 90-009
TECHNICAL SPECIFICATION VIOLATION DUE TO INOPERABLE SOURCE RANGE NEUTRON FLUX MONITORS DURING REACTOR CORE RELOAD.
EVENT DATE: 072390 REPORT DATE: 082190 NSSS: WE TYPE: PWR

(NSIC 219373) ON JULY 23, 1990, WITH UNIT 2 IN MODE 6, REFUELING, AT

APPROXIMATELY 0730 HOURS, THE SOURCE RANGE NEUTRON FLUX (SRNF) MONITORS WERE DISCOVERED TO BE OPERATING WITHOUT AN AUDIBLE COUNT RATE INDICATION AS SPECIFIED IN TECHNICAL SPECIFICATIONS (T/S). THIS DISCREPANCY WAS DISCOVERED WHEN THE ON-COMING OPERATIONS SHIFT SUPERVISOR (SS) REALIZED THAT THE AUDIBLE COUNT RATE INDICATION WAS NOT PRESENT IN THE CONTROL ROOM (C/R) AND IN CONTAINMENT. A SUBSEQUENT INVESTIGATION BY THE CONTROL ROOM OPERATORS (CROS) REVEALED THAT THE "AUDIBLE MULTIPLIER" SWITCH WAS IN THE "OFF" POSITION. THE CROS IMMEDIATELY POSITIONED THE AUDIBLE MULTIPLIER SWITCH TO THE "X10" POSITION AND AN AUDIBLE COUNT RATE INDICATION WAS OBTAINED. FURTHER INVESTIGATION OF THIS INCIDENT REVEALED THAT CORE ALTERATIONS WERE IN PROGRESS FOR APPROXIMATELY TWO HOURS WITHOUT THE REQUIRED AUDIBLE COUNT RATE IN THE C/R AND IN CONTAINMENT. THIS INCIDENT IS ATTRIBUTED TO INAPPROPRIATE ACTION DUE TO THE CRO AND SS FAILURE TO ENSURE THAT THE SRNF MONITORS' "AUDIBLE MULTIPLIER SWITCH" WAS POSITIONED "ON" AS REQUIRED BY THE PROCEDURE. CORRECTIVE ACTIONS WILL INCLUDE PROCEDURE ENHANCEMENT TO ELIMINATE MULTIPLE ACTIONS AND REQUIREMENTS WITHIN A PROCEDURE STEP AND TRAINING ON WHAT CONSTITUTES PROPER COMPLETION OF PERIODIC TESTS.

[39] CATAWBA 2 DOCKET 50-414 LER 90-007
POTENTIAL LOSS OF RESIDUAL HEAT REMOVAL SAFETY FUNCTION AS A RESULT OF A LOW VOLTAGE CELL IN DIESEL GENERATOR BATTERY BANK.
EVENT DATE: 081190 REPORT DATE: 090790 NSSS: WE TYPE: PWR

(NSIC 219514) ON AUGUST 11, 1990, WITH UNIT 2 IN MODE 6, REFUELING, DIESEL GENERATOR BATTERY BANK 2A WAS FOUND TO HAVE A CELL BELOW THE TECHNICAL SPECIFICATION REQUIRED VOLTAGE. DIESEL GENERATOR (D/G) 2A WAS DECLARED INOPERABLE AT 1620 HOURS. D/G 2B HAD PREVIOUSLY BEEN DECLARED INOPERABLE FOR THE PERFORMANCE OF OUTAGE MAINTENANCE. TO ENABLE THE REPAIR WORK TO BE IMPLEMENTED, THE BATTERY OUTPUT BREAKER FOR D/G 2A WAS TAGGED OPEN. TECHNICAL SPECIFICATIONS REQUIRE A 125 VDC BATTERY AND CHARGER TO BE CONNECTED TO THE DIESEL GENERATOR CONTROL LOADS. THE DECISION WAS MADE TO JUMPER THE CELL OUT OF SERVICE BY A TEMPORARY STATION MODIFICATION (TSM) INSTEAD OF REPLACING THE CELL WITH A SPARE BATTERY. TECHNICAL SPECIFICATIONS ALSO REQUIRE TWO INDEPENDENT RESIDUAL HEAT REMOVAL (ND) LOOPS TO BE OPERABLE WITH AT LEAST ONE ND LOOP IN OPERATION. THE RESIDUAL HEAT REMOVAL SAFETY FUNCTION WOULD HAVE BEEN TEMPORARILY LOST IN THE EVENT OF A POSTULATED LOSS OF OFF-SITE POWER, BECAUSE NEITHER D/G WAS IMMEDIATELY AVAILABLE AS AN EMERGENCY POWER SOURCE. PERSONNEL WERE PRESENT AT D/G 2A TO START THE DIESEL IF NEEDED. THIS INCIDENT IS ATTRIBUTED TO EQUIPMENT FAILURE DUE TO THE DEGRADATION OF THE BATTERY CELL. THE CELL WAS JUMPERED OUT BY A TSM AND 2A D/G WAS DECLARED OPERABLE AT 2150 HOURS.

[40] CLINTON 1 DOCKET 50-461 LER 90-004 REV 01
UPDATE ON POTENTIAL INABILITY OF SAFETY-RELATED AIR OPERATED VALVES TO FUNCTION DUE TO FAILURE TO FULLY COMMUNICATE AND EVALUATE AIR REGULATOR FULL-OPEN FAILURE.
EVENT DATE: 030790 REPORT DATE: 082990 NSSS: GE TYPE: BWR
VENDOR: ASCO VALVES
FISHER CONTROLS CO.
WATTS REGULATOR COMPANY

(NSIC 219442) INVESTIGATION OF INFORMATION NOTICE 88-24 REVEALED THAT THE MAXIMUM OPERATING PRESSURE DIFFERENTIAL (MOPD) OF 73 SOLENOID OPERATED VALVES (SOVS) SUPPLYING AIR TO ACTIVE SAFETY-RELATED AIR OPERATED VALVES (AOVS) AND DAMPERS WAS LESS THAN THE MAXIMUM INSTRUMENT AIR SYSTEM PRESSURE (MAX IAP). THE SOVS WOULD BE SUBJECTED TO MAX IAP IN THE EVENT OF AIR REGULATOR (AR) FAILURE. DURING THE THIRD PLANNED OUTAGE (PO-3), 3 SOVS WERE FOUND TO BE ACCEPTABLE, 3 SOVS WERE REPLACED, AND A FISHER MODEL 67 AR WAS INSTALLED UPSTREAM OF ANOTHER. AN EVALUATION WAS PERFORMED TO PROVIDE REASONABLE ASSURANCE OF THE RELIABILITY OF FISHER MODEL 67 ARS, ASSOCIATED WITH THE REMAINING SOVS, FROM PO-3 THROUGH THE SECOND REFUELING OUTAGE, WHEN THE SOVS WERE TO BE REPLACED. DURING PREPARATION OF THE FIELD ALTERATION TO REPLACE THE SOVS, THE POTENTIAL OVERPRESSURIZATION OF THE AOV COMPONENTS, AS A RESULT OF AR FAILURE, WAS IDENTIFIED. BOTH OF THE OVERPRESSURIZATIONS (AOV COMPONENTS AND SOVS) MAY RESULT IN FAILURE OF THE AOVS TO REPOSITION TO THEIR SAFETY POSITIONS. BOTH ARE THE RESULT OF VENDOR ERROR AND MISCOMMUNICATION WITH THE ARCHITECT ENGINEER. CORRECTIVE ACTIONS INCLUDE INSTALLING RUPTURE DISKS AND/OR HIGH MOPD SOVS. THE POTENTIAL OVERPRESSURIZATION

OF THE AOVS HAS BEEN DETERMINED TO BE REPORTABLE IN ACCORDANCE WITH THE PROVISIONS OF 10CFR21.

[41] CLINTON 1 DOCKET 50-461 LER 90-015
INADVERTENT SHORTING OF CIRCUIT CARD EDGE CONNECTORS DURING CARD REMOVAL RESULTS IN INITIATION OF NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM ISOLATION SIGNAL AND A HALF SCRAM.
EVENT DATE: 081690 REPORT DATE: 091090 NSSS: GE TYPE: BWR

(NSIC 219319) ON AUGUST 16, 1990, THE PLANT WAS IN POWER OPERATION AT 87 PERCENT REACTOR POWER. CONTROL AND INSTRUMENTATION (C&I) MAINTENANCE TECHNICIANS WERE REPLACING A FAULTY DIGITAL SIGNAL CONDITIONER CIRCUIT CARD IN THE NUCLEAR SYSTEMS PROTECTION SYSTEM. WHILE HAVING DIFFICULTY IN REMOVING THE CARD FROM ITS FRONT PLANE CONNECTOR, A TECHNICIAN INADVERTENTLY SHORTED TOGETHER THE CARD EDGE CONTACTS AND CAUSED A DIVISION II NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NS4) ISOLATION SIGNAL AND A DIVISION II REACTOR PROTECTION SYSTEM SCRAM SIGNAL. THE ISOLATION SIGNAL CAUSED VARIOUS DIVISION II SYSTEM/COMPONENT ACTUATIONS. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO C&I MAINTENANCE TECHNICIAN ERROR. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDE BRIEFING C&I MAINTENANCE TECHNICIANS ON THIS EVENT TO ENHANCE THEIR SENSITIVITY TO CIRCUIT CARD EDGE CONNECTOR REMOVAL/INSERTION AND INCORPORATING CIRCUIT CARD EDGE CONNECTOR REMOVAL TECHNIQUES INTO THE TROUBLESHOOTING TRAINING LESSON PLAN.

[42] COMANCHE 1 DOCKET 50-445 LER 90-019
SOURCE RANGE FLUX DOUBLING ACTUATION DUE TO SPURIOUS SOURCE RANGE SIGNAL DECREASE.
EVENT DATE: 072690 REPORT DATE: 082490 NSSS: WE TYPE: PWR

(NSIC 219383) AT 0929 ON JULY 26, 1990, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 WAS MANUALLY TRIPPED FROM 100 PERCENT POWER AS PART OF THE INITIAL STARTUP TEST PROGRAM. NEUTRON FLUX LEVEL DECREASED AS EXPECTED. AT 0943, BOTH SOURCE RANGE NUCLEAR INSTRUMENT CHANNELS (SR N31 AND SR N32) WERE AUTOMATICALLY ENERGIZED AND WERE INDICATING EXPECTED LEVELS. AT 1005, THE SR N32 SIGNAL BEGAN A GRADUAL DECREASE BELOW EXPECTED LEVELS. AT 1035, SR N32 INDICATED 150 COUNTS PER SECOND (CPS); THE EXPECTED LEVEL WAS 500 CPS AS INDICATED ON SR N31. AT 1036, SR N32 SUDDENLY RETURNED TO THE EXPECTED LEVEL RESULTING IN A SOURCE RANGE FLUX DOUBLING (SRFD) ACTUATION. THE REACTOR OPERATOR IMMEDIATELY ACKNOWLEDGED THE SRFD ACTUATION, BLOCKED AND RESET THE SRFD SIGNAL, AND RESTORED AFFECTED COMPONENTS TO THEIR ORIGINAL POSITION. AFTER HAVING INVESTIGATED THE APPARENT POTENTIAL CAUSES FOR THIS EVENT, THERE HAS BEEN NO REPRODUCIBLE OCCURRENCE WHICH CAN BE IDENTIFIED AS THE CAUSE OF THE EVENT. SINCE THE EVENT, PROPER OPERATION OF SR N32 HAS BEEN OBSERVED, AND TROUBLESHOOTING RESULTS INDICATE THAT SR N32 IS FUNCTIONING CORRECTLY. NO FURTHER PROBLEMS HAVE BEEN FOUND.

[43] COMANCHE 1 DOCKET 50-445 LER 90-020
SAFETY INJECTION ACTUATION RESULTING FROM A SEQUENCING ERROR WHILE HANGING A CLEARANCE ON THE MAIN STEAM ISOLATION VALVES.
EVENT DATE: 072690 REPORT DATE: 082490 NSSS: WE TYPE: PWR

(NSIC 219382) ON JULY 26, 1990, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 WAS IN MODE 3, HOT STANDBY, WITH THE MAIN STEAM ISOLATION VALVES (MSIVS) CLOSED. A CLEARANCE WAS BEING APPLIED TO THE MSIVS IN PREPARATION FOR MAINTENANCE ACTIVITIES ON VARIOUS COMPONENTS IN THE SECONDARY SIDE OF THE PLANT. WHEN CONTROL POWER FUSES WERE PULLED, MSIVS NUMBER 4 AND 2 OPENED; THE RESULTANT RATE COMPENSATED LOW MAIN STEAMLINE PRESSURE CAUSED A SAFETY INJECTION ACTUATION. THE CAUSES OF THE EVENT ARE CLEARANCE PROCESS WEAKNESSES WHICH LED TO A SEQUENCING ERROR IN THE APPLICATION OF THE CLEARANCE. CORRECTIVE ACTIONS INCLUDE PROGRAM ENHANCEMENTS AND TRAINING.

[44] COMANCHE 1 DOCKET 50-445 LER 90-021
SAFETY INJECTION DUE TO WATER ACCUMULATION IN MAIN STEAM LINE.
EVENT DATE: 073090 REPORT DATE: 082990 NSSS: WE TYPE: PWR
VENDOR: FISHER CONTROLS CO.

(NSIC 219440) ON JULY 30, 1990 AT 2137 CDT, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 WAS IN MODE 3, HOT STANDBY. A SAFETY INJECTION ACTUATION OCCURRED FROM RATE COMPENSATED LOW STEAM LINE PRESSURE ACTUATION LOGIC. THE ACTUATION SIGNAL WAS GENERATED BY A RAPID OPENING OF STEAM GENERATOR (SG) NUMBER (NO.) 3 ATMOSPHERIC RELIEF VALVE (ARV) DUE TO ACCUMULATION OF WATER IN MAIN STEAM LINE NO. 3. WATER ACCUMULATION IN THE LINE WAS DUE TO SG NO. 3 BEING STAGNANT FOR 12 HOURS WITH THE MAIN STEAM ISOLATION VALVE, ASSOCIATED BYPASS VALVE AND DRIPPOT DRAIN VALVE CLOSED. THE CAUSE OF THE EVENT WAS IDENTIFIED TO BE WEAKNESSES IN THE WORK ORDER AND CLEARANCE IMPACT REVIEW PROCESS WHICH FAILED TO IDENTIFY THE POSSIBILITY OF WATER ACCUMULATION IN MODE 3 AND AFFECT ON ARV. CORRECTIVE ACTIONS INCLUDE PROCEDURE ENHANCEMENTS.

[45] COMANCHE 1 DOCKET 50-445 LER 90-022
FAILURE TO COMPLY WITH TECHNICAL SPECIFICATION ACTION STATEMENT DUE TO INADEQUATE POST TRIP REVIEW.
EVENT DATE: 080790 REPORT DATE: 090690 NSSS: WE TYPE: PWR

(NSIC 219525) ON 7/26/90 AND 7/30/90, COMANCHE PEAK STEAM ELECTRIC STATION (CPSES) UNIT 1 EXPERIENCED SAFETY INJECTION (SI) ACTUATIONS DUE TO RATE COMPENSATED LOW MAIN STEAM LINE PRESSURE SIGNALS. THE DETAILS OF THE SI ACTUATIONS ARE ADDRESSED IN LER 90-020 AND LER 90-021, RESPECTIVELY. FOLLOWING EACH EVENT, THE TECH SPEC ACTION STATEMENT FOR THE PRESSURIZER HEATUP LIMIT WAS NOT ENTERED. IT WAS NOT REALIZED THAT THE PRESSURIZER HEATUP LIMIT HAD BEEN EXCEEDED UNTIL 8/7/90. THE ROOT CAUSE WAS DETERMINED TO BE AN INADEQUATE POST TRIP REVIEW. PRESSURIZER THERMAL STRATIFICATION WAS IDENTIFIED AS A CONTRIBUTING FACTOR. CORRECTIVE ACTIONS INCLUDE REVISIONS TO STATION PROCEDURES AND TRAINING.

[46] CONNECTICUT YANKEE DOCKET 50-213 LER 90-010
INSTALLED TEST EQUIPMENT RESULTS IN ACTUATION OF REACTOR PROTECTION SYSTEM.
EVENT DATE: 072790 REPORT DATE: 082490 NSSS: WE TYPE: PWR

(NSIC 219386) ON JULY 27, 1990, AT 2150 HOURS, WITH THE PLANT IN MODE 3, (HOT STANDBY) AN INADVERTENT ACTUATION OF THE REACTOR PROTECTION SYSTEM OCCURRED DUE TO A TEST PROCEDURE USED AT AN INAPPROPRIATE TIME THAT INSTALLED TEST PRESSURE TRANSMITTERS IN PLACE OF ALL FOUR CHANNELS OF STEAM LINE BREAK (SLB) CIRCUITRY. THE SLB CIRCUIT WAS, THEREFORE, NOT IN SERVICE AS REQUIRED BY THE TECHNICAL SPECIFICATIONS. WHEN STEAM GENERATOR PRESSURE WAS RAISED TO A POINT WHERE THE SLB OUTPUT WAS AT THE TRIP SETPOINT, THE CIRCUIT ACTUATED A TRIP OF THE REACTOR TRIP BREAKERS AND MAIN STEAM TRIP VALVES. SINCE THE CONTROL RODS WERE ALREADY FULLY INSERTED AND THE FOUR MAIN STEAM LINE TRIP VALVES WERE ALREADY SHUT, THE ONLY PLANT RESPONSE TO THIS EVENT WAS THE OPENING OF THE TRIP BREAKERS. THE ROOT CAUSE OF THIS EVENT WAS THE FAILURE TO IDENTIFY THE INCOMPATIBILITY OF THE TEST PROCEDURE WITH NEW TECHNICAL SPECIFICATION REQUIREMENTS. CORRECTIVE ACTION CONSISTS OF REVISING THE TEST PROCEDURE PRIOR TO THE NEXT USE AND STRENGTHENING THE REVIEW PROCESS TO PREVENT RECURRENCE.

[47] CONNECTICUT YANKEE DOCKET 50-213 LER 90-011
POTENTIAL FOR LOSS OF SUMP RECIRCULATION DUE TO BUS UNDERVOLTAGE.
EVENT DATE: 080290 REPORT DATE: 083090 NSSS: WE TYPE: PWR

(NSIC 219396) ON AUGUST 2, 1990, AT 1115 HOURS, WITH THE PLANT IN MODE 3 (HOT STANDBY) AN ENGINEERING EVALUATION OF EMERGENCY OPERATING PROCEDURES (EOP) DETERMINED THAT RESTARTING A HIGH PRESSURE SAFETY INJECTION (HPSI) PUMP DURING A CERTAIN ACCIDENT SCENARIO COULD RESULT IN THE LOAD SHEDDING OF A RESIDUAL HEAT REMOVAL (RHR) PUMP DUE TO A BUS UNDERVOLTAGE CONDITION. SINCE THE RHR PUMP WOULD BE SUPPLYING THE SUCTION OF THE HPSI PUMP, DAMAGE TO THE HPSI PUMP COULD OCCUR. THE ROOT CAUSE OF THIS EVENT IS A DEFICIENCY IN THE TECHNICAL AND SAFETY REVIEW PROCESS FOR EOP'S. SHORT TERM CORRECTIVE ACTION CONSISTED OF LOWERING THE BUS UNDERVOLTAGE SETPOINTS AND REVIEWING ALL APPROPRIATE EOP'S. THIS REVIEW CONFIRMED THAT THIS CONDITION WAS LIMITED TO TWO PROCEDURES. ADDITIONALLY, THE EOP REVIEW PROCESS WILL BE STRENGTHENED TO PREVENT RECURRENCE. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(V)(D) SINCE A CONDITION EXISTED WHICH ALONE

COULD HAVE PREVENTED THE FULFILLMENT OF A SAFETY FUNCTION OF A SYSTEM NEEDED TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT.

[48] CONNECTICUT YANKEE DOCKET 50-213 LER 90-012
CHARGING PUMP DECLARED INOPERABLE DUE TO GAS BINDING.
EVENT DATE: 080290 REPORT DATE: 083090 NSSS: WE TYPE: PWR

(NSIC 219397) ON AUGUST 2, 1990, AT 1945 HOURS WITH THE PLANT IN MODE 4 (HOT SHUTDOWN) CONTROL ROOM OPERATORS STARTED THE "A" CHARGING PUMP AND SUBSEQUENTLY DETERMINED THROUGH CONTROL BOARD INDICATION THAT THE PUMP WAS POTENTIALLY GAS BOUND. IMMEDIATE CORRECTIVE ACTION WAS TO SHUT DOWN THE PUMP. THE "B" CHARGING PUMP, WHICH WAS RUNNING AT THE TIME, WAS UNAFFECTED. THE ROOT CAUSE OF THIS EVENT IS UNKNOWN. AN ENGINEERING REVIEW OF THE DESIGN OF THE PUMP'S SUCTION VENTS DETERMINED THE VENT SYSTEM TO BE OPERABLE. LONG TERM CORRECTIVE ACTION CONSISTS OF REVIEWING OPERATING PROCEDURES TO DETERMINE IF THEY ARE ADEQUATE TO PREVENT GAS BINDING OF THE PUMPS. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(V)(D) SINCE THIS CONDITION ALONE COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF A SYSTEM NEEDED TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT.

[49] CONNECTICUT YANKEE DOCKET 50-213 LER 90-013
INCORRECT O-RING MATERIAL INSTALLED IN FOUR MOTOR OPERATED VALVES.
EVENT DATE: 080690 REPORT DATE: 090490 NSSS: WE TYPE: PWR
VENDOR: CRANE TELIDYNE CHEM-PUMP DIVISION

(NSIC 219398) ON AUGUST 6, 1990, AT 1010 HOURS, WITH THE PLANT IN MODE 5 (COLD SHUTDOWN) AN ENGINEERING EVALUATION DETERMINED THAT THE INSTALLATION OF INCORRECT O-RING MATERIAL IN FOUR SAFETY RELATED MOTOR OPERATED VALVES (MOV) COULD HAVE RENDERED THE VALVES INOPERABLE. SPECIFICALLY, REPLACEMENT O-RING GREASE SEALS INSTALLED DURING THE PREVIOUS REFUELING OUTAGE WERE MADE OF A MATERIAL NOT SUITABLE FOR USE WITH PETROLEUM OIL PRODUCTS. SWELLING OF THE O-RINGS COULD HAVE RESULTED IN INCREASED FRICTION AND THIS COULD HAVE CAUSED THE MOV'S TO STALL. THE ROOT CAUSE OF THIS EVENT WAS A COMBINATION OF PROCEDURAL INADEQUACIES AND HUMAN ERROR DURING THE PROCUREMENT PROCESS. CORRECTIVE ACTION CONSISTED OF REPLACING THE O-RINGS WITH THOSE OF APPROPRIATE MATERIAL AND REMOVING ALL OTHER INCORRECT O-RINGS FROM STOCK. PROCUREMENT PROCEDURES HAVE BEEN SIGNIFICANTLY UPGRADED SINCE 1986 AND THE CURRENT PROCEDURES WILL PREVENT RECURRENCE. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(V)(D) SINCE THIS CONDITION ALONE COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF A SYSTEM NEEDED TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT.

[50] CONNECTICUT YANKEE DOCKET 50-213 LER 90-014
INCORRECT TORQUE SWITCH SETTINGS ON RCS LOOP ISOLATION VALVES.
EVENT DATE: 080690 REPORT DATE: 090490 NSSS: WE TYPE: PWR
VENDOR: LIMITORQUE CORP.

(NSIC 219399) ON AUGUST 6, 1990, AT 1010 HOURS, WITH THE PLANT IN MODE 5 (COLD SHUTDOWN) AN ENGINEERING EVALUATION DETERMINED THAT THE REACTOR COOLANT LOOP ISOLATION MOTOR OPERATED VALVES (MOV) COULD HAVE BEEN PREVENTED FROM FULLY CLOSING DURING A STEAM GENERATOR TUBE RUPTURE DUE TO THEIR TORQUE SWITCH SETTINGS BEING SET BELOW THE MANUFACTURER'S RECOMMENDED SETPOINT. THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE SETPOINT CONTROL. CORRECTIVE ACTION CONSISTED OF SETTING THE TORQUE SWITCHES TO THE MANUFACTURER'S RECOMMENDED SETTING AND INCLUDING THESE SETPOINTS IN THE SETPOINT CONTROL PROGRAM. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(II)(B) SINCE IT RESULTED IN A CONDITION THAT WAS OUTSIDE THE DESIGN BASIS OF THE PLANT. IT IS ALSO REPORTABLE UNDER 10CFR50.73(A)(2)(V)(C) AND (D) SINCE THIS EVENT ALONE COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF A SYSTEM NEEDED TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT AND CONTROL THE RELEASE OF RADIOACTIVE MATERIAL.

[51] CONNECTICUT YANKEE DOCKET 50-213 LER 90-015
FAILURE TO ADD NEW FIRE DOOR TO SURVEILLANCE PROCEDURE.
EVENT DATE: 081790 REPORT DATE: 091490 NSSS: WE TYPE: PWR

(NSIC 219504) ON 8/17/90, AT 0815 HOURS, WITH THE PLANT IN MODE 1 AT 8% POWER, THE OPERATIONS SHIFT SUPERVISOR DETERMINED THAT A FIRE DOOR IN A NEWLY DESIGNATED FIRE BARRIER WAS NOT BEING INSPECTED ON A WEEKLY BASIS AS REQUIRED BY THE PLANT'S TECHNICAL SPECIFICATIONS. THE FIRE DOOR IS IN A WALL SEPARATING A PORTION OF THE PRIMARY AUXILIARY BUILDING MAIN CORRIDOR FROM THE PRIMARY SIDE AUXILIARY OPERATOR'S OFFICE. THE FIRE DOOR WAS ADDED TO THE LIST OF TECHNICAL SPECIFICATION BARRIERS ON 7/30/90 BUT WAS NOT INSPECTED UNTIL 8/17/90. THE ROOT CAUSE OF THIS EVENT WAS THE FAILURE TO INITIATE ALL PROCEDURE REVISIONS REQUIRED AS A RESULT OF ADDING NEW FIRE DOORS TO THE LIST OF TECH SPEC BARRIERS. IMMEDIATE CORRECTIVE ACTION INCLUDED A VERIFICATION THAT THE DOOR WAS IN ITS PROPER POSITION AND ADDING THE FIRE DOOR TO THE SURVEILLANCE PROCEDURE. THE PROCEDURE WHICH IMPLEMENTS THE FIRE PROTECTION PROGRAM WAS REVISED TO PREVENT RECURRENCE. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B) SINCE IT RESULTED IN A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[52] COOK 1 DOCKET 50-315 LER 90-004 REV 01
UPDATE ON INADVERTENT OPERATION OF THE WRONG CONTROL SWITCH DUE TO PERSONNEL ERROR RESULTED IN OPENING OF THE ICE CONDENSER LOWER INLET DOORS.
EVENT DATE: 050890 REPORT DATE: 083190 NSSS: WE TYPE: PWR

(NSIC 219423) THIS REVISION UPDATES REPORT REGARDING CLASSIFICATION OF THE EVENT AS AN ESF ACTUATION AND RESCHEDULES COMPLETION DATE FOR PREVENTIVE ACTION. ON MAY 8, 1990, AT 1555, A REACTOR OPERATOR INADVERTENTLY STARTED A CONTAINMENT RECIRCULATION FAN (CEQ) INSTEAD OF THE INTENDED HYDROGEN RECOMBINER DURING A SURVEILLANCE TEST. THE FAN OPERATION CAUSED SUFFICIENT DIFFERENTIAL PRESSURE ACROSS THE ICE CONDENSER LOWER INLET DOORS TO OPEN THEM. VARIOUS VENTILATION ALIGNMENTS WERE ATTEMPTED TO RECLOSE THE INLET DOORS, BUT WITHOUT SUCCESS. THE ICE CONDENSER WAS DECLARED INOPERABLE AT 2129 WHEN IT WAS DETERMINED THAT THE TS 3.6.5.1 MAXIMUM ICE BED TEMPERATURE OF 27F HAD BEEN EXCEEDED. POWER WAS DECREASED TO EIGHT PERCENT AT 1125 ON MAY 9, 1990, TO ALLOW LOWER CONTAINMENT ENTRY AND MANUAL CLOSURE OF THE LOWER ICE CONDENSER INLET DOORS. THE INLET DOORS WERE CLOSED AND DECLARED OPERABLE AT 1248. THE ICE BED TEMPERATURES WERE DETERMINED TO BE WITHIN THE TS LIMITS AT 1325 ON MAY 19, 1990. THEREFORE, THE TS 3.6.5.1 REQUIREMENT TO RESTORE THE ICE BED TO OPERABLE STATUS WITHIN 48 HOURS WAS MET. THE REQUIRED WORK PRACTICES TO PREVENT RECURRENCE OF A SIMILAR EVENT WERE REVIEWED WITH THE INVOLVED REACTOR OPERATOR. MANAGEMENT'S EXPECTATIONS CONCERNING WORK PRACTICES TO PREVENT SIMILAR EVENTS WILL BE COMMUNICATED TO PLANT PERSONNEL BY DECEMBER 6, 1990.

[53] COOK 1 DOCKET 50-315 LER 90-008
DEFICIENCIES RESULTING IN POTENTIAL FOR LOSS OF AUTO START OF SERVICE WATER PUMPS DUE TO INCORRECT IMPLEMENTATIONS OF DESIGN CHANGE.
EVENT DATE: 060190 REPORT DATE: 083190 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)

(NSIC 219426) ON JUNE 19, 1990 WITH UNIT ONE OPERATING AT 83 PERCENT POWER AND UNIT TWO OPERATING AT 90 PERCENT POWER, IT WAS DISCOVERED THAT THE ISOLATION RELAY CIRCUITRY FOR THE LOW HEADER PRESSURE AUTO START SWITCH FOR THE ESSENTIAL SERVICE WATER (ESW) PUMPS HAD BEEN INSTALLED INCORRECTLY. BECAUSE OF A FUSE COORDINATION PROBLEM BETWEEN THE PRESSURE SWITCH AND THE REMAINDER OF THE CONTROL CIRCUIT, IT WAS POSSIBLE FOR A PRESSURE SWITCH WIRING SHORT TO DISABLE THE ESW PUMPS AUTO START CIRCUIT. ON JUNE 20, 1990, THE SAME CONDITION WAS FOUND FOR THE ISOLATION RELAY CIRCUIT FOR THE LOW HEADER PRESSURE AUTO START SWITCH FOR THE COMPONENT COOLING WATER PUMPS. APPENDIX R OF 10 CFR 50 REQUIRES THAT WHEN CABLES OF REDUNDANT EQUIPMENT NECESSARY TO ACHIEVE AND MAINTAIN HOT SHUTDOWN CONDITIONS ARE LOCATED IN THE SAME AREA, STEPS MUST BE TAKEN TO ENSURE THAT ONE OF THE REDUNDANT TRAINS IS FREE OF FIRE DAMAGE. THE INSTALLED CONFIGURATION FOR THE PRESSURE SWITCH ISOLATION CIRCUITRY DID NOT MEET THIS REQUIREMENT. THE IMMEDIATE CORRECTIVE ACTION TAKEN WAS TO REPLACE THE ISOLATION RELAY CIRCUIT FUSES WITH

SMALLER VALUE FUSES TO PROVIDE PROPER FUSE COORDINATION. A PLANT MODIFICATION PACKET HAS BEEN INITIATED TO MODIFY THE CIRCUITS TO THE CORRECT CONFIGURATION.

[54] COOK 1 DOCKET 50-315 LER 90-005
TECHNICAL SPECIFICATION CALIBRATION INTERVAL EXCEEDED DUE TO INCORRECT ENTRY INTO COMPUTERIZED SCHEDULER PROGRAM.
EVENT DATE: 072490 REPORT DATE: 082390 NSSS: WE TYPE: PWR

(NSIC 219350) DURING A ROUTINE QA AUDIT COMPLETED ON JULY 24, 1990, IT WAS DISCOVERED THAT THE TECHNICAL SPECIFICATION (T/S) TIME LIMIT OF 18 MONTHS, PLUS 25 PERCENT GRACE PERIOD HAD BEEN EXCEEDED BY 28 DAYS BETWEEN JULY 7, 1986 AND JUNE 20, 1988 FOR CALIBRATION OF THE AUXILIARY BUILDING VENTILATION SYSTEM UNIT VENT SAMPLER FLOW RATE MEASURING DEVICE (VFS-1521). PRIOR TO BECOMING OVERDUE, THE SCHEDULER IDENTIFIED THE NEED FOR RECALIBRATION, BUT DID NOT ADEQUATELY IDENTIFY IT AS BEING REQUIRED TO SATISFY A T/S REQUIREMENT. ALTHOUGH THE EXACT REASON FOR DELAY OF THE CALIBRATION COULD NOT BE DETERMINED, IT IS BELIEVED TO HAVE BEEN ALLOWED DUE TO REVIEW OF THE SCHEDULER-SUPPLIED INFORMATION. ONCE CALIBRATED, SCHEDULER UPDATING WOULD HAVE NOTED THE MISSED DUE DATE, BUT NOT THE RELATIONSHIP TO T/S COMPLIANCE. THE CALIBRATION WAS COMPLETED ON JUNE 20, 1988. NO ADVERSE CONDITIONS HAD RESULTED DUE TO THE EXTENDED INTERVAL AS THE INSTRUMENT WAS FOUND WITHIN ACCEPTABLE TOLERANCES. DURING REVIEW OF THE SCHEDULER PROGRAM FOR SIMILAR PROBLEMS, THE EQUIVALENT INSTRUMENT FOR THE UNIT 2 VENT SYSTEM, VFS-2521, WAS ALSO FOUND TO BE INADEQUATELY IDENTIFIED TO T/S REQUIREMENTS, BUT THE CALIBRATIONS HAD BEEN DONE ON TIME. THE SCHEDULER WAS REPROGRAMMED.

[55] COOK 1 DOCKET 50-315 LER 90-006
INADEQUATE EMERGENCY LIGHTING DUE TO FAILURE TO IDENTIFY NECESSITY FOR REVISION.
EVENT DATE: 073090 REPORT DATE: 082890 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)

(NSIC 219424) SINCE JANUARY, 1990, THREE SEPARATE WALKDOWNS WERE PERFORMED TO DETERMINE THE ADEQUACY OF THE EMERGENCY LIGHTING ASSOCIATED WITH APPENDIX R. AS A RESULT OF THESE WALKDOWNS, IT WAS DETERMINED THAT IN SOME AREAS THE EMERGENCY LIGHTING HAD TO BE IMPROVED TO FACILITATE THE ACCOMPLISHMENT OF THE EMERGENCY REMOTE SHUTDOWN (ERS) PROCEDURES. AS CORRECTIVE ACTION, AN EXPEDITED DESIGN CHANGE WAS ISSUED TO INSTALL ADDITIONAL EMERGENCY LIGHTING BY AUGUST 30, 1990. IN ADDITION, AS AN INTERIM MEASURE, MINER'S HARD HATS WITH BATTERY-POWERED LIGHTS WERE ORDERED AND ARE NOW PLACED NEAR THE CONTROL ROOMS FOR UNITS 1 AND 2. THE PREVENTIVE ACTIONS TAKEN TO PRECLUDE RECURRENCE OF THE EVENT WILL INCLUDE A DETAILED REVIEW OF ALL FUTURE CHANGES TO THE ERS PROCEDURES FOR THEIR IMPACT ON THE APPENDIX R EMERGENCY LIGHTING. THE EVENT DOES NOT HAVE ANY SIGNIFICANT SAFETY CONSEQUENCES OR IMPLICATIONS BECAUSE IT IS BELIEVED THAT, BY THE USE OF FLASHLIGHTS IN THE AREAS OF INADEQUATE LIGHTING, THE ERS PROCEDURES CAN BE FOLLOWED DURING AN APPENDIX R FIRE. THEREFORE, THE EVENT DOES NOT PRESENT A SIGNIFICANT HAZARD TO THE PUBLIC HEALTH AND SAFETY.

[56] COOK 1 DOCKET 50-315 LER 90-007
FAILURE TO INSTALL FIREPROOFING PER APPENDIX A TO BRANCH TECHNICAL POSITION 9.5-1, AS A RESULT OF INADEQUATE CONTROLS.
EVENT DATE: 082290 REPORT DATE: 083190 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)

(NSIC 219425) ON AUGUST 2, 1990, IT WAS DETERMINED THAT DUE TO THE LACK OF FIREPROOFING MATERIAL ON EXPOSED STRUCTURAL STEEL, WITHIN FIVE LUBE OIL STORAGE ROOMS, A CONDITION EXISTS THAT IS IN VIOLATION OF APPENDIX A OF BTP 9.5-1. RECORDS INDICATE THAT FIREPROOFING MATERIAL WAS NEVER APPLIED DURING CONSTRUCTION AS REQUIRED. THE CAUSE FOR NOT INITIALLY INSTALLING THE FIREPROOFING COULD NOT BE DECISIVELY DETERMINED. SUBSEQUENT DESIGN CHANGE MODIFICATIONS AND UPDATING OF THE FIRE PROTECTION DOCUMENTS DID NOT CAPTURE THIS DEFICIENCY. CURRENT CONTROLS REGARDING THE DESIGN CHANGE PROCESS AND UPDATING OF THE FIRE PROTECTION DOCUMENTS WILL PREVENT RECURRENCE; THEREFORE, NO PREVENTIVE ACTION IS PLANNED. CORRECTIVE ACTION INVOLVES THE INSTALLATION OF THE REQUIRED FIREPROOFING MATERIAL AND UPDATING OF DRAWINGS. UNTIL THE FIREPROOFING MATERIAL IS INSTALLED, THE PLANT

WILL RELY ON THE AUTOMATIC FIRE DETECTION AND SUPPRESSION SYSTEMS FOUND IN THE ROOMS, COUPLED WITH THE RESPONSE OF THE FIRE BRIGADE, TO PRECLUDE ANY JEOPARDY TO THE HEALTH AND SAFETY OF THE PUBLIC.

[57] COOPER DOCKET 50-298 LER 90-009
INOPERABILITY OF THE REACTOR CORE ISOLATION COOLING SYSTEM DUE TO FAILURE OF THE MOTOR OPERATED STEAM SUPPLY VALVE ATTRIBUTED TO SPRING PACK HYDRAULIC LOCK.
EVENT DATE: 080890 REPORT DATE: 083090 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 219418) AT APPROX. 3:00 PM, ON 8/8/90, WITH THE PLANT OPERATING AT FULL POWER UNDER NORMAL STEADY STATE CONDITIONS, RCIC STEAM SUPPLY VALVE, RCIC-MOV-M0131 FAILED IN ITS NORMALLY CLOSED POSITION UPON BEING CLOSED FOLLOWING SUCCESSFUL COMPLETION OF THE PUMP OPERABILITY SURVEILLANCE TEST. THE RCIC SYSTEM WAS OUT OF SERVICE AS PART OF A PREPLANNED EVOLUTION TO ENABLE TROUBLESHOOTING AND CORRECTION OF AN UNRELATED PROBLEM. THE SYSTEM WAS NOT CONSIDERED TO BE OPERABLE. SURVEILLANCE TEST WAS BEING CONDUCTED AS A POST MAINTENANCE TEST, PRIOR TO RETURNING THE SYSTEM TO AN OPERABLE CONDITION. INVESTIGATION REVEALED THAT THE MOTOR FOR THE MOTOR OPERATED VALVE (MOV) HAD FAILED DUE TO FAILURE OF THE CLOSE TORQUE SWITCH TO OPEN. APPARENT CAUSE OF THE MOTOR FAILURE APPEARS TO HAVE BEEN DUE TO HYDRAULIC LOCK OF THE SPRING PACK. THIS IS BASED UPON AN INSPECTION OF THE SPRING PACK CAVITY WHICH WAS FOUND TO BE FULL OF GREASE. IT IS POSTULATED THAT THE GREASE PREVENTED SPRING PACK COMPRESSION, RESULTING IN THE TORQUE SWITCH REMAINING CLOSED WHICH SUBSEQUENTLY CAUSED MOTOR FAILURE. THE MOTOR WAS REPLACED, THE SPRING PACK CLEANED, AND THE TORQUE SWITCHES WERE VERIFIED TO OPERATE PROPERLY. THIS IS THE FIRST INSTANCE OF FAILURE OF A MOV AT CNS THAT COULD POTENTIALLY BE TRACED TO A HYDRAULIC LOCK CONDITION IN THE SPRING PACK.

[58] COOPER DOCKET 50-298 LER 90-010
TWO LOCAL POWER RANGE MONITOR DETECTOR OUTPUTS DISCOVERED REVERSED DURING REVIEW OF INCORE FLUX MEASUREMENTS.
EVENT DATE: 081490 REPORT DATE: 091390 NSSS: GE TYPE: BWR

(NSIC 219539) ON AUGUST 14, 1990, FOLLOWING A REVIEW OF INCORE FLUX MEASUREMENTS, IT WAS DETERMINED THAT TWO LOCAL POWER RANGE MONITOR (LPRM) DETECTORS WERE DISPLAYING REVERSED INDICATIONS. A REVIEW OF ADDITIONAL INFORMATION, TO DETERMINE ANY ADVERSE CONSEQUENCES, INDICATED THAT THE NUMBER OF LPRM INPUTS TO ELEVATION B OF AVERAGE POWER RANGE MONITOR (APRM) CHANNEL D WAS REDUCED TO ONE FROM JUNE 30 THROUGH JULY 6, 1989. ONLY ONE DETECTOR FOR ELEVATION C OF APRM CHANNEL F WAS OPERABLE FROM JUNE 18, 1989 THROUGH MAY 16, 1990. THESE CONDITIONS VIOLATE THE TECHNICAL SPECIFICATION REQUIREMENTS. UPON DISCOVERY, THE TWO LPRM DETECTORS WERE BYPASSED, AND ALL OTHER DETECTORS WERE VERIFIED TO BE OPERATING PROPERLY. IT HAS BEEN VERIFIED THAT NO THERMAL LIMITS WERE VIOLATED. THE DETECTOR STRING WAS REPLACED DURING THE RELOAD 12 REFUELING OUTAGE WHICH ENDED JUNE 16, 1989. THE CAUSE OF THIS EVENT CANNOT BE DETERMINED UNTIL DRYWELL ACCESS IS PERMITTED. IN THE INTERIM, ADMINISTRATIVE CONTROLS FOR MAINTENANCE AND RECEIPT INSPECTION ON THE LPRMS WILL BE REVIEWED AND ENHANCED AS REQUIRED. A SUPPLEMENTAL REPORT WILL BE SUBMITTED WHEN THE ROOT CAUSE HAS BEEN DETERMINED. ACCORDING TO THE GENERAL ELECTRIC COMPANY, THE REVERSED DETECTOR OUTPUTS MAY HAVE SLIGHTLY REDUCED THE MARGIN ASSUMED IN THE APRM CHANNEL RESPONSE TO A HIGH FLUX CONDITION. THIS REDUCTION IS NOT CONSIDERED SIGNIFICANT.

[59] CRYSTAL RIVER 3 DOCKET 50-302 LER 90-001 REV 01
UPDATE ON PLANT SHUTDOWN REQUIRED BY TECHNICAL SPECIFICATIONS DUE TO EXCESSIVE REACTOR COOLANT SYSTEM UNIDENTIFIED LEAKAGE CAUSED BY VALVE PACKING FAILURE.
EVENT DATE: 012290 REPORT DATE: 082890 NSSS: BW TYPE: PWR
VENDOR: VELAN VALVE CORP.

(NSIC 219348) CRYSTAL RIVER UNIT 3 WAS OPERATING IN MODE 1 (POWER OPERATION) AT 98% FULL POWER ON JANUARY 22, 1990. REACTOR COOLANT SYSTEM (RCS) LEAKAGE CALCULATIONS COMPLETED EARLIER THAT DAY SHOWED THAT UNIDENTIFIED LEAKAGE WAS 0.3 GPM. AT 1105, REACTOR COOLANT SYSTEM LEAKAGE CALCULATIONS INDICATED THAT UNIDENTIFIED LEAKAGE HAD INCREASED TO 1.3 GPM. THIS VALUE EXCEEDED TECHNICAL

SPECIFICATION LIMITS. AT 1203 OPERATORS BEGAN PLANT SHUTDOWN DUE TO EXCESS LEAKAGE. THE PLANT ENTERED AN UNUSUAL EVENT DUE TO EXCESS RCS UNIDENTIFIED LEAKAGE, IN ACCORDANCE WITH THE PLANT EMERGENCY PLAN. PLANT SHUTDOWN WAS COMPLETED AT 1350. AT 1700, OPERATORS ISOLATED THE LEAK. THE SOURCE OF RCS LEAKAGE WAS IDENTIFIED AS FAILED PACKING ON THE BLOCK VALVE ASSOCIATED WITH THE PILOT OPERATED RELIEF VALVE. THE ROOT CAUSE FOR THE PACKING FAILURE WAS DETERMINED TO BE EXCESSIVE TORQUE APPLIED TO THE PACKING FLANGE GLAND BOLTS.

[60] CRYSTAL RIVER 3 DOCKET 50-302 LER 90-012 REV 01
UPDATE ON PERSONNEL ERROR LEADS TO INCOMPLETE QUARTERLY SURVEILLANCE AND TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 070990 REPORT DATE: 092890 NSSS: BW TYPE: PWR

(NSIC 219509) ON 7/9/90, WHILE OPERATING AT 98% POWER, FLORIDA POWER CORPORATION DISCOVERED THE QUARTERLY CALIBRATION OF THE HYDROGEN CHANNEL OF THE WASTE GAS DECAY TANK (WGDT) EXPLOSIVE GAS MONITORING INSTRUMENTATION HAD NOT BEEN PERFORMED AS REQUIRED BY TECH SPEC 4.3.3.10 PRIOR TO DECLARING THE SYSTEM OPERABLE ON 6/2/90. THE IMMEDIATE CAUSE OF THIS NONCONFORMANCE WAS PERSONNEL ERROR. CRYSTAL RIVER UNIT 3 TECHNICIANS WERE NOT ADEQUATELY NOTIFIED OF THE NEED TO PERFORM THE QUARTERLY SURVEILLANCE. ADDITIONALLY, THE CALIBRATION THAT WAS PERFORMED WAS INCORRECTLY COMMUNICATED TO THE OPERATING SHIFT AS A COMPLETE CALIBRATION. TO PREVENT RECURRENCE, SURVEILLANCE PROCEDURES (SPS), OR OTHER TESTS USED TO RETURN EQUIPMENT TO SERVICE, WILL BE REVIEWED BY THE RESPONSIBLE PROCEDURE SUPERVISOR PRIOR TO RETURNING THE EQUIPMENT TO OPERABLE STATUS.

[61] CRYSTAL RIVER 3 DOCKET 50-302 LER 90-013
PERSONNEL ERROR LEADS TO A MISSED HYDROSTATIC TEST RESULTING IN A TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 080390 REPORT DATE: 090490 NSSS: BW TYPE: PWR

(NSIC 219419) ON NOVEMBER 16, 1989, CRYSTAL RIVER UNIT 3 WAS IN MODE 1 (POWER OPERATIONS). THE AUTHORIZED NUCLEAR INSERVICE INSPECTOR DISCOVERED THAT A WORK REQUEST PACKAGE TO REPLACE A DECAY HEAT CLOSED CYCLE COOLING SYSTEM HEAT EXCHANGER DRAIN VALVE (DCV-181) (BI,SHV) DID NOT INCLUDE DOCUMENTATION OF A HYDROSTATIC TEST. A NONCONFORMANCE WAS INITIATED AND EVALUATED AS NON-REPORTABLE, SINCE IT WAS NOT KNOWN IF THE TEST RESULTS WERE MISPLACED OR THE TEST WAS NOT COMPLETED. SUBSEQUENT INVESTIGATION DETERMINED THAT THE TEST HAD NOT BEEN PERFORMED. THE HYDROSTATIC TEST WAS THEN COMPLETED, BUT THE NONCONFORMANCE WAS RE-EVALUATED FOR REPORTABILITY. ON AUGUST 3, 1990, THE NONCONFORMANCE WAS RE-EVALUATED AS A VIOLATION OF PLANT TECHNICAL SPECIFICATIONS, REPORTABLE IN ACCORDANCE WITH 10CFR50.73(A)(2)(I)(B). THIS INCIDENT IS ATTRIBUTABLE TO PERSONNEL ERROR. THE MAINTENANCE SUPERVISOR DID NOT CONTACT THE TEST GROUP AFTER COMPLETING THE WORK. THIS ACTION VIOLATES PROCEDURE REQUIREMENTS. A WORK REQUEST WAS WRITTEN TO PERFORM THE REQUIRED HYDROSTATIC TEST ON DCV-181 (BI,SHV) AND THE TEST WAS COMPLETED ON JANUARY 16, 1990. ADDITIONALLY, THE MECHANICAL MAINTENANCE SHOP HAS ASSUMED THE RESPONSIBILITY FOR PERFORMING POST-MAINTENANCE HYDROSTATIC TESTS.

[62] DAVIS-BESSE 1 DOCKET 50-346 LER 90-010 REV 01
UPDATE ON INADVERTENT SAFETY FEATURES ACTUATION WITH INJECTION OF 1,000 GALLONS OF BORATED WATER.
EVENT DATE: 051890 REPORT DATE: 081790 NSSS: BW TYPE: PWR

(NSIC 219361) ON MAY 18, 1990, AT 2121 HOURS, THE STATION EXPERIENCED AN INADVERTENT SFAS LEVEL 1 THROUGH 4 ACTUATION INCLUDING THE INJECTION OF 1,000 GALLONS OF WATER FROM THE BORATED WATER STORAGE TANK (BWST) INTO THE REACTOR VIA LOW PRESSURE INJECTION (LPI) TRAIN 1-2. THE PLANT WAS IN MODE 5. AT 2126 HOURS, THE ACTUATION WAS BLOCKED. BY 2129 HOURS, DECAY HEAT REMOVAL FLOW WAS RE-ESTABLISHED. THE CAUSE OF THE ACTUATION WAS THE LOSS OF 120VAC BUS Y3 WHICH DE- ENERGIZED SFAS CHANNEL 3 WITH SFAS CHANNEL 1 HIGH CONTAINMENT PRESSURE MODULE PREVIOUSLY REMOVED FOR MAINTENANCE. Y3 DE-ENERGIZED UNEXPECTEDLY WHILE COMPLETING THE PREREQUISITES FOR DB-SC-04053, 4160 V SYSTEM TRANSFER AND LOCKOUT TEST-BUSES C1 AND C2. BUS Y3 WAS LOST DUE TO THE LACK OF A FORMAL TURNOVER FOR

THE COMPLETION OF A PHASE OF ONGOING MODIFICATION (MOD)86-0272. UNCERTAINTIES BY OPERATIONS PERSONNEL CONCERNING ACTUAL PLANT CONFIGURATION ALSO CONTRIBUTED TO THE EVENT. APPLICABLE PLANT PERSONNEL INVOLVED IN THE MODIFICATION PROCESS WILL BE RETRAINED ON THE APPLICABLE REQUIREMENTS FOR IMPLEMENTING MODIFICATIONS. ADDITIONAL ENHANCEMENTS WILL INCLUDE A WRITTEN TURNOVER ACKNOWLEDGMENT BY SHIFT PERSONNEL. APPROPRIATE IMMEDIATE NOTIFICATIONS TO THE NRC STATE AND LOCAL OFFICIALS WERE MADE.

[63] DAVIS-BESSE 1 DOCKET 50-346 LER 90-013
18 MONTH FIRE BARRIER INSPECTIONS NOT COMPLETED DUE TO PROCEDURAL DEFICIENCIES.
EVENT DATE: 072390 REPORT DATE: 082290 NSSS: BW TYPE: PWR

(NSIC 219362) ON JULY 23, 1990, FIRE PROTECTION PERSONNEL, PERFORMING A REVIEW OF THE 18 MONTH FIRE BARRIER TEST PROCEDURES, DETERMINED THAT THE TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS 4.7.10A AND 4.7.10C WERE NOT BEING COMPLETELY SATISFIED. DB-FP-03023, THE 18 MONTH RATED BARRIER VISUAL INSPECTION, AND DB-FP-03025, THE 18 MONTH TEN PERCENT PENETRATION SEAL VISUAL INSPECTION, ALLOW BARRIERS TO BE EXEMPTED FROM INSPECTION WHEN UNRESOLVABLE INACCESSIBILITIES OCCUR FOR ALARA OR OTHER REASONS. THE TECHNICAL SPECIFICATION DOES SPECIFICALLY EXEMPT THREE BARRIERS THAT CANNOT BE INSPECTED ON EITHER SIDE, BUT IT DOES NOT PROVIDE GUIDANCE FOR THE ONE-SIDED INSPECTIONS THAT ARE ALLOWED BY THE INSPECTION PROCEDURES IN CERTAIN INSTANCES. THE EXEMPTION OF THE ANNULUS SIDE OF THE SHIELD BUILDING BARRIERS IN THE PROCEDURES HAS BEEN DETERMINED TO BE INCORRECT. ALSO, DURING THE PERFORMANCE OF THE LAST TEST, ONE-SIDED INSPECTIONS WERE ALLOWED INCORRECTLY FOR OTHER BARRIERS THAT WERE NOT DEEMED ACCESSIBLE AT THAT TIME. THE BARRIERS IN QUESTION WERE DECLARED INOPERABLE AND APROPRIATE FIRE WATCHES ESTABLISHED. A LETTER HAS BEEN SENT TO THE NRC ON AUGUST 15, 1990 TO OBTAIN THEIR CONCURRENCE WITH TOLEDO EDISON'S CLARIFICATION OF THE CRITERIA FOR ONE-SIDED INSPECTIONS. IN ADDITION, THE 18 MONTH INSPECTION PROCEDURES WILL BE REVISED TO REFLECT THE CLARIFICATION.

[64] DIABLO CANYON 2 DOCKET 50-323 LER 90-005 REV 01
UPDATE ON CLOSURE OF THE MAIN STEAMLINE BYPASS VALVES DUE TO INADEQUATE BACKFILLING OF TRANSMITTER SENSING LINES.
EVENT DATE: 042290 REPORT DATE: 091490 NSSS: WE TYPE: PWR

(NSIC 219450) ON APRIL 22, 1990, AT 0536 PDT, WITH UNIT 2 IN MODE 4 (HOT SHUTDOWN), AN INADVERTENT MAIN STEAMLINE ISOLATION SIGNAL WAS GENERATED, WHICH RESULTED IN CLOSURE OF THE MAIN STEAMLINE ISOLATION BYPASS VALVES. THIS ACTION CONSTITUTED AN ENGINEERED SAFETY FEATURE ACTUATION. THE 4-HOUR, NON-EMERGENCY REPORT REQUIRED BY 10 CFR 50.72(B)(2)(II) WAS MADE ON APRIL 22, 1990, AT 0710 PDT. THE MAIN STEAMLINE ISOLATION SIGNAL OCCURRED DUE TO ACTUATION OF TWO OUT OF FOUR STEAMLINE HIGH STEAM FLOW BISTABLES WHEN BOTH STEAM LOOP 2 AND LOOP 3 TRIPPED UPON REACHING THEIR SETPOINTS WITH NEGLIGIBLE STEAM FLOW IN THE STEAMLINES. THE FLOW TRANSMITTERS PROVIDING INPUT TO THE BISTABLES WERE FOUND TO BE READING HIGHER THAN ACTUAL STEAM FLOW DUE TO VOIDS IN THE SENSING LINES OF THE TRANSMITTERS. THE CAUSE OF THE VOIDS IN THE SENSING LINES COULD NOT BE POSITIVELY IDENTIFIED. THE MOST LIKELY CAUSE WAS DETERMINED TO BE INADEQUATE BACKFILLING OF THE SENSING LINES AFTER THE REPLACEMENT OF THE FLOW TRANSMITTERS DURING THE REFUELING OUTAGE. TO REDUCE FUTURE OCCURRENCES OF THIS EVENT, STP I-12B7, "CALIBRATION TRANSMITTERS STEAM GENERATOR STEAM FLOW," WILL BE REVISED TO INCLUDE A REQUIREMENT TO BACKFILL THE TRANSMITTERS AND SENSING LINES AFTER EACH CALIBRATION.

[65] DRESDEN 2 DOCKET 50-237 LER 90-005
UNPLANNED PRIMARY CONTAINMENT GROUP V ISOLATION DUE TO PROCEDURE DEFICIENCY.
EVENT DATE: 073090 REPORT DATE: 082490 NSSS: GE TYPE: BWR

(NSIC 219401) ON JULY 30, 1990 AT 1653 HOURS WITH UNIT 2 IN THE RUN MODE AT 84% OF RATED CORE THERMAL POWER, WHILE REPLACING A BURNED OUT LIGHT BULB ON THE CONTROL ROOM POSITION INDICATION FOR LOW PRESSURE COOLANT INJECTION (LPCI) SYSTEM INBOARD MANUAL ISOLATION VALVE 2-1501-26A, THE LIGHT BULB CAUSED A SHORT CIRCUIT AND CAUSED SUPPLY POWER FUSE 595-714B TO OPEN. SIMULTANEOUSLY, A PRIMARY

CONTAINMENT GROUP V ISOLATION WAS RECEIVED, AND THE APPROPRIATE ISOLATION CONDENSER ISOLATION VALVES CLOSED AS DESIGNED. THE CAUSE OF THIS EVENT WAS ATTRIBUTED TO INADVERTENT INSTALLATION OF A BULB OF IMPROPER VOLTAGE RATING DUE TO PROCEDURE DEFICIENCY. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL SINCE THE ISOLATION SIGNAL WAS RESET IN APPROXIMATELY 17 MINUTES AND ALL ACTIVE COMPONENTS OF THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM REMAINED OPERABLE DURING THE EVENT. TO PREVENT RECURRENCE OF THIS EVENT, IMPROVEMENTS TO ADMINISTRATIVE CONTROLS FOR ISSUANCE OF REPLACEMENT BULBS WERE IMPLEMENTED, AND THE BULB CHANGE-OUT PROCEDURE WAS REVISED. A PREVIOUS EVENT INVOLVING AN UNPLANNED PRIMARY CONTAINMENT GROUP V ISOLATION WAS REPORTED BY LER 89-021 ON DOCKET 050237.

[66] DRESDEN 2 DOCKET 50-237 LER 90-006
TARGET ROCK SAFETY-RELIEF VALVE FAILS OPEN DUE TO A STEAM CUT PILOT VALVE DISC.
EVENT DATE: 080290 REPORT DATE: 082890 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 219402) ON AUGUST 2, 1990, DURING NORMAL POWER OPERATION IN THE RUN MODE, THE 2-203-3A MAIN STEAM TARGET ROCK SAFETY-RELIEF VALVE (TRSRV) ACOUSTIC MONITOR ALARMED, INDICATING THAT THE VALVE HAD SPURIOUSLY OPENED AND WAS RELIEVING REACTOR PRESSURE TO THE SUPPRESSION CHAMBER. THE REACTOR WAS SUBSEQUENTLY MANUALLY SCRAMMED FROM 87% POWER AT 0116 HOURS. ALL CONTAINMENT COOLING SERVICE WATER AND LOW PRESSURE COOLANT INJECTION PUMPS WERE MANUALLY STARTED FOR MAXIMUM SUPPRESSION CHAMBER COOLING. THE MAXIMUM AVERAGE COOLDOWN RATE WHEN AVERAGED OVER A ONE HOUR PERIOD REACHED 129.3 DEGREES F/HR, AND MAXIMUM BULK SUPPRESSION CHAMBER TEMPERATURE WAS 122 DEGREES F. THE OPENING OF THE TRSRV WAS APPARENTLY CAUSED BY STEAM CUTS ON THE FIRST STAGE PILOT VALVE DISC. ANALYSES WERE PERFORMED TO VERIFY THAT THE COOLDOWN RATE AND THE BULK SUPPRESSION CHAMBER TEMPERATURE ATTAINED DURING THIS EVENT WERE WITHIN DESIGN LIMITS. A SATISFACTORILY LEAK TESTED, REBUILT TRSRV WAS INSTALLED. THE TECHNICAL STAFF WILL MONITOR THE TRSRV TAIL PIPE TEMPERATURES TO VERIFY PROPER PILOT VALVE OPERATION. IN ADDITION, ANY TRSRV PILOT VALVE OF GREATER THAN EIGHT MONTHS SERVICE WILL BE REPLACED DURING FUTURE SHORT UNIT OUTAGES WITH PRIMARY CONTAINMENT DRYWELL ACCESSIBILITY. THE TRSRVS ARE ROUTINELY REPLACED AT EACH REFUEL OUTAGE. A PREVIOUS TRSRV FAILURE EVENT WAS REPORTED BY LER 50-237/76-34.

[67] DRESDEN 2 DOCKET 50-237 LER 90-007
UNPLANNED PRIMARY CONTAINMENT GROUP V ISOLATION DUE TO UNKNOWN CAUSE.
EVENT DATE: 080290 REPORT DATE: 082890 NSSS: GE TYPE: BWR

(NSIC 219403) ON AUGUST 2, 1990 AT 0741 HOURS WITH UNIT 2 IN THE SHUTDOWN MODE WITH ALL CONTROL RODS FULLY INSERTED AND REACTOR WATER TEMPERATURE AT APPROXIMATELY 180 DEGREES F, AN UNPLANNED PRIMARY CONTAINMENT GROUP V ISOLATION OCCURRED. NO ABNORMALITIES, OPEN FUSES OR OTHER ELECTRICAL PROBLEMS WERE FOUND IN THE CIRCUITRY AND THE ISOLATION SIGNAL WAS RESET AFTER VERIFICATION THAT THE SIGNAL WAS SPURIOUS. OPERATIONS PERSONNEL WERE ALSO DISPATCHED TO THE AREA OF THE DIFFERENTIAL PRESSURE INSTRUMENTATION WHICH INITIATES THE ISOLATION. NO PERSONNEL WERE IDENTIFIED AS HAVING INADVERENTLY JARRED THE INSTRUMENTS IN QUESTION; HOWEVER THIS INSTRUMENTATION IS BELIEVED TO BE VIBRATION SENSITIVE. AS A CORRECTIVE ACTION, WARNING SIGNS WERE PLACED ON THE INSTRUMENT RACK TO CAUTION AGAINST SENSITIVITY OF THE FLOW SWITCHES TO VIBRATION. VIBRATION TESTING OF THIS INSTRUMENT RACK WILL ALSO BE PERFORMED. A PREVIOUS EVENT DUE TO UNRELATED CAUSES WAS REPORTED BY LER 90-06/050237.

[68] DRESDEN 2 DOCKET 50-237 LER 90-008
FAILURE OF HPCI STEAM LINE HIGH FLOW ISOLATION DIFFERENTIAL PRESSURE TRANSMITTER DUE TO UNKNOWN CAUSE.
EVENT DATE: 082090 REPORT DATE: 091190 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 219551) ON 8/20/90 AT 0645 HOURS, WITH UNIT 2 AT 82% RATED CORE THERMAL POWER, DURING PERFORMANCE OF ROUTINE SURVEILLANCE TESTING, DIFFERENTIAL PRESSURE TRANSMITTER (DPT) 2352, WHICH MONITORS HIGH PRESSURE COOLANT INJECTION (HPCI)

STEAM SUPPLY LINE FLOW, WAS FOUND TO BE OUTSIDE OF THE ESTABLISHED TOLERANCE LIMITS. WITH ONLY ONE OPERABLE INSTRUMENT CHANNEL, WHEN TWO ARE REQUIRED BY THE TECH SPECS, THE HPCI SYSTEM WAS DECLARED INOPERABLE, AND A SEVEN-DAY LIMITING CONDITION FOR OPERATION (LCO) WAS ENTERED. THE FAILED TRANSMITTER WAS REPLACED AND THE LCO TERMINATED ON 8/22/90. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL, AS THE REDUNDANT TRANSMITTER WAS DETERMINED TO BE OPERABLE, AND WOULD HAVE PROVIDED THE REQUIRED PRIMARY CONTAINMENT ISOLATION SIGNAL. THE ROOT CAUSE OF THE TRANSMITTER FAILURE WILL BE DETERMINED BY THE MANUFACTURER. A SIMILAR EVENT OCCURRED WHEN THE SAME TRANSMITTER FAILED IN 1988, DUE TO A LOSS OF TRANSMITTER FILL OIL. THAT EVENT WAS DOCUMENTED IN LER 88-15, DOCKET 050237.

[69] DRESDEN 3 DOCKET 50-249 LER 90-002 REV 01
UPDATE ON HIGH PRESSURE COOLANT INJECTION SYSTEM FLOW TRANSMITTER 3-2358 INDICATION DISCREPANCY DUE TO AMPLIFIER FAILURE.
EVENT DATE: 022490 REPORT DATE: 031990 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219391) DURING PERFORMANCE OF DRESDEN OPERATING SURVEILLANCE (DOS) 2300-3, HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM OPERABILITY VERIFICATION, ON 2/24/90 AT 1158 HOURS, THE NUCLEAR STATION OPERATOR (NSO) OBSERVED THAT FLOW INDICATING CONTROLLER (FIC) 3-2340-1 WAS INDICATING NO FLOW WHILE THE HPCI PUMP WAS OPERATING AND LINED UP TO THE CONDENSATE STORAGE TANK (CST) THROUGH THE TEST RETURN LINE. THE HPCI SYSTEM WAS DECLARED INOPERABLE AS A CONSERVATIVE MEASURE AT 1158 HOURS PENDING FURTHER INVESTIGATION, AND UNIT 3 SUBSEQUENTLY ENTERED INTO A 7 DAY LIMITING CONDITION FOR OPERATION (LCO) TIME CLOCK AS REQUIRED BY TECH SPECS SECTION 3.5.C.2.A. THE ROOT CAUSE OF THIS EVENT WAS ATTRIBUTED TO FAILURE OF FLOW TRANSMITTER (FT) 3-2358 AMPLIFIER DUE TO MOISTURE INTRUSION. THE AMPLIFIER WAS REPLACED AND RECALIBRATED. DOS 2300-3 WAS PERFORMED SUCCESSFULLY ON 2/25/90 AND HPCI WAS DECLARED OPERABLE AT 1045 HOURS. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL SINCE FUNCTIONAL OPERABILITY OF HPCI WAS UNAFFECTED AND SINCE ALL EMERGENCY CORE COOLING SYSTEMS (ECCSS) REQUIRED BY TECH SPEC SECTION 3.5.C.2.A WERE OPERABLE. THIS WAS THE FIRST OCCURRENCE OF THIS TYPE INVOLVING A HPCI FLOW TRANSMITTER FAILURE.

[70] FERMI 2 DOCKET 50-341 LER 89-038 REV 01
UPDATE ON REACTOR SCRAMMED WHEN FIRE OCCURRED IN THE VICINITY OF THE MAIN TURBINE.
EVENT DATE: 122389 REPORT DATE: 083190 NSSS: GE TYPE: BWR

(NSIC 219392) ON DECEMBER 23, 1989, SMOKE WAS DISCOVERED IN THE VICINITY OF THE HIGH PRESSURE TURBINE. THE FIRE BRIGADE WAS DISPATCHED, AND REACTOR WAS MANUALLY SCRAMMED. THE BRIGADE DISCOVERED THE LAGGING INSULATION PADS WERE BURNING AND EXTINGUISHED THE FIRE. INVESTIGATION HAS DETERMINED THAT THE INSULATION PADS WERE SOAKED WITH OIL AND THE TURBINE CASING HAD HEATED UP TO THE OIL'S FLASH POINT. AN OIL SPILL HAD OCCURRED SEVERAL WEEKS EARLIER DURING A FLUSH OF THE TURBINE LUBE OIL SYSTEM DURING TURBINE REASSEMBLY. APPARENTLY, THE SPILL WAS NOT ADEQUATELY CLEANED UP. THE DAMAGED INSULATION PADS WERE REPLACED AND AN INSPECTION FOR OTHER DAMAGE PERFORMED. NO OTHER DAMAGE WAS NOTED. AN ACCOUNTABILITY MEETING WAS HELD BETWEEN MANAGEMENT AND THE PERSONNEL INVOLVED IN THE MAINTENANCE ON THE TURBINE. PROCEDURES HAVE BEEN REVISED TO ASSURE THAT OIL LEAKS ARE DETECTED/CLEANED UP AND TO PRESCRIBE APPROPRIATE INSPECTIONS. THIS INCIDENT COUPLED WITH OTHER INCIDENTS WHICH OCCURRED DURING STARTUP FROM THE FIRST REFUELING OUTAGE LED TO FERMI 2 DEVELOPING AN ACCOUNTABILITY ACTION PLAN. THE ACCOUNTABILITY ACTION PLAN WAS IMPLEMENTED BY SITE ORGANIZATIONS AND EVALUATED BY QUALITY ASSURANCE AS COMMITTED TO IN DETROIT EDISON LETTER NRC-89-0300, DATED DECEMBER 26, 1989.

[71] FERMI 2 DOCKET 50-341 LER 90-006
UNANALYZED CONDITION RESULTING FROM A SINGLE ACTIVE FAILURE.
EVENT DATE: 072690 REPORT DATE: 082790 NSSS: GE TYPE: BWR

(NSIC 219432) A SITE SAFETY SYSTEM FUNCTIONAL INSPECTION REVIEW OF THE CONTROL CENTER HEATING, VENTILATION AND AIR CONDITIONING (CCHVAC) SYSTEM REVEALED THAT A PREVIOUSLY UNANALYZED CONDITION COULD RESULT FROM A SINGLE ACTIVE FAILURE OF

EITHER ONE OF TWO CCHVAC CHLORINE DETECTORS FOLLOWED BY A LOCA WITH A RADIATION RELEASE. AN ENGINEERING FUNCTIONAL ANALYSIS WAS PERFORMED THAT DETERMINED THAT THE CCHVAC SYSTEM WAS OPERABLE. THE IMMEDIATE ACTIONS TAKEN WERE TO INSTRUCT THE OPERATORS ON SPECIFIC ACTIONS REQUIRED WHILE IN THE CHLORINE MODE SHOULD A LOCA SIGNAL BE RECEIVED; AND TO REVISE APPLICABLE PROCEDURES. CURRENTLY, CONSIDERATION IS BEING GIVEN TO LONG TERM CORRECTIVE ACTION OPTIONS, SUCH AS, MODIFYING EXISTING LOGIC CIRCUITRY OR REPLACING SITE CHLORINE WITH AN ALTERNATE BIOCIDE.

[72] FITZPATRICK DOCKET 50-333 LER 90-012 REV 01
UPDATE ON NORMAL AND EMERGENCY SERVICE WATER SYSTEM INSPECTION RESULTS SAFETY CONCERNS DUE TO SILT AND CORROSION PRODUCT BUILD-UP.
EVENT DATE: 040490 REPORT DATE: 082990 NSSS: GE TYPE: BWR
VENDOR: CHAPMAN VALVE & MFG
PACIFIC VALVES, INC.
VELAN VALVE CORP.

(NSIC 219453) DURING THE 1990 REFUEL OUTAGE, 61 CHECK VALVES IN THE NORMAL (NSW) (KG) AND EMERGENCY SERVICE WATER (ESW) (BI) WERE OPENED AND VISUALLY INSPECTED. THIRTY-SEVEN OF THE VALVES WERE FROM THE ASME SECTION XI IST PROGRAM AND 24 WERE FROM THE CHECK VALVE PREVENTIVE MAINTENANCE (PM) PROGRAM. TWENTY IST AND 10 PM PROGRAM VALVES WERE INITIALLY DECLARED INOPERABLE DUE TO FAILING THE VISUAL INSPECTION CRITERIA. HOWEVER, THE IST CHECK VALVES WERE SHOWN TO BE OPERABLE BY ACTUAL FLOW TEST OR CALCULATION. OTHER EFFORTS INCLUDED INTERNAL INSPECTION OF 500' OF SMALL BORE PIPING AND 10 SAFETY-RELATED COOLERS AND AIR HANDLING UNITS (AHUS). OF THESE COOLERS AND AHUS, 2 WERE FOUND TO HAVE 25% TUBE PLUGGING WITH SILT/SAND, BUT SHOWN ABLE TO REMOVE DESIGN BASIS HEAT LOAD. OF THE 500' OF PIPING, 200' WERE FOUND 10%-30% RESTRICTED IN CROSS-SECTIONAL AREA, BUT A CALCULATION DEMONSTRATED THAT FLOW CONTROL VALVES WERE HYDRAULICALLY LIMITING. IN ALL, THE ESW SYSTEM WAS CONSIDERED CAPABLE OF PERFORMING THE DESIGN SAFETY FUNCTION. THE AFFECTED VALVES, COOLERS, AND PIPING WERE CLEANED OR REPLACED AS NECESSARY AND RETURNED TO SERVICE. THE INTAKE BAYS WERE ALSO CLEANED. PERIODIC FLUSHING AND PERFORMANCE TESTING SHOULD PREVENT RECURRENCE. LERS 88-005, 88-009, AND 89-015 ARE RELATED.

[73] FITZPATRICK DOCKET 50-333 LER 90-018 REV 01
UPDATE ON REACTOR SAFETY RELIEF VALVE PILOT ASSEMBLY SETPOINT DRIFT.
EVENT DATE: 061590 REPORT DATE: 090490 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 219429) DURING THE REFUELING OUTAGE BEGINNING 03/31/90, THE ACTUATING TOPWORKS MECHANISM FOR EIGHT SAFETY RELIEF VALVES (SRVS) (AD) WERE REMOVED FOR TESTING. TWO VALVES ACTUATED AT PRESSURES WHICH DEVIATED FROM THE NAMEPLATE SETPOINT BEYOND THE +/-1.0% TOLERANCE ALLOWED BY TECHNICAL SPECIFICATIONS. ONE SRV LIFTED 1.1% BELOW THE SETPOINT, THE OTHER LIFTED 2.3% ABOVE THE SETPOINT. NO CAUSE WAS DETERMINED. A PLANT SPECIFIC ANALYSIS STATES THAT AN SRV SETPOINT TOLERANCE OF +/-3% WILL HAVE NO SIGNIFICANT SAFETY IMPACT ON VESSEL OVERPRESSURE MARGIN, THERMAL LIMITS, ECCS/LOCA PERFORMANCE, HPCI/RCIC OPERABILITY, CONTAINMENT PRESSURE, OR CONTAINMENT INTEGRITY. THERE WAS AN ADEQUATE MARGIN BETWEEN THE AS-FOUND SRV SETTINGS AND NORMAL REACTOR PRESSURES DURING POWER OPERATION. CORRECTIVE ACTION INCLUDED REPLACING THE FAILED SRV WITH A RECERTIFIED VALVE, CONTINUED PARTICIPATION IN THE B OWNERS' GROUP TO RESOLVE SRV ISSUES, AND PREVIOUS SUBMISSION TO THE NRC OF PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS TO TAKE CREDIT FOR EXCESS INSTALLED SRV CAPACITY. LER-85-009, 85-013, 87-004, 88-004, 88-010, AND 89-026 ARE SIMILAR EVENTS INVOLVING SRV SETPOINT DRIFT.

[74] GINNA DOCKET 50-244 LER 90-008 REV 01
UPDATE ON SAFEGUARDS BUSES DEGRADED VOLTAGE RELAYS MISCALIBRATED DUE TO PROCEDURE INADEQUACY CAUSES A CONDITION PROHIBITED BY PLANT TECHNICAL SPECIFICATIONS.
EVENT DATE: 052490 REPORT DATE: 082090 NSSS: WE TYPE: PWR
VENDOR: ITE IMPERIAL CORPORATION

(NSIC 219340) ON MAY 24, 1990 AT 1604 EDST WITH THE REACTOR AT APPROXIMATELY 98%

FULL POWER, AN EVALUATION OF UNDERVOLTAGE RELAY TEST DATA REVEALED THAT FIVE (5) OF THE EIGHT (8) DEGRADED VOLTAGE RELAYS ON 480 VOLT SAFEGUARD BUSES WERE CALIBRATED SUCH THAT RELAY ACTUATION WOULD NOT OCCUR WITHIN THE LIMITS OF TECHNICAL SPECIFICATIONS. AS THIS WAS A SETPOINT PROBLEM WITH THE DEGRADED VOLTAGE RELAYS, NO IMMEDIATE PLANT OR OPERATOR RESPONSES WERE NECESSARY. THE UNDERLYING CAUSE OF THE EVENT WAS ATTRIBUTED TO THE WRONG SETPOINT AND SETPOINT TOLERANCE BEING SPECIFIED IN A CALIBRATION PROCEDURE. IMMEDIATE CORRECTIVE ACTION WAS TO CALIBRATE THE DEGRADED VOLTAGE RELAYS TO A SETPOINT ON THE CONSERVATIVE SIDE OF THE TOLERANCE. SUBSEQUENT TO THE CALIBRATION, THE DEGRADED VOLTAGE RELAYS WERE TESTED SATISFACTORILY AND RETURNED TO SERVICE.

[75] GRAND GULF 1 DOCKET 50-416 LER 89-002 REV 02
UPDATE ON CONTAINMENT ISOLATION VALVES EXCEED TECHNICAL SPECIFICATION LEAKAGE LIMITS.
EVENT DATE: 033189 REPORT DATE: 083190 NSSS: GE TYPE: BWR
VENDOR: ATWOOD & MORRILL CO., INC.
POWELL, WILLIAM COMPANY, THE

(NSIC 219393) DURING ROUTINE LOCAL LEAK RATE TESTING, THE LEAKAGE RATES OF FIVE CONTAINMENT ISOLATION VALVES COULD NOT BE QUANTIFIED. THIS WAS DUE TO EXCESSIVE LEAKAGE BEYOND THE CAPABILITY OF THE TEST EQUIPMENT TO PRESSURIZE THE TEST VOLUME TO THE REQUIRED TEST PRESSURE. THUS, THE LEAKAGE IS CONSIDERED TO HAVE EXCEEDED TECHNICAL SPECIFICATION ALLOWABLE LEAKAGE LIMIT. THE SUBJECT VALVES ARE: MAIN STEAM LINE "C" INBOARD ISOLATION VALVE, B21F022C; FEEDWATER LINE "A" INBOARD AND OUTBOARD ISOLATION CHECK VALVES, B21F010A AND B21F032A; FEEDWATER LINE "B" INBOARD AND OUTBOARD ISOLATION CHECK VALVES, B21F010B AND B21F032B. ALL CONDITIONS WHICH ATTRIBUTED TO THE EXCESSIVE LEAKAGE HAVE BEEN CORRECTED AND THE VALVES HAVE BEEN RETESTED SATISFACTORILY. AN EVALUATION OF THE FEEDWATER CHECK VALVE SEAT FAILURE WAS NOT SUCCESSFUL IN POSITIVELY IDENTIFYING THE CAUSE OF THE FAILURE OF THE E692-75 MATERIAL. THE INSTALLED SR 740-70 RESILIENT SEATS WILL CONTINUE TO BE USED AND WILL BE REPLACED EACH REFUELING OUTAGE.

[76] GRAND GULF 1 DOCKET 50-416 LER 90-011
REACTOR SCRAM ON HIGH REACTOR WATER LEVEL.
EVENT DATE: 072490 REPORT DATE: 082390 NSSS: GE TYPE: BWR
VENDOR: DAHL, GEORGE W., COMPANY INCORPORATED

(NSIC 219375) ON JULY 24, 1990 AT 1355, THE REACTOR TRIPPED ON HIGH REACTOR WATER LEVEL. REACTOR POWER WAS IN THE PROCESS OF BEING REDUCED IN AN ATTEMPT TO CONTROL "B" REACTOR FEED PUMP TURBINE (RFPT) OSCILLATIONS CAUSED BY A MALFUNCTION OF THE "B" RFPT CONTROLLER. THE CAUSE OF THE MALFUNCTIONING "B" RFPT CONTROLLER WAS DETERMINED TO BE AN INTERMITTENT FAILURE OF THE LINEAR VARIABLE DIFFERENTIAL TRANSFORMER (LVDT) AND THE ASSOCIATED CIRCUIT BOARD TO THE ELECTRONIC AUTOMATIC POSITIONER (EAP). THE EAP DAHL CONTROLLER WAS CALIBRATED IN ACCORDANCE WITH THE VENDOR RECOMMENDED SETTING. THE CALIBRATION PROCEDURE FOR THE REACTOR FEED PUMP TURBINE "A" AND "B" EAP DAHL CONTROL CIRCUIT HAS BEEN IMPROVED AND THE CALIBRATION FREQUENCY HAS BEEN INCREASED. ALL PLANT SAFETY SYSTEMS PERFORMED AS EXPECTED. THE MINIMUM WATER LEVEL REACHED WAS APPROXIMATELY 126.7 INCHES ABOVE THE TOP OF ACTIVE FUEL. THE RCIC WAS MANUALLY STARTED AND FUNCTIONED AS DESIGNED.

[77] GRAND GULF 1 DOCKET 50-416 LER 90-012
DEFICIENCIES IN HPCS 125 VDC SYSTEM.
EVENT DATE: 072490 REPORT DATE: 082390 NSSS: GE TYPE: BWR

(NSIC 219376) ON JULY 24, 1990 DURING A DIVISION III 125 VDC SYSTEM REVIEW, CALCULATIONS COULD NOT DEMONSTRATE THAT ALL LOADS WOULD RECEIVE THE MANUFACTURER'S MINIMUM VOLTAGE REQUIREMENTS IMMEDIATELY FOLLOWING THE LOSS OF THE CLASS 1E BATTERY CHARGER. ADDITIONALLY, CALCULATIONS DID NOT DEMONSTRATE THAT THE DIVISION III LOAD PROFILE, AS STATED IN GGNS TECHNICAL SPECIFICATIONS, WAS GREATER THAN THE ACTUAL EMERGENCY LOADS FOR ALL PERIODS. A MATERIAL NONCONFORMANCE REPORT WAS GENERATED TO DOCUMENT THE NONCONFORMANCE. THE DIVISION III 125 VDC DISTRIBUTION CIRCUITS WERE MODIFIED TO ENSURE ADEQUATE VOLTAGE LEVELS WOULD BE ACHIEVED. THE GGNS UFSAR AND TECHNICAL SPECIFICATIONS WILL BE CHANGED

TO REFLECT THE CORRECT LOAD PROFILE. AN UPDATE REPORT WILL BE SUBMITTED TO PRESENT RESULTS FROM ADDITIONAL REVIEWS.

[78] GRAND GULF 1 DOCKET 50-416 LER 90-013
NEUTRON MONITORING SYSTEM CAUSES SCRAM DUE TO PERSONNEL ERROR.
EVENT DATE: 072390 REPORT DATE: 082390 NSSS: GE TYPE: BWR

(NSIC 219377) WHILE PERFORMING MAINTENANCE ON THE INTERMEDIATE RANGE NEUTRON MONITORING SYSTEM, PLANT PERSONNEL DISCONNECTED THE INSTRUMENT CABLE FROM THE 'B' DETECTOR RATHER THAN THE 'C' DETECTOR. THIS SENT A TRIP SIGNAL TO THE ASSOCIATED TRIP SYSTEM. THIS RESULTED IN A REACTOR SCRAM DUE TO THE OTHER TRIP SYSTEM BEING IN THE TRIPPED CONDITION. THE CIRCUMSTANCES LEADING TO THE INCIDENT WERE REVIEWED BY THE INCIDENT REVIEW BOARD. THE PERSONNEL INVOLVED WERE COUNSELED ON THE APPROPRIATE METHODS FOR SELF VERIFICATION. THIS LICENSEE EVENT REPORT WILL BE SENT TO THE REQUIRED READING COORDINATOR FOR DISTRIBUTION TO APPROPRIATE DISCIPLINES FOR REVIEW.

[79] GRAND GULF 1 DOCKET 50-416 LER 90-014
FAILURE TO RETEST SECONDARY CONTAINMENT ISOLATION VALVE FOLLOWING MAINTENANCE.
EVENT DATE: 081090 REPORT DATE: 090790 NSSS: GE TYPE: BWR

(NSIC 219515) DURING A REVIEW OF COMPLETED MAINTENANCE WORK ORDER PACKAGES, IT WAS DISCOVERED THAT THE TECHNICAL SPECIFICATION REQUIRED RETEST WAS NOT PERFORMED ON SECONDARY CONTAINMENT ISOLATION VALVE 1P71F307 PRIOR TO RETURNING THE VALVE TO THE OPERABLE STATUS. ON MARCH 22, 1990 A LEAK WAS FOUND IN THE AIR CONTROL CIRCUIT OF SECONDARY CONTAINMENT ISOLATION VALVE 1P71F307. THE FILTER REGULATOR WAS REPLACED AND THE VALVE WAS RETURNED TO SERVICE PRIOR TO PERFORMING THE REQUIRED STROKE TIME RETEST. THE FAILURE TO RETEST IS ATTRIBUTED TO PROGRAMMATIC WEAKNESSES IN THE ADMINISTRATIVE CONTROLS OF POST-MAINTENANCE RETESTS. SUBSEQUENTLY, THE VALVE WAS DETERMINED TO HAVE A STROKE TIME WELL WITHIN THE SPECIFIED LIMIT. THEREFORE, THERE WAS NO REDUCTION OF THE SAFETY FUNCTION PROVIDED BY THE VALVE. PLANT PROCEDURES ARE BEING CHANGED TO MORE CLEARLY SPECIFY THE RESPONSIBILITIES FOR SPECIFYING POST-MAINTENANCE RETESTS.

[80] HATCH 1 DOCKET 50-321 LER 90-015
COMPONENT FAILURE CAUSES HIGH PRESSURE COOLANT INJECTION SYSTEM INOPERABILITY.
EVENT DATE: 073090 REPORT DATE: 082790 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219353) ON 07/30/90, AT APPROXIMATELY 0650 CDT, UNIT 1 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2436 CMWT (APPROXIMATELY 100 PERCENT RATED THERMAL POWER). AT THAT TIME, THE HIGH PRESSURE COOLANT INJECTION (HPCI, EIIS CODE BJ) SYSTEM WAS DECLARED INOPERABLE DUE TO A MALFUNCTIONING FLOW CONTROLLER. THE CONTROLLER WAS SUBSEQUENTLY REPAIRED AND RETURNED TO SERVICE. AT APPROXIMATELY 1425 CDT, THE HPCI SYSTEM WAS DECLARED OPERABLE FOLLOWING SUCCESSFUL COMPLETION OF PROCEDURE 34SV-E41-002-1S, HPCI PUMP OPERABILITY. THE CAUSE OF THE FLOW CONTROLLER MALFUNCTION WAS A DEFECTIVE CONTROL AMPLIFIER. THE SOLID STATE CONTROL AMPLIFIER FAILED TO GENERATE A 10 TO 50 MA D-C CONTROL SIGNAL NECESSARY TO OPERATE THE HPCI TURBINE IN AUTOMATIC MODE. THE CORRECTIVE ACTIONS WERE TO REPLACE THE CONTROL AMPLIFIER WITH A NEW ONE FROM WAREHOUSE STOCK, INITIATE SCHEDULED REPLACEMENT OF THE FAILED COMPONENT ON BOTH UNITS AS A PREVENTIVE MAINTENANCE, AND TO SEND THE FAILED CONTROL AMPLIFIER BOARD TO THE MANUFACTURER FOR A MORE DETAILED FAILURE ANALYSIS.

[81] HATCH 1 DOCKET 50-321 LER 90-016
COMPONENT FAILURE AND HUMAN FACTORS RESULT IN UNPLANNED ESF ACTUATION.
EVENT DATE: 082190 REPORT DATE: 091390 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 2 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219473) ON 8/21/90 AT APPROXIMATELY 1305 CDT, UNIT 1 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2436 CMWT (APPROXIMATELY 100% OF RATED THERMAL

POWER) AND UNIT 2 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2436 CMWT (APPROXIMATELY 100% OF RATED THERMAL POWER). AT THAT TIME, THE "B" TRAINS OF BOTH UNITS' STANDBY GAS TREATMENT (SBGT) SYSTEMS AUTOMATICALLY STARTED AND THE UNIT 1 AND UNIT 2 REFUELING FLOOR AND UNIT 1 REACTOR BUILDING VENTILATION SYSTEMS ISOLATED. THE ACTUATION OF THE SBGT SYSTEM AND THE REFUELING FLOOR/REACTOR BUILDING VENTILATION SYSTEM OCCURRED DUE TO A FAILED RELAY. A SECOND ESF ACTUATION OCCURRED DURING THE REPLACEMENT OF THE RELAY. A FUSE BLEW RESULTING IN THE ISOLATION OF PRIMARY CONTAINMENT ISOLATION VALVES (PCIVS) 1D11-F052, F053, AND F072 AND, CONSEQUENTLY, THE FISSION PRODUCTS MONITORING SYSTEM. THE CAUSE OF THE FIRST EVENT WAS COMPONENT FAILURE. THE COIL IN RELAY 1C61-K76 FAILED, ACTUATING THE LOGIC FOR THE "B" TRAINS OF THE SBGT SYSTEMS AND THE UNIT 1 AND UNIT 2 REFUELING FLOOR AND UNIT 1 REACTOR BUILDING VENTILATION SYSTEMS. THE CAUSE OF THE SECOND EVENT WAS LESS THAN OPTIMUM HUMAN FACTORS. DURING WORK ACTIVITIES FOR THE REPLACEMENT OF THE FAILED RELAY, A JUMPER HAD TO BE INSTALLED ON A TERMINAL OF A RELAY WHICH DOES NOT ALLOW EASY JUMPER INSTALLATION OR PROVIDE A SECURE JUMPER CONNECTION POINT.

[82] HATCH 2 DOCKET 50-366 LER 90-006
SPURIOUS ACTUATION AND INADEQUATE MOUNTING OF TIME DELAY RELAY CAUSE PARTIAL REACTOR WATER CLEANUP ISOLATION.
EVENT DATE: 080290 REPORT DATE: 082790 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219364) ON 8/2/90, AT APPROXIMATELY 2027 CDT, UNIT 2 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2436 MWT (APPROXIMATELY 100% RATED THERMAL POWER). AT THAT TIME, PLANT OPERATORS WERE OPENING THE REACTOR WATER CLEANUP (RWCU, EIIS CODE CE) DEMINERALIZER ISOLATION VALVE 2G31-F053B WHEN THEY RECEIVED ANNUNCIATION THAT RWCU WAS EXPERIENCING HIGH DIFFERENTIAL FLOW, INDICATING THE POSSIBILITY OF A LEAK IN THE SYSTEM. RWCU EXPERIENCED AN IMMEDIATE, PARTIAL PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS, EIIS CODE JM) ACTUATION INVOLVING ONLY THE OUTBOARD ISOLATION VALVE 2G31-F004 (EIIS CODE ISV). NORMAL SYSTEM RESPONSE SHOULD HAVE BEEN ISOLATION OF BOTH VALVES 2G31-F001 (INBOARD ISOLATION VALVE) AND 2G31-F004 FOLLOWING A 45-SECOND TIME DELAY. THE ROOT CAUSES OF THIS EVENT WERE SPURIOUS ACTUATION AND A LESS THAN ADEQUATE MOUNTING FOR RELAY 2G31-R616D. TIME DELAY RELAY 2G31-R616D EXHIBITED INTERMITTENT CONTACT FAILURE WHICH CAUSED THE RELAY TO ACTUATE WITHOUT A TIME DELAY. CORRECTIVE ACTIONS FOR THIS EVENT WILL INCLUDE REPLACING THE RELAY WITH A TIME DELAY RELAY OF A DIFFERENT DESIGN, AND IMPROVING THE RELAY MOUNTING. THIS ACTION WILL BE COMPLETE BY THE END OF THE UNIT 2 1991 REFUELING OUTAGE, CURRENTLY SCHEDULED TO BEGIN 3/21/91, OR DURING THE NEXT OUTAGE OF SUFFICIENT DURATION.

[83] HOPE CREEK 1 DOCKET 50-354 LER 90-014
DESIGN CALCULATION ERROR RESULTS IN TECHNICAL SPECIFICATION LIMIT FOR ULTIMATE HEAT SINK (DELAWARE RIVER) BEING ESTABLISHED UNCONSERVATIVELY HIGH.
EVENT DATE: 081790 REPORT DATE: 091290 NSSS: GE TYPE: BWR

(NSIC 219427) ON 8/17/90 AT 0845, NUCLEAR ENGINEERING DEPARTMENT PERSONNEL INFORMED THE SENIOR NUCLEAR SHIFT SUPERVISOR THAT THE TECH SPEC MINIMUM OPERABILITY VALUE SPECIFIED FOR THE ULTIMATE HEAT SINK (DELAWARE RIVER) TEMPERATURE WAS HIGH (90.5F). THIS DETERMINATION WAS MADE AFTER REVIEWING PLANT CONSTRUCTION DESIGN CALCULATIONS WHICH WERE UTILIZED TO VERIFY DESIGN CRITERIA FOR THE STATION SERVICE WATER (SSWS) PUMPS. AN ADMINISTRATE MAXIMUM LIMIT OF 85 F FOR THE ULTIMATE HEAT SINK (UHS) HAS BEEN ESTABLISHED. EXCEEDING THIS LIMIT WILL REQUIRE ENTRY INTO REQUIRED ACTIONS AS SPECIFIED IN TECH SPECS. A REVIEW OF ALL TIMES THAT UHS TEMPERATURE HAD EXCEEDED 85F WAS CONDUCTED. RESULTS OF THE ANALYSIS INDICATE THAT 85F HAD BEEN EXCEEDED ON 10 DAYS BETWEEN 1988 A 1990. THE HIGHEST RECORDED TEMPERATURE DURING THIS TIMEFRAME WAS 86.8F. SSWS PUMP OPERATIONAL DATA WAS REVIEWED TO DETERMINE IF ALL SSWS PUMP WOULD HAVE MET THEIR DESIGN BASIS FLOW REQUIREMENTS DURING THE OCCASION WHERE UHS TEMPERATURE EXCEEDED 85F. THE PSE&G LICENSING DEPARTMENT CONDUCTED A REVIEW OF THE POTENTIAL REPORTABILITY OF THIS CONDITION UNDER 10CFR21. IT WAS DETERMINED THAT DUE TO THE SITE SPECIFIC MISAPPLICATION OF THE DESIGN CALCULATIONS, THE SITUATION IS NOT CLEARLY REPORTABLE UNDER THIS PART. PSE&G FINDS IT PRUDENT TO NOTE THAT THIS EVALUATION WAS CONDUCTED.

[84] HOPE CREEK 1 DOCKET 50-354 LER 90-015
AUTO CLOSURE OF PRIMARY CONTAINMENT ISOLATION VALVE DURING PERFORMANCE OF I&C FUNCTIONAL TEST PROCEDURE DUE TO PERSONNEL ERRORS.
EVENT DATE: 082190 REPORT DATE: 091490 NSSS: GE TYPE: BWR

(NSIC 219458) ON 8/21/90 AT 1000, DURING THE PERFORMANCE OF AN I&C DEPARTMENT FUNCTIONAL TEST PROCEDURE ON THE STEAM LEAK DETECTION (SLD) SYSTEM, AN ISOLATION OF THE REACTOR CORE ISOLATION COOLING (RCIC) INBOARD STEAM ISOLATION VALVE (FC-HV-F007) OCCURRED. AFTER ASCERTAINING THE CAUSE OF THE ISOLATION CONTROL ROOM PERSONNEL RESET THE ISOLATION LOGIC AND RE-OPENED THE VALVE. THE PRIMARY CAUSE OF THIS EVENT WAS A PERSONNEL ERROR ON THE PART OF I&C TECHNICIANS PERFORMING THE SURVEILLANCE IN THAT AN ISOLATION BYPASS SWITCH WAS NOT PLACED IN THE PROPER POSITION (BYPASS) PRIOR TO PERFORMING THE TEST. THE PROCEDURE STEP CONTROLLING THIS EVOLUTION WAS NOT PROPERLY COMPLETED. CORRECTIVE ACTIONS CONSIST OF COUNSELLING FOR THE TECHNICIAN RESPONSIBLE FOR PERFORMING THE FUNCTIONAL TEST PROCEDURE.

[85] HOPE CREEK 1 DOCKET 50-354 LER 90-016
DISCOVERY OF INADEQUATE DIESEL FUEL OIL ANALYSIS METHODS BY CONTRACT LABORATORY RESULTS IN ENTRY TO TECHNICAL SPECIFICATION 3.0.3.
EVENT DATE: 082490 REPORT DATE: 091490 NSSS: GE TYPE: BWR

(NSIC 219459) 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 ONE OF THE ANALYSIS METHODS FOR EDG FUEL OIL SAMPLES WAS INADEQUATE. TECHNICAL SPECIFICATION SURVEILLANCE 4.8.1.1.2.F.2 REQUIRES THAT EDG FUEL OIL SAMPLES BE TESTED FOR AN IMPURITY LEVEL OF LESS THAN 2MG OF INSOLUBLES PER 100ML WHEN TESTED IN ACCORDANCE WITH ASTM-D2274-70. ON 8/23/90, THE SYSTEM ENGINEER DETERMINED THAT THE CONTRACT TESTING LABORATORY WAS NOT TESTING IAW ASTM-D2274-70, BUT RATHER, TESTING FOR PARTICULATES IAW ASTM-D2276. WHEN INFORMED OF THE PROBLEM AT 1630, THE SNSS INVOKED THE HOUR DELAY OF IMPLEMENTING TECHNICAL SPECIFICATION ACTION REQUIREMENTS, ALLOWED BY TECHNICAL SPECIFICATION 4.0.3 TO ALLOW FOR PROPER COMPLETION TESTING. HOWEVER, GIVEN THAT THE TESTING UNDER D2274 REQUIRES IN EXCESS OF 20 HOURS TO COMPLETE, THE ALLOWED 24 HOURS WAS NOT SUFFICIENT TO COMPLETE THE TESTING, GIVEN TRANSPORT TIME AND CONTRACTING WITH ANOTHER LAB TO PERFORM THE TESTS. THEREFORE, A WAIVER OF COMPLIANCE FOR TECH SPEC 4.0.3 WAS REQUESTED FROM NRC REGION I. THIS WAIVER WAS GRANTED FOR PERIOD OF 48 HOURS, EFFECTIVE AT 1730, 8/24/90, THUS DELAYING THE NEED FOR PLANT SHUTDOWN IAW TECHNICAL SPECIFICATION 3.0.3.

[86] INDIAN POINT 2 DOCKET 50-247 LER 90-003
ESF ACTUATION DUE TO ELECTRICAL SPIKES.
EVENT DATE: 071390 REPORT DATE: 081290 NSSS: WE TYPE: PWR

(NSIC 219341) ON JULY 13, 1990 THE INDIAN POINT 2 CONTAINMENT RADIOGAS MONITOR, R-12, EXPERIENCED AN ELECTRICAL SPIKE THAT CAUSED A CONTAINMENT VENTILATION ISOLATION SIGNAL TO BE ACTUATED. THIS IN TURN ISOLATED THE CONTAINMENT VENTILATION SYSTEM AND ACTIVATED A PORTION OF THE WELD CHANNEL AND CONTAINMENT PENETRATION PRESSURIZATION SYSTEM (WCCPPS). THE WCCPPS IS CLASSIFIED AS AN ENGINEERED SAFETY FEATURE (ESF).

[87] INDIAN POINT 2 DOCKET 50-247 LER 90-004
CONTAINMENT RADIOGAS MONITOR ESF ACTUATION DUE TO ELECTRICAL SPIKE.
EVENT DATE: 073190 REPORT DATE: 083090 NSSS: WE TYPE: PWR

(NSIC 219404) DURING THE PERFORMANCE OF A PRESSURE RELIEF OF CONTAINMENT ON JULY 31, 1990, WITH THE PLANT AT 96% POWER, THE CONTAINMENT RADIOGAS MONITOR (R-12) 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, RADIATION MONITOR R-12 WAS RESET AND PRESSURE

RELIEF WAS REINSTITUTED. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED BY THIS EVENT.

[88] INDIAN POINT 2 DOCKET 50-247 LER 90-005
FAILURE TO MONITOR SERVICE WATER INLET TEMPERATURE.
EVENT DATE: 080190 REPORT DATE: 083090 NSSS: WE TYPE: PWR
VENDOR: PYCO

(NSIC 219485) ON AUGUST 1, 1990, WITH THE REACTOR AT 96% OF RATED POWER, IT WAS DETERMINED THAT THE SERVICE WATER INLET TEMPERATURE WAS AT OR ABOVE 80F AND NOT BEING MONITORED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.3.F.5. THE TEMPERATURE RECORDER IN THE CENTRAL CONTROL ROOM WAS READING APPROXIMATELY 8F LESS THAN MANUAL MEASUREMENTS OF UNIT 2 INLET TEMPERATURE. BASED ON UNIT 3 TEMPERATURE RECORDS, THE 24 HOUR AVERAGE TEMPERATURE HAD REACHED 80F ON JULY 21, 1990. THE REQUIRED MONITORING WAS THEREFORE NOT PERFORMED OVER AN ELEVEN DAY PERIOD. UPON DISCOVERY, REQUIRED FOUR HOUR ALTERNATIVE MEASUREMENTS WERE INITIATED. AT NO TIME DID THE TEMPERATURE EXCEED THE 95F MAXIMUM ALLOWED BY THE TECHNICAL SPECIFICATIONS. THUS, THERE WAS NO REDUCTION IN OVERALL PLANT SAFETY.

[89] INDIAN POINT 2 DOCKET 50-247 LER 90-006
WELD CHANNEL/CONTAINMENT PENETRATION PRESSURIZATION ACTUATION.
EVENT DATE: 081390 REPORT DATE: 091290 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 219486) ON AUGUST 13, 1990 WHILE THE PLANT WAS AT FULL POWER OPERATION, THE CONTAINMENT PARTICULATE RADIATION MONITOR ALARMED. THE PLANT WAS IN THE PROCESS OF RELIEVING CONTAINMENT PRESSURE. THE CONTAINMENT PRESSURE RELIEF VALVES AUTOMATICALLY CLOSED DUE TO A CONTAINMENT HIGH RADIOACTIVITY SIGNAL AND PARTIAL ACTUATION OF THE WELD CHANNEL AND CONTAINMENT PENETRATION PRESSURIZATION SYSTEM (WCCPPS), CLASSIFIED AS AN ESF, OCCURRED. THE HIGH CONTAINMENT RADIOACTIVITY SIGNAL RESULTED FROM A PACKING LEAK FROM AN INSTRUMENT ROOT VALVE CONNECTED TO THE PRESSURIZER.

[90] KEWAUNEE DOCKET 50-305 LER 90-008
QUALITY ASSURANCE AUDIT OF THE SIEMENS ENERGY AND AUTOMATION FACILITY IN RALEIGH, NC FINDS INADEQUATE 10 CFR 50 APPENDIX B PROGRAM.
EVENT DATE: 050290 REPORT DATE: 090590 NSSS: WE TYPE: PWR

(NSIC 219420) AT 1500 ON MAY 2, 1990, WITH THE PLANT AT 96% POWER, WISCONSIN PUBLIC SERVICE CORPORATION (WPSC) RECEIVED A QUALITY ASSURANCE (QA) AUDIT REPORT ON THE SIEMENS ENERGY AND AUTOMATION FACILITY LOCATED IN RALEIGH, NORTH CAROLINA. THE AUDIT WAS CONDUCTED BY GASSER ASSOCIATES, INCORPORATED FOR WPSC. THE AUDIT FOUND THAT WHILE THE SIEMENS FACILITY HAD BEEN ACCEPTING PURCHASE ORDERS FOR SAFETY RELATED EQUIPMENT AND SERVICES THEY WERE NOT IMPLEMENTING AN ADEQUATE 10 CFR 50 APPENDIX B QA PROGRAM. THE EVENT WAS CAUSED BY A BREAKDOWN IN THE QUALITY ASSURANCE PROGRAM AT THE SIEMENS FACILITY THEREFORE, WPSC WAS NOT ABLE TO DETERMINE THE ROOT CAUSE OF THE EVENT. THE FOLLOWING CORRECTIVE ACTIONS HAVE BEEN OR WILL BE TAKEN. THE SIEMENS FACILITY WAS REMOVED FROM KEWAUNEE'S LIST OF QUALIFIED SUPPLIERS. EQUIPMENT PURCHASED FROM SIEMENS SINCE THE LAST AUDIT WAS IDENTIFIED. THE EQUIPMENT PURCHASED AS SAFETY RELATED WAS LOCATED. EQUIPMENT PURCHASED AS SAFETY RELATED AND INSTALLED ON SAFETY RELATED EQUIPMENT WAS EVALUATED. SAFETY RELATED EQUIPMENT PURCHASED FROM SIEMENS AND LOCATED IN THE WAREHOUSE WAS PLACED IN THE HOLD AREA. THE SIEMENS EQUIPMENT IN THE HOLD AREA WILL NOT BE RELEASED UNTIL IT CAN BE SHOWN THAT THE EQUIPMENT MEETS COMMERCIAL GRADE DEDICATION REQUIREMENTS FOR SAFETY RELATED COMPONENTS OR IS RECLASSIFIED FOR NON-SAFETY RELATED USE ONLY.

[91] LA SALLE 1 DOCKET 50-373 LER 90-011
FAILURE OF REACTOR CORE ISOLATION COOLING STEAM LINE HIGH FLOW STATIC-O-RING DIFFERENTIAL PRESSURE SWITCH DUE TO TORN DIAPHRAGM.
EVENT DATE: 080190 REPORT DATE: 083190 NSSS: GE TYPE: BWR
VENDOR: STATIC-O-RING

(NSIC 219446) ON 8/1/90 AT 2000 HOURS, DURING PERFORMANCE OF LASALLE INSTRUMENT SURVEILLANCE, LIS-RI-301, "UNIT 1 STEAM LINE HIGH FLOW REACTOR CORE ISOLATION COOLING (RCIC) ISOLATION FUNCTION TEST," PRESSURE DIFFERENTIAL SWITCH (PDS) 1E31-N013BA WAS FOUND FAILED, APPARENTLY FROM A TORN DIAPHRAGM. UNIT 1 WAS IN OPERATIONAL CONDITION 1 (RUN) AT 84% POWER LEVEL. THIS SWITCH FUNCTIONS TO PROVIDE A DIVISION II (INBOARD) ISOLATION OF THE RCIC STEAM LINE AND TO INITIATE A RCIC TURBINE TRIP UNDER A HIGH STEAM FLOW CONDITION. THE SWITCH PDS 1E31-N013BA WAS MADE NON-FUNCTIONAL BY THE RUPTURED DIAPHRAGM, THEREFORE THE DIVISION II ISOLATION AND RCIC TRIP ASSOCIATED WITH THIS SWITCH WAS UNAVAILABLE. HOWEVER, PRESSURE DIFFERENTIAL SWITCH PDS 1E31-N013AA WAS STILL AVAILABLE TO PROVIDE DIVISION I (OUTBOARD) ISOLATION AND TURBINE TRIP. RCIC SYSTEM WAS DECLARED INOPERABLE ON 8/1/90 AT 1803 HOURS, IN ORDER TO PERFORM THE REQUIRED SURVEILLANCE. THE HIGH PRESSURE CORE SPRAY REMAINED OPERABLE THROUGHOUT THE DURATION OF THIS EVENT. A REPLACEMENT SWITCH WAS INSTALLED, CALIBRATED, AND FUNCTIONALLY TESTED SATISFACTORILY. RCIC SYSTEM WAS DECLARED OPERABLE ON 8/3/90 AT 0950 HOURS. THIS EVENT IS REPORTED TO THE NRC AS A LER IN ACCORDANCE WITH 10CFR50.73(A)(2)(V) DUE TO RCIC BEING DECLARED INOPERABLE AND LOSS OF A SAFETY SYSTEM FUNCTION.

[92] LIMERICK 1 DOCKET 50-352 LER 90-013
PROBLEMS WITH 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: 071290 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 218821) ON 6/11/90, BASED ON THE REVIEW OF THE DC ELECTRICAL DISTRIBUTION SYSTEM, IT WAS 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 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 ISOLATION PROTECTION. FURTHER INVESTIGATION ON 6/13/90 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 RESULTING FROM THE UNDER-RATED DC FUSES FAILING TO ISOLATE HIGH OVERLOAD CURRENT CONDITIONS. IMMEDIATE CORRECTIVE ACTIONS WERE TAKEN TO ESTABLISH HOURLY FIRE WATCHES IN THE AFFECTED UNIT 2 FIRE AREAS UNTIL 6/26/90 WHEN A MODIFICATION WAS COMPLETED REPLACING UNDER-RATED FUSES. THE AFFECTED UNIT 1 FIRE AREA WAS NOT FIRE WATCHED SINCE UNIT 1 WAS IN COLD SHUTDOWN AT THE TIME. APPROXIMATE CAUSES OF THESE CONDITIONS ARE ERRORS MADE DURING THE ORIGINAL DESIGN WHEN WE INCORRECTLY ASSUMED THAT DOUBLE FUSING WAS SUFFICIENT ISOLATION.

[93] LIMERICK 1 DOCKET 50-352 LER 90-015
REACTOR WATER CLEANUP SYSTEM ISOLATION RESULTING FROM HIGH REGENERATIVE HEAT EXCHANGER ROOM TEMPERATURE CAUSED BY LEAKING SYSTEM VENT VALVES.
EVENT DATE: 081390 REPORT DATE: 091290 NSSS: GE TYPE: BWR

(NSIC 219545) ON AUGUST 13, 1990, AT 1935 HOURS, AND ACTUATION OF THE GROUP III PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM (PCRVICS) OCCURRED, AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION, INITIATING A REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATION. THE PCRVICS ISOLATION WAS INITIATED WHEN THE STEAM LEAK DETECTION SYSTEM (DIVISIONS 1 AND 4) DETECTED HIGH TEMPERATURES IN THE RWCU SYSTEM REGENERATIVE HEAT EXCHANGER ROOM. OPERATIONS PERSONNEL RESET THE PCRVICS ISOLATION AT 0024 HOURS ON AUGUST 14, 1990. THE CAUSE OF THE HIGH AREA TEMPERATURE WAS FOUND TO BE STEAM LEAKAGE PAST THE SEATS OF TWO NORMALLY CLOSED VENT LINE DRAIN VALVES IN THE RWCU SYSTEM. A FOUR HOUR NOTIFICATION WAS MADE TO THE NRC AT 2127 HOURS ON AUGUST 13, 1990 IN ACCORDANCE WITH THE REQUIREMENTS OF 10CFR 50.72(B)(2)(II) SINCE THIS EVENT RESULTED IN AN AUTOMATIC ACTUATION OF AN ESF. THE LEAKING VALVES WERE REPLACED AND THE RWCU SYSTEM WAS RETURNED TO SERVICE. THE RWCU SYSTEM ISOLATED AS DESIGNED IN RESPONSE TO THE HIGH RWCU SYSTEM REGENERATIVE HEAT EXCHANGER ROOM TEMPERATURE AS SENSED BY THE STEAM LEAK DETECTION SYSTEM. THE RWCU SYSTEM WAS ISOLATED FOR 2 DAYS AND 12 HOURS. THE REACTOR COOLANT CONDUCTIVITY AND CHLORIDE CONCENTRATION INCREASED BUT REMAINED

WITHIN THE LIMITS SPECIFIED BY THE LIMERICK GENERATING STATION UNIT 1 TECHNICAL SPECIFICATIONS.

[94] LIMERICK 1 DOCKET 50-352 LER 90-016
VIOLATION OF TECHNICAL SPECIFICATIONS BY FIREWATCH EMPLOYEES DUE TO THE FAILURE TO PROPERLY PERFORM FIREWATCH SURVEILLANCE PROCEDURE.
EVENT DATE: 081590 REPORT DATE: 091490 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 219456) ON 8/15/90, STATION PERSONNEL IDENTIFIED A CONDITION WHEREBY A TECH SPECS (TS) SURVEILLANCE REQUIREMENT HAD NOT BEEN SATISFIED DUE TO A COGNITIVE PERSONNEL ERROR, AND AS A RESULT, THE ASSOCIATED TS LIMITING CONDITION FOR OPERATION (LCO) ACTION HAD NOT BEEN IMPLEMENTED WITHIN THE REQUIRED TIME PERIOD. ON 8/14/90, PHILADELPHIA ELECTRIC COMPANY CORPORATE SECURITY PERSONNEL WERE INFORMED OF AN ALLEGATION INDICATING THAT A VENDOR EMPLOYEE MAY HAVE BEEN INVOLVED IN THE FALSIFICATION OF SURVEILLANCE TEST (ST) PROCEDURES REGARDING FIRE DOOR POSITION VERIFICATION. INVESTIGATION INTO THE ALLEGATION REVEALED THAT THE FIRE WATCH EMPLOYEE HAD INDEED BEEN FALSIFYING RECORDS. DURING AN INTERVIEW, THE FIREWATCH EMPLOYEE ADMITTED TO DELIBERATELY FALSIFYING THE ST RECORDS. THE CAUSE OF THIS EVENT WAS A COGNITIVE PERSONNEL ERROR. THE FIREWATCH EMPLOYEE HAD DELIBERATELY FALSIFIED FIRE DOOR STS ON NUMEROUS OCCASIONS AND KNOWINGLY CIRCUMVENTED ESTABLISHED PROCEDURES. THE FIREWATCH EMPLOYEE INVOLVED IN THE FALSIFICATION OF RECORDS WAS IMMEDIATELY REMOVED FROM THE JOB-SITE AND WAS TERMINATED. RANDOM SAMPLING OF DOOR STS PERFORMED BY ALL THE OTHER FIREWATCH EMPLOYEES WAS DONE TO VERIFY THE VALIDITY OF THEIR TESTS. ONE ADDITIONAL VIOLATION WAS IDENTIFIED DURING THIS INVESTIGATION AND THE ASSOCIATED EMPLOYEE WAS ALSO TERMINATED.

[95] MAINE YANKEE DOCKET 50-309 LER 90-001 REV 01
UPDATE ON FAILURE OF ENVIRONMENTALLY QUALIFIED LIMIT SWITCH.
EVENT DATE: 020790 REPORT DATE: 090590 NSSS: CE TYPE: PWR
VENDOR: NAMCO CONTROLS

(NSIC 219466) WHILE OPERATING AT 98% POWER, MAINE YANKEE IDENTIFIED A FAILED ENVIRONMENTALLY QUALIFIED (EQ) LIMIT SWITCH LOCATED INSIDE THE CONTAINMENT. THE LIMIT SWITCH PROVIDES MAIN CONTROL BOARD POSITION INDICATION OF AN ISOLATION VALVE FOR THE PRIMARY SAMPLING SYSTEM. THE VALVE CLOSES ON A SAFETY INJECTION OR CONTAINMENT ISOLATION SIGNAL, OR MAY BE OPERATED FROM THE MAIN CONTROL BOARD. INVESTIGATION REVEALED THAT MOISTURE ENTERED THE LIMIT SWITCH, DUE TO INADEQUATE SEALING OF THE CONDUIT, AND CORRODED A TERMINAL. THE LIMIT SWITCH PROVIDES POSITION INDICATION ONLY AND DOES NOT AFFECT VALVE OPERATION. THE VALVE REMAINED OPERABLE AT ALL TIMES. THE SWITCH WAS REPLACED WITH AN IDENTICAL MODEL. THE NEW SWITCH AND CONDUIT WERE BENCH ASSEMBLED AND SEALED TO ENSURE PROPER CONSTRUCTION. THE CONDUIT ORIENTATION WAS CHANGED TO ENTER BELOW THE LIMIT SWITCH AND HOLES WERE DRILLED IN THE CONDUIT SUCH THAT ANY MOISTURE IN THE CONDUIT WOULD DRAIN OUT. MAINE YANKEE SUBSEQUENTLY UPGRADED THIRTY-SIX EQ LIMIT SWITCH CONDUIT SEALS WITH NEW PRE-SEALED CONNECTOR ASSEMBLIES DURING THE 1990 REFUELING OUTAGE. FOURTEEN OF THESE THIRTY-SIX SEALS HAD PREVIOUSLY BEEN MADE IN THE SAME TIME PERIOD AS THE ONE FOUND FAILED IN FEBRUARY 1990; THE SECOND SWITCH FOR THAT VALVE WAS NOT REWORKED AS IT HAD BEEN INSPECTED AND FOUND SATISFACTORY.

[96] MAINE YANKEE DOCKET 50-309 LER 90-002 REV 01
UPDATE ON INADVERTENT SAFETY INJECTION ACTUATION WHILE SWAPPING VITAL AC BUSSES.
EVENT DATE: 041490 REPORT DATE: 090790 NSSS: CE TYPE: PWR

(NSIC 219467) ON 4/14/90 WHILE IN A REFUELING SHUTDOWN CONDITION, AN INADVERTENT ACTUATION OF ENGINEERED SAFEGUARD FEATURES OCCURRED. WHILE RETURNING AN INVERTER TO SERVICE, OPERATORS INCORRECTLY OPENED THE OUTPUT BREAKER OF AN OPERATING INVERTER. THIS INVERTER WAS SUPPLYING POWER TO ITS OWN BUS AS WELL AS ALTERNATE POWER FOR THE INVERTER WHICH HAD BEEN OUT OF SERVICE. OPENING THE OUTPUT BREAKER CAUSED A LOSS OF POWER TO THE TWO VITAL AC BUSSES CONNECTED TO IT. DUE TO THE FAILSAFE DESIGN OF THE SAFETY INJECTION ACTUATION SYSTEM (SIAS) LOGIC, THE SYSTEM ACTUATED UPON LOSS OF POWER TO TWO CHANNELS.

[97] MAINE YANKEE DOCKET 50-309 LER 90-00?
MISPOSITIONED NUCLEAR INSTRUMENTATION SWITCH.
EVENT DATE: 072590 REPORT DATE: 082490 NSSS: CE TYPE: PWR

(NSIC 219421) ON JULY 25, 1990, TECHNICIANS PERFORMING A FUNCTIONAL TEST OF THE POWER RANGE SAFETY CHANNELS DISCOVERED AN IMPROPER SWITCH ALIGNMENT WHICH MAY HAVE RENDERED AN INPUT TO THREE CHANNEL "D" REACTOR PROTECTIVE SYSTEM (RPS) FUNCTIONAL UNITS INOPERABLE. THE AFFECTED UNITS WERE VARIABLE OVER POWER (VOP), THERMAL MARGIN/LOW PRESSURE (TM/LP) AND AXIAL FLUX OFFSET (S/O). DURING THE PERIOD THAT THE SWITCH WAS MISPOSITIONED, IT IS CONSERVATIVELY ASSUMED THAT THE TECHNICAL SPECIFICATION REQUIREMENT TO MAINTAIN FOUR (4) OPERABLE CHANNELS WAS NOT MET. THE SWITCH WAS RETURNED TO ITS NORMAL POSITION. A CALORIMETRIC ADJUSTMENT WAS PERFORMED TO ENSURE PROPER CHANNEL OUTPUT. AN EXACT TIME THAT THE SWITCH WAS MISPOSITIONED COULD NOT BE DETERMINED. PERSONNEL OVERSIGHT DURING MAINTENANCE IS BELIEVED TO BE THE CAUSE OF THE EVENT. A REVIEW OF RPS SURVEILLANCE PROCEDURES WILL BE CONDUCTED BY OCTOBER 31, 1990 TO ENSURE ADEQUATE COMPONENT POSITION VERIFICATION. ADDITIONALLY, SPECIAL COUNSELLING WAS CONDUCTED FOR INVOLVED PERSONNEL.

[98] MCGUIRE 1 DOCKET 50-369 LER 90-004 REV 01
UPDATE ON HOLES WERE LEFT IN THE AUXILIARY SHUTDOWN PANEL IN VIOLATION OF TECHNICAL SPECIFICATIONS BECAUSE OF UNKNOWN REASONS.
EVENT DATE: 020890 REPORT DATE: 031990 NSSS: WE TYPE: PWR

(NSIC 219365) ON FEBRUARY 8, 1990, QUALITY ASSURANCE (QA) PERSONNEL PERFORMED A RANDOM WALK THROUGH INSPECTION OF ELECTRICAL EQUIPMENT IN THE AUXILIARY BUILDING. THE INSPECTION WAS PERFORMED TO DETERMINE IF THE STATION WAS IN COMPLIANCE WITH THE ENVIRONMENTAL SEAL PROGRAM. QA PERSONNEL INSPECTED THE UNIT 1 AUXILIARY SHUTDOWN PANEL (ASP) AND FOUND SEVERAL HOLES IN THE PANEL. QA PERSONNEL WROTE PROBLEM INVESTIGATION REPORT (PIR) 0-M90-0038 TO DOCUMENT THE DISCOVERY OF THE HOLES IN THE ASP. SUBSEQUENTLY, COMPLIANCE PERSONNEL REQUESTED THAT DESIGN ENGINEERING PERSONNEL PERFORM AN OPERABILITY EVALUATION TO DETERMINE THE STATUS OF THE ASP. ON FEBRUARY 15, 1990, DESIGN ENGINEERING PERSONNEL ISSUED AN OPERABILITY EVALUATION AND DETERMINED THAT BECAUSE THE HOLES IN THE ASP WOULD ALLOW WATER SPRAY INTO THE CABINET THE OPERABILITY OF THE COMPONENTS CONTROLLED FROM THE ASP IS INDETERMINATE; THEREFORE, THEY ARE CONSIDERED INOPERABLE. THIS EVENT IS ASSIGNED A CAUSE OF UNKNOWN. IT COULD NOT BE DETERMINED DURING THIS INVESTIGATION WHEN OR UNDER WHAT DIRECTION THE HOLES WERE LEFT IN THE PANEL. THE MCGUIRE SAFETY REVIEW GROUP WILL CONTINUE TO INVESTIGATE THIS EVENT AND WILL WRITE AN ADDENDUM IF THE CAUSE IS DISCOVERED. CONSTRUCTION AND MAINTENANCE DIVISION (CMD) PERSONNEL HAVE BEGUN TO PLUG THE HOLES IN THE ASP. UNIT 1 WAS IN MODE 6 (REFUELING) AT THE TIME THIS EVENT WAS DISCOVERED.

[99] MCGUIRE 1 DOCKET 50-369 LER 90-012 REV 02
UPDATE ON LOOSE MATERIAL WAS LOCATED IN UPPER CONTAINMENT DURING UNIT OPERATION BECAUSE OF AN INAPPROPRIATE ACTION.
EVENT DATE: 052290 REPORT DATE: 062190 NSSS: WE TYPE: PWR

(NSIC 219435) LOOSE MATERIAL WAS DISCOVERED IN THE UNIT 1 UPPER CONTAINMENT ON MAY 22, 1990 AT 1100. UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT THE TIME OF THE EVENT DISCOVERY. UNIT 1 HAD ENTERED MODE 3 (HOT STANDBY) ON MAY 14, 1990 AT 1339. THE LOOSE MATERIAL PRESENT IN CONTAINMENT RESULTED IN A SURVEILLANCE REQUIREMENT OF TECHNICAL SPECIFICATION 3/4.5.2 NOT BEING MET. THE LOOSE MATERIAL WAS IMMEDIATELY REMOVED. SUBSEQUENTLY, AN INSPECTION WAS CONDUCTED OF THE UNIT 1 LOWER CONTAINMENT OUTSIDE OF THE CRANE WALL. NO LOOSE MATERIAL WAS FOUND. THIS EVENT IS ASSIGNED A CAUSE OF INAPPROPRIATE ACTION RESULTING FROM FAILURE TO FOLLOW PROCEDURE. THIS EVENT WILL BE COVERED WITH APPROPRIATE PERSONNEL. APPLICABLE PROCEDURES WILL BE REVIEWED AND CHANGED AS NECESSARY.

[100] MCGUIRE 1 DOCKET 50-369 LER 90-017 REV 01
UPDATE ON BOTH EMERGENCY DIESEL GENERATORS WERE INOPERABLE DUE TO EQUIPMENT FAILURE CAUSED BY INADEQUATE WORK CONTROL AND MANAGEMENT INTERFACE, AND AN INAPPROPRIATE ACTION.

EVENT DATE: 062690 REPORT DATE: 072690 NSSS: WE TYPE: PWR

(NSIC 219437) ON 6/26/90, AT 0905, DURING PERFORMANCE OF A DIESEL GENERATOR (D/G) OPERABILITY TEST, D/G 1A FAILED TO REACH THE REQUIRED VOLTAGE (4160 VAC) IN THE ALLOTTED TIME (11 SEC.). AT 1000, OPERATIONS PERSONNEL DECLARED D/G 1A INOPERABLE. INVESTIGATION REVEALED PAINT OVERSPRAY ON THE D/G 1A EXCITER COMMUTATOR RING. AT 1006, A SECOND START OF D/G 1A WAS ATTEMPTED. A VALID FAILURE OCCURRED DUE TO AN UNSUCCESSFUL LOADING ATTEMPT. OPS PERSONNEL AND MAINTENANCE PERSONNEL CONTINUED TO INVESTIGATE THE CAUSE OF THE FAILURES AND, AT APPROX. 1100, PAINT WAS NOTED ON THE BACK SIDE OF THE FUEL RACK PIVOT POINTS FOR D/G 1A WHICH CAUSED THEM TO BIND. INVESTIGATION REVEALED THE SAME PROBLEM ON D/G 1B. AT 1134, OPS PERSONNEL DECLARED D/G 1B INOPERABLE TO REMOVE PAINT FROM THE FUEL RACKS AND COMMUTATOR RING. OPS PERSONNEL ENTERED TECH SPEC (TS) ACTION STATEMENTS 3.0.3 AND 3.8.1.1F FOR UNIT 1. AT 1137, REMOVAL OF THE PAINT WAS COMPLETED ON D/G 1A. A THIRD START OF D/G 1A WAS ATTEMPTED AND THE OPERABILITY TEST WAS COMPLETED SATISFACTORILY. AT 1325, OPS PERSONNEL DECLARED D/G 1A OPERABLE AND EXITED UNIT 1 FROM TSS 3.0.3 AND 3.8.1.1F. UNIT 1 WAS IN MODE 1 AT 100% POWER. THIS INCIDENT IS ASSIGNED A CAUSE OF MANAGEMENT DEFICIENCY RESULTING FROM INADEQUATE WORK CONTROL AND MANAGEMENT INTERFACE, AND A CAUSE OF INAPPROPRIATE ACTION.

[101] MCGUIRE 1 DOCKET 50-369 LER 90-019 REV 01
UPDATE ON A DAILY CHANNEL CHECK FOR THE CONTAMINATED PARTS WAREHOUSE VENTILATION RADIATION MONITOR WAS MISSED BECAUSE OF INAPPROPRIATE ACTIONS.
EVENT DATE: 071190 REPORT DATE: 081090 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 219461) ON JUNE 1, 1990, A REVISION TO PROCEDURE PT/1/A/4600/03B, DAILY SURVEILLANCE ITEMS, WAS PUT INTO AFFECT. DURING THE REVISION PROCESS OF THIS PROCEDURE THE DAILY CHANNEL CHECK FOR EMF 53, RADIATION MONITOR FOR THE CONTAMINATED PARTS WAREHOUSE VENTILATION SYSTEM, WAS INADVERTENTLY DELETED FROM THE DAILY SURVEILLANCE ITEMS PROCEDURE CHECKLIST. NEITHER THE PREPARER OF THE PROCEDURE REVISION NOR THE QUALIFIED REVIEWER FOR THE PROCEDURE REVISION NOTICED THAT THE CHANNEL CHECK FOR EMF 53 WAS MISSING FROM THE CHECKLIST. ON JULY 11, 1990, DURING PERFORMANCE OF THE DAILY SURVEILLANCE ITEMS PROCEDURE, IT WAS DETERMINED THAT EMF 53 HAD BEEN INADVERTENTLY LEFT OFF THE CHECKLIST. THIS EVENT IS ASSIGNED CAUSES OF INAPPROPRIATE ACTIONS BECAUSE OF A LACK OF ATTENTION TO DETAIL BY OPERATIONS PERSONNEL. UNIT 1 AND UNIT 2 WERE IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER AT THE TIME OF DISCOVERY OF THIS EVENT. UNIT 1 VARIED IN POWER LEVELS FROM 5 PERCENT TO 100 PERCENT POWER WHILE THE TECHNICAL SPECIFICATION (TS) SURVEILLANCE WAS NOT BEING PERFORMED FOR EMF 53. UNIT 2 VARIED IN POWER FROM 93 PERCENT POWER TO 100 PERCENT POWER DURING THIS SAME TIME PERIOD. OPERATIONS PROCEDURE GROUP PERSONNEL IMMEDIATELY REVISED THE DAILY SURVEILLANCE ITEMS PROCEDURE BY ADDING THE DAILY CHANNEL CHECK FOR EMF 53 BACK ONTO THE CHECKLIST.

[102] MCGUIRE 1 DOCKET 50-369 LER 90-022
BOTH TRAINS OF THE RESIDUAL HEAT REMOVAL SYSTEM WERE INOPERABLE DURING QUARTERLY VALVE STROKE TIME TESTING BECAUSE OF IMPROPER SCHEDULING.
EVENT DATE: 071690 REPORT DATE: 091790 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 219510) ON JULY 11 1990, PERFORMANCE (PRF) PERSONNEL PREPARED A REISSUE OF PROCEDURE PT/2/A/4206/02B, SAFETY INJECTION (NI) TRAIN B VALVE STROKE TIMING QUARTERLY. ON JULY 16, 1990, PRF ENGINEER A PERFORMED THE NORMAL REVIEW OF THE PROCEDURE REISSUE. DURING THE COURSE OF THE REVIEW, PRF ENGINEER A IN CONJUNCTION WITH OPERATIONS PERSONNEL DISCOVERED THAT CYCLING VALVES 1 AND 2NI-136B, RESIDUAL HEAT REMOVAL (ND) HEAT EXCHANGER (HX) 1B AND 2B TO NI PUMP 1B AND 2B, WHILE AT POWER COULD DEGRADE ND SYSTEM OPERATION IN THE EVENT OF A LARGE BREAK LOSS OF COOLANT ACCIDENT (LBLOCA). DEGRADATION COULD OCCUR WHEN THE VALVES WERE CYCLED TO THE OPEN POSITION. THESE VALVES HAVE ROUTINELY BEEN STROKE TIME TESTED IN MODES 1 (POWER OPERATION), 2 (STARTUP), 3 (HOT STANDBY), AND 4 (HOT SHUTDOWN) SINCE PLANT STARTUP FOR EACH UNIT RESPECTIVELY. SUBSEQUENTLY, IT WAS DETERMINED THAT THE SAME SITUATION WAS ALSO TRUE FOR VALVES 1 AND 2NS-38, ND PUMP

B TO CONTAINMENT SPRAY (NS) NOZZLES CONTAINMENT ISOLATION, AND 1 AND 2NS-43, ND PUMP A TO NS NOZZLES CONTAINMENT ISOLATION, DURING PERFORMANCE OF PROCEDURES PT/1 AND 2/A/4208/02A AND B, NS TRAINS A AND B VALVE STROKE TIMING QUARTERLIES. THIS INCIDENT IS ASSIGNED A CAUSE OF IMPROPER SCHEDULING OF THE VALVE STROKE TIMING FOR THESE VALVES. BOTH UNITS WERE IN MODE 1 AT 100 PERCENT POWER AT THE TIME OF THE EVENT DISCOVERY.

[103] MCGUIRE 1 DOCKET 50-369 LER 90-020
BOTH TRAINS OF THE CONTROL ROOM VENTILATION SYSTEM WERE INOPERABLE BECAUSE OF AN UNKNOWN.
EVENT DATE: 080290 REPORT DATE: 090390 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 219438) ON 8/2/90, AT 0400, OPERATIONS (OPS) PERSONNEL VERIFIED THAT CONTROL ROOM VENTILATION (VC) VALVES 1VC-1, 2, 3, 4, 9, 10, 11, AND 12 WERE OPEN. AT 0639, 1EMF43B, CONTROL ROOM AIR RADIATION MONITOR FOR TRAIN B, ALARMED ON A LOSS OF SAMPLE FLOW. ALARM WAS ACKNOWLEDGED BY A CONTROL ROOM OPERATOR; IT CLEARED, THEN REALARMED. ONCE AGAIN, THE ALARM WAS ACKNOWLEDGED. SAMPLE FLOW WAS REESTABLISHED AND THE ALARM CLEARED. OPS PERSONNEL CONTACTED BOTH RADIATION PROTECTION (RP) AND INSTRUMENT AND ELECTRICAL (IAE) PERSONNEL TO DETERMINE IF ANY IN PROGRESS WORK ACTIVITIES COULD HAVE CAUSED THE ALARMS. NO ACTIVITIES WERE IDENTIFIED. AT 0650, OPS PERSONNEL WERE INFORMED BY AN NRC REPRESENTATIVE THAT HE HAD OBSERVED CONTROL ROOM AIR INTAKE VALVES 1VC-9 THROUGH 12 IN CLOSED POSITION. OPS PERSONNEL REOPENED VALVES, AND AN INVESTIGATION AS TO WHY THE VALVES WERE CLOSED WAS INITIATED. IT WAS NOTED THAT 1EMF43B HAD EXPERIENCED AN APPARENT HIGH RADIATION CONDITION OF 3.99 E+03 COUNTS PER MINUTE (CPM) AT THE SAME TIME THAT NORMAL FLOW WAS RESTORED TO THE RADIATION MONITOR (EMF). THE TRIP 2 ALARM SETPOINT TO CLOSE THE VALVES IS 9.0 E+02 CPM. IAE PERSONNEL FOUND NO PROBLEM WITH THE EMF AND WERE UNABLE TO DUPLICATE THE CLOSING OF VALVES 1VC-9 THROUGH 12. UNITS 1 AND 2 WERE IN MODE 1 (POWER OPERATION) AT 100% PERCENT POWER. THIS EVENT IS ASSIGNED A CAUSE OF UNKNOWN, POSSIBLE EQUIPMENT FAILURE/MALFUNCTION.

[104] MCGUIRE 1 DOCKET 50-369 LER 90-021
A TECHNICAL SPECIFICATION SURVEILLANCE ON THE CONTROL AREA VENTILATION SYSTEM WAS MISSED BECAUSE OF AN INAPPROPRIATE ACTION.
EVENT DATE: 080690 REPORT DATE: 090590 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 219462) ON AUGUST 5, 1990, BETWEEN 1600 AND 1900, A CONTROL ROOM OPERATOR INADVERTENTLY USED THE UNIT 2 SEMI-DAILY SURVEILLANCE ITEMS PROCEDURE CHECKLIST TO COMPLETE THE UNIT 1 SEMI-DAILY SURVEILLANCE REQUIREMENTS. OPERATIONS PERSONNEL DISCOVERED THIS ON AUGUST 6, 1990, AT APPROXIMATELY 0800. THERE ARE TWO ITEMS NOT LISTED ON THE UNIT 2 SEMI-DAILY SURVEILLANCE PROCEDURE CHECKLIST THAT ARE LISTED ON THE UNIT 1 PROCEDURE. THESE ITEMS ARE THE CONTROL ROOM AIR TEMPERATURE LESS THAN 90 DEGREES FAHRENHEIT VERIFICATION AND THE POSITION OF THE CONTROL AREA VENTILATION (VC) VALVES 1VC-1, 2, 3, 4, 9, 10, 11, AND 12 VERIFICATION. THE VC OUTSIDE AIR INTAKE VALVES WERE BEING VERIFIED IN THE OPEN POSITION HOURLY, WHEN THIS EVENT OCCURRED. THEREFORE, THE ONLY MISSED SURVEILLANCE WAS VERIFICATION OF THE CONTROL ROOM AIR TEMPERATURE. UNIT 1 AND UNIT 2 WERE IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER WHEN THIS EVENT OCCURRED. THIS EVENT WAS ASSIGNED A CAUSE OF INAPPROPRIATE ACTION BECAUSE OF LACK OF ATTENTION TO DETAIL. THIS EVENT WILL BE COVERED DURING A SHIFT SUPERVISORS MEETING TO REITERATE THE IMPORTANCE OF THE CONTROL OF APPROVED PROCEDURES.

[105] MCGUIRE 1 DOCKET 50-369 LER 90-023
BOTH TRAINS OF THE CONTROL ROOM VENTILATION SYSTEM WERE INOPERABLE BECAUSE OF AN EQUIPMENT FAILURE/MALFUNCTION.
EVENT DATE: 082190 REPORT DATE: 092090 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 219566) ON AUGUST 20, 1990, AT 0230, TRAIN B OF THE CONTROL ROOM

VENTILATION (VC) SYSTEM WAS DECLARED INOPERABLE FOR MODIFICATION AND FUNCTIONAL TESTING. MECHANICAL MAINTENANCE (MNT) PERSONNEL CONDUCTED A CHECK OF THE TRAIN A VC AIR HANDLING UNIT (AHU) AT APPROXIMATELY 1630, IN PREPARATION FOR MODIFICATION OF THAT TRAIN. TRAIN A VC WAS IN SERVICE AT THIS TIME. THIS CHECK REVEALED THAT THE TRAIN A VC AHU MOTOR BASE TENSIONING BOLTS WERE SHEARED OFF. THE TRAIN A VC AHU WAS SUBSEQUENTLY DECLARED INOPERABLE AT 1705. AT THAT TIME, BOTH TRAINS OF THE VC SYSTEM WERE INOPERABLE AND TS 3.0.3 WAS ENTERED. WORK ON TRAIN B VC SYSTEM WAS COMPLETED AND THE TRAIN WAS DECLARED OPERABLE ON AUGUST 21, 1990, AT 1755. REPAIRS WERE MADE TO THE TRAIN A VC SYSTEM AND IT WAS DECLARED OPERABLE ON AUGUST 22, 1990, AT 1845. UNITS 1 AND 2 WERE IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER AT THE TIME OF THIS EVENT. THIS EVENT IS ASSIGNED A CAUSE OF EQUIPMENT FAILURE/MALFUNCTION. PRODUCTION SUPPORT DEPARTMENT (PSD) PERSONNEL CONDUCTED AN ANALYSIS OF THE FAILED PARTS TO DETERMINE THE CAUSE OF THE FAILURE. DESIGN ENGINEERING (DE) PERSONNEL WILL EVALUATE THIS FAILURE AND RECOMMEND SOLUTIONS TO THIS PROBLEM.

[106] MCGUIRE 1 DOCKET 50-369 LER 90-024
INLEAKAGE ON TRAIN A OF THE CONTROL ROOM VENTILATION SYSTEM EXCEEDED ACCEPTABLE LEVELS BECAUSE OF AN UNKNOWN.
EVENT DATE: 082290 REPORT DATE: 092190 NSSS: WE TYPE: PWR

(NSIC 219547) ON AUGUST 22, 1990, AT APPROXIMATELY 1445, MECHANICAL MAINTENANCE (MNT) PERSONNEL DISCOVERED AIR INLEAKAGE ON THE CONTROL ROOM VENTILATION (VC) SYSTEM TRAIN A AIR HANDLING UNIT (AHU). THIS INLEAKAGE WAS DISCOVERED AFTER MNT PERSONNEL NOTICED AUDIBLE LEAKAGE ON THE VC SYSTEM TRAIN A AHU. THE INLEAKAGE WAS ESTIMATED BY MNT PERSONNEL TO BE SEVERAL HUNDRED CUBIC FEET PER MINUTE (CFM). THIS UNFILTERED AIR INLEAKAGE VIOLATED THE LEAKAGE ASSUMPTIONS FOR THE POST-ACCIDENT OPERATOR DOSE ACCORDING TO TECHNICAL SPECIFICATION (TS) REQUIREMENTS. THIS EVENT IS ASSIGNED A CAUSE OF UNKNOWN BECAUSE IT CANNOT BE DETERMINED HOW THE AHU FAN HOUSING BECAME DAMAGED, CAUSING THE UNFILTERED AIR INLEAKAGE TO OCCUR. ALSO, IT COULD NOT BE DETERMINED WHEN THE UNFILTERED AIR INLEAKAGE ON TRAIN A OF THE VC SYSTEM AHU FAN HOUSING BECAME UNACCEPTABLE. THEREFORE, THE VC SYSTEM TRAIN A WAS INOPERABLE FOR AN INDETERMINABLE TIME PERIOD. UNIT 1 AND UNIT 2 WERE IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER WHEN THIS VC SYSTEM INLEAKAGE WAS DISCOVERED. MNT PERSONNEL IMMEDIATELY SEALED THE AHU FAN HOUSING WITH RTV SEALANT. MAINTENANCE ENGINEERING SERVICES (MES) PERSONNEL WILL INVESTIGATE FURTHER A PROBABLE FAILURE OR DEFICIENCY CAUSE FOR THE FAN HOUSING DAMAGE.

[107] MCGUIRE 2 DOCKET 50-370 LER 90-007
FORCED SHUTDOWN DUE TO AN UNIDENTIFIED REACTOR COOLANT LEAK CAUSED BY AN UNKNOWN.
EVENT DATE: 081590 REPORT DATE: 091790 NSSS: WE TYPE: PWR
VENDOR: KEROTEST MANUFACTURING CORP.

(NSIC 219548) ON AUGUST 16, 1990 AT 0200, PROCEDURE RP/0/A/5700/01, NOTIFICATION OF UNUSUAL EVENT WAS IMPLEMENTED DUE TO AN UNIDENTIFIED REACTOR COOLANT LEAK >1 GALLON PER MINUTE (GPM) ON UNIT 2. THIS DETERMINATION WAS BASED UPON THE COMPLETION OF 2 REACTOR COOLANT LEAKAGE CALCULATIONS. THESE CALCULATIONS SHOWED REACTOR COOLANT LEAKAGE OF APPROXIMATELY 1.5 GPM. AT 0210, OPS PERSONNEL NOTIFIED THE NRC AND COMMENCED A FORCED LOAD REDUCTION ON UNIT 2. THE UNIT DECREASED POWER AT 25 PERCENT AN HOUR AND AT 0613 ENTERED MODE 3 (HOT STANDBY). IT WAS LATER DETERMINED THAT A UNIT 2 SAFETY INJECTION SYSTEM VENT VALVE, 2NI-458, WAS LEAKING REACTOR COOLANT WATER INTO UNIT 2 CONTAINMENT FLOOR AND EQUIPMENT SUMP B. THE LEAKAGE FROM VALVE 2NI-458 WAS SECURED AND ON AUGUST 17, 1990, AT 0135, UNIT 2 ENTERED MODE 1 (POWER OPERATION). PRIOR TO THIS EVENT, UNIT 2 WAS IN MODE 1 AT 100 PERCENT POWER. THIS EVENT HAS BEEN ASSIGNED A CAUSE OF UNKNOWN SINCE IT IS NOT KNOWN HOW THE VENT VALVE BECAME PARTIALLY OPENED. TO PREVENT THIS EVENT FROM RECURRING, PROJECT SERVICES PERSONNEL WILL EVALUATE REROUTING THE DISCHARGE OF NI VENT VALVES TO THE PRESSURIZER RELIEF TANK AND OPS PERSONNEL WILL EVALUATE ADDING DOUBLE ISOLATION VALVES AND/OR PIPE CAPS TO EXISTING VENT VALVES LOCATED NEAR THE SAFETY INJECTION/REACTOR COOLANT INTERFACE.

[108] MILLSTONE 1 DOCKET 50-245 LER 90-013
STANDBY GAS TREATMENT SYSTEM INITIATION DUE TO INADVERTENT DE-ENERGIZATION OF POWER SUPPLY TO THE CHANNEL 2 PROCESS RADIATION MONITORING SYSTEM.
EVENT DATE: 080690 REPORT DATE: 090590 NSSS: GE TYPE: BWR

(NSIC 219507) ON 8/6/90, AT 1055 HOURS, WITH THE PLANT AT 100% POWER (530F AND 1030 PSIG), THE STANDBY GAS TREATMENT SYSTEM AUTOMATICALLY INITIATED AND THE REACTOR BUILDING VENTILATION SYSTEM ISOLATED WHEN THE COMMON POWER SUPPLY FOR THE CHANNEL 2 PROCESS RADIATION MONITORING SYSTEM WAS INADVERTENTLY DE-ENERGIZED. POWER WAS IMMEDIATELY RESTORED TO CHANNEL 2 OF THE PROCESS RADIATION MONITORING CIRCUIT. OPERATIONS PERSONNEL RESET THE INITIATION TRIP LOGIC, SECURED THE STANDBY GAS TREATMENT SYSTEM AND RESTORED REACTOR BUILDING VENTILATION. THE STANDBY GAS TREATMENT SYSTEM FUNCTIONED AS REQUIRED FOLLOWING THE DOWNSCALE TRIP ON THE REACTOR BUILDING VENTILATION EXHAUST RADIATION MONITOR AND THE REFUEL FLOOR RADIATION MONITOR. NO SAFETY CONSEQUENCES RESULTED FROM THE EVENT.

[109] MILLSTONE 2 DOCKET 50-336 LER 89-008 REV 01
UPDATE ON INCOMPLETE SURVEILLANCE REQUIREMENTS FOR TESTING OF ANNUNCIATOR CIRCUITS.
EVENT DATE: 100289 REPORT DATE: 091890 NSSS: CE TYPE: PWR

(NSIC 219508) ON 10/2/89 AT 1300 HOURS WHILE OPERATING AT 92% NORMAL POWER OPERATION, A MANAGEMENT REVIEW OF A REPORTABLE OCCURRENCE SUBMITTED BY ANOTHER LICENSEE DETERMINED THAT MILLSTONE 2 WAS NOT PERFORMING A FUNCTIONAL CHECK OF CONTROL BOARD ANNUNCIATORS DURING TECH SPEC "CHANNEL FUNCTIONAL TESTS". THIS WAS CONSIDERED A CONDITION PROHIBITED BY THE PLANT'S TECH SPECS. THE FAILURE TO PERFORM A FUNCTIONAL TEST OF THE ANNUNCIATORS HAS NO SAFETY CONSEQUENCES SINCE THE TRIP AND OPERATIONAL FUNCTIONS OF THE EQUIPMENT WERE BEING VERIFIED AS REQUIRED. ADDITIONALLY, THE ANNUNCIATOR PORTION OF THE CIRCUIT PERFORMS NO SAFETY FUNCTION. ACTION WAS TAKEN TO COMMENCE A REVIEW OF ALL APPLICABLE INSTRUMENTATION AND CONTROL PROCEDURES. THE REVIEW OF PROCEDURES IDENTIFIED SIX PROCEDURES THAT REQUIRED REVISIONS, TO ENSURE PROPER TESTING OF ANNUNCIATORS. EACH OF THESE PROCEDURES HAS BEEN REVISED TO INCLUDE THE REQUIRED ANNUNCIATOR TESTING.

[110] MILLSTONE 3 DOCKET 50-423 LER 90-013 REV 01
UPDATE ON MANUAL REACTOR TRIP DUE TO IMMINENT LOSS OF CONDENSER VACUUM.
EVENT DATE: 041690 REPORT DATE: 081390 NSSS: WE TYPE: PWR

(NSIC 219378) ON APRIL 16, 1990, AT 1201 HOURS WITH THE PLANT AT 48% POWER IN MODE 1, A MANUAL REACTOR TRIP WAS INITIATED BECAUSE OF AN ANTICIPATED TURBINE TRIP DUE TO LOSS OF CONDENSER VACUUM. CIRCULATING WATER PUMP 3CWS-P1B WAS PROVIDING COOLING WATER FOR CONDENSER WATERBOXES "A" AND "B". A RAPID BUILDUP OF SEAWEED ON "B" TRAVELING SCREEN RESULTED IN AN AUTOMATIC TRIP OF 3CWS-P1B DUE TO HIGH SCREEN DIFFERENTIAL LEVEL. AS THE LOSS OF COOLING TO TWO CONDENSER BAYS WOULD HAVE RESULTED IN A LOW CONDENSER VACUUM, A REACTOR TRIP WAS INITIATED. THE REACTOR TRIP CAUSED A MAIN TURBINE AND GENERATOR TRIP IN ACCORDANCE WITH DESIGN. THE ROOT CAUSE OF THE EVENT WAS INADEQUATE ADMINISTRATIVE GUIDANCE IN THAT DEBRIS WAS ALLOWED TO COLLECT ON THE TRASH RACK. DURING TRASH RACK RAKING, SEAWEED BROKE FREE AND CLOGGED "B" TRAVELING SCREEN. HIGHER THAN NORMAL SEAWEED INFLUX AND 130% FLOW DUE TO CROSS CONNECTING A & B WATERBOXES CONTRIBUTED TO THE HIGH WATER PUMP TO SUPPLY ONLY THE ASSOCIATED WATERBOX DURING SEVERE WEATHER. FOR CORRECTIVE \CTION, THE TRASH RA HIGH DIFFERENTIAL LEVEL ALARM POINT HAS BEEN LOWERED FROM 15 INCHES TO 6 INCH . PERSONNEL HAVE BEEN INSTRUCTED TO CLOSELY MONITOR THE TRASH RACK WATER LEV LS AND CLEAN THE RACKS BEFORE LEVELS EXCEED 4 INCHES.

[111] MILLSTONE 3 DOCKET 50-423 LER 90-018 REV 01
UPDATE ON IMPROPERLY ESTABLISHED FIRE WATCH DUE TO MISCOMMUNICATION.
EVENT DATE: 06[illegible]190 REPORT DATE: 081390 NSSS: WE TYPE: PWR

(NSIC 219389) ON 6/2/90, AT 1245 HOURS, WITH THE PLANT AT 100% POWER (MODE 1), THE SHIFT SUPERVISOR (SS) DISCOVERED THAT AN HOURLY FIRE WATCH PATROL HAD NOT

BEEN PROPERLY ESTABLISHED IN THE ESF SUMP AREA AFTER ASSOCIATED FIRE RATED ASSEMBLIES HAD BEEN DECLARED INOPERABLE. THE DURATION OF THE EVENT WAS APPROXIMATELY 26 HOURS. ON 6/1/90 AT APPROXIMATELY 1100 HOURS, FOUR FIRE STOP AND SEAL PENETRATIONS WERE DECLARED INOPERABLE IN THE ESF BUILDING SUMP AND THE "A" TRAIN CONTAINMENT RECIRCULATION SYSTEM (RSS) PIPE TUNNEL AREAS IN ASSOCIATED WITH A TECH SPEC SURVEILLANCE. AN HOURLY FIREWATCH PATROL WAS ESTABLISHED IN THE 4 FT. 6 IN. ELEVATION OF THE ESF BUT DID NOT ENCOMPASS THE ESF SUMP AREA. THE ROOT CAUSE OF THE EVENT WAS MISCOMMUNICATION BETWEEN SHIFT PERSONNEL WHICH RESULTED IN AN HOURLY FIRE WATCH PATROL SIGNATURE SHEET BEING PLACED IN AN INCORRECT LOCATION. IMMEDIATE CORRECTIVE ACTION WAS TO ESTABLISH AN HOURLY FIREWATCH PATROL IN THE ESF SUMP AREA. THE SS HAS BEEN COUNSELED IN THE IMPORTANCE OF VERIFYING COMMUNICATIONS WITH SHIFT PERSONNEL.

[112] MILLSTONE 3 DOCKET 50-423 LER 90-026
HYDROGEN MONITOR TEMPERATURE PROFILE/OPERATING SPECIFICATION INCONSISTENT DUE TO INADEQUATE DESIGN ENGINEERING INTERFACE.
EVENT DATE: 062590 REPORT DATE: 072590 NSSS: WE TYPE: PWR

(NSIC 219074) AT 1530 HOURS ON 6/25/90, WHILE OPERATING IN MODE 1, AT 100% POWER, 587F AND 2260 PSIA, THE "B" TRAIN CONTAINMENT HYDROGEN MONITOR FAILED A CALIBRATION SURVEILLANCE. THE APPLICABLE TECH SPECS LIMITED CONDITION FOR OPERATION (LCO) HAD BEEN ENTERED AT 0720 HOURS ON 6/25/90, AND THE PLANT REMAINED IN THE LCO UNTIL THE "B" TRAIN HYDROGEN MONITOR CALIBRATION WAS SUCCESSFULLY COMPLETED AT 1630, ON 6/26/90. ON 7/22/90, AN INCONSISTENCY WAS DISCOVERED BETWEEN THE ANTICIPATED AMBIENT TEMPERATURE PROFILE AT THE CONTAINMENT HYDROGEN MONITORS, AND THE OPERATING TEMPERATURE REQUIREMENTS OF THE MONITORS. DURING SOME CONDITIONS, THE AMBIENT TEMPERATURE COULD EXCEED THAT RECOMMENDED BY THE MANUFACTURER. IT WAS IMMEDIATELY DIRECTED THAT THE RECOMBINER BUILDING VENTILATION SYSTEM BE MAINTAINED IN CONTINUOUS SERVICE. THE CONTAINMENT HDYROGEN MONITORS WERE REVIEWED FOR OPERABILITY AND WERE DETERMINED TO BE OPERABLE PROVIDED THAT THE AMBIENT TEMPERATURE IS MAINTAINED BELOW 90F. THIS FINDING WAS DOCUMENTED IN A JUSTIFICATION FOR CONTINUED OPERATION ON 7/25/90. THE ROOT CAUSE OF THE EVENT IS INADEQUATE ENGINEERING INTERFACE BETWEEN DESIGN ORGANIZATIONS FOR THE RECOMBINER BUILDING AND THE CONTAINMENT HYDROGEN MONITORS.

[113] MILLSTONE 3 DOCKET 50-423 LER 90-027
BLOCKED OPEN FIRE DOOR WITHOUT COMPENSATORY FIRE WATCH DUE TO INADEQUATE TRAINING.
EVENT DATE: 081190 REPORT DATE: 091090 NSSS: WE TYPE: PWR

(NSIC 219524) ON 8/11/90, AT 0840, WITH THE PLANT AT 99% POWER (MODE 1), AT 586F AND 2250 PSIA, A NON-LICENSED OPERATOR (PEO) PERFORMING ROUTINE ROUNDS DISCOVERED A TECH SPEC FIRE DOOR BLOCKED OPEN. AN HOURLY FIRE WATCH WAS NOT ESTABLISHED. THE DURATION OF THE EVENT WAS APPROXIMATELY 3 HOURS. THE ROOT CAUSE OF THE EVENT WAS INADEQUATE TRAINING OF FIRE WATCH PERSONNEL. THE FIRE WATCH SUPERVISOR WAS AWARE THE DOOR WAS LEFT IN ITS OPEN POSITION BUT DID NOT CONVEY THIS INFORMATION TO THE SHIFT SUPERVISOR. IMMEDIATE CORRECTIVE ACTION WAS TO CLOSE THE SUBJECT DOOR. AS ACTION TO PREVENT RECURRENCE, FIRE WATCH PERSONNEL MANAGEMENT HAS ISSUED A MEMORANDUM TO ITS PERSONNEL REINFORCING THE IMPORTANCE OF RELATING RELEVANT FIRE WATCH PATROL INFORMATION TO SHIFT MANAGEMENT. THIS INFORMATION WILL BE INCORPORATED INTO THE FIRE WATCH PERSONNEL INDOCTRINATION TRAINING MODULE BY 10/30/90. IN ADDITION, THE FIRE WATCH SERVICES SITE SUPERINTENDENT HAS BEEN REPLACED DUE TO RECOGNIZED FIRE WATCH ADMINISTRATION DEFICIENCIES.

[114] MONTICELLO DOCKET 50-263 LER 90-001 REV 02
UPDATE ON POTENTIAL EMERGENCY FILTER TRAIN SYSTEM INOPERABILITY DUE TO INTERACTION WITH NON-SAFETY RELATED EQUIPMENT.
EVENT DATE: 031390 REPORT DATE: 091490 NSSS: GE TYPE: BWR

(NSIC 219492) THE REPORT CONCERNS THE DISCOVERY OF SYSTEM DESIGN DEFICIENCIES. NO OPERATIONAL EVENTS, EQUIPMENT FAILURES OR PROCEDURE VIOLATIONS OCCURRED. ON MARCH 13, 1990, DESIGN DEFICIENCIES IN THE EMERGENCY FILTER TRAIN SYSTEM, AND SYSTEMS WHICH INTERACT WITH THE EMERGENCY FILTER TRAIN SYSTEM WERE DISCOVERED DURING A SPECIAL TEST. AN IN-DEPTH INVESTIGATION OF THE EMERGENCY FILTER TRAIN

DESIGN WAS INITIATED WHICH UNCOVERED ADDITIONAL DESIGN DEFICIENCIES AND SAFETY RELATED/NON-SAFETY RELATED SYSTEM INTERACTIONS. IMMEDIATE CORRECTIVE ACTIONS WERE TAKEN TO ISOLATE AND SECURE VARIOUS VENTILATION UNITS AND DUCTWORK TO PREVENT SAFETY RELATED/NON-SAFETY RELATED SYSTEMS INTERACTION AND ENSURE OPERABILITY OF THE EMERGENCY FILTER TRAIN SYSTEM. FOLLOWING COMPLETION OF FUNCTIONAL AND SAFETY ANALYSES, AND ISSUANCE OF PROCEDURE CHANGES, SOME OF THE ISOLATED EQUIPMENT HAS BEEN RETURNED TO SERVICE. INVESTIGATION INTO KNOWN DESIGN DEFICIENCIES IS CONTINUING.

[115] MONTICELLO DOCKET 50-263 LER 90-009
INOPERABLE FIRE BARRIER PENETRATION SEAL DUE TO NON-COMPLIANCE WITH APPROVED PLANT PROCEDURES.
EVENT DATE: 071990 REPORT DATE: 081890 NSSS: GE TYPE: BWR

(NSIC 219342) ON JULY 19, 1990 WITH THE PLANT OPERATING AT 100% POWER AN UNSEALED PENETRATION BETWEEN TWO FIRE AREAS WAS DISCOVERED BY AN OPERATOR DURING NORMAL ROUNDS. THIS IS A CONDITION WHICH IS CONTRARY TO THE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.13.G. IT IS BELIEVED THAT DUE TO A COGNITIVE PERSONNEL ERROR, THE PENETRATION WAS OPENED AND NOT RE-SEALED. THIS WAS NOT IN COMPLIANCE WITH APPROVED PLANT PROCEDURES. CORRECTIVE ACTIONS INCLUDED SEALING OF THE PENETRATION AND DEVELOPMENT OF SEVERAL LONG TERM CORRECTIVE ACTIONS TO CHANGE THE OVERALL APPROACH TO PENETRATION SEAL INVENTORY AND CONTROL AT MONTICELLO. THE CORRECTIVE ACTIONS TO PREVENT RECURRENCE INVOLVE TECHNICAL STAFF TRAINING ON THE EVENT, REVISION OF INSPECTION AND CONTROL PROCEDURES, AND INITIATING A PROJECT TO IDENTIFY AND CATALOG ALL PENETRATIONS BASED ON FIELD VERIFICATIONS.

[116] MONTICELLO DOCKET 50-263 LER 90-010
SPURIOUS SIGNAL FROM SPENT FUEL POOL RADIATION MONITOR CAUSING A PARTIAL GROUP II ISOLATION, STANDBY GAS TREATMENT INITIATION AND REACTOR BUILDING ISOLATION.
EVENT DATE: 080790 REPORT DATE: 090690 NSSS: GE TYPE: BWR

(NSIC 219493) ON AUGUST 7, 1990, WITH THE PLANT OPERATING AT 100% POWER, A PARTIAL GROUP II ISOLATION, STANDBY GAS TREATMENT SYSTEM INITIATION, AND REACTOR BUILDING VENTILATION ISOLATION OCCURRED. THIS WAS THE RESULT OF A SPURIOUS SIGNAL FROM THE SPENT FUEL POOL RADIATION MONITOR DURING A ROUTINE FUNCTIONAL TEST OF THE MONITOR. THE MONITOR INDICATOR WAS CHECKED AND FOUND TO BE AT THE NORMAL RADIATION LEVEL. THE TRIP WAS RESET AND THE GROUP II VALVES, STANDBY GAS TREATMENT SYSTEM AND REACTOR BUILDING VENTILATION WERE RETURNED TO NORMAL. THE MONITOR WAS SATISFACTORILY FUNCTIONALLY TESTED. INITIAL TROUBLE SHOOTING OF THE RADIATION MONITOR, POWER SUPPLY, AND TRIP AND INDICATOR UNIT REVEALED NO PROBLEM WITH THE UNITS, AND NO REASON FOR THE SPURIOUS TRIP. A LIKELY CAUSE OF THE TRIP IS THE JARRING OF THE TRIP AND INDICATOR UNIT WHICH OCCURS WHILE SLIDING THE UNIT BACK INTO THE PANEL NEAR THE END OF THE FUNCTIONAL TEST. THE TEST WILL BE REVISED TO ELIMINATE UNNECESSARY TRIP AND INDICATOR UNIT MOVEMENTS. EFFORTS ARE CONTINUING TO IDENTIFY ANY EQUIPMENT MALFUNCTION THAT MAY HAVE RESULTED IN THE SPURIOUS TRIP.

[117] MONTICELLO DOCKET 50-263 LER 90-011
ENGINEERED SAFETY FEATURE ACTUATION DUE TO PERSONNEL ERROR DURING SURVEILLANCE TESTING.
EVENT DATE: 082290 REPORT DATE: 092190 NSSS: GE TYPE: BWR

(NSIC 219536) DURING STEADY STATE POWER OPERATION, BOTH TRAINS OF THE CONTROL ROOM EMERGENCY FILTRATION TRAIN SYSTEM (EFT) WERE TRIPPED INTO THE EMERGENCY MODE OF OPERATION. THE CAUSE OF THE EVENT WAS A PERSONNEL ERROR DURING PERFORMANCE OF THE CONTROL ROOM INTAKE RADIATION MONITOR MONTHLY SURVEILLANCE TEST. ONCE THE CAUSE OF THE EVENT WAS DETERMINED, CONTROL ROOM PERSONNEL RESTORED BOTH TRAINS OF THE EMERGENCY FILTRATION TRAIN SYSTEM TO THEIR NORMAL LINEUP. THE I&C SPECIALIST RESPONSIBLE FOR PERFORMING THE TEST FAILED TO FOLLOW THE APPROVED PROCEDURE; HE WAS DISCIPLINED UNDER THE COMPANY DISCIPLINE PROGRAM.

[118] NINE MILE POINT 1 DOCKET 50-220 LER 90-006
UNVERIFIED ASSUMPTION IN APPENDIX R SAFE SHUTDOWN ANALYSIS.
EVENT DATE: 102789 REPORT DATE: 081090 NSSS: GE TYPE: BWR

(NSIC 219339) ON OCTOBER 27, 1989, WITH NINE MILE POINT UNIT 1 (NMP1) IN COLD SHUTDOWN AND THE CORE OFF-LOADED, IT WAS DISCOVERED THAT AN ASSUMPTION MADE IN THE APPENDIX R SAFE SHUTDOWN ANALYSIS (SSA) COULD NOT BE VERIFIED. THIS CONDITION WAS IDENTIFIED DURING CORRECTIVE ACTIONS BEING CARRIED OUT AS PART OF THE NMP1 RESTART ACTION PLAN (RAP), SPECIFIC ISSUE 18 (125 VDC SYSTEM CONCERNS). NIAGARA MOHAWK POWER CORPORATION (NMPC) FEELS IT PRUDENT TO REPORT THIS ISSUE AS A VOLUNTARY LER. NMPC IS UNABLE TO PROVIDE CONCLUSIVE DOCUMENTATION FOR THE ACTIONS ASSUMED IN THE APPENDIX R SSA. THE CAUSE OF THIS CONDITION WAS THE FIRE PROTECTION PROGRAM'S FAILURE TO PROVIDE DETAILED PROCEDURAL INSTRUCTIONS FOR IMPLEMENTING OPERATOR ACTIONS CREDITED BY THE APPENDIX R SSA. CORRECTIVE ACTIONS INCLUDED REVISION AND DEVELOPMENT OF NEW PROCEDURES, DEVELOPMENT OF STATION BATTERY LOAD PROFILES AND REPLACEMENT OF STATION BATTERIES.

[119] NINE MILE POINT 1 DOCKET 50-220 LER 90-016
UNUSUAL EVENT CLASSIFICATION AND REACTOR SHUTDOWN DUE TO EXCESS DRYWELL LEAKAGE RESULTING FROM UNADJUSTED EMERGENCY RELIEF VALVE PILOT VALVES.
EVENT DATE: 073090 REPORT DATE: 082990 NSSS: GE TYPE: BWR
VENDOR: TECHNO CORP.

(NSIC 219505) ON 7/30/90, AT 1116 HOURS, DURING A REACTOR STARTUP AT NINE MILE POINT UNIT 1 (NMP1), AN UNUSUAL EVENT EMERGENCY CLASSIFICATION WAS DECLARED AND AN ORDERLY REACTOR SHUTDOWN WAS COMPLETED DUE TO REACTOR COOLANT SYSTEM INTEGRITY BEING BREACHED AS INDICATED BY AN INCREASE IN UNIDENTIFIED LEAKAGE BEYOND THAT ALLOWED BY TECH SPECS. POST-SHUTDOWN INVESTIGATION REVEALED THAT A COMBINATION OF EQUIPMENT MALFUNCTION AND EQUIPMENT FAILURE LED TO THE UNIDENTIFIED LEAKAGE CONDITION. LEAKAGE FROM TWO UNADJUSTED EMERGENCY RELIEF VALVE (ERV) PILOT VALVES AND A FAILED OPEN VACUUM BREAKER ON ONE OF THE ERV'S DOWNCOMERS RESULTED IN STEAM AND WATER VAPOR COLLECTING AND CONDENSING IN THE DRYWELL, WHICH GENERATED THE UNIDENTIFIED LEAKAGE CONDITION. THE ROOT CAUSE FOR THE UNADJUSTED ERV PILOT VALVES WAS A PROCEDURAL DEFICIENCY WHICH RESULTED IN MAINTENANCE PERSONNEL FAILING TO PERFORM CERTAIN CRITICAL STEPS DURING RECENT MAINTENANCE ACTIVITIES. THE CAUSE FOR THE OPEN VACUUM BREAKER WAS A FAILED INTERNAL SPRING THAT NORMALLY HOLDS THE VACUUM BREAKER IN THE CLOSED POSITION. FOLLOWING THE REACTOR SHUTDOWN, CORRECTIVE ACTION WAS TAKEN TO READJUST THE ERV PILOT VALVES AND REPAIR THE DOWNCOMER VACUUM BREAKER. ADDITIONAL CORRECTIVE ACTIONS WILL BE TAKEN TO CORRECT THE SPECIFIC PROCEDURAL DEFICIENCY AND ITS POTENTIAL IMPACT ON MAINTENANCE PROCEDURES.

[120] NINE MILE POINT 1 DOCKET 50-220 LER 90-017
TURBINE TRIP/MANUAL SCRAM DUE TO TURBINE GENERATOR BEARING FAILURE.
EVENT DATE: 080690 REPORT DATE: 090490 NSSS: GE TYPE: BWR
VENDOR: GEN ELEC CO (STEAM TURB/ENGRD PROD)

(NSIC 219483) ON AUGUST 6, 1990, WITH THE REACTOR MODE SWITCH IN THE RUN POSITION, REACTOR POWER AT 19 PERCENT, NINE MILE POINT UNIT 1 (NMP1) EXPERIENCED TURBINE BEARING VIBRATION PROBLEMS WHEN STARTING UP THE TURBINE-GENERATOR. THE TURBINE WAS MANUALLY TRIPPED, CONDENSER VACUUM WAS BROKEN, AND A MANUAL REACTOR SCRAM WAS INSERTED. CONSEQUENTLY, A HIGH PRESSURE COOLANT INJECTION (MODE OF FEEDWATER CONTROL) SIGNAL WAS RECEIVED DUE TO LOW REACTOR WATER LEVEL (53 INCHES) AND A MAIN STEAM ISOLATION VALVE ISOLATION OCCURRED ON DECREASING CONDENSER VACUUM (7 INCHES MERCURY HG). THE IMMEDIATE CAUSE OF THIS EVENT WAS THE FAILURE OF TURBINE BEARING #5 DUE TO A BLANK FOUND INSTALLED IN THE BEARING OIL SUPPLY LINE. AN INCIDENT INVESTIGATION DETERMINED THAT THE CAUSE OF THE EVENT WAS INADEQUATE REVIEW OF A POLICY CHANGE AND PERSONNEL ERROR. IMMEDIATE CORRECTIVE ACTIONS INCLUDED REPAIR OF THE BEARING, INSPECTION OF THE OTHER BEARING OIL SUPPLY LINES, INSPECTION/CLEANUP OF THE TURBINE LUBE OIL TANK AND SCREENS, AND REVISING THE PROCEDURE/PROCESS THAT CONTROLS THE BLANK INSTALLATION. ADDITIONALLY, A REVIEW OF THE NMPC TURBINE MAINTENANCE LOG WAS PERFORMED TO IDENTIFY ANY OTHER POTENTIAL WEAKNESSES.

[121] NINE MILE POINT 1 DOCKET 50-220 LER 90-018
BREACH OF SECONDARY CONTAINMENT INTEGRITY DUE TO MOMENTARY OPENING OF DOUBLE AIRLOCK DOORS.
EVENT DATE: 080690 REPORT DATE: 091290 NSSS: GE TYPE: BWR

(NSIC 219484) ON AUGUST 6, 1990, AT APPROXIMATELY 1330 HOURS WITH THE MODE SWITCH IN THE "RUN" POSITION, IT WAS REPORTED THAT REACTOR BUILDING SECONDARY CONTAINMENT INTEGRITY HAD BEEN MOMENTARILY BREACHED. THIS EVENT INVOLVED THE INADVERTENT SIMULTANEOUS OPENING OF A DOUBLE DOORED ACCESSWAY FROM THE TURBINE BUILDING TO THE REACTOR BUILDING. ALL ACCESS OPENINGS TO THE REACTOR BUILDING HAVE AS A MINIMUM TWO AIRLOCK DOORS IN SERIES. TECHNICAL SPECIFICATION (T.S.) CONDITIONS REQUIRE THAT ONLY ONE DOOR IN EACH OF THE DOUBLE DOORED ACCESSWAYS SHALL BE OPENED AT ONE TIME. THE ROOT CAUSE FOR THIS EVENT IS A DOOR POSITION INDICATING SYSTEM THAT IS NOT DESIGNED TO PRECLUDE THE SIMULTANEOUS OPENING OF BOTH DOORS OF A DOUBLE DOORED ACCESSWAY. INITIAL CORRECTIVE ACTION INCLUDED THE IMMEDIATE CLOSURE OF THE TURBINE BUILDING SIDE AIR LOCK DOOR. ADDITIONAL ACTIONS INCLUDE: VERIFYING PROPER DOOR POSITION INDICATOR SWITCH SETTING; PURSUING THE POSSIBILITY OF SUBMITTING A TECHNICAL SPECIFICATION AMENDMENT; AND REVISING SECURITY MAINTENANCE PROCEDURE FOR "SITE DOORS".

[122] NINE MILE POINT 2 DOCKET 50-410 LER 90-005 REV 01
UPDATE ON ESF ACTUATIONS CAUSED BY HIGH DIFFERENTIAL FLOW DUE TO PERSONNEL ERROR.
EVENT DATE: 020690 REPORT DATE: 081690 NSSS: GE TYPE: BWR

(NSIC 219369) THIS SUPPLEMENT IS A COMPLETE REVISION OF LER 90-05 AND ADDRESSES ONE OF THE TWO EVENTS ADDRESSED IN LER 90-05. AN ADDITIONAL SUPPLEMENT WILL BE SUBMITTED TO ADDRESS THE REMAINING EVENT. ON FEBRUARY 6, 1990, AT 0019 HOURS, NINE MILE POINT UNIT 2 EXPERIENCED THE ACTUATION OF AN ENGINEERED SAFETY FEATURE, SPECIFICALLY, ISOLATION OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM. AT THE TIME OF THE EVENT, THE PLANT WAS AT FOUR (4) PERCENT POWER WITH THE MODE SWITCH IN THE "START UP" POSITION (OPERATIONAL CONDITION 2) AND AT APPROXIMATELY 522 DEGREES FAHRENHEIT AND 900 POUNDS PER SQUARE INCH GAUGE. THE ISOLATION, INITIATED BY A HIGH DIFFERENTIAL FLOW SIGNAL, OCCURRED AS THE PLANT OPERATORS ATTEMPTED TO MANIPULATE THE RWCU SYSTEM. THE ROOT CAUSE FOR THIS EVENT HAS BEEN DETERMINED TO BE A PERSONNEL ERROR DUE TO A KNOWLEDGE DEFICIENCY. IMMEDIATE CORRECTIVE ACTIONS WERE FOR THE LICENSED OPERATORS TO VERIFY THE AUTOMATIC RESPONSE OF THE RWCU SYSTEM, VERIFY THE PLANT STATUS AS NORMAL, RESET THE ISOLATION AND RESTORE THE RWCU SYSTEM TO SERVICE. OTHER CORRECTIVE ACTIONS INCLUDE ISSUING A WORK REQUEST TO TROUBLESHOOT INSTRUMENTATION, GENERATING A LESSONS LEARNED TRANSMITTAL, AND PROVIDING ADDITIONAL TRAINING FOR APPROPRIATE PERSONNEL.

[123] NINE MILE POINT 2 DOCKET 50-410 LER 90-008 REV 01
UPDATE ON TECHNICAL SPECIFICATION VIOLATION, MISSED CHEMISTRY SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 040290 REPORT DATE: 082790 NSSS: GE TYPE: BWR

(NSIC 219512) AT APPROXIMATELY 0900 HOURS ON APRIL 2, 1990, WITH THE PLANT OPERATING AT 100% RATED THERMAL POWER AND THE MODE SWITCH IN THE "RUN" POSITION, IT WAS DISCOVERED THAT THE MONTHLY OFFSITE DOSE CALCULATION REQUIRED TO BE COMPLETED BY MIDNIGHT APRIL 1, 1990, HAD BEEN MISSED. THIS IS A VIOLATION OF STATION TECHNICAL SPECIFICATIONS SECTION 4.11.2.2, 4.11.2.3 AND TABLE 4.11.2-1.2. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. CORRECTIVE ACTIONS TAKEN FOR THIS EVENT INCLUDED 1) COMPLETION OF THE SURVEILLANCES BY 1000 HOURS, APRIL 2, 1990; 2) DISCIPLINARY ACTION FOR THE INDIVIDUALS INVOLVED; 3) REPLACEMENT OF THE PART TIME CHEMISTRY PLANNER WITH A FULL TIME PLANNER; 4) ADDITIONAL INSTRUCTION TO CHEMISTRY DEPARTMENT MEMBERS ON THE CONDUCT AND SCHEDULING OF SURVEILLANCES, AND 5) GENERATION OF A LESSONS LEARNED TRANSMITTAL.

[124] NORTH ANNA 2 DOCKET 50-339 LER 90-002
2H EMERGENCY DIESEL GENERATOR START DUE TO "B" RESERVE STATION SERVICE TRANSFORMER ISOLATION ON PHASE DIFFERENTIAL RELAY TRIPS.
EVENT DATE: 080290 REPORT DATE: 083090 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219433) AT 0220 HOURS ON AUGUST 2, 1990, WITH UNIT 1 OPERATING AT 100 PERCENT POWER (MODE 1) AND UNIT 2 OPERATING AT APPROXIMATELY 75 PERCENT POWER (MODE 1), THE "B" RESERVE STATION SERVICE TRANSFORMER (RSST) ISOLATED BECAUSE OF PHASE DIFFERENTIAL RELAY TRIPS ON ALL THREE PHASES. THE LOSS OF NORMAL POWER FROM THE "B" RSST CAUSED THE UNIT 2 "H" EMERGENCY DIESEL GENERATOR (EDG) TO AUTOSTART AND RE-ENERGIZE THE UNIT 2 "H" EMERGENCY BUS. THE CAUSE OF THE "B" RSST ISOLATION WAS MOST LIKELY A FAULT IN THE WINDINGS OF THE TRANSFORMER. A FOUR-HOUR REPORT WAS MADE TO THE NRC OPERATIONS CENTER IN ACCORDANCE WITH 10CFR50.72(B)(2)(II) AT 0405 HOURS. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV) AS AN AUTOMATIC ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF). THE "B" RSST WAS REPLACED AND RESTORED AS THE POWER SUPPLY TO THE UNIT 2 "H" EMERGENCY BUS, THE 2H EDG WAS SHUTDOWN, AND THE ACTION STATEMENT OF TS 3.8.1.1.A WAS CLEARED AT 0213 HOURS, AUGUST 4, 1990. THE 2H EDG FUNCTIONED PROPERLY TO SUPPLY POWER TO THE UNIT 2 "H" EMERGENCY BUS. ALL OTHER AUTOMATIC ACTUATIONS ALSO FUNCTIONED PROPERLY. THE 2J EDG WAS VERIFIED OPERABLE IN ACCORDANCE WITH TECHNICAL SPECIFICATION (TS) 4.8.1.1.2.A.4. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED AT ANY TIME DURING THIS EVENT.

[125] NORTH ANNA 2 DOCKET 50-339 LER 90-003
AUXILIARY FEEDWATER PUMPS AUTO-STARTED WHEN MAIN FEEDWATER PUMP "C" BREAKERS OPENED ON HI-HI LEVEL IN "A" STEAM GENERATOR DUE TO PERSONNEL ERROR AND PROCEDURAL INADEQUACY.
EVENT DATE: 082190 REPORT DATE: 091990 NSSS: WE TYPE: PWR

(NSIC 219544) AT 2020 HOURS ON AUGUST 21, 1990, WITH UNIT 1 OPERATING AT 100 PERCENT POWER (MODE 1) AND UNIT 2 IN HOT SHUTDOWN (MODE 4) PREPARING FOR A REFUELING OUTAGE, THE UNIT 2 AUXILIARY FEEDWATER PUMPS (AFWP) AUTO-STARTED WHEN THE BREAKERS FOR THE "C" MAIN FEEDWATER PUMP (FWP) OPENED DUE TO A HI-HI LEVEL IN THE "A" STEAM GENERATOR (SG). THE "A" SG HI-HI LEVEL WAS CAUSED BY RE-OPENING THE "A" MAIN STEAM (MS) NON-RETURN VALVE (NRV) WITH A LARGE DIFFERENTIAL PRESSURE ACROSS IT WHICH INDUCED A SWELL IN SG INVENTORY. THE EVENT WAS CAUSED BY PERSONNEL ERROR AND PROCEDURAL INADEQUACY. A FOUR-HOUR REPORT WAS MADE IN ACCORDANCE WITH 10CFR50.72(B)(2)(II) AT 2219 HOURS. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV) AS AN AUTOMATIC ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF). THE AFWP'S FUNCTIONED PROPERLY TO SUPPLY COOLING WATER TO THE SG'S. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED AT ANY TIME DURING THIS EVENT.

[126] OCONEE 1 DOCKET 50-269 LER 90-012
POTENTIAL OVERLOAD CONDITIONS MAY RESULT IN INADEQUATE ON-SITE EMERGENCY POWER SOURCE DURING A LOCA/LOOP EVENT DUE TO DESIGN DEFICIENCY.
EVENT DATE: 073190 REPORT DATE: 082990 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 219409) WHILE DEVELOPING A DESIGN BASIS DOCUMENT FOR THE KEOWEE EMERGENCY POWER SYSTEM, DESIGN ENGINEERING DETERMINED THAT TWO POSTULATED ACCIDENT SCENARIOS EXISTED THAT COULD PREVENT KEOWEE HYDRO STATION FROM PROVIDING ADEQUATE EMERGENCY POWER TO OCONEE NUCLEAR STATION DURING A LOCA/LOOP OR LOOP EVENT. BOTH SCENARIOS WOULD REQUIRE ONE KEOWEE UNIT TO BE GENERATING POWER TO THE GRID, PRIOR TO A LOCA/LOOP OR LOOP EVENT. THE FIRST SCENARIO, IDENTIFIED ON JULY 31, 1990, WAS THAT A KEOWEE OVERLOAD CONDITION WAS POSSIBLE DUE TO THE RECLOSURE OF THE KEOWEE GENERATOR BREAKERS PRIOR TO THE TRIPPING OF THE REACTOR COOLANT PUMP BREAKERS. THE SECOND SCENARIO, IDENTIFIED ON AUGUST 2, 1990, WAS THAT A FAILURE OF A KEOWEE GENERATOR AIR CIRCUIT BREAKER COULD EXPOSE A KEOWEE UNIT TO A SIMILAR OVERLOAD CONDITION. BOTH CONDITIONS WERE DISCOVERED WITH ALL THREE OCONEE UNITS AT 100% FULL POWER. THE ROOT CAUSE OF THIS EVENT IS CLASSIFIED DESIGN DEFICIENCY, UNANTICIPATED INTERACTION OF SYSTEMS. IMMEDIATE CORRECTIVE ACTIONS WERE TO ADMINISTRATIVELY PROHIBIT THE USE OF KEOWEE FOR GENERATION TO THE GRID AND A SUBSEQUENT CORRECTIVE ACTION WAS TO IMPLEMENT A MODIFICATION THAT CORRECTED THE OVERLOAD PROBLEMS.

[127] OYSTER CREEK DOCKET 50-219 LER 90-010
ELECTROMATIC RELIEF VALVE HIGH PRESSURE RELIEF SETPOINTS EXCEEDED TECHNICAL SPECIFICATION LIMIT DUE TO DRIFT.
EVENT DATE: 071890 REPORT DATE: 082090 NSSS: GE TYPE: BWR

(NSIC 219338) ON JULY 18, 1990 WHILE PERFORMING AN ELECTROMATIC RELIEF VALVE (EMRV) PRESSURE SENSOR SURVEILLANCE, THE "AS FOUND" TRIP SETPOINT FOR THE HIGH PRESSURE RELIEF FUNCTION ON THREE OUT OF FIVE EMRVS WAS ABOVE THAT SPECIFIED IN THE TECHNICAL SPECIFICATIONS. ADDITIONALLY, A REVIEW OF RECORDS FOR THIS SURVEILLANCE REVEALED THAT ON APRIL 14, 1988 AND JUNE 13, 1990, ONE EMRV HAD A HIGH PRESSURE SETPOINT THAT WAS ABOVE THE LIMIT. THE CAUSE OF THESE OCCURRENCES IS ATTRIBUTED TO SETPOINT REPEATABILITY AND INSTRUMENT DRIFT. THE DESIGN SETPOINT REPEATABILITY CAN TOLERATE INSTRUMENT DRIFT WITHIN 2.5 PSIG OF THE TECHNICAL SPECIFICATION LIMIT. PREVIOUS SURVEILLANCE RECORDS INDICATE THAT THESE INSTRUMENTS FREQUENTLY UNDERGO ADDITIONAL DRIFT WITHIN TECHNICAL SPECIFICATION LIMITS DUE TO CHANGING PLANT AND ENVIRONMENTAL CONDITIONS. THIS OCCURRENCE IS CONSIDERED TO HAVE MINIMAL SAFETY SIGNIFICANCE AS THE AUTOMATIC DEPRESSURIZATION FUNCTION OF THE EMRVS IS NOT AFFECTED BY THESE PRESSURE SWITCHES, ALL FIVE EMRVS WOULD HAVE ACTUATED TO RELIEVE PRESSURE, AND THE ISOLATION CONDENSER SYSTEM AND TURBINE BYPASS VALVES WERE FULLY OPERABLE. THE PRESSURE SWITCHES WERE ADJUSTED TO ACTUATE WITHIN THE TECHNICAL SPECIFICATION LIMIT. A NEW PRESSURE SENSING SYSTEM IS INCLUDED IN THE OYSTER CREEK INTEGRATED SCHEDULE.

[128] OYSTER CREEK DOCKET 50-219 LER 90-011
UNQUALIFIED OPERATORS ON SHIFT DUE TO INADEQUACIES IN EXAM PROCESS RESULTS IN VIOLATION OF TECH SPEC SHIFT MANNING REQUIREMENTS.
EVENT DATE: 072390 REPORT DATE: 082190 NSSS: GE TYPE: BWR

(NSIC 219400) IN JUNE AND JULY OF 1990, AS A RESULT OF AN NRC AUDIT OF THE OPERATOR TRAINING PROGRAM, 1989 BIENNIAL REQUALIFICATION EXAMS WERE FOUND TO HAVE BEEN INCORRECTLY GRADED. A REGRADING EFFORT RESULTED IN THE FAILURE OF TWO LICENSED OPERATORS. THE OPERATORS WERE IMMEDIATELY REMOVED FROM LICENSED DUTIES AND ENTERED INTO AN ACCELERATED REQUALIFICATION PROGRAM. THESE OPERATORS HAD BEEN PERFORMING LICENSED DUTIES DURING THE PERIOD FROM THE REQUAL EXAM UP TO THE REGRADING OF THE EXAM. SINCE THESE TWO OPERATORS WERE RETROACTIVELY DISQUALIFIED, THERE WERE 89 SHIFTS DURING THIS PERIOD WITH LESS THAN TWO CONTROL ROOM OPERATORS AS REQUIRED BY TECHNICAL SPECIFICATIONS. THE CAUSE OF THIS OCCURRENCE IS ATTRIBUTED TO PERSONNEL ERROR AS A RESULT OF PROGRAMMATIC INADEQUACIES IN THE EXAM PROCESS. AN INVESTIGATION AND CRITIQUE OF THIS INCIDENT REVEALED THAT THESE INADEQUACIES LED TO ERRORS IN THE PREPARATION, ADMINISTRATION AND GRADING OF THE 1989 WRITTEN REQUALIFICATION EXAM. THESE INADEQUACIES CAUSED THE GRADING ANOMALIES IDENTIFIED. TO PREVENT RECURRENCE, AN EXAMINATION PROCEDURE WILL BE DEVELOPED TO PROVIDE GUIDANCE FOR THE PREPARATION, ADMINISTRATION AND GRADING OF EXAMS.

[129] OYSTER CREEK DOCKET 50-219 LER 90-012
AN ERROR IN A FEEDWATER FLOW CALCULATION EQUATION MAY HAVE RESULTED IN OPERATION OF THE REACTOR IN EXCESS OF THE LICENSE LIMIT.
EVENT DATE: 080190 REPORT DATE: 083190 NSSS: GE TYPE: BWR

(NSIC 219481) A DECREASE IN PLANT PERFORMANCE HAD BEEN NOTED FOR THE CURRENT OPERATING CYCLE. A LEAK IN THE HIGH PRESSURE FEEDWATER REHEATERS WAS POSTULATED AND INVESTIGATED. VISUAL INSPECTIONS REVEALED NO LEAKS. A DETAILED REVIEW OF PLANT DATA WAS INITIATED ON JULY 11, 1990 TO LOCATE THE SOURCE OF THE PERFORMANCE DECREASE. ON AUGUST 1, 1990, IT WAS NOTED THAT A PROCEDURE REVISION (APPROVED ON FEBRUARY 9, 1987) TO THE FEEDWATER FLOW CALIBRATION CALCULATION HAD RESULTED IN APPROXIMATELY A 2% CORRECTION IN INDICATED FEEDWATER FLOW. THIS CHANGE RESULTED IN A DECREASE IN ALLOWED PLANT POWER. THEREFORE, ALTHOUGH INDICATED REACTOR POWER HAS BEEN CORRECT SINCE FEBRUARY 9, 1987, PRIOR TO THAT DATE THE REACTOR MAY HAVE BEEN OPERATED SLIGHTLY IN EXCESS OF ITS LICENSE LIMIT OF 1930 MEGAWATTS THERMAL. THE MAGNITUDE OF THE POWER ANOMALY AND ANY RESULTANT EFFECTS ON SAFETY SIGNIFICANCE ARE PRESENTLY BEING REVIEWED. A SUPPLEMENTAL LICENSEE EVENT REPORT WILL BE SUBMITTED WHEN THE ONGOING REVIEW IS COMPLETED.

[130] PALISADES DOCKET 50-255 LER 90-013
OVERTIME LIMIT EXCEEDED DUE TO SCHEDULING OVERSIGHT.
EVENT DATE: 081890 REPORT DATE: 091790 NSSS: CE TYPE: PWR

(NSIC 219554) ON 8/18/90 AT 2005 HOURS IT WAS IDENTIFIED A SHIFT ENGINEER (SE) HAD WORKED 28 HOURS IN A 48 HOUR PERIOD. THE PLANT WAS OPERATING AT 80% POWER AT THIS TIME. TECH SPEC 6.2.2.F.B STATES THAT WORKERS WHO PERFORM SAFETY-RELATED FUNCTIONS SHALL NOT BE PERMITTED TO WORK MORE THAN 24 HOURS IN ANY 48 HOUR PERIOD. THE SE, WHO WAS INVOLVED, IDENTIFIED THIS MATTER AND PROMPTLY REPORTED IT TO THE SHIFT SUPERVISOR (SS). ARRANGEMENTS WERE THEN MADE FOR RELIEVING THE SE FROM DUTY AND THE SE WAS SUBSEQUENTLY RELIEVED. ADDITIONALLY, LESSONS LEARNED FROM THIS OCCURRENCE WILL BE INCORPORATED INTO OPERATIONS TRAINING PROGRAMS. THE VIOLATION OF THE OVERTIME RESTRICTIONS HAS BEEN ATTRIBUTED TO PERSONNEL ERROR ON THE PART OF THE OPERATIONS SCHEDULER WHO OFFERED THE OVERTIME AND THE SE WHO HAS THE ULTIMATE RESPONSIBILITY FOR ASSURING REQUIREMENTS ARE MET.

[131] PALO VERDE 1 DOCKET 50-528 LER 90-006
REACTOR TRIP FOLLOWING MANUAL TURBINE TRIP.
EVENT DATE: 081490 REPORT DATE: 091390 NSSS: CE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC COMPANY (ELEV. DIV)

(NSIC 219528) ON 8/14/90 PALO VERDE UNIT 1 WAS OPERATING IN MODE 1 AT 100% POWER. AT APPROXIMATELY 2159 MST IT WAS DISCOVERED THAT THE 'B' PHASE OF THE MAIN TRANSFORMER HAD LOST FORCED COOLING. A RAPID POWER REDUCTION WAS INITIATED IN ORDER TO UNLOAD THE TRANSFORMER WITHIN 30 MINUTES AS REQUIRED BY THE ALARM RESPONSE PROCEDURE FOR A LOSS OF TRANSFORMER COOLING AND TO MINIMIZE THE TRANSIENT ON THE PLANT. THE MAIN TURBINE WAS MANUALLY TRIPPED AT APPROXIMATELY 2223 WITH THE REACTOR AT APPROXIMATELY 65% POWER. APPROXIMATELY 30 SECONDS AFTER THE MAIN TURBINE TRIP, THE REACTOR TRIPPED ON HIGH PRESSURIZER PRESSURE. ALL SYSTEMS FUNCTIONED AS DESIGNED AND THE PLANT WAS STABILIZED IN MODE 3 (HOT STANDBY). THE LOSS OF COOLING TO THE MAIN TRANSFORMER WAS CAUSED BY THE FAILURE OF A CONTROL POWER TRANSFORMER. THE REACTOR TRIP HAS BEEN DETERMINED TO BE THE EXPECTED RESULT OF A LOAD REJECT WITH THE REACTOR AT 65% POWER WITH STEAM BYPASS CONTROL CONFIGURED FOR NORMAL (100% POWER) OPERATION. THE CONTROL POWER TRANSFORMER WAS REPLACED. THE ALARM RESPONSE PROCEDURE FOR THE MAIN TRANSFORMER WILL BE REVISED TO ENHANCE DIRECTIONS FOR THE "NO VOLTAGE ALARM". ENHANCEMENTS TO THE STEAM BYPASS CONTROL SYSTEM ARE CURRENTLY UNDER EVALUATION.

[132] PALO VERDE 2 DOCKET 50-529 LER 90-004 REV 01
UPDATE ON PRESSURIZER SAFETY RELIEF VALVE SETPOINTS OUT OF TOLERANCE.
EVENT DATE: 050490 REPORT DATE: 091190 NSSS: CE TYPE: PWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV

(NSIC 219550) AT APPROXIMATELY 0700 MST ON MAY 4, 1990, PALO VERDE UNIT 2 WAS IN A REFUELING OUTAGE WITH THE CORE OFFLOADED TO THE SPENT FUEL POOL WHEN APS WAS INFORMED BY AN OFFSITE TESTING LAB THAT ALL FOUR PRESSURIZER CODE SAFETY VALVES' AS-FOUND SETPOINTS WERE OUT OF THE TECHNICAL SPECIFICATION (TS) TOLERANCE OF 2500 POUNDS PER SQUARE INCH ABSOLUTE (PSIA) PLUS OR MINUS ONE (1) PERCENT (25 PSI). THE CAUSE OF THE EVENT FOR THREE OF THE VALVES IS A PERFORMANCE LIMITATION OF THE PRESSURIZER SAFETY VALVES. INDUSTRY TESTING HAS SHOWN THAT RELIEF AND SAFETY VALVES, OF THE SIZE AND APPLICATION OF THE PALO VERDE PRESSURIZER SAFETY VALVES, HAVE A LIFT SETTING REPEATABILITY OF PLUS OR MINUS THREE (3) PERCENT. THREE OF THE PRESSURIZER SAFETY VALVES LIFTED WITHIN THREE (3) PERCENT OF THE REQUIRED LIFT SETTING. ONE OF THE PRESSURIZER SAFETY VALVES LIFTED AT APPROXIMATELY 3.6 PERCENT ABOVE THE REQUIRED LIFT SETTING. THE CAUSE OF THE VALVE EXCEEDING THE THREE (3) PERCENT PERFORMANCE LIMITATION IS SETPOINT DRIFT. AS IMMEDIATE CORRECTIVE ACTION, THE VALVES WERE ADJUSTED AND RETESTED SATISFACTORILY. A PREVIOUS SIMILAR LER (528/89-007-01) DESCRIBED AN EVENT WHEREIN TWO OF FOUR PRESSURIZER SAFETY VALVES IN UNIT 1 WERE OUT OF THE TS TOLERANCE LIMITS.

[133] PALO VERDE 3 DOCKET 50-530 LER 90-005
INCORRECT BACKUP RING MATERIAL IN FEEDWATER ISOLATION VALVE 4-WAY VALVES.
EVENT DATE: 080390 REPORT DATE: 090490 NSSS: CE TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 219521) ON AUGUST 3, 1990, AT APPROXIMATELY 1400 MST, PALO VERDE UNIT 3 WAS IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER WHEN A SYSTEM ENGINEER (SE), WHILE PERFORMING A ROOT CAUSE OF FAILURE EVALUATION OF AN ECONOMIZER FEEDWATER ISOLATION VALVE (FWIV), CONCLUDED THAT FWIV 4-WAY VALVE REBUILD KITS CONTAINED NON-QUALIFIED BACKUP RING MATERIAL. THIS RESULTED IN A CONDITION THAT COULD HAVE PREVENTED THE FULFILLMENT OF A SAFETY FUNCTION. ON JULY 5, 1990 AN ECONOMIZER FWIV WOULD NOT EXERCISE PROPERLY DURING ROUTINE SURVEILLANCE TESTING. FOLLOWING TROUBLESHOOTING, A 4-WAY VALVE WAS REPLACED AND A ROOT CAUSE OF FAILURE EVALUATION WAS INITIATED. DURING DISASSEMBLY OF THE 4-WAY VALVE, THE SE DISCOVERED THAT THE BACKUP RINGS WERE SWOLLEN AND DID NOT APPEAR TO BE MADE OF THE REQUIRED MATERIAL (VITON). AN INDEPENDENT LABORATORY ANALYSIS COMPLETED ON JULY 23, 1990 CONFIRMED THIS DISCOVERY. ON JULY 27, 1990 AT APPROXIMATELY 1700 MST, APS INITIATED THE REPLACEMENT AND INSPECTION OF SUSPECT BACKUP RINGS INSTALLED IN THE UNIT 1 AND 3 FWIV AND MAIN STEAM ISOLATION VALVE 4-WAY VALVES. SUBSEQUENT TESTING REVEALED THAT A TOTAL OF FOUR (4) FWIV 4-WAY REBUILD KITS HAD NON-VITON BACKUP RINGS. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED PURSUANT TO 10CFR50.73 BY APS. HOWEVER, ANCHOR/DARLING ISSUED A 10CFR21 REPORT ON JANUARY 26, 1990, NOTIFYING THE NRC THAT SOME BACKUP RINGS WERE OF POTENTIALLY INCORRECT MATERIAL.

[134] PALO VERDE 3 DOCKET 50-530 LER 90-006
REACTOR SHUTDOWN REQUIRED BY TECHNICAL SPECIFICATION FOR MISALIGNED CONTROL ROD.
EVENT DATE: 080590 REPORT DATE: 082990 NSSS: CE TYPE: PWR
VENDOR: RCA SOLID STATE DIVISION

(NSIC 219445) ON AUGUST 5, 1990, AT APPROXIMATELY 1450 MST, PALO VERDE UNIT 3 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 100 PERCENT POWER WHEN A REGULATING GROUP CONTROL ELEMENT ASSEMBLY (CEA) SLIPPED INTO THE CORE DURING ROUTINE CONTROL ROD TESTING. POWER REDUCTION WAS BEGUN IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS (TS) LIMITING CONDITIONS FOR OPERATION (LCO). THE MISALIGNED CEA COULD NOT BE RESTORED TO OPERABLE STATUS (WITHDRAWN) WITHIN ONE HOUR AND AN ORDERLY PLANT SHUTDOWN WAS INITIATED IN ACCORDANCE WITH THE TS LCO. NO SAFETY SYSTEM RESPONSES OCCURRED AND NONE WERE REQUIRED. AT APPROXIMATELY 2009 MST ON AUGUST 5, 1990, THE PLANT WAS STABILIZED IN MODE 3 (HOT STANDBY) AT NORMAL TEMPERATURE AND PRESSURE. THE EVENT WAS DETERMINED TO BE AN UNCOMPLICATED NORMAL PLANT SHUTDOWN. THE CAUSE OF THE CEA DROPPING INTO THE CORE WAS A COIL DRIVER ACTUATING LOGIC CARD MALFUNCTION. THE MALFUNCTION WAS DETERMINED TO BE A RANDOM COMPONENT FAILURE. AS CORRECTIVE ACTION, THE COIL DRIVER ACTUATING LOGIC CARD WAS REPLACED AND SUCCESSFULLY RETESTED. SIMILAR EVENTS WERE REPORTED IN UNIT 1 LERS 85-088, 88-020 AND 88-026 AND UNIT 3 LER 90-004.

[135] PEACH BOTTOM 2 DOCKET 50-277 LER 90-016
TWO CONTROL ROOM EMERGENCY VENTILATION SYSTEM ACTUATIONS RESULTING FROM SPURIOUS HIGH RADIATION SIGNALS FROM THE "B" CONTROL ROOM VENTILATION RADIATION MONITOR.
EVENT DATE: 071890 REPORT DATE: 081790 NSSS: GE TYPE: BWR

(NSIC 219346) ON 7/18/90 A CONTROL ROOM EMERGENCY VENTILATION (CREV) ACTUATION OCCURRED AT 1020 AND 1405 HOURS. INITIATION OF CREV IS AN ENGINEERED SAFETY FEATURE ACTUATION. THE ACTUATION RESULTED FROM A HIGH RADIATION SIGNAL FROM THE "B" CONTROL ROOM VENTILATION SYSTEM (CRVS) RADIATION MONITOR. THE CRVS SERVES THE PBAPS CONTROL ROOM WHICH IS COMMON TO BOTH UNITS. AFTER VERIFYING THAT THE SIGNAL WAS NOT VALID, THE CRVS WAS RESTORED TO THE NORMAL STANDBY ALIGNMENT. THE CAUSE OF THE SPURIOUS HIGH RADIATION SIGNAL IS CURRENTLY UNDER INVESTIGATION AND IS BELIEVED TO BE RELATED TO A NOISE IN THE SIGNAL LINE OR A PROBLEM IN THE CIRCUITRY OF THE RADIATION MONITOR. THE POWER SUPPLY LINES WERE CHECKED FOR ELECTRICAL NOISE AND NO SIGNIFICANT LEVELS WERE FOUND. DIAGNOSTIC EQUIPMENT WAS INSTALLED TO AID IN DETERMINING THE CAUSE OF THE SPURIOUS SIGNALS. THERE WERE FOUR PREVIOUS SIMILAR EVENTS IDENTIFIED.

[136] PEACH BOTTOM 2 DOCKET 50-277 LER 90-017
HIGH PRESSURE COOLANT INJECTION SYSTEM DECLARED INOPERABLE DUE TO 2B BATTERY CHARGER TRANSIENT.
EVENT DATE: 072490 REPORT DATE: 082290 NSSS: GE TYPE: BWR
VENDOR: EXIDE INDUSTRIAL DIV

(NSIC 219347) ON 7/24/90 AT 0322 HOURS THE "2DB HPCI 250 VDC BUS LOW VOLTAGE" ALARM WAS RECEIVED IN THE CONTROL ROOM. AT THIS TIME THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM WAS DECLARED INOPERABLE. THE 2DB003 125 VDC BATTERY CHARGER EXPERIENCED A TRANSIENT WHICH CAUSED AN UNDERVOLTAGE CONDITION ON THE 2DB HPCI 250 VDC BUS. THE HPCI SYSTEM WAS IN AN INOPERABLE STATE FOR LESS THAN TEN MINUTES. THREE ADDITIONAL BATTERY CHARGER TRANSIENTS OCCURRED WHICH RESULTED IN DECLARING HPCI INOPERABLE. THESE OCCURRED ON 8/3/90 AT 1145 HOURS AND ON 8/4/90 AT 1033 HOURS AND 2122 HOURS. HPCI WAS INOPERABLE LESS THAN TWO MINUTES IN THESE THREE CASES. THE CAUSE OF THESE EVENTS WAS DUE TO THE DEGRADATION OF A FOAM RUBBER ELECTRICAL MODULE SUPPORT PIECE, WHICH LED TO POOR ELECTRICAL CONNECTIONS BETWEEN THE MODULES AND THE SOCKETS. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. THE FOAM WHICH SUPPORTED THE CIRCUIT CARD WAS REPLACED IN THE 2BD003 BATTERY CHARGER. PROCEDURES WILL BE UPGRADED TO INCLUDE PERIODIC REPLACEMENT OF THE FOAM RUBBER SUPPORT PIECE. THERE WERE NO PREVIOUS SIMILAR EVENTS IDENTIFIED.

[137] PEACH BOTTOM 2 DOCKET 50-277 LER 90-018
CONTROL ROOM EMERGENCY VENTILATION SYSTEM ACTUATION RESULTING FROM SPURIOUS HIGH RADIATION SIGNALS FROM THE "B" CONTROL ROOM VENTILATION RADIATION MONITOR.
EVENT DATE: 073090 REPORT DATE: 082990 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 219412) ON 7/30/90 A CONTROL ROOM EMERGENCY VENTILATION (CREV) ACTUATION OCCURRED AT 0925 HOURS. INITIATION OF CREV IS AN ENGINEERED SAFETY FEATURE ACTUATION. THE ACTUATION RESULTED FROM A HIGH RADIATION SIGNAL FROM THE "B" CONTROL ROOM VENTILATION SYSTEM (CRVS) RADIATION MONITOR. THE CRVS SERVES THE PBAPS CONTROL ROOM WHICH IS COMMON TO BOTH UNITS. AFTER VERIFYING THAT THE SIGNAL WAS NOT VALID, THE CRVS WAS RESTORED TO THE NORMAL STANDBY ALIGNMENT. THE CAUSE OF THE SPURIOUS HIGH RADIATION SIGNAL IS CURRENTLY UNDER INVESTIGATION AND IS BELIEVED TO BE RELATED TO A NOISE IN THE SIGNAL LINE OR A PROBLEM IN THE CIRCUITRY OF THE RADIATION MONITOR. THE POWER SUPPLY LINES WERE CHECKED FOR ELECTRICAL NOISE AND NO SIGNIFICANT LEVELS WERE FOUND. DIAGNOSTIC EQUIPMENT WAS INSTALLED TO AID IN DETERMINING THE CAUSE OF THE SPURIOUS SIGNALS. THERE WERE FIVE PREVIOUS SIMILAR EVENTS IDENTIFIED.

[138] PEACH BOTTOM 2 DOCKET 50-277 LER 90-019
CONTROL ROOM EMERGENCY VENTILATION SYSTEM ACTUATIONS DUE TO A DIRTY CONNECTOR AND AGE OF THE ELECTRONIC COMPONENTS IN THE RADIATION MONITOR.
EVENT DATE: 081790 REPORT DATE: 091490 NSSS: GE TYPE: BWR
VENDOR: NUCLEAR RESEARCH CORP.

(NSIC 219501) ON 8/17/90 A CONTROL ROOM EMERGENCY VENTILATION (CREV) ACTUATION OCCURRED AT 2130. ANOTHER CREV ACTUATION OCCURRED ON 8/20/90 AT 1135 HOURS. INITIATION OF CREV IS AN ENGINEERED SAFETY FEATURE ACTUATION. THE ACTUATIONS RESULTED FROM A HIGH RADIATION SIGNAL FROM THE "B" CONTROL ROOM VENTILATION SYSTEM (CRVS) RADIATION MONITOR. THE CRVS SERVES THE PBAPS CONTROL ROOM WHICH IS COMMON TO BOTH UNITS. AFTER VERIFYING THAT THE SIGNALS WERE NOT VALID, THE CRVS WAS RESTORED TO THE NORMAL STANDBY ALIGNMENT. THE CAUSE OF THE SPURIOUS HIGH RADIATION SIGNALS IS BELIEVED TO BE DIRT AND DUST WHICH ACCUMULATED ON AN INTERNAL BULKHEAD CONNECTOR ALONG WITH THE AGE OF THE RADIATION MONITOR. THE CONNECTIONS WERE CLEANED AND A PREVENTIVE MAINTENANCE TASK WILL BE ESTABLISHED FOR FUTURE INSPECTIONS OF RADIATION MONITOR INTERNALS. THERE WERE SIX PREVIOUS SIMILAR EVENTS IDENTIFIED.

[139] PEACH BOTTOM 2 DOCKET 50-277 LER 90-020
HPCI INOPERABLE DUE TO LOW 2B BATTERY VOLTAGE DURING MAINTENANCE CAUSED BY PERSONNEL ERROR.
EVENT DATE: 082190 REPORT DATE: 091990 NSSS: GE TYPE: BWR

(NSIC 219537) ON 8/21/90 AT 0920 HOURS THE 2D BATTERY CHARGER WAS MADE INOPERABLE FOR UNDERVOLTAGE ALARM RELAY CALIBRATION. ALTHOUGH THE BATTERY CHARGER WAS CORRECTLY DECLARED INOPERABLE AT THIS TIME, THE BATTERY AND THE ASSOCIATED LOADS WERE NOT. THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM SHOULD ALSO HAVE BEEN DECLARED INOPERABLE. THE 2D BATTERY CHARGER WAS DECLARED OPERABLE AT 1055 HOURS. AT 1100 HOURS THE 2B BATTERY CHARGER WAS DECLARED INOPERABLE IN ORDER TO PERFORM THE SAME CALIBRATION. THE 2B BATTERY CHARGER WAS DECLARED OPERABLE AT 1420 HOURS. AT 1445 HOURS IT WAS REALIZED THAT HPCI SHOULD HAVE BEEN DECLARED INOPERABLE PER A TECHNICAL SPECIFICATION PLANT OPERATIONS REVIEW COMMITTEE (PORC) POSITION ON BATTERY OPERABILITY. ALTHOUGH THE WORK ON THE BATTERIES WAS PREPLANNED, THE FACT THAT HPCI WOULD ALSO BE INOPERABLE WAS NOT RECOGNIZED, THEREFORE IT WAS REPORTED. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR BY SHIFT SUPERVISION. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. SHIFT SUPERVISION WAS COUNSELED ON THE EVENT. NO PREVIOUS SIMILAR LERS HAVE BEEN IDENTIFIED.

[140] PEACH BOTTOM 3 DOCKET 50-278 LER 90-008
MANUAL SCRAM DUE TO LOSS OF CONDENSER VACUUM FOLLOWING ISOLATION OF OFFGAS RECOMBINER CAUSED BY A COMPONENT FAILURE.
EVENT DATE: 072790 REPORT DATE: 082790 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 2 (BWR)
VENDOR: FISHER CONTROLS CO.

(NSIC 219413) ON 7/27/90 AT 0350 HOURS, WITH UNIT 3 OPERATING AT 100% POWER, AN OFFGAS RECOMBINER ISOLATION OCCURRED CAUSING THE MAIN CONDENSER VACUUM TO BEGIN DECREASING. A FAST REACTOR POWER REDUCTION WAS INITIATED IN ACCORDANCE WITH THE PROCEDURE FOR A LOSS OF MAIN CONDENSER VACUUM. AT 0403 HOURS, WITH UNIT 3 AT APPROXIMATELY 80% POWER, A MANUAL SCRAM WAS INITIATED BY PLACING THE MODE SWITCH IN SHUTDOWN FOLLOWING THE RECEIPT OF AN 'A' CHANNEL REACTOR AUTO HALF SCRAM SIGNAL. A GROUP II AND III ISOLATION OCCURRED AS A RESULT OF THE MANUAL SCRAM. OTHER SAFETY SYSTEMS PERFORMED AS DESIGNED. AN OFFSITE RADIOACTIVE GASEOUS RELEASE AMOUNTING TO 23.5% OF TECHNICAL SPECIFICATION LIMITS OCCURRED DURING POST SCRAM RECOVERY. THERE WERE NO ADVERSE HEALTH THREATS TO ONSITE PERSONNEL OR THE GENERAL PUBLIC. THE CAUSE OF THE FAILURE APPEARS TO BE A COMPONENT/SYSTEM FAILURE IN THE OFFGAS RECOMBINER CONDENSATE COOLING WATER PRESSURE CONTROL SYSTEM. THE DESIGN OF THE SYSTEM WILL BE EVALUATED TO DETERMINE THE EXACT CAUSE OF THE FAILURE AND THE APPROPRIATE MEASURES REQUIRED TO CORRECT THE PROBLEM. THERE WAS ONE PREVIOUS SIMILAR EVENT.

[141] PEACH BOTTOM 3 DOCKET 50-278 LER 90-009
TECHNICAL SPECIFICATION VIOLATIONS RESULTING FROM A FAILURE TO LOG REQUIRED TEMPERATURES AND EXCEEDING A HEATUP RATE DUE TO PROCEDURAL WEAKNESS AND PERSONNEL ERROR.
EVENT DATE: 072790 REPORT DATE: 082790 NSSS: GE TYPE: BWR

(NSIC 219414) ON 7/27/90 AT 1100 HOURS, WITH UNIT 3 AT 0% POWER IT WAS DISCOVERED THAT THE 'B' RECIRCULATION (RECIRC) LOOP TEMPERATURE HAD INCREASED FROM 105 DEGREES F TO 266 DEGREES F OVER AN 18 MINUTE PERIOD. THIS INCREASE EXCEEDED THE HEATUP RATE ALLOWED BY TECH SPEC 3.6.A.1. THIS OCCURRED AS A RESULT OF A SUDDEN REVERSE FLOW INTO THE 'B' RECIRC LOOP DURING PERFORMANCE OF THE PROCEDURE TO START THE IDLE 'B' RECIRC PUMP. THE CAUSE OF THE EVENT WAS A LACK OF PROCEDURAL GUIDANCE TO CONTROL THE HEATUP DURING REACTOR VESSEL DEPRESSURIZATION. APPROPRIATE PROCEDURAL CONTROLS WILL BE REVISED TO ADD GUIDANCE CONCERNING THE HEATUP OF THE IDLE RECIRC LOOP DURING DEPRESSURIZATION AND A CAUTION WILL BE ADDED INFORMING THE REACTOR OPERATOR OF THE POTENTIAL FOR A SUDDEN REVERSE FLOW IN THE LOOP DURING REACTOR VESSEL DEPRESSURIZATION. DURING THE INVESTIGATION OF THIS EVENT, IT WAS DISCOVERED THAT THE TEMPERATURE READING FOR THE 'B' RECIRC LOOP HAD NOT BEEN RECORDED EVERY 15 MINUTES FOLLOWING THE SHUTDOWN OF UNIT 3 ON 7/27/90 AS REQUIRED BY TECHNICAL SPECIFICATION 4.6.A.1. THE CAUSE WAS A

COMBINATION OF PERSONNEL ERROR AND PROCEDURAL DEFICIENCY. ST 9.12 WILL BE REVISED TO ADD AN ADDITIONAL NOTE TO THE TOP OF THE DATA SHEET TO FURTHER EMPHASIZE THE REQUIREMENT TO RECORD THESE READINGS. THE INDIVIDUAL INVOLVED WAS COUNSELLED.

[142] PEACH BOTTOM 3 DOCKET 50-278 LER 90-010
HIGH PRESSURE COOLANT INJECTION MADE INOPERABLE DUE TO FAILURE OF THE MANUAL AND OVERSPEED TRIP TAPPET ASSEMBLY.
EVENT DATE: 080490 REPORT DATE: 090690 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219502) ON 8/4/90 AT 1600 HOURS DURING THE PERFORMANCE OF A SURVEILLANCE TEST, THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM WAS DECLARED INOPERABLE SINCE THE TURBINE STOP VALVE WOULD NOT STAY IN THE OPENED POSITION DURING SYSTEM OPERATION. FURTHER INVESTIGATION REVEALED THAT THE MANUAL AND OVERSPEED TRIP TAPPET ASSEMBLY SPRING DID NOT HAVE ADEQUATE SPRING TENSION TO MAINTAIN THE TAPPET IN THE RESET POSITION. THE SPRING TENSION WAS ADJUSTED AND HPCI WAS RETURNED TO AN OPERABLE STATUS ON 8/5/90 AT 1400 HOURS. THE CAUSE OF THE SPRING TENSION FORCE NOT BEING ENOUGH TO RESET THE TAPPET IS BELIEVED TO BE A DESIGN PROBLEM INVOLVING TAPPET SWELLING, WHICH CHANGES ITS CLEARANCES AS DESCRIBED IN GENERAL ELECTRIC (GE) COMPANY RAPID INFORMATION COMMUNICATION SERVICE INFORMATION LETTER (RICSIL) 04, RICSIL 37, SIL 392, AND NRC INFORMATION NOTICE 88-067. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. A NEW DESIGN, WHICH IS BEING PROVIDED BY GE, WILL REPLACE THE EXISTING TAPPET ASSEMBLY. NO PREVIOUS SIMILAR LERS WERE IDENTIFIED.

[143] PERRY 1 DOCKET 50-440 LER 90-004 REV 01
UPDATE ON FAILURE TO PERFORM REQUIRED SNUBBER EXAMINATIONS DUE TO PROGRAM INADEQUACY CAUSES TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 030290 REPORT DATE: 090790 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PERRY 2 (BWR)

(NSIC 219439) ON MARCH 2, 1990, AT 1815, IT WAS DETERMINED THAT TWELVE UNIT 2 SNUBBERS WERE NOT INSPECTED AS REQUIRED BY TECHNICAL SPECIFICATION 4.7.4. THE SNUBBERS INVOLVED WERE FOR PIPING IN UNIT 2 FUEL TRANSFER, FUEL POOL COOLING AND CLEANUP, AND EMERGENCY CLOSED COOLING SYSTEMS. ON MARCH 27, 1990 AT 1307, AS PART OF THE CONTINUING INVESTIGATION, TWO UNIT 1 FUEL TRANSFER SYSTEM SNUBBERS WERE IDENTIFIED AS NOT BEING INSPECTED AS REQUIRED BY TECHNICAL SPECIFICATION 4.7.4. FOR BOTH EVENTS, OPERATORS TOOK THE ACTIONS AS REQUIRED BY TECHNICAL SPECIFICATIONS AND NONE OF THE EVENTS AFFECTED UNIT 1 POWER OPERATION. THE CAUSE OF THESE EVENTS WAS PROGRAM INADEQUACY. THE MARCH 2 EVENT IS RELATED TO THE INITIAL DEVELOPMENT OF THE PROGRAM. UNIT 2 DESIGNATED EQUIPMENT THAT IS REQUIRED FOR UNIT 1 OPERATION WAS NOT CONSIDERED DURING THE INITIAL REVIEW FOR INCLUSION IN THE TECHNICAL SPECIFICATION SNUBBER PROGRAM. FOR THE MARCH 27 EVENT, THE UNIT 1 SNUBBERS WERE IMPROPERLY REMOVED FROM THE SNUBBER PROGRAM DUE TO INADEQUATE STANDARDS FOR SNUBBER CLASSIFICATION AND SELECTION. TO PREVENT RECURRENCE OF THESE EVENTS, APPROPRIATE PROCEDURAL CHANGES WERE IMPLEMENTED TO ENSURE THE INCLUSION OF THE SNUBBERS IDENTIFIED BY THESE EVENTS INTO PERRY'S SNUBBER PROGRAM. A COMPREHENSIVE EVALUATION OF PERRY'S SNUBBER PROGRAM WAS PERFORMED TO DETERMINE POTENTIAL IMPROVEMENTS.

[144] PERRY 1 DOCKET 50-440 LER 90-017
THERMAL POWER EXCEEDS OPERATING LICENSE LIMITS DUE TO FEEDWATER HEATER LEVEL CONTROL FAILURE.
EVENT DATE: 081490 REPORT DATE: 091390 NSSS: GE TYPE: BWR
VENDOR: COPES-VULCAN, INC.

(NSIC 219549) ON AUGUST 14, 1990 AT 0635 HOURS, REACTOR THERMAL POWER LEVEL EXCEEDED 102 PERCENT OF THE MAXIMUM POWER LEVEL AUTHORIZED IN THE FACILITY OPERATING LICENSE. THE TRANSIENT EVENT OCCURRED WHEN THE 5A AND 6A FEEDWATER HEATERS UNEXPECTEDLY ISOLATED DUE TO A FAILURE OF THE 5A FEEDWATER HEATER LEVEL CONTROL VALVE, RESULTING IN A DECREASE IN REACTOR FEEDWATER TEMPERATURE. PRIOR TO THE TRANSIENT, THE PLANT WAS IN OPERATIONAL CONDITION 1 (POWER OPERATION) AT

100 PERCENT OF RATED THERMAL POWER. THE REACTOR PRESSURE VESSEL [RPV] WAS AT SATURATED CONDITIONS AT APPROXIMATELY 1015 PSIG. OPERATORS RECOVERED FROM THE TRANSIENT BY MANUALLY ADJUSTING REACTOR RECIRCULATION FLOW IN ACCORDANCE WITH APPROVED PLANT INSTRUCTIONS. THERMAL POWER EXCEEDED 102 PERCENT FOR 117 SECONDS, WITH A PEAK THERMAL POWER OF 104.24 PERCENT. THE ROOT CAUSE OF THIS EVENT WAS COMPONENT FAILURE WITH A CONTRIBUTING PROCEDURAL DEFICIENCY. FAILURE OF THE 5A FEEDWATER HEATER LEVEL CONTROL VALVE CAUSED THE 5A FEEDWATER HEATER TO ISOLATE ON HIGH WATER LEVEL. THIS ISOLATION IN TURN CAUSED THE 6A FEEDWATER HEATER TO ISOLATE ON HIGH WATER LEVEL. CORRECTIVE ACTIONS TAKEN INCLUDE MODIFICATIONS TO MANUALLY THROTTLE THE NORMAL DRAIN VALVE UNTIL REPAIRS CAN BE MADE DURING THE REFUELING OUTAGE CURRENTLY IN PROGRESS. ADDITIONALLY, ENHANCEMENTS WERE MADE TO THE APPLICABLE OFF NORMAL INSTRUCTION.

[145] POINT BEACH 1 DOCKET 50-266 LER 90-008
REACTOR COOLANT SYSTEM LEAKAGE.
EVENT DATE: 072090 REPORT DATE: 082090 NSSS: WE TYPE: PWR
VENDOR: KEROTEST MANUFACTURING CORP.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 219343) POINT BEACH NUCLEAR PLANT UNIT 1 WAS SHUT DOWN FROM 100 PERCENT POWER ON JULY 20, 1990, AT 0839 CDT TO REPAIR LEAKS IN THE REACTOR COOLANT SYSTEM WITH AN AVERAGE TOTAL LEAKAGE OF APPROXIMATELY 0.27 GALLONS PER MINUTE. REACTOR COOLANT WAS LEAKING THROUGH A CANOPY SEAL WELD ON CONTROL ROD DRIVE MECHANISM I-3 AND THE UPSTREAM WELD ON B STEAM GENERATOR CHANNEL HEAD DRAIN LINE ISOLATION VALVE 1RC-526B. BOTH LEAKS WERE REPAIRED BY WELDING AND THE PLANT WAS BROUGHT ON LINE JULY 29, 1990, AT 0536 CDT.

[146] POINT BEACH 1 DOCKET 50-266 LER 90-009
REACTOR PROTECTION SYSTEM ACTUATION.
EVENT DATE: 072390 REPORT DATE: 082090 NSSS: WE TYPE: PWR

(NSIC 219344) ON JULY 23, 1990, AT 1:30 P.M., WHILE IN COLD SHUTDOWN WITH THE REACTOR TRIP BREAKERS OPEN AND RACKED OUT, AN AUTOMATIC REACTOR TRIP SIGNAL WAS GENERATED DURING ROUTINE INTERMEDIATE RANGE INSTRUMENTATION TESTING. DURING THE TESTING, A CONTROL POWER FUSE FAILED RESULTING IN A LOSS OF POWER TO THE CIRCUITS WHICH BYPASS THE REACTOR TRIPS AND IN TURN, RESULTED IN THE TRIP SIGNAL.

[147] POINT BEACH 1 DOCKET 50-266 LER 90-010
AXIAL FLUX DIFFERENTIAL OUTSIDE THE LIMITS ALLOWED BY TECH SPEC.
EVENT DATE: 081690 REPORT DATE: 091390 NSSS: WE TYPE: PWR

(NSIC 219494) AT 7:07 P.M. ON AUGUST 16, 1990, UNIT 1 EXPERIENCED A LOAD REJECTION OF APPROXIMATELY 100 MWE DUE TO A MALFUNCTION OF THE TURBINE ELECTRO-HYDRAULIC (EH) GOVERNOR CONTROL. AS A RESULT OF THE LOAD REJECTION THE PLANT PROCESS COMPUTING SYSTEM (PPCS) INDICATED BY ALARM THAT THE AXIAL FLUX DIFFERENTIAL WAS OUTSIDE PRESCRIBED LIMITS FOR 17 MINUTES. CONTROL BOARD INDICATION SHOWED THE AXIAL FLUX DIFFERENTIAL TO BE OUTSIDE THE PRESCRIBED LIMITS FOR LESS THAN 15 MINUTES. IN ACCORDANCE WITH OUR REACTOR ENGINEERING INSTRUCTIONS (REIS) THE PPCS IS TO BE USED FOR PRIMARY INDICATION AND CONTROL BOARD INDICATIONS WILL BE USED IF THE PPCS FAILS. CONTRARY TO TECHNICAL SPECIFICATION, POWER WAS NOT REDUCED AFTER THE LAPSE OF 15 MINUTES WITH THE "DELTA FLUX - OUTSIDE ENVELOPE" ALARM ON THE PPCS.

[148] PRAIRIE ISLAND 1 DOCKET 50-282 LER 90-009
AUTOMATIC START OF A COMPONENT COOLING PUMP DUE TO MOMENTARY LOW PRESSURE DURING SURVEILLANCE TEST.
EVENT DATE: 081190 REPORT DATE: 091090 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)

(NSIC 219468) ON 8/11/90, UNIT 1 WAS OPERATING AT 100% POWER. ROUTINE TESTING OF THE RESIDUAL HEAT REMOVAL PUMPS WAS IN PROGRESS. DURING THE PUMP TESTS, THE STANDBY COMPONENT COOLING WATER PUMP IS OPERATED. AT THE END OF THE TEST, THE

STANDBY COMPONENT COOLING WATER PUMP IS SHUT DOWN. ON THIS OCCASION, AT 0251, WHEN THE STANDBY COMPONENT COOLING WATER PUMP WAS SHUT DOWN, IT RESTARTED AUTOMATICALLY DUE TO MOMENTARY LOW PRESSURE IN THE COMPONENT COOLING WATER SYSTEM. THIS WAS A NON-ESF ACTUATION OF ESF EQUIPMENT. CAUSE OF THE EVENT WAS MOMENTARY LOW PRESSURE IN THE COMPONENT COOLING WATER SYSTEM. PRESSURE OSCILLATIONS IN THE SYSTEM NORMALLY OCCUR WHEN SHUTTING DOWN A PUMP. PROCEDURES WILL BE REVISED TO WARN THE OPERATOR OF COMPONENT COOLING SYSTEM PRESSURE OSCILLATIONS AND THE POSSIBILITY OF AUTOMATIC PUMP STARTS. SIMILAR EVENTS CAN BE PREVENTED IF THE PUMP CONTROL SWITCH IS HELD IN THE STOP POSITION A FEW MOMENTS UNTIL SYSTEM PRESSURE OSCILLATIONS ABATE.

[149] PRAIRIE ISLAND 1 DOCKET 50-282 LER 90-012
DISCOVERY THAT SEVERAL INSTRUMENTS USED FOR SURVEILLANCE ACCEPTANCE ARE NOT ROUTINELY CALIBRATED.
EVENT DATE: 081590 REPORT DATE: 091490 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)

(NSIC 219448) DURING ROUTINE INDEPENDENT AUDITS, IT WAS DETERMINED THAT SEVERAL INSTRUMENTS USED FOR JUDGING THE ACCEPTABILITY OF CERTAIN SURVEILLANCE TESTS WERE NOT ON A ROUTINE CALIBRATION SCHEDULE. THESE INSTRUMENTS INCLUDE PRESSURE GAUGES USED FOR MEASURING SUCTION AND DISCHARGE PRESSURE FOR ASME SECTION XI TESTING OF CERTAIN PUMPS, AND PRESSURE GAUGES AND FLOWMETERS USED TO PERFORM AIRLOCK DOOR SEAL TESTS. ALL THE INSTRUMENTS IN QUESTION WILL BE CALIBRATED. AS-FOUND CALIBRATION DATA WILL BE USED TO JUDGE ACCEPTABILITY OF PREVIOUS TEST DATA. ALL THE INSTRUMENTS WILL BE PLACED ON A ROUTINE CALIBRATION SCHEDULE.

[150] QUAD CITIES 1 DOCKET 50-254 LER 89-026 REV 01
UPDATE ON CONTROL ROOM HVAC ISOLATION DUE TO DRIED OUT CHLORINE PROBE CAUSED BY COLD DRY WEATHER.
EVENT DATE: 122589 REPORT DATE: 081590 NSSS: GE TYPE: BWR

(NSIC 219335) ON 12/25/89, AT 2028 HOURS, A HIGH CHLORINE CONCENTRATION SIGNAL WAS NOTED ON THE TOXIC GAS MONITORING PORTION OF THE CONTROL ROOM HEATING, VENTILATION, AND AIR CONDITIONING SYSTEM. THIS RESULTED IN A CONTROL ROOM HVAC ENGINEERED SAFETY FEATURE (ESF). A 4 HOUR EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE NOTIFICATION WAS MADE AT 2050 HOURS IN ACCORDANCE WITH 10CFR50.72(B)(2)(II). ON DECEMBER 26, 1989, THE MONITOR'S CHLORINE ANALYZER PROBE WAS INJECTED WITH AN ELECTROLYTIC SOLUTION IN ACCORDANCE WITH QIP 5700-2, FILLING PROCEDURE FOR THE CHLORINE ANALYZER PROBE. ON DECEMBER 27, 1989, THE CHLORINE CIRCUIT OF THE TOXIC GAS ANALYZER WAS CALIBRATED IN ACCORDANCE WITH QIS 79-S1, GAS ANALYZER CALIBRATION AND FUNCTIONAL TEST DATA SHEET, AND THE SYSTEM WAS RETURNED TO THE NORMAL OPERATING MODE. THE APPARENT CAUSE OF THE EVENT WAS A DRYING OUT OF THE CHLORINE PROBE WHICH RESULTED IN A FALSE READING. THIS EVENT WAS DETERMINED TO BE AN ISOLATED INCIDENT, HOWEVER, THE INSPECTION OF THE ELECTROLYTE LEVEL DURING COLD, DRY WEATHER WILL BE INCREASED. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV).

[151] QUAD CITIES 1 DOCKET 50-254 LER 90-016
LOSS OF DIESEL FIRE PUMP DUE TO FAILURE OF THE STARTING RELAYS, WITH A DIESEL FIRE PUMP OUT OF SERVICE FOR MAINTENANCE.
EVENT DATE: 080990 REPORT DATE: 091090 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)
VENDOR: CUMMINS ENGINE CO., INC.

(NSIC 219488) ON AUGUST 9, 1990, UNITS ONE AND TWO WERE IN THE RUN MODE AT 36 PERCENT AND 100 PERCENT OF RATED CORE THERMAL POWER, RESPECTIVELY. AT 1110 HOURS WITH THE 1/2B DIESEL FIRE PUMP (FP) OUT OF SERVICE FOR MAINTENANCE ON THE DISCHARGE HEADER RELIEF VALVE, THE 1/2A DIESEL FIRE PUMP RECEIVED AN AUTO START SIGNAL, BUT FAILED TO START DUE TO FAILURE OF THE STARTING RELAYS. AT 1140 HOURS, THE SERVICE WATER CROSSTIE VALVE WAS OPENED AND A BACKUP DIESEL DRIVEN PUMP WAS CONNECTED TO THE FIRE MAIN IN ORDER TO PROVIDE BACK UP FIRE SUPPRESSION WATER. THE IDENTICAL RELIEF VALVE FROM THE 1/2A DIESEL FIRE PUMP WAS INSTALLED ON THE 1/2B DIESEL FIRE PUMP IN ORDER TO RESTORE THE 1/2B SYSTEM TO AN OPERABLE

STATUS ON AUGUST 9, 1990, AT 2212 HOURS. THE 1/2A DIESEL FIRE PUMP WAS REPAIRED ON AUGUST 16, 1990 BY REPLACING THE FAULTY RELAYS AND ITS RELIEF VALVE. THIS REPORT IS SUBMITTED AS A VOLUNTARY REPORT.

[152] QUAD CITIES 1 DOCKET 50-254 LER 90-017
EXCESSIVE LEAKAGE THROUGH HPCI STEAM EXHAUST CHECK VALVE DUE TO DETERIORATION OF SEAT.
EVENT DATE: 081190 REPORT DATE: 091090 NSSS: GE TYPE: BWR
VENDOR: MARLIN MFG.

(NSIC 219489) ON AUGUST 11, 1990, AT 0818 HOURS, UNIT ONE WAS IN THE RUN MODE AT 82 PERCENT OF RATED CORE THERMAL POWER. A LOCAL LEAK RATE TEST (LLRT) WAS PERFORMED ON HIGH PRESSURE COOLANT INJECTION (HPCI) STEAM EXHAUST CHECK VALVE 1-2301-45. THE VALVE FAILED THE LLRT WITH A LEAKAGE VALUE OF 993 STANDARD CUBIC FEET PER HOUR (SCFH). A ONE-HOUR EMERGENCY NOTIFICATION SYSTEM (ENS) NOTIFICATION FOR THIS EVENT WAS COMPLETED AT 0913 HOURS IN ACCORDANCE WITH 10CFR50.72(B)(1)(II). THE CHECK VALVE FAILED TO PASS THE LLRT DUE TO DEGRADATION OF THE NORDEL ELASTOMER SEAT MATERIAL. IT APPEARS THAT AS THE CHECK VALVE CYCLED OPEN AND CLOSED, THE EDGE OF THE VALVE DISKS WORE AGAINST THE RAISED RIBS ON THE SEAT CAUSING GAPS IN THE SEAL TO OCCUR. REPEATED CYCLING OF THE CHECK VALVE OCCURS DURING LOW SPEED TURBINE OPERATION. THE STEAM EXHAUST CHECK VALVE WAS REPLACED LIKE-FOR-LIKE WITH A SPARE CHECK VALVE. FUTURE CORRECTIVE ACTIONS INCLUDE CHANGING THE SEAT MATERIAL TO ONE MORE RESISTANT TO HIGH TEMPERATURES AND MODIFYING THE HPCI SURVEILLANCE PROCEDURES TO REDUCE THE AMOUNT OF STEAM EXHAUST CHECK VALVE CYCLING.

[153] RIVERBEND 1 DOCKET 50-458 LER 86-023 REV 01
UPDATE ON DIESEL GENERATOR FUEL OIL VALVE MISALIGNMENT.
EVENT DATE: 031786 REPORT DATE: 083190 NSSS: GE TYPE: BWR

(NSIC 219478) ON 03/17/86 AT 1526 WITH THE UNIT AT 44 PERCENT POWER AND DURING A SURVEILLANCE TEST OF THE DIVISION II DIESEL GENERATOR, THE DIESEL BEGAN TO LOSE SPEED AND WAS MANUALLY TRIPPED. INVESTIGATION REVEALED THAT A MISALIGNED FUEL OIL STRAINER VALVE RESTRICTED FUEL OIL FLOW TO THE ENGINE. FURTHER INVESTIGATION REVEALED THAT THE DIESEL GENERATOR MAY HAVE BEEN INOPERABLE SINCE 2/17/86. TECHNICAL SPECIFICATIONS REQUIRES A PLANT SHUTDOWN IF ONE DIESEL GENERATOR IS INOPERABLE FOR GREATER THAN 72 HOURS. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO RESTORE OPERABILITY TO THE DIESEL GENERATOR BY 03/18/86. BECAUSE OF A SIMILAR PROBLEM EARLIER WITH THE DIVISION I DIESEL GENERATOR, ADDITIONAL CORRECTIVE ACTION WAS TAKEN TO PREVENT RECURRENCE. THERE WAS NO ADVERSE EFFECT ON THE HEALTH AND SAFETY OF THE PUBLIC.

[154] RIVERBEND 1 DOCKET 50-458 LER 90-002 REV 01
UPDATE ON LOSS OF BOTH DIVISIONS OF THE CONTROL BUILDING VENTILATION SYSTEM DUE TO THE FAILURE OF TWO CHILLERS TO START.
EVENT DATE: 020290 REPORT DATE: 090490 NSSS: GE TYPE: BWR
VENDOR: CARRIER CORP.

(NSIC 219526) AT 0956 HOURS ON 02/02/90, WITH THE UNIT AT 100% POWER, BOTH DIVISIONS OF THE CONTROL BUILDING VENTILATION SYSTEM WERE DECLARED INOPERABLE DUE TO THE LOSS OF CHILLERS 1HVK*CHL1A (DIVISION I) AND 1HVK*CHL1D (DIVISION II) (NOTE THAT EACH DIVISION CONSISTS OF (2) 100% CAPACITY CHILLERS). THIS OCCURRED AS A RESULT OF AN UNSUCCESSFUL ATTEMPT TO SWAP THE OPERATING DIVISION FROM DIVISION II TO DIVISION I. THIS WAS FOLLOWED BY UNSUCCESSFUL ATTEMPTS TO RETURN DIVISION II TO SERVICE AFTER IT HAD BEEN SECURED. THE REMAINING CHILLER IN EACH DIVISION WAS INOPERABLE DUE TO ANNUAL INSPECTION ACTIVITIES. THEREFORE, ALL FOUR CHILLERS WERE INOPERATIVE FOR THE DURATION OF THE EVENT. TECH SPEC 3.0.3 WAS ENTERED SINCE THE LIMITING CONDITION FOR OPERATION (LCO) OF TECH SPEC 3.7.2 COULD NOT BE COMPLIED WITH. THEREFORE, THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(I)(B). A PLANT SHUTDOWN WAS INITIATED IN ACCORDANCE WITH TECH SPEC 3.0.3. HOWEVER, NRC REGION IV GRANTED 6 HOURS OF DISCRETIONARY ENFORCEMENT, PROVIDING RELIEF FROM THE SIX HOUR HOT SHUTDOWN REQUIREMENT. THE INOPERABLE DIVISION WAS RESTORED WITHIN APPROX. 4 HOURS AND THE PLANT SHUTDOWN WAS

TERMINATED. CORRECTIVE ACTIONS INCLUDED PROCEDURE REVISIONS AND REBALANCING THE FLOWS IN THE CHILLERS TO PROVIDE ADEQUATE FLOW. GSU'S EVALUATION TO DATE INDICATES THAT CAUSE OF TRIP IS INDETERMINATE.

[155] RIVERBEND 1 DOCKET 50-458 LER 90-005 REV 01
UPDATE ON CONDITIONS AFFECTING THE OFFGAS PRETREATMENT RADIATION MONITOR.
EVENT DATE: 030290 REPORT DATE: 083190 NSSS: GE TYPE: BWR
VENDOR: STONE & WEBSTER ENGINEERING CORP.

(NSIC 219516) ON 03/02/90 AND ON 03/06/90 WITH THE REACTOR OPERATING AT 100 PERCENT POWER (OPERATIONAL CONDITION 1), CONDITIONS AFFECTING THE OFFGAS PRETREATMENT RADIATION MONITOR (PTRM) WERE IDENTIFIED BY ENGINEERING PERSONNEL. THESE CONDITIONS WERE (1) A NON-CONSERVATIVE VALUE FOR THE HIGH ALARM SETPOINT, AND (2) PERIODS OF INOPERABILITY IN THE PAST IN WHICH THE MONITOR HAS BEEN INOPERABLE DUE TO INADEQUATE SAMPLE FLOW. THESE ARE REPORTABLE AS (1) A CONDITION OUTSIDE OF THE DESIGN BASIS, AND (2) OPERATION PROHIBITED BY THE TECHNICAL SPECIFICATIONS, RESPECTIVELY. THE ERRONEOUS HIGH ALARM SETPOINT HAS BEEN CORRECTED. GSU HAS ALSO IDENTIFIED SHORT AND LONG TERM CORRECTIVE ACTIONS TO ASSURE ADEQUATE SAMPLE FLOW IS MAINTAINED TO THE MONITOR AND THUS ASSURE OPERABILITY. DUE TO THE PRESENCE OF REDUNDANT, INDEPENDENT MONITORS AND ALARMS, ADEQUATE ASSURANCE EXISTS THAT RBS HAS NOT EXCEEDED THE REVISED ALARM SETPOINT. FOR THE SAME REASON, ALTERNATIVE MEANS OF MONITORING RADIOACTIVITY IS PROVIDED IN THE RBS DESIGN EVEN WITH THE PTRM INOPERABLE. THEREFORE, THESE CONDITIONS HAVE NOT ADVERSELY AFFECTED THE HEALTH AND SAFETY OF THE PUBLIC.

[156] RIVERBEND 1 DOCKET 50-458 LER 90-024
MISSED FIRE WATCH PATROL DUE TO MISCOMMUNICATION AND INADEQUATE VERIFICATION OF FIRE WATCH LOG.
EVENT DATE: 071990 REPORT DATE: 082090 NSSS: GE TYPE: BWR

(NSIC 219380) AT APPROXIMATELY 1700 HOURS ON 7/19/90, IT WAS DISCOVERED BY FIRE WATCH PERSONNEL THAT THE REQUIRED HOURLY FIRE WATCH PATROL SCHEDULED FOR 1600 FOR THE AUXILIARY BUILDING, FUEL BUILDING, STANDBY COOLING TOWER, AND VARIOUS PARTS OF THE TUNNELS HAD NOT BEEN SIGNED FOR IN THE LOG. FURTHER INVESTIGATION REVEALED THE INSPECTION HAD NOT BEEN DONE AS REQUIRED BY RIVER BEND STATION TECHNICAL SPECIFICATION 3/4.7.7. THEREFORE, THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(I)(B) AS OPERATION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. CORRECTIVE ACTION TO PREVENT RECURRENCE WILL CONSIST OF THE USE OF A NEW LOG BOOK. THIS LOG BOOK IS DESIGNED TO BE PASSED DOWN TO EACH SUCCEEDING SHIFT AND WILL PROVIDE DETAILS OF ANY CHANGES, FIRE WATCH PATROLS COMPLETED AND SIGNED OFF, AND THE IDENTIFICATION OF PROBLEMS. ALL FIRE PROTECTION SYSTEMS WERE OPERATIONAL IN THE AFFECTED AREAS AND THERE WAS NO FIRE IN THE SUBJECT AREA DURING THE PERIOD OF NONCOMPLIANCE. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT ADVERSELY AFFECTED BY THIS EVENT.

[157] RIVERBEND 1 DOCKET 50-458 LER 90-025
FAILURE TO PERFORM SURVEILLANCES ON DRYWELL ISOLATION VALVE DUE TO OMISSION OF VALVE FROM TECHNICAL SPECIFICATION.
EVENT DATE: 080990 REPORT DATE: 090790 NSSS: GE TYPE: BWR

(NSIC 219517) ON 08/09/90 WITH THE PLANT AT 100% POWER GSU DETERMINED THAT A REPORTABLE CONDITION EXISTED CONCERNING A MANUAL DRYWELL ISOLATION VALVE (1HVN*V343). SURVEILLANCE REQUIREMENTS PER TECH SPEC (TS) 4.6.2.1-A HAD NOT BEEN PERFORMED ON THIS VALVE DUE TO ITS OMISSION FROM TS TABLE 3.6.4-1. ON 01/23/90 IT WAS DISCOVERED THAT 1HVN*V343 HAD BEEN INADVERTENTLY OMITTED FROM TECHNICAL SPECIFICATION (TS) TABLE 3.6.4-1, "CONTAINMENT AND DRYWELL ISOLATION VALVES." THIS VALVE ISOLATES CHILLED WATER RETURNING FROM THE DRYWELL. THE CHILLED WATER SUPPLY LINE ISOLATION VALVE (1HVN*V342) IS INCLUDED IN THE TS TABLE. THE INITIAL REPORTABILITY CONSIDERATION DETERMINED THAT THIS EVENT WAS NOT REPORTABLE BECAUSE OF COMPLIANCE WITH THE REQUIREMENTS OF TS 3/4.6.4. HOWEVER, AFTER FURTHER INVESTIGATION, DURING PREPARATION OF A LICENSE AMENDMENT REQUEST, IT WAS DISCOVERED THAT THE SURVEILLANCE REQUIREMENT 4.6.2.1-A FOR TS 3/4.6.2.1, "DRYWELL INTEGRITY," HAD NOT BEEN PERFORMED FOR VALVE 1HVN*V343. THIS REQUIRES THAT

DRYWELL ISOLATION VALVES BE VERIFIED AS CLOSED ONCE EVERY 31 DAYS. THEREFORE, THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73 (A)(2)(I)(B) AS OPERATION PROHIBITED BY THE TECH SPECS. THE MONTHLY OPERATING LOG, WAS REVISED TO REQUIRE THAT 1HVN*V543 VERIFIED AS CLOSED ONCE EVERY 31 DAYS, THUS SATISFYING THE INTENT OF SURVEILLANCE REQUIREMENT.

[158] RIVERBEND 1 DOCKET 50-458 LER 90-026
REACTOR WATER CLEANUP SYSTEM ISOLATION DUE TO SHORT CIRCUIT DURING JUMPER MANIPULATION.
EVENT DATE: 081090 REPORT DATE: 090790 NSSS: GE TYPE: BWR
VENDOR: TOPAZ ELECTRONICS

(NSIC 219518) AT APPROXIMATELY 0202 ON 08/10/90 WITH THE UNIT AT FULL POWER IN OPERATIONAL CONDITION 1, THE REACTOR WATER CLEANUP SYSTEM (RWCU) ISOLATED. MAINTENANCE WAS INSTALLING JUMPERS TO ALLOW REPLACEMENT OF A FAULTY OPTICAL ISOLATOR CARD WHEN A SHORT CIRCUIT OCCURRED. THIS SHORT CAUSED TWO FUSES TO BLOW, RESULTING IN A LOSS OF POWER TO SEVERAL BALANCE-OF-PLANT (BOP) OPTICAL ISOLATOR CARDS. LOSS OF THESE ISOLATOR CARDS DE-ENERGIZED NORMALLY ENERGIZED RELAYS FOR THE HVAC - COOLING WATER SYSTEM (HVN). THIS CAUSED AN ISOLATION OF THE COOLING WATER TO THE CONTAINMENT UNIT COOLERS. WITH COOLING WATER ISOLATED, AREAS IN THE CONTAINMENT SERVED BY THESE UNITS BEGAN TO HEAT UP. WHEN THE RWCU HEAT EXCHANGER ROOM AMBIENT TEMPERATURE REACHED THE ISOLATION SETPOINT, AN ISOLATION OF THE SYSTEM OCCURRED. THEREFORE THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV) AS ESF ACTUATION. A SHORT CIRCUIT DURING JUMPER MANIPULATION RESULTED IN THE RWCU ISOLATION. INSTALLATION OF JUMPERS ON THE OPTICAL ISOLATOR CARDS IS COMPLICATED BY THE RELATIVE LOCATIONS OF THE POSITIVE AND NEGATIVE TERMINALS, WHICH APPROXIMATELY 1/4" APART WITH NO BARRIERS BETWEEN. AS CORRECTIVE ACTION, THIS REPORT WILL BE REQUIRED READING FOR APPLICABLE MAINTENANCE PERSONNEL BY 10/31/90. SINCE ALL PLANT SYSTEMS PERFORMED AS DESIGNED, THIS EVENT DID NOT ADVERSELY AFFECT THE HEALTH AND SAFETY OF THE PUBLIC.

[159] ROBINSON 2 DOCKET 50-261 LER 90-005 REV 01
UPDATE ON FAILURE TO TEST RPS LOGIC CHANNELS IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS.
EVENT DATE: 030290 REPORT DATE: 083090 NSSS: WE TYPE: PWR

(NSIC 219407) ON MARCH 2, 1990, SITE MAINTENANCE PERSONNEL IDENTIFIED A PROCEDURAL DEFICIENCY IN THAT MONTHLY TESTING OF LOGIC CHANNELS ASSOCIATED WITH CERTAIN REACTOR PROTECTION SYSTEM (RPS) FEATURES WAS NOT BEING PERFORMED IN ACCORDANCE WITH TECHNICAL SPECIFICATION TABLE 4.1-1, ITEM 27. THE LOGIC CHANNELS AFFECTED ARE ONLY APPLICABLE DURING PLANT STARTUP AND REDUCED POWER OPERATION. THE CAUSE OF THIS PROCEDURAL DEFICIENCY HAS BEEN ATTRIBUTED TO THE CONFIGURATION OF ORIGINALLY INSTALLED EQUIPMENT FOR TESTING OF RPS LOGIC CHANNELS, AND A MISINTERPRETATION OF THE TECHNICAL SPECIFICATIONS. CHANGES TO APPLICABLE PROCEDURES WERE PROMPTLY IMPLEMENTED FOR TESTING OF LOGIC CHANNELS FOR RPS FEATURES WHICH ARE CREDITED FOR MITIGATION OF ANALYZED ACCIDENTS. THE LOGIC CHANNELS FOR THESE FEATURES WERE SUCCESSFULLY TESTED ON MARCH 14 AND 15, 1990. THE REMAINING AFFECTED RPS LOGIC CHANNELS HAVE EITHER BEEN INCORPORATED INTO MONTHLY TEST PROCEDURES, OR HAVE BEEN ADDRESSED BY AN AMENDMENT TO THE TECHNICAL SPECIFICATIONS. THIS LICENSEE EVENT REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(I)(B) AS AN OPERATION OR CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[160] ROBINSON 2 DOCKET 50-261 LER 90-011
TECHNICAL SPECIFICATION VIOLATION DUE TO INOPERABLE FIRE BARRIER PENETRATION.
EVENT DATE: 080290 REPORT DATE: 082990 NSSS: WE TYPE: PWR

(NSIC 219408) ON AUGUST 2, 1990, WITH H. B. ROBINSON UNIT NO. 2 OPERATING AT ONE HUNDRED PERCENT POWER, AN HVAC DAMPER WHICH CONSTITUTES A FIRE BARRIER PENETRATION WAS DISCOVERED IN THE OPEN (INOPERABLE) POSITION DURING SCHEDULED SURVEILLANCE TESTING. THE DAMPER WAS DECLARED INOPERABLE AT 0430 HOURS, AND A WORK REQUEST WAS INITIATED TO RETURN THE DAMPER TO THE CLOSED (OPERABLE) POSITION. COMPENSATORY ACTIONS WERE TAKEN IN ACCORDANCE WITH TECHNICAL

SPECIFICATION 3.14.7.2. THE DAMPER WAS RETURNED TO SERVICE AT 1330 HOURS ON AUGUST 2, 1990. AN INVESTIGATION INTO THE CIRCUMSTANCES SURROUNDING THE DAMPER MISPOSITIONING WAS INITIATED, BUT THE ROOT CAUSE COULD NOT BE DETERMINED. HOWEVER, SINCE IT CAN BE ESTABLISHED THAT THIS FIRE BARRIER PENETRATION HAD BEEN INOPERABLE FOR A TIME PERIOD WHICH EXCEEDED THE REQUIREMENTS OF THE TECHNICAL SPECIFICATION LIMITING CONDITION FOR OPERATION, THIS LICENSEE EVENT REPORT IS SUBMITTED PURSUANT TO 10CFR50.73 (A)(2)(I)(B) AS A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[161] SALEM 1 DOCKET 50-272 LER 90-026
ASME CODE 3 PIPING LEAKAGE DUE TO EQUIPMENT FAILURE.
EVENT DATE: 061190 REPORT DATE: 090690 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 219497) THIS LICENSEE EVENT REPORT (LER) ADDRESSES SEVERAL OCCURRENCES OF ASME CODE 3 PIPING LEAKAGE. BASED UPON DISCUSSION WITH THE NUCLEAR REGULATORY COMMISSION (NRC) NOTIFICATIONS WERE MADE WITHIN FOUR (4) HOURS OF EACH OCCURRENCE. THIS WAS DONE AS AGREED IN ACCORDANCE WITH CODE OF FEDERAL REGULATIONS 10CFR 50.72. IN ALL CASES, SALEM UNIT 1 TECHNICAL SPECIFICATION 3.4.10.1 ACTION C AND SALEM UNIT 2 TECHNICAL SPECIFICATION 3.4.11.1 ACTION C WERE COMPLIED WITH. THE SALEM UNIT 1 AND UNIT 2 TECHNICAL SPECIFICATION FOR "STRUCTURAL INTEGRITY" ARE IDENTICAL EXCEPT FOR THEIR NUMBER (I.E., 3.4.10.1 VS. 3.4.11.1). THE ROOT CAUSE OF THE LISTED ASME CODE III COMPONENT LEAKAGE HAS BEEN ATTRIBUTED TO EQUIPMENT FAILURE. THE COMPONENT LEAKS WERE THE RESULT OF EROSION/CORROSION FACTORS. THE COMPONENTS WHICH EXHIBITED LEAKAGE WERE DECLARED INOPERABLE IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS. THE COMPONENTS WERE NOT DECLARED OPERABLE UNTIL COMPLETION OF REPAIRS, WHICH WERE DONE IN ACCORDANCE WITH THE ASME CODE FOR CLASS CODE III COMPONENTS. THE REQUIREMENTS OF THE TECHNICAL SPECIFICATIONS WERE COMPLIED WITH IN ALL CASES.

[162] SALEM 1 DOCKET 50-272 LER 90-023
TECH SPEC NONCOMPLIANCE DUE TO INADEQUATE ADMINISTRATIVE CONTROLS.
EVENT DATE: 072590 REPORT DATE: 082390 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 219410) ON 7/25/90, IT WAS DISCOVERED THAT THE REQUIRED SURVEILLANCE OF THE UNITS 1 AND 2 SERVICE WATER SYSTEM SW51 CHECK VALVES HAVE NOT BEEN COMPLETED IN ACCORDANCE WITH THE REQUIREMENTS OF THE INSERVICE TESTING PROGRAM MANUAL, WHICH REQUIRES TESTING OF ASME CODE COMPONENTS. SPECIFICALLY, THE IST PROGRAM REQUIRED DISASSEMBLY AND INSPECTION OF THE SW51 VALVES WHEN THE UNIT ENTERS COLD SHUTDOWN IF NOT DONE WITHIN THE LAST 3 MONTHS. CONTRARY TO THIS, THE VALVES HAD BEEN SCHEDULED TO BE PERFORMED EVERY 18 MONTHS. A REVIEW SHOWS THAT THESE SURVEILLANCES WERE COMPLETED DURING REFUELING OUTAGES. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INADEQUATE ADMINISTRATIVE CONTROLS. AN ADDITIONAL CONTRIBUTING FACTOR ASSOCIATED WITH THIS EVENT WAS THAT THE COMPUTER BASED WORK ACTIVITY SYSTEM, MANAGED MAINTENANCE INFORMATION SYSTEM (MMIS), IDENTIFIED THE SW51 VALVE SURVEILLANCE RECURRING TASK AS A "PM" INSTEAD OF CORRECTLY IDENTIFYING IT AS AN "ST". AS A RESULT OF A RECENT REVIEW OF NRC GENERIC LETTER NO. 89-04, THE IST PROGRAM HAS SINCE BEEN MODIFIED TO REQUIRE THE IST SURVEILLANCE OF THE SW51 VALVES TO BE PERFORMED EVERY REFUELING. THE RECURRING TASKS, IN THE MMIS, HAVE BEEN CORRECTED ACCORDINGLY. A VERIFICATION AUDIT OF THE TECH SPEC SURVEILLANCES, INITIATED IN 1989, IS CONTINUING. THE PROCEDURE UPGRADE PROJECT IS CONDUCTING A THOROUGH TECHNICAL VERIFICATION OF PROCEDURES IN USE.

[163] SALEM 1 DOCKET 50-272 LER 90-024
TECHNICAL SPECIFICATION SURVEILLANCE NONCOMPLIANCE FOR P-10 AND P-12 INTERLOCK DUE TO INADEQUATE ADMINISTRATIVE CONTROLS.
EVENT DATE: 072890 REPORT DATE: 082190 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 219411) ON 7/28/90, DURING THE COURSE OF THE TECH. SPEC. SURVEILLANCE AUDIT PROJECT (REF. UNIT 2 LER 311/89-015-00), IT WAS DISCOVERED THAT SOME OF THE INTERLOCK FUNCTIONS FOR THE UNITS 1 AND 2 P-10 AND P-12 PERMISSIVES ARE NOT FULLY

TESTED IN ACCORDANCE WITH TECH. SPEC. SURVEILLANCE 4.3.1.1.2. THE P-10 PERMISSIVE FUNCTION THAT WAS NOT FULLY TESTED (PER THE SURVEILLANCE) INVOLVES THE NUCLEAR INSTRUMENTATION SYSTEM (NIS) INTERMEDIATE RANGE CHANNEL ROD STOP. THE P-12 PERMISSIVE FUNCTION THAT WAS NOT FULLY TESTED (PER THE SURVEILLANCE) INVOLVES THE STEAM DUMP SYSTEM (JI) BLOCK. THE APPARENT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INADEQUATE ADMINISTRATIVE CONTROLS OF THE TECHNICAL SPECIFICATION SURVEILLANCE TESTING PROGRAM WHEN THE PROGRAM WAS FIRST INITIATED. THE NIS FUNCTIONAL PROCEDURES, FOR THE INTERMEDIATE RANGE CHANNELS, HAVE BEEN REVISED TO ADDRESS TESTING OF THE P-10 PERMISSIVE FOR ROD STOP MANUAL BLOCK. NEW PROCEDURES HAVE BEEN ISSUED TO ADDRESS TESTING THE P-12 PERMISSIVE STEAM DUMP SYSTEM BLOCK FUNCTION. A RECURRING TASK, TO ENSURE THE COMPLETION OF THE P-12 PERMISSIVE STEAM DUMP SYSTEM BLOCK FUNCTION SURVEILLANCE WITHIN THE REQUIRED TIME FRAME, HAS BEEN ADDED TO THE MMIS (MANAGED MAINTENANCE INFORMATION SYSTEM), A COMPUTERIZED WORK TRACKING SYSTEM. THE P-10 PERMISSIVE SURVEILLANCE IS ALREADY ADDRESSED IN THE MMIS.

[164] SALEM 1 DOCKET 50-272 LER 90-025
CONTROL ROOM VENTILATION SWAP DUE TO 1R1B CHANNEL EQUIPMENT FAILURE.
EVENT DATE: 080690 REPORT DATE: 082890 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)
VENDOR: LFE CORP.

(NSIC 219496) ON 8/6/90 AT 0915 HOURS, CONTROL ROOM AIR INTAKE RADIATION MONITORING SYSTEM (RMS) MONITOR (1R1B) ALARM CIRCUITRY SPIKED HIGH RESULTING IN AUTOMATIC SWITCHING OF CONTROL ROOM VENTILATION TO THE ACCIDENT MODE OF OPERATION (ESF) FOR BOTH SALEM UNIT 1 AND SALEM UNIT 2. THE 1R1B CHANNEL SPIKE OCCURRED WHILE A MAINTENANCE-I&C TECHNICIAN WAS TROUBLESHOOTING THE 1R13B CHANNEL. THE 1R13B CHANNEL CONTROL CIRCUITRY IS LOCATED IN THE SAME CABINET AS THE 1R1B CHANNEL CONTROL CIRCUITRY. IT WAS ASSUMED THAT THE SPIKES WERE RELATED TO THE TECHNICIANS WORK WITH THE 1R13B CHANNEL. ADDITIONAL 1R1B CHANNEL SPIKES OCCURRED ON 8/7/90 AT 0807 HOURS AND ON 8/8/90 AT 0350 HOURS AND 1239 HOURS RESULTING IN ESF ACTUATIONS. AT THE TIME OF THESE CHANNEL SPIKES, WORK WAS NOT BEING PERFORMED ON THE 1R13B CONTROL CIRCUITRY. ROOT CAUSE OF THE 1R1B CHANNEL ESF ACTUATIONS HAS BEEN ATTRIBUTED TO EQUIPMENT FAILURE. TWO (2) FAILED OPEN CAPACITORS, IN THE POWER SUPPLY OF THE CONTROLS CIRCUITRY, WERE FOUND. A FUNCTION OF THESE CAPACITORS IS TO FILTER ELECTRICAL NOISE. THE THIRD AND FOURTH 1R1B CHANNEL SPIKES COULD HAVE BEEN AVOIDED. AFTER THE SECOND SPIKE, OPERATIONS PERSONNEL SHOULD HAVE EXPEDITED THE INVESTIGATIVE PROCESS. THE FOURTH ACTUATION COULD HAVE BEEN AVOIDED HAD MAINTENANCE-I&C SUPERVISION ENSURED A MORE TIMELY RESPONSE. THE 1R1B CHANNEL FAILED CAPACITORS WERE REPLACED AND A CHANNEL FUNCTIONAL TEST WAS COMPLETED.

[165] SALEM 1 DOCKET 50-272 LER 90-027
MAIN STEAMLINE ISOLATION DUE TO EQUIPMENT/DESIGN CONCERN.
EVENT DATE: 081290 REPORT DATE: 090690 NSSS: WE TYPE: PWR
VENDOR: ROSEMOUNT, INC.

(NSIC 219498) ON 8/12/90 AT 0831 HOURS, A MAIN STEAMLINE ISOLATION ACTUATION OCCURRED. AT THE TIME OF THE EVENT, THE UNIT WAS IN MODE 4 AND HEATING UP IN PREPARATION FOR STARTUP. THE MAIN STEAMLINE ISOLATION SIGNAL OCCURRED UPON RECEIPT OF A HIGH STEAMLINE FLOW SIGNAL COINCIDENT WITH A LOW STEAMLINE PRESSURE SIGNAL. IN MODE 4, THE BISTABLES FOR LOW STEAMLINE PRESSURE ARE TRIPPED PROVIDING HALF THE LOGIC SIGNAL REQUIRED FOR MAIN STEAMLINE ISOLATION. THE HIGH STEAMLINE FLOW LOGIC WAS COMPLETED WHEN THE NO. 11 STEAM GENERATOR (S/G) CHANNEL I AND THE NO. 12 S/G CHANNEL II HIGH STEAMLINE FLOW BISTABLES TRIPPED. MAIN STEAMLINE ISOLATION EVENT OCCURRED ON 6/3/90 (REFERENCE LER 272/90-019-00). ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO POSSIBLE EQUIPMENT/DESIGN CONCERNS ASSOCIATED WITH THE MAIN STEAMLINE FLOW TRANSMITTER SENSING LINES. INITIAL INVESTIGATION BY MAINTENANCE PERSONNEL, TO DETERMINE WHY THE NOS. 11 AND 12 S/G HIGH STEAMLINE FLOW BISTABLES TRIPPED, DID NOT IDENTIFY ANY FAILED COMPONENTS. A CHANNEL FUNCTIONAL TEST, OF THE SUBJECT STEAM FLOW CHANNELS, WAS SUCCESSFULLY COMPLETED. STEAMLINE FLOW MEASURING DESIGN CONCERNS HAVE PREVIOUSLY BEEN IDENTIFIED VIA LER 272/88-017-01 (I.E., STEAMLINE FLOW MEASUREMENT DRIFT CONCERN). ENGINEERING BELIEVES THAT THE DRIFT AND THIS RECENT EVENT APPEAR TO HAVE A RELATED CAUSE.

DESIGN MODIFICATIONS, TO CORRECT THE MAIN STEAMLINE FLOW INSTRUMENTATION CONCERNS ARE UNDER DEVELOPMENT.

[166] SALEM 1 DOCKET 50-272 LER 90-028
MORE THAN ONE ANALOG ROD POSITION INDICATOR PER BANK INOPERABLE DUE TO SYSTEM DESIGN.
EVENT DATE: 081490 REPORT DATE: 091290 NSSS: WE TYPE: PWR

(NSIC 2194[illegible]) ON 8/14/90 AT 2355 HOURS, THE CONTROL ROOM OPERATOR OBSERVED THAT TWO INDIVIDUAL ROD POSITION INDICATIONS (IRPIS) IN ONE CONTROL BANK AND TWO IRPIS EACH IN TWO DIFFERENT SHUTDOWN BANKS HAD TWO IRPI WITH A GREATER THAN +/- 12 STEP DEVIATION, FROM ITS GROUP DEMAND COUNTER. CONTROL BANK IRPIS 1B1 AND 1B2 AND SHUTDOWN BANK IRPIS 2SA2, 2SA4, 1SD1 AND 1SD3 WERE INVOLVED. TECH SPEC 3.1.3.2.1 ADDRESSES THE OPERABILITY REQUIREMENT OF THE "REACTIVITY CONTROL SYSTEM'S" POSITION INDICATING SYSTEMS. THE ACTIONS REQUIRED, IF MORE THAN ONE ANALOG ROD POSITION INDICATOR (ARPI) PER BANK IS INOPERABLE, IS NOT DIRECTLY PROVIDED FOR IN TECH SPEC 3.1.3.2.1 THEREFORE, THE ACTIONS ASSOCIATED WITH TECH SPEC 3.0.3 APPLIES (AND WAS CONSEQUENTLY ENTERED). ON 8/15/90 AT 0057 HOURS ALL CONTROL ROD BANKS WERE INSERTED INTO THE CORE AND THE UNIT ENTERED MODE 3 (HOT STANDBY). SINCE TECH SPEC 3.1.3.2.1 IS APPLICABLE ONLY IN MODES 1 AND 2, TECH SPEC 3.0.3 WAS EXITED. THE ROOT CAUSE OF THE IRPIS HAVING GREATER THAN +/- 12 STEP DEVIATION FROM THEIR GROUP DEMAND COUNTER, FOR THE SIX (6) CONTROL RODS, HAS BEEN ATTRIBUTED TO SYSTEM DESIGN. THE ARPI SYSTEM ELECTRONICS SETTINGS WILL HAVE SOME DRIFT DUE TO THE INHERENT NATURE OF THE ANALOG STACK COILS SUSCEPTIBILITY TO TEMPERATURE CHANGES ASSOCIATED WITH MODE CHANGE. THE IRPI FOR THE SUBJECT CONTROL RODS WAS SUCCESSFULLY RECALIBRATED.

[167] SALEM 1 DOCKET 50-272 LER 90-029
REACTOR TRIP FROM 25% POWER DUE TO EQUIPMENT FAILURE.
EVENT DATE: 081790 REPORT DATE: 091290 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219300) ON 8/17/90, DURING POWER ASCENSION, THE REACTOR TRIPPED DUE TO NO. 14 STEAM GENERATOR (S/G) LOW-LOW LEVEL. THE TRIP OCCURRED DURING THE TRANSFER OF THE FOUR GROUP BUSSES FROM THE STATION POWER TRANSFORMER (SPT) TO THE AUX. POWER TRANSFORMER (APT). AFTER SUCCESSFULLY TRANSFERRING THE 1H AND 1E GROUP BUSSES, 1G GROUP BUS WAS BEING TRANSFERRED. THE OPERATOR INITIATED CLOSURE OF THE 1BGGD CIRCUIT BREAKER (APT SIDE). THE 12GSD BREAKER OPENED; HOWEVER, THE 1BGGD BREAKER FAILED TO CLOSE RESULTING IN DEENERGIZATION OF THE 1G GROUP BUS. CONSEQUENTLY THE NO. 14 REACTOR COOLANT PUMP (RCP) CIRCUIT BREAKER OPENED ON UNDERVOLTAGE. THE REACTOR SUBSEQUENTLY TRIPPED ON NO. 14 S/G LOW-LOW S/G LEVEL. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO EQUIPMENT FAILURE. THE 1BGGD CIRCUIT BREAKER FAILED TO CLOSE DURING TRANSFER DUE TO ITS POSITIVE INTERLOCK SWITCH NOT CLOSING THEREBY PREVENTING ENERGIZATION OF THE BREAKER CLOSE COIL. THE POSITIVE INTERLOCK ASSEMBLY WAS FOUND TO BE OUT OF ADJUSTMENT/WORN PREVENTING THE REQUIRED CLOSING OF THE BREAKER'S POSITIVE INTERLOCK SWITCH AFTER THE BREAKER WAS RACKED INTO ITS NORMAL OPERATING POSITION. THE 1BGGD CIRCUIT BREAKER ENCLOSURE POSITIVE INTERLOCK ASSEMBLY DEFICIENCIES WERE CORRECTED.

[168] SALEM 2 DOCKET 50-311 LER 90-032
TECH SPEC 4.0.5 NONCOMPLIANCE DUE TO INADEQUATE ADMINISTRATIVE CONTROL.
EVENT DATE: 070290 REPORT DATE: 090790 NSSS: WE TYPE: PWR

(NSIC 219477) ON 8/9/90, IT WAS DISCOVERED BY OPERATIONS DEPARTMENT PERSONNEL THAT THE INSERVICE TESTING (IST) PROGRAM, PER TECH SPEC SURVEILLANCE 4.0.5, HAD NOT BEEN COMPLIED WITH. THE NOS. 21 COMPONENT COOLING (CC) PUMP, 23 CC PUMP AND 26 SERVICE WATER (SW) PUMP SURVEILLANCE FREQUENCY HAD NOT BEEN INCREASED AS PER THE IST PROGRAM RESULTING IN A SURVEILLANCE NOT BEING PERFORMED WITHIN THE REQUIRED TIME FRAME. WHEN A PUMP IS PLACED IN "ALERT", THE INTERVAL BETWEEN SURVEILLANCES IS HALVED. THIS WAS NOT DONE FOR THESE PUMPS AND CONSEQUENTLY, A SURVEILLANCE WAS MISSED. THE CC PUMPS WERE LAST SURVEILLED ON 5/6/90 THEREBY REQUIRING THE NEXT SURVEILLANCE TO BE COMPLETED ON OR BEFORE 8/3/90. THE ROOT

CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INADEQUATE ADMINISTRATIVE CONTROL. CONTRIBUTING TO THE CAUSE OF THE SW PUMP MISSED SURVEILLANCE WAS PERSONNEL ERROR. A SURVEILLANCE FREQUENCY CHANGE FORM HAD NOT BEEN INITIATED FOR THE NO. 26 SW PUMP AFTER ITS 6/7/90 SURVEILLANCE. THE RECURRING TASKS FOR THE CC PUMPS AND THE SW PUMP WERE CORRECTED AS REQUIRED. THE NOS. 21 AND 23 CC PUMPS WERE SUCCESSFULLY SURVEILLED ON 8/9/90 AND 8/11/90. REVIEW OF THE ADMINISTRATIVE CONTROLS, WHICH ENSURE COMPLIANCE WITH SURVEILLANCE FREQUENCY CHANGES, HAS BEEN INITIATED. UNTIL COMPLETION OF THIS REVIEW AND IMPLEMENTATION OF PROGRAMMATIC MODIFICATIONS, INTERIM PROCEDURAL MODIFICATIONS HAVE BEEN MADE.

[169] SALEM 2 DOCKET 50-311 LER 90-031
MS18 VALVE MAIN STEAMLINE ISOLATION CONTROL CIRCUIT DESIGN CONCERNS DUE TO PERSONNEL ERROR.
EVENT DATE: 080290 REPORT DATE: 090490 NSSS: WE TYPE: PWR
VENDOR: MARLO CORP.

(NSIC 219469) ON 7/9/90, DURING STARTUP OPERATIONS, MANUAL MAIN STEAMLINE ISOLATION INITIATION TESTING WAS INITIATED (PER PROCEDURE SP(O)4.3.2.1D, "ENGINEERED SAFETY FEATURES - STEAM LINE ISOLATION"). WHEN MANUAL PUSHBUTTONS FOR THE SOLID STATE PROTECTION SYSTEM (SSPS) MAIN STEAMLINE ISOLATION WERE DEPRESSED 21MS18, 23MS18 AND 24MS18 VALVES (MAIN STEAMLINE BYPASS VALVES) DID NOT CLOSE. TECH SPEC ACTION STATEMENT 3.6.3.A WAS ENTERED. ON 7/9/90, THE THREE (3) MS18 VALVES WERE CLEARED AND TAGGED CLOSED AND TECH SPEC ACTION STATEMENT 3.6.3.A WAS EXITED AND ACTION 3.6.3.B WAS ENTERED. MAINTENANCE-I&C TROUBLESHOOTING DID NOT IDENTIFY ANY CIRCUIT FUNCTIONAL PROBLEMS. THE THREE (3) MS18 VALVES WERE DECLARED OPERABLE AND TECH SPEC 3.6.3 ACTION STATEMENT[illegible]WERE EXITED. ON 8/2/90, A CONTINUING INVESTIGATION IDENTIFIED THAT THE [illegible] CIRCUITRY WAS NOT WIRED AS PER DESIGN CAUSING THE 21MS18 VALVE CONTROL CIRCUIT [illegible] COIL TO BE MAINTAINED ENERGIZED FOR APPROX. 1 SEC. AFTER INITIATION OF A[illegible] MAIN STEAMLINE ISOLATION SIGNAL (MANUAL OR AUTOMATIC). INVESTIGATION REVEALED THAT A REVISION TO A DESIGN CHANGE (DCP NO. 2EC-0904) HAD NOT BEEN IMPLEMENTED CIRCA 1980. SPECIFIC DETAILS OF WHY DCP REVISION WAS NOT IMPLEMENTED COULD NOT BE DETERMINED; HOWEVER, THE CAUSES HAVE BEEN ATTRIBUTED TO PERSONNEL ERROR.

[170] SALEM 2 DOCKET 50-311 LER 90-033
CONTAINMENT VENTILATION ISOLATION DUE TO DESIGN CONCERNS.
EVENT DATE: 080990 REPORT DATE: 090690 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 219470) THIS LER ADDRESSES CONTAINMENT PURGE/PRESSURE-VACUUM RELIEF SYSTEM (CP/P-VRS) ISOLATION SIGNALS (DUE TO CHANNEL SPIKES) FROM THE 2R41C PLANT VENT RADIOACTIVE NOBLE GAS MONITOR (8/9/90) AND THE 2R11A CONTAINMENT PARTICULATE RADIATION MONITOR (8/17, 24, & 29/90) RADIATION MONITORING SYSTEM (RMS) CHANNELS. AT THE TIME OF THE SIGNALS, THE VALVES ASSOCIATED WITH CP/P-VRS WERE CLOSED. THE APPROPRIATE TECH SPEC ACTION STATEMENT REQUIREMENTS WERE COMPLIED WITH IN ALL CASES. INVESTIGATION OF THE FIRST 2R11A CHANNEL ESF ACTUATION SIGNAL DID NOT IDENTIFY ANY FAILED COMPONENTS; ALTHOUGH, THE CHANNEL MICROPROCESSOR WAS FOUND "LOCKED UP". SUBSEQUENTLY, THE MEMORY WAS CLEARED, THE SETPOINTS REENTERED AND A CHANNEL FUNCTIONAL TEST WAS SUCCESSFULLY COMPLETED. ON 8/24/90, THE SECOND ESF SIGNAL OCCURRED. THE THIRD ACTUATION OCCURRED DURING FUNCTIONAL TESTING OF THE 2R11A CHANNEL. THE ROOT CAUSE OF THE CP/P-VRS ACTUATIONS IS ATTRIBUTED TO DESIGN/EQUIPMENT. AS INDICATED IN PRIOR LERS, THE SALEM UNIT 2 RMS IS PRONE TO VOLTAGE TRANSIENTS. INVESTIGATION OF THE 2R41A CHANNEL ESF ACTUATION DID NOT IDENTIFY ANY FAILED COMPONENTS. INVESTIGATION OF THE SUBSEQUENT 2R11A CHANNEL ESF ACTUATIONS IDENTIFIED SEVERAL FAILED COMPONENTS INCLUDING A DAMAGED BACKPLANE PIN AND A BROKEN FOIL ON THE CIRCUIT BOARD. ON 8/9/90, A 2R41C CHANNEL FUNCTIONAL TEST WAS SUCCESSFULLY COMPLETED AND THE CHANNEL DECLARED OPERABLE.

[171] SALEM 2 DOCKET 50-311 LER 90-034
PARTIAL MAIN STEAMLINE ISOLATION DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 081990 REPORT DATE: 091890 NSSS: WE TYPE: PWR

(NSIC 219471) ON 8/19/90 AT 0850 HOURS, DURING STARTUP OPERATIONS, A PARTIAL MAIN

STEAMLINE ISOLATION OCCURRED. MAIN STEAMLINE ISOLATION IS AN ENGINEERED SAFETY FEATURE (ESF). AT THE TIME OF THE EVENT, THE SURVEILLANCE PROCEDURE FOR DEMONSTRATING THE OPERABILITY OF THE SOLID STATE PROTECTION SYSTEM (SSPS) SLAVE RELAYS WAS IN PROGRESS. SPECIFICALLY, TESTING OF THE MS18 VALVE (MAIN STEAMLINE BYPASS VALVE) AND THE MS7 VALVE (MAIN STEAM DRAIN VALVE) SLAVE RELAYS WAS BEING PERFORMED. THE 21 - 24MS167 VALVES (MAIN STEAM ISOLATION VALVES) ISOLATION FUNCTION WAS "BLOCKED" PRIOR TO INITIATING THE TEST. WHEN THE TEST SWITCH (TS623) WAS PLACED TO THE "OPERATE OUTPUT" POSITION, THE 23 AND 24 MS7 AND MS18 VALVES CLOSED AND NO. 23 AND 24 MAIN STEAMLINE ISOLATION (TRAINS A AND B) SIGNALS WERE RECEIVED AS REQUIRED; HOWEVER, THE 24MS167 VALVE ALSO CLOSED. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO PROCEDURAL INADEQUACY. THE PROCEDURE DID NOT DETAIL SPECIFIC GUIDANCE TO THE OPERATOR TO ENSURE THAT TEST EQUIPMENT USED BY THE MAINTENANCE-I&C TECHNICIAN WAS DISCONNECTED PRIOR TO TURNING THE TS623 SWITCH. THE TECHNICIAN'S MULTIMETER WAS NOT DISCONNECTED WHEN THE SWITCH POSITION WAS CHANGED. A CONTRIBUTING FACTOR TO THIS EVENT WAS INADEQUATE COMMUNICATIONS. THE OPERATOR DID NOT INFORM THE I&C TECHNICIAN PRIOR TO HIS TURNING THE TS623 SWITCH.

[172] SALEM 2 DOCKET 50-311 LER 90-035
TECHNICAL SPECIFICATION SURVEILLANCE NONCOMPLIANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 082490 REPORT DATE: 091890 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 1 (PWR)

(NSIC 219555) ON 8/24/90, IT WAS DISCOVERED THAT THE TECH SPEC SURVEILLANCE TABLE 4.3-1, "REACTOR TRIP SYSTEM INSTRUMENTATION", REQUIREMENTS FOR "SAFETY INJECTION INPUT FROM SSPS" (FUNCTIONAL UNIT #19) IS NOT FULLY PERFORMED AT A FREQUENCY OF ONCE EVERY 31 DAYS. THE SURVEILLANCE WAS BEING PERFORMED EVERY 62 DAYS ON A STAGGERED BASIS. THIS SURVEILLANCE NONCOMPLIANCE WAS DISCOVERED BY THE TECH SPEC SURVEILLANCE AUDIT PROJECT WHICH HAD BEEN INITIATED AS A CORRECTIVE ACTION TO THE SALEM UNIT 2 LER 311/89-015-00. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO PERSONNEL ERROR. THE TECH SPEC SURVEILLANCE TEST FREQUENCY HAS HISTORICALLY BEEN PERFORMED EVERY 62 DAYS ON A STAGGERED TEST BASIS. THE RECURRING TASKS, FOR COMPLETION OF THE SUBJECT SURVEILLANCES, HAVE BEEN CORRECTED (BOTH UNITS) TO REQUIRE PERFORMANCE OF THE SURVEILLANCE EVERY 31 DAYS. THE VERIFICATION AUDIT OF THE UNIT 1 AND UNIT 2 TECH SPEC SURVEILLANCES ENSURING ALL REQUIREMENTS ARE MET IS CONTINUING. THE AUDIT INCLUDES IDENTIFICATION OF ALL REQUIRED SURVEILLANCES. THOSE SURVEILLANCES WITH PERFORMANCE FREQUENCIES GREATER THAN 7 DAYS WILL BE VERIFIED TO HAVE SPECIFIC RECURRING TASKS AND THAT THOSE TASKS CONTAIN THE CORRECT RELEVANT INFORMATION. ADDITIONALLY, THE PROCEDURE UPGRADE PROJECT IS TASKED TO CONDUCT A THOROUGH REVIEW AND TECHNICAL VERIFICATION OF PROCEDURES IN USE.

[173] SAN ONOFRE 1 DOCKET 50-206 LER 90-016
SUSCEPTIBILITIES TO EMERGENCY CORE COOLING SYSTEM SINGLE FAILURES DUE TO SINGLE FAILURE ANALYSIS DEFICIENCIES.
EVENT DATE: 072790 REPORT DATE: 082490 NSSS: WE TYPE: PWR

(NSIC 219395) IN A LETTER TO NRC REGION V DATED 3/17/89, ENTITLED "TECHNICAL ISSUES IMPACTING SAN ONOFRE UNIT 1 RESTART", SCE COMMITTED TO A REANALYSIS OF THE 1976 SINGLE FAILURE ANALYSIS (SFA) PERFORMED ON THE SAN ONOFRE UNIT 1 EMERGENCY CORE COOLING SYSTEM (ECCS) AND SUPPORTING SYSTEMS. AT 1330 ON 7/27/90, WITH UNIT 1 SHUT DOWN AND DEFUELED, SINGLE FAILURE SUSCEPTIBILITIES WHICH COULD HAVE POTENTIALLY IMPACTED THE PERFORMANCE OF SOME ECCS FUNCTIONS WERE CONFIRMED IN THIS REANALYSIS EFFORT. THESE FINDINGS WERE PROMPTLY REPORTED TO NRC. IN THE PRESENT SHUTDOWN CONDITION, THESE POTENTIAL SUSCEPTIBILITIES HAVE NO SAFETY SIGNIFICANCE. DETAILS OF THE 1990 SFA, INCLUDING METHODOLOGY AND CRITERIA USED, ALONG WITH PROPOSED CORRECTIVE ACTIONS FOR ALL IDENTIFIED DEFICIENCIES WERE PROVIDED TO THE NRC IN AN INTERIM REPORT DATED 7/31/90, "ECCS SINGLE FAILURE ANALYSIS, SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1." THE REPORT ALSO IDENTIFIES OTHER SINGLE FAILURE ISSUES WHICH REMAIN UNDER REVIEW AND NOTES THAT AN EFFORT IS UNDERWAY TO ENSURE THE STUDY ASSUMPTIONS CORRECTLY REFLECT THE AS-BUILT CONDITION OF THE PLANT. THE 1976 ASSESSMENT WAS LIMITED IN SCOPE AND WAS PRODUCED IN THE SAME TIME PERIOD AS WERE THE RELEVANT GUIDANCE DOCUMENTS

(IEEE 279-1971 AND ANSI N658-1976). ADDITIONAL REGULATORY GUIDANCE AND INTERPRETATION HAS BEEN PROVIDED FOR SINGLE FAILURE ASSESSMENTS.

[174] SAN ONOFRE 3 DOCKET 50-362 LER 90-011
AUXILIARY FEEDWATER VALVE INOPERABLE FOR MAIN STEAM ISOLATION RESULTING IN A TECHNICAL SPECIFICATION 3.0.3 ENTRY.
EVENT DATE: 072290 REPORT DATE: 082190 NSSS: CE TYPE: PWR
VENDOR: PAUL-MUNROE HYDAULICS INC.

(NSIC 219363) AT 0315 ON 7/22/90, WITH UNIT 3 IN MODE 2, WHILE FILLING THE STEAM GENERATORS, AUX. FEEDWATER (AFW) BYPASS CONTROL VALVE 3HV-4763 FAILED TO CLOSE UPON DEMAND. PER TS 3.3.2 THIS VALVE HAS A MAIN STEAM ISOLATION SIGNAL (MSIS) RESPONSE TIME REQUIREMENT TO CLOSE. AS THE VALVE WAS NOT CAPABLE OF AUTOMATIC CLOSURE BY A MSIS SIGNAL WITHIN THE MINIMUM RESPONSE TIME REQUIRED BY TS 3.3.2, IT WAS DECLARED INOPERABLE. SINCE THERE ARE NO TS ACTION STATEMENTS WHICH ADDRESS THE CONDITION WHERE AN AFW VALVE CAN NOT CLOSE ON A MSIS SIGNAL, TS 3.0.3 WAS INVOKED. AT APPROX. 0325, TS 3.0.3 WAS EXITED WHEN OPERATORS MANUALLY CLOSED THE VALVE. CAUSE OF THIS EVENT WAS INABILITY OF 3HV-4763 TO MEET ITS MSIS RESPONSE TIME DUE TO A FAILED SOLENOID VALVE. ROOT CAUSE OF THE SOLENOID VALVE FAILURE IS UNKNOWN AT THIS TIME AND UNDER INVESTIGATION. ACTUATOR HAS BEEN SENT TO THE MANUFACTURER FOR AN OVERHAUL WHICH WILL INCLUDE REPLACEMENT OF THE DEFECTIVE SOLENOID VALVE. A CONTRIBUTING CAUSE OF THIS EVENT IS THAT EXISTING TSS DO NOT INCLUDE A LIMITING CONDITION FOR OPERATION AND ACCOMPANYING ACTION STATEMENT APPLICABLE TO THIS COMPONENT. IT IS THEREFORE NECESSARY TO INVOKE TS 3.0.3 WHEN A RESPONSE TIME REQUIREMENT IS NOT MET. IN ADDITION, A TS AMENDMENT REQUEST WILL BE SUBMITTED TO PROVIDE AN APPROPRIATE ACTION STATEMENT WHICH WILL PRECLUDE SIMILAR ENTRIES INTO TS 3.0.3.

[175] SAN ONOFRE 3 DOCKET 50-362 LER 90-012
MISSED AC OFFSITE POWER SOURCES VERIFICATION DUE TO PERSONNEL ERROR.
EVENT DATE: 080390 REPORT DATE: 090490 NSSS: CE TYPE: PWR

(NSIC 219460) AT 0200 ON AUGUST 1, 1990, WITH UNIT 3 AT 100% POWER, THE TRAIN "A" DIESEL GENERATOR (DG) 3G002 WAS REMOVED FROM SERVICE FOR ROUTINE MAINTENANCE. IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS (TS) 3.8.1.1, ACTION STATEMENT A2, THE OPERABILITY OF THE REMAINING AC OFFSITE SOURCES WAS REQUIRED TO BE VERIFIED BY PERFORMING SURVEILLANCE REQUIREMENT 4.8.1.1.A WITHIN 1 HOUR AND EVERY 8-HOURS THEREAFTER. AT 1840 ON AUGUST 3, 1990, IT WAS IDENTIFIED THAT THE 8-HOUR AC OFFSITE SOURCES VERIFICATION DUE AT 1524, HAD NOT BEEN PERFORMED. THE SURVEILLANCE WAS IMMEDIATELY INITIATED AND SATISFACTORILY COMPLETED AT 1917. THE CAUSE OF THIS EVENT IS COGNITIVE PERSONNEL ERROR. THE OPERATORS WERE AWARE THAT THE SURVEILLANCE WAS DUE AT 1524; HOWEVER, THEY FAILED TO ENSURE THAT THE SURVEILLANCE WAS COMPLETED WITHIN THE TIME LIMITS REQUIRED BY TS 3.8.1.1. FOR CORRECTIVE ACTIONS: 1) APPROPRIATE DISCIPLINARY ACTION HAS BEEN ADMINISTERED TO THE PERSONNEL INVOLVED IN THIS EVENT, 2) THE EVENT WILL BE DISCUSSED WITH APPROPRIATE OPERATIONS PERSONNEL, 3) THE USE OF SIGNAGE OR ADDITIONAL ENHANCEMENTS AS A FURTHER MEANS FOR PROMPTING THE OPERATORS TO PERFORM NONROUTINE ACTION STATEMENT SURVEILLANCES WILL BE EVALUATED, AND 4) A HUMAN PERFORMANCE ENHANCEMENT SYSTEM (HPES) EVALUATION OF THIS EVENT WILL BE PERFORMED.

[176] SEABROOK 1 DOCKET 50-443 LER 90-020
NONCOMPLIANCE WITH TECHNICAL SPECIFICATION BECAUSE OF UNSECURED HIGH RADIATION AREA.
EVENT DATE: 080190 REPORT DATE: 083090 NSSS: WE TYPE: PWR

(NSIC 219447) ON 8/1/90 AT 7:18 A.M. EDT, WHILE IN MODE 1, IT WAS DISCOVERED THAT A DOOR LEADING TO A HIGH RADIATION AREA, DEMIN ALLEY, WAS UNSECURED AND HAD BEEN UNSECURED FOR APPROXIMATELY TWENTY-ONE HOURS AND THIRTY MINUTES. THIS OCCURRED CONTRARY TO SEABROOK STATION TECH SPEC 6.11.2. ON 7/31/90 AT 9:48 A.M., A HEALTH PHYSICS TECHNICIAN ENTERED THE DEMIN ALLEY THROUGH DOOR P303. UPON CLOSURE, HOWEVER, THE DOOR LATCH DID NOT COMPLETELY ENGAGE AND THE DOOR WAS CONSEQUENTLY LEFT SLIGHTLY AJAR. THE INTRUSION ALARM TO ALERT PERSONNEL OF THIS CONDITION FAILED TO OPERATE DUE TO DOOR ALARM MISALIGNMENT. ADDITIONALLY, AT 4:50 P.M. A

DAILY SURVEILLANCE WAS CONDUCTED ON THE DOOR TO ENSURE THE DOOR WAS PROPERLY SECURED BUT FAILED TO IDENTIFY THE CONDITION. THE ROOT CAUSE HAS BEEN ATTRIBUTED TO PERSONNEL ERROR INVOLVING A LACK OF ATTENTION TO DETAIL. CORRECTIVE ACTIONS INCLUDE ADJUSTMENTS TO THE INTRUSION ALARM SWITCH, DOOR LATCH AND CLOSURE MECHANISM. FUNCTIONAL TESTS OF ALL LOCKED HIGH RADIATION AREA DOORS, OUTSIDE OF CONTAINMENT, WERE PERFORMED TO ENSURE PROPER OPERATION. THE PERSONNEL INVOLVED IN THE EVENT WERE COUNSELLED REGARDING THE SERIOUSNESS OF THE EVENT AND THE NEED FOR INCREASED ATTENTION TO DETAIL. A MEETING WAS HELD WITH HEALTH PHYSICS OPERATIONS TECHNICIANS TO DISCUSS THE EVENT.

[177] SEABROOK 1 DOCKET 50-443 LER 90-021
GROUP "A" PRESSURIZER BACKUP HEATERS NOT OPERABLE DURING LOSS OF OFF-SITE POWER.
EVENT DATE: 080190 REPORT DATE: 083090 NSSS: WE TYPE: PWR

(NSIC 219441) ON AUGUST 1, 1990 AT 10:00 A.M., WHILE IN MODE 3, DURING STARTUP TEST PROCEDURE ST-39, "LOSS OF OFF-SITE POWER", IT WAS DISCOVERED THAT THE GROUP "A" PRESSURIZER BACKUP HEATERS COULD NOT BE MANUALLY ENERGIZED FROM THE MAIN CONTROL BOARD. INVESTIGATION SHOWED THAT THESE HEATERS HAD BEEN MISWIRED SINCE BEFORE THE PLANT'S FIRST ENTRY INTO MODE 3 IN 1987. THESE HEATERS ARE REQUIRED TO BE OPERABLE IN MODES 1, 2 AND 3 BY SEABROOK STATION TECHNICAL SPECIFICATION 3/4.4.3. THIS CONDITION WAS INITIALLY DISCOVERED IN 1986 DURING A PREOPERATIONAL TEST. HOWEVER, THE INVESTIGATION AND CORRECTIVE ACTIONS DID NOT FIND OR CORRECT THE ACTUAL PROBLEM. FINAL CORRECTIVE ACTIONS FOR THE AUGUST 1, 1990 EVENT, INCLUDED A DESIGN CHANGE TO CORRECT THE WIRING PROBLEM AND ADEQUATE FUNCTIONAL CHECKS TO ENSURE THE CONDITION WAS CORRECTED. ALSO A REVIEW OF TEST EXCEPTIONS AND A DESIGN REVIEW OF THE WIRING OF SIMILAR LOADS WILL BE CONDUCTED.

[178] SEQUOYAH 1 DOCKET 50-327 LER 90-013 REV 01
UPDATE OF MAIN CONTROL ROOM FIRE DETECTORS INOPERABLE AS A RESULT OF INADEQUATE DESIGN REVIEW.
EVENT DATE: 060190 REPORT DATE: 091090 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 219451) THIS LER IS BEING REVISED TO PROVIDE ADDITIONAL CORRECTIVE ACTIONS FOR THE EVENT. ON JUNE 1, 1990, AT 0930 EASTERN DAYLIGHT TIME WITH UNIT 1 IN MODE 2, AND UNIT 2 IN MODE 1, IT WAS DISCOVERED THAT THE REMOVAL OF THE LIGHTING DIFFUSER PANELS IN THE MAIN CONTROL ROOM AFFECTED THE AIR FLOW PATTERN SUCH THAT THE IONIZATION-TYPE SMOKE DETECTORS ARE EFFECTIVELY INOPERABLE. LIMITING CONDITION FOR OPERATION (LCO) 3.3.3.8 WAS ENTERED AT 0930 EDT ON JUNE 1, 1990, AND AN HOURLY FIRE WATCH WAS ESTABLISHED. THE LIGHTING DIFFUSERS WERE REMOVED IN DECEMBER 1989 TO BE CLEANED AND A TEMPORARY ALTERATION CHANGE FORM (TACF) WAS INITIATED TO LEAVE THEM OUT BECAUSE OF THE IMPROVED LIGHTING EFFECT. A PERMANENT DESIGN CHANGE WAS REQUESTED IN JANUARY 1990, WHICH RESULTED IN DESIGN CHANGE NOTICE 2178 BEING ISSUED ON MARCH 23, 1990. THE ROOT CAUSE OF THE EVENT IS INADEQUATE DOCUMENTATION OF INITIAL DETECTOR DESIGN ASSUMPTIONS AND INADEQUATE DISCIPLINE DESIGN REVIEW WHEN THE DECISION WAS MADE TO REMOVE THE DIFFUSER PERMANENTLY. CORRECTIVE ACTIONS INCLUDED CLARIFICATION OF DISCIPLINE FIRE PROTECTION REVIEW RESPONSIBILITIES AND WILL INCLUDE INSTALLATION OF CLEAR PLASTIC PANELS IN THE MAIN CONTROL ROOM BY FEBRUARY 15, 1991.

[179] SEQUOYAH 1 DOCKET 50-327 LER 90-015
LIMITING CONDITION FOR OPERATION 3.0.3 ENTERED WHEN TWO PRESSURIZER PRESSURE CHANNELS WERE DECLARED INOPERABLE WHILE VENTING A COMMON CONDENSATE POT FOR MAINTENANCE.
EVENT DATE: 081490 REPORT DATE: 091090 NSSS: WE TYPE: PWR

(NSIC 219452) ON AUGUST 14, 1990, AT 0006 EASTERN DAYLIGHT TIME (EDT) WITH UNIT 1 IN MODE 1 (100 PERCENT POWER), LIMITING CONDITION FOR OPERATION (LCO) 3.0.3 WAS ENTERED WHEN PRESSURIZER PRESSURE TRANSMITTERS 1-PT-68-322 AND 1-PT-68-323 WERE DECLARED INOPERABLE TO PERFORM MAINTENANCE ACTIVITIES FOR PRESSURIZER LEVEL TRANSMITTER 1-LT-68-320, WHICH HAD DRIFTED HIGH. THIS ACTIVITY INVOLVED VENTING A COMMON CONDENSATE POT RESULTING IN THE PRESSURE CHANNELS BEING DECLARED INOPERABLE. AT 0008 EDT THE VENTING WAS COMPLETED, AND LCO 3.0.3 WAS EXITED.

[180] SEQUOYAH 2 DOCKET 50-328 LER 90-011
CONTAINMENT ISOLATION VALVES WERE RETURNED TO SERVICE WITHOUT A PROPER POSTMAINTENANCE TEST BECAUSE OF PROCEDURAL INADEQUACIES.
EVENT DATE: 072890 REPORT DATE: 082790 NSSS: WE TYPE: PWR

(NSIC 219356) ON JULY 28, 1990, AT 1625 EASTERN DAYLIGHT TIME (EDT) WITH UNIT 2 IN MODE 1 (94 PERCENT POWER, 2,240 POUNDS PER SQUARE INCH GAUGE, AND 576 DEGREES FAHRENHEIT), IT WAS DETERMINED THAT AFTER MAINTENANCE HAD BEEN PERFORMED ON THE POWER CIRCUITRY OF CONTAINMENT ISOLATION VALVE (CIV) 2-FCV-67-131, THE VALVE HAD BEEN RETURNED TO SERVICE WITHOUT A PROPER VERIFICATION OF ISOLATION TIME. THE SHIFT OPERATIONS SUPERVISOR WAS NOTIFIED AND LIMITING CONDITION FOR OPERATIONS (LCO) 3.0.3 WAS ENTERED. THE VALVE WAS APPROPRIATELY STROKE-TIME TESTED IN ACCORDANCE WITH SURVEILLANCE INSTRUCTION 166.6 AND RETURNED TO SERVICE. AT 1642 EDT, LCO 3.0.3 WAS EXITED. THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE PROCEDURES. CORRECTIVE ACTION INCLUDED A REVIEW OF WORK REQUESTS (WRS) DETAILING SIMILAR REPLACEMENT MAINTENANCE ON CIVS REQUIRING POSTMAINTENANCE TESTING. OTHER WRS THAT DID NOT REQUIRE AN ADEQUATE POSTMAINTENANCE TEST FOR THE CIVS SUBSEQUENTLY RECEIVED THE APPROPRIATE STROKE-TIME TEST. CORRECTIVE ACTION INCLUDES REPLANNING OF ASSOCIATED WRS AND REVISION OF AFFECTED PROCEDURES.

[181] SHEARON HARRIS 1 DOCKET 50-400 LER 90-010 REV 01
UPDATE ON ADMINISTRATIVE CONTROL OF REACTOR AUXILIARY BUILDING EMERGENCY EXHAUST PRESSURE BOUNDARY NOT IMPLEMENTED.
EVENT DATE: 040590 REPORT DATE: 090590 NSSS: WE TYPE: PWR

(NSIC 219523) ON 2/7/90, DURING INVESTIGATION OF A SAFETY-RELATED DAMPER PROBLEM IN THE REACTOR AUX. BLDG. EMERGENCY EXHAUST SYSTEM (RABEES), CONCERNS WERE IDENTIFIED WITH DECLARING THE SYSTEM INOPERABLE DUE TO ISOLATION DAMPER FAILURES, WHEN DOORS AND HATCHES WHICH PROVIDE A SIMILAR ISOLATION FUNCTION TO THE VENTILATED ROOMS WERE NOT CONTROLLED, AND COULD BE BLOCKED OPEN FOR MAINTENANCE ACTIVITIES. FURTHER EVALUATION IDENTIFIED COMMITMENTS IN THE FSAR FOR ADMINISTRATIVE CONTROL OF THOSE DOORS WHICH ACCESS THE EXHAUSTED ROOMS. WHILE THESE DOORS ARE SUBJECT TO CERTAIN ADMINISTRATIVE CONTROLS IN ACCORDANCE WITH THE FIRE PROTECTION PROGRAM, SPECIFIC CONTROLS FOR RABEES PRESSURE BOUNDARY WERE NOT ESTABLISHED. ADMINISTRATIVE CONTROL WAS ESTABLISHED ON AFFECTED DOORS ON 2/21, WHILE AN ENGINEERING EVALUATION OF THE RABEES AND IMPACT OF OPEN DOORS AND HATCHES WAS CONDUCTED. ON 4/5, IT WAS CONCLUDED THAT WITH OPEN DOORS AND HATCHES, THE RABEES MIGHT NOT BE CAPABLE OF PERFORMING ITS DESIGN FUNCTION TO MAINTAIN A -1/8 INWG PRESSURE IN EXHAUSTED ROOMS, WITH THE REQUIRED AIR FLOW RATES OF THE FILTER BANKS. FURTHER TESTING IS PLANNED TO DETERMINE THE EFFECTS OF OPEN DOORS AND HATCHES ON THE SYSTEM OPERABILITY. IT IS NOT KNOWN WHEN PRIOR CONDITIONS OF OPEN DOORS HAS ACTUALLY OCCURRED, ALTHOUGH TWO HATCHES TO A NON-EXHAUSTED AREA HAD BEEN ROUTINELY OPEN DURING OPERATION.

[182] SHOREHAM DOCKET 50-322 LER 90-005
FAILURES OF PRIMARY CONTAINMENT ELECTRICAL PENETRATION ASSEMBLY MODULES.
EVENT DATE: 041890 REPORT DATE: 082990 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 219427) SEVEN TIMES OVER THE COURSE OF THE LAST ELEVEN YEARS, LILCO HAS EXPERIENCED COMPONENT INOPERABILITY DUE TO THE LOSS OR REDUCTION OF INSULATION RESISTANCE OR LOSS OF CONTINUITY FOR INDIVIDUAL CONDUCTORS IN EIGHT DIFFERENT MODULES CONTAINED IN SIX PRIMARY CONTAINMENT ELECTRICAL PENETRATION ASSEMBLIES. INDIVIDUAL CONDUCTOR FAILURES OCCURRED ON 12/5/79, 8/8/83 AND 12/18/85 AND WERE CORRECTED BY UTILIZING SPARE CONDUCTORS IN THE SAME MODULE. FAILURES ALSO OCCURRED ON 12/1/86, 7/3/87, 1/19/89, AND 5/10/89 AND THESE WERE CONSIDERED "TOTAL MODULE FAILURES" BECAUSE NOT ENOUGH SPARE CONDUCTORS WERE AVAILABLE IN THE MODULE TO CONTINUE TO UTILIZE THE MODULE. IT IS OUR DETERMINATION THAT SINCE COLLECTIVELY THESE FAILURES REPRESENT AN ADVERSE PERFORMANCE TREND, OF SAFETY-RELATED EQUIPMENT, LILCO IS HEREBY NOTIFYING THE NRC STAFF. CORRECTIVE ACTIONS INCLUDED PERFORMANCE TRENDING OF TWO MODULES IN TWO DIFFERENT ELECTRICAL PENETRATION ASSEMBLIES. THIS PERFORMANCE TRENDING, HOWEVER, HAS NOW BEEN DEFERRED INDEFINITELY DUE TO THE NEW YORK STATE - LILCO SHOREHAM SETTLEMENT AGREEMENT.

[183] SOUTH TEXAS 1 DOCKET 50-498 LER 90-020
REACTOR TRIP CAUSED BY BOTH LOGIC TRAINS OF THE SOLID STATE PROTECTION SYSTEM BEING IN THE URGENT ALARM CONDITION.
EVENT DATE: 071690 REPORT DATE: 081690 NSSS: WE TYPE: PWR
VENDOR: GRAYHILL

(NSIC 219381) AT 0236, ON JULY 16, 1990 WITH UNIT 1 AT 100% POWER, A REACTOR TRIP OCCURRED DURING PERFORMANCE OF A SOLID STATE PROTECTION SYSTEM (SSPS) SURVEILLANCE TEST. THE TRIP WAS CAUSED BY A MALFUNCTION IN A TEST SWITCH WHICH RESULTED IN ONE TRAIN OF SSPS REMAINING IN THE URGENT ALARM CONDITION, COUPLED WITH COMPLETION OF A SUBSEQUENT PROCEDURAL STEP THAT PLACED THE OTHER SSPS TRAIN IN THE URGENT ALARM CONDITION. THE SSPS IS DESIGNED SUCH THAT IF BOTH LOGIC TRAINS ARE PLACED IN THE URGENT ALARM CONDITION, A REACTOR TRIP OCCURS. THE TEST SWITCH WILL BE REPLACED PRIOR TO STARTUP FROM THE NEXT REFUELING OUTAGE. IN ADDITION, TEST PROCEDURES HAVE BEEN REVISED TO ENSURE THAT URGENT ALARM CONDITIONS ARE SATISFACTORILY CLEARED PRIOR TO CONTINUING WITH PROCEDURAL STEPS THAT COULD RESULT IN A REACTOR TRIP.

[184] SOUTH TEXAS 1 DOCKET 50-498 LER 90-006
MANUAL REACTOR TRIP DUE TO FULL CLOSURE OF A FEEDWATER ISOLATION VALVE DURING PARTIAL STROKE TESTING.
EVENT DATE: 073090 REPORT DATE: 083190 NSSS: WE TYPE: PWR

(NSIC 219444) ON 7/30/90, UNIT 1 WAS IN MODE 1 AT 100% POWER. AT APPROXIMATELY 1946, FEEDWATER ISOLATION VALVE 1A FULLY CLOSED DURING A PARTIAL STROKE SURVEILLANCE TEST. THE RESULTANT LOSS OF FEEDWATER FLOW CAUSED A DECREASE IN STEAM GENERATOR LEVEL AND THE REACTOR WAS MANUALLY TRIPPED. THE UNIT WAS STABILIZED WITH THE EXCEPTION OF LEVEL IN STEAM GENERATOR 1A WHICH DID NOT RECOVER DUE TO A MISPOSITIONED RECIRCULATION TEST VALVE IN THE TRAIN A AUXILIARY FEEDWATER SYSTEM (AFW). THE RECIRCULATION TEST VALVE WAS RETURNED TO THE REQUIRED POSITION AND STEAM GENERATOR 1A LEVEL WAS RECOVERED. THE FEEDWATER ISOLATION VALVE CLOSURE WAS CAUSED BY A TECHNICIAN INADVERTENTLY CONTACTING THE WRONG TERMINAL WITH A TEST JUMPER. THE CAUSE OF THE MISPOSITIONED RECIRCULATION TEST VALVE COULD NOT BE CONCLUSIVELY ESTABLISHED; HOWEVER, IT IS LIKELY THAT THE VALVE WAS NOT CORRECTLY REPOSITIONED DURING A SURVEILLANCE TEST PRIOR TO THE EVENT, AND THIS ERROR WAS NOT DISCOVERED DUE TO A LACK OF ADEQUATE INDEPENDENT VERIFICATION. CORRECTIVE ACTIONS INCLUDE: ISSUANCE OF TRAINING BULLETINS CONCERNING USE OF JUMPERS; EVALUATION OF ALTERNATIVE DESIGNS TO OBVIATE THE NEED TO PERFORM THE PARTIAL STROKE TEST WITH JUMPERS; AND, ISSUANCE OF A MEMORANDUM TO OPERATIONS PERSONNEL TO REENFORCE THE REQUIREMENTS PERTAINING TO INDEPENDENT VERIFICATION.

[185] SURRY 1 DOCKET 50-280 LER 90-008
RCS LEAKAGE EXCEEDS 10 GALLONS PER MINUTE DUE TO GAGE SENSING LINE BREAK.
EVENT DATE: 072790 REPORT DATE: 082490 NSSS: WE TYPE: PWR

(NSIC 219384) ON JULY 27, 1990, AT 0530 HOURS, WITH UNIT 1 AT 99% REACTOR POWER, THE UNIT 1 REACTOR OPERATOR (RO) DETERMINED THAT REACTOR COOLANT SYSTEM (RCS) LEAKAGE WAS GREATER THAN 10 GPM AS INDICATED BY THE LEVEL DROP INDICATED ON THE VOLUME CONTROL TANK (VCT) LEVEL RECORDER. THIS WAS CONTRARY TO TECHNICAL SPECIFICATION 3.1.C.5 WHICH LIMITS THE AMOUNT OF TOTAL RCS LEAKAGE. THE SOURCE OF THE LEAK WAS A BROKEN GAGE SENSING LINE IN THE LETDOWN SYSTEM. THE SYSTEM WAS ISOLATED IN ACCORDANCE WITH PROCEDURES, AND NO RADIATION MONITOR ALARMS WERE RECEIVED. THE SENSING LINE HAS BEEN REPAIRED. APPROPRIATE FOLLOW-UP ACTIONS WILL BE INITIATED PENDING DETERMINATION OF THE FAILURE MECHANISM OF THE TUBING. THE FAILURE OF THE SENSING LINE OCCURRED AFTER FLOW WAS DIVERTED TO A PREVIOUSLY ISOLATED PORTION OF THE SYSTEM. THE PREVIOUS SHIFT HAD CLOSED MANUAL ISOLATION VALVES FOR THE UNIT'S DEBORATING DEMINERALIZERS (DEBORATOR) IN ACCORDANCE WITH AN OPTIONAL SECTION OF AN APPROVED PROCEDURE. THE RELIEF RO, DURING THE EVENT, ATTEMPTED TO PLACE THE DEBORATOR IN SERVICE USING THE DIVERT VALVE ONLY. OPERATING PROCEDURES WILL BE CLARIFIED TO ELIMINATE CONFUSION ON OPERATING PRACTICES.

[186] SURRY 1 DOCKET 50-280 LER 90-009
INOPERABLE CONTAINMENT VACUUM PUMP FLOW PATHS DUE TO FAILURES OF PROCESS VENT NORMAL RANGE RADIATION MONITOR.
EVENT DATE: 081090 REPORT DATE: 090690 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)
VENDOR: KAMAN SCIENCES CORP.

(NSIC 219463) ON 8/10/90 AT 1940 HOURS AND ON 8/29/90 AT 1830 HOURS, THE CONTAINMENT VACUUM SYSTEM FLOWPATHS FOR UNITS 1 AND 2 BECAME INOPERABLE WHEN THE CONTAINMENT VACUUM PUMP DISCHARGE VALVES FCV-GW-160 AND FCV-GW-260 (EIIS-FCV) AUTOMATICALLY CLOSED AS A RESULT OF THE FAILURE AND LOCKOUT OF THE PROCESS VENT NORMAL RANGE RADIATION MONITOR, RM-GW-130-1 (EIIS-IL). THIS IS CONTRARY TO TECH SPEC 3.15B WHICH REQUIRES THAT ONE MECHANICAL VACUUM PUMP AND ONE ASSOCIATED FLOW PATH BE OPERABLE. DURING THE AUGUST 10 EVENT, UNIT 1 WAS AT 94% POWER AND UNIT 2 WAS AT 100% POWER. DURING THE AUGUST 29 EVENT, UNIT 1 WAS AT 79% POWER AND UNIT 2 WAS AT 88% POWER. IN BOTH EVENTS, THE MONITOR'S HIGH ALARM SETPOINT FAILED TO ZERO, CAUSING A SPURIOUS HIGH ALARM AND CLOSURE OF THE CONTAINMENT VACUUM PUMP DISCHARGE VALVES AS DESIGNED. LOCKOUT OF THE OPERATOR KEYBOARD PREVENTED RESETTING THE ALARM AND OPENING OF THE VALVES. THE AUGUST 10 EVENT RESULTED FROM A DEFECTIVE FLOW SWITCH, WHILE THE AUGUST 29 EVENT WAS CAUSED BY A FAILED CPU CARD. THE FAILED COMPONENTS WERE REPLACED.

[187] SURRY 2 DOCKET 50-281 LER 89-019 REV 01
UPDATE ON SERVICE WATER MOTOR OPERATED VALVES TO THE RECIRCULATION SPRAY HEAT EXCHANGERS INOPERABLE DUE TO PERSONNEL ERROR IN REMOVING FLOOD PROTECTION.
EVENT DATE: 112289 REPORT DATE: 082490 NSSS: WE TYPE: PWR

(NSIC 219387) ON NOVEMBER 22, 1989 AT 1500 HOURS, UNIT 2 WAS AT HOT SHUTDOWN. DURING ROUTINE WALKDOWNS IN THE UNIT 2 TURBINE BUILDING, IN PREPARATION FOR A UNIT STARTUP, IT WAS NOTED THAT THE FLOOD PROTECTION DIKES HAD BEEN REMOVED FROM ONE SIDE OF THE VALVE PITS FOR THE SERVICE WATER (SW) SUPPLY MOTOR OPERATED VALVES (MOVS) TO THE RECIRCULATION SPRAY HEAT EXCHANGERS (RSHXS). WORK WAS IN PROGRESS NEAR THE VALVE PIT AREA FOR REPLACEMENT OF THE SW PIPING. THE AFFECTED VALVES, MOV-SW-203A/B/C/D, WERE DECLARED INOPERABLE AND A 30 HOUR ACTION STATEMENT TO COLD SHUTDOWN WAS ENTERED. A FOUR HOUR NON-EMERGENCY REPORT WAS MADE TO THE NUCLEAR REGULATORY COMMISSION (NRC) PURSUANT TO 10CFR50.72, PARAGRAPH (B)(2)(I) ON NOVEMBER 22, 1989 EVENT. THE DIKES WERE REPLACED ON NOVEMBER 22, 1989 AT 2053. ON DECEMBER 13, 1989 WITH UNITS 1 AND 2 AT 100% POWER, IT WAS DISCOVERED THAT BACKFLOW PREVENTERS WERE NOT IN PLACE IN THE TWO EMERGENCY SWITCHGEAR ROOM FLOOR DRAINS. A FLOOD WATCH WAS ALREADY IN PLACE AT THE TIME OF DISCOVERY OF THE MISSING DEVICES BECAUSE OF OTHER CONCERNS IDENTIFIED DURING ENGINEERING REVIEW. THE BACKFLOW PREVENTERS WERE REINSTALLED ON JANUARY 31, 1990.

[188] SUSQUEHANNA 1 DOCKET 50-387 LER 90-013
POSTULATED SINGLE FAILURE COULD PLACE THE PLANT IN CONDITION OUTSIDE DESIGN BASIS.
EVENT DATE: 072090 REPORT DATE: 081790 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)

(NSIC 219366) ON JULY 20, 1990, WITH BOTH UNIT 1 AND UNIT 2 OPERATING AT 100% POWER, EVALUATION OF AN ENGINEERING-IDENTIFIED POSTULATED ACCIDENT SCENARIO CONCLUDED THAT THE SCENARIO, IF ASSUMED TO OCCUR, COULD PLACE THE PLANT IN A CONDITION THAT WOULD BE OUTSIDE OF ITS ANALYZED DESIGN BASIS. THE SCENARIO INVOLVES A POSTULATED INDEPENDENT SINGLE FAILURE OF A 125 V DC BATTERY CHANNEL COINCIDENT WITH OR PRIOR TO A LOSS OF COOLANT ACCIDENT (LOCA) AND A LOSS OF OFFSITE POWER (LOOP). THE LOSS OF THE A (OR B) 125 V DC BATTERY CHANNEL WOULD LEAD TO THE LOSS OF CONTROL POWER TO ESSENTIAL LOADS ON ITS ASSOCIATED 4.16 KV BUS, AS PER THE DESIGN BASIS, BUT WOULD ALSO LEAD TO LOSS OF CONTROL POWER TO CERTAIN LOADS ON THE OTHER 4.16 KV BUS [C (OR D)]IN THE SAME DIVISION. THIS RESULTS IN FAILURE TO SHED THESE NON-ESSENTIAL LOADS ON THE RECEIPT OF A LOCA SIGNAL, RESULTING IN ADDITIONAL UNANALYZED LOADING ON THE ASSOCIATED EMERGENCY DIESEL GENERATOR [C (OR D)] WHICH COULD LEAD TO AN OVERLOADED CONDITION AND POSSIBLE LOSS OF THE DIESEL GENERATOR AND ASSOCIATED 4.16 KV BUS. THE FINAL CONDITION WOULD BE THE LOSS OF TWO 4.16 KV BUSES IN THE SAME DIVISION. THIS CONDITION COULD OCCUR IN EITHER UNIT 1 OR UNIT 2. THE CONDITION WAS IDENTIFIED

DURING CONTINUING EFFORTS TO RESOLVE SUSQUEHANNA'S DESIGN BASIS FOR DEGRADED GRID VOLTAGE CONDITIONS AND WAS PREVIOUSLY REPORTED PURSUANT TO 10CFR50.9 IN PLA-3395, DATED 6/7/90.

[189] SUSQUEHANNA 1 DOCKET 50-387 LER 90-016
EQUIPMENT EXCEEDED QUALIFIED LIFE PER ENVIRONMENTAL QUALIFICATION PROGRAM.
EVENT DATE: 072490 REPORT DATE: 082390 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)

(NSIC 219368) ON JUNE 29, 1990 PP&L RECEIVED INFORMATION THAT CALLED INTO QUESTION THE ENVIRONMENTAL QUALIFICATION OF FOUR NH90 HYDRAULIC ACTUATORS IN THE STANDBY GAS TREATMENT AND EMERGENCY SWITCHGEAR ROOM COOLING SYSTEMS. FURTHER EVALUATION AND VALIDATION OF THIS INFORMATION WAS BEGUN IN PARALLEL WITH PLANS TO REFURBISH THE EQUIPMENT IN QUESTION. THE PROBLEM WAS DUE TO AN INCORRECT TEMPERATURE ENTRY IN THE EQ INDEX DATABASE WHICH RESULTED IN PP&L'S FAILING TO RE-EVALUATE THIS EQUIPMENT WHEN THE POSTULATED POST-LOCA REACTOR BUILDING TEMPERATURE WERE REVISED IN 1989. THE PROBLEM WAS DISCOVERED WHILE PERFORMING WORK TO UPGRADE OUR EQ FILES. THIS CONDITION WAS DETERMINED TO BE REPORTABLE PER 10CFR50.73 (A)(2)(V). THERE WERE NO SIGNIFICANT SAFETY CONSEQUENCES OR COMPROMISES TO THE PUBLIC HEALTH OR SAFETY. APPROPRIATE PARTS WERE REPLACED WITH QUALIFIED MATERIAL. ALL EQUIPMENT IN THE EQ BINDERS (FILES) THAT COULD BE AFFECTED BY A SIMILAR ERROR WAS REVIEWED. EIGHT OTHER SIMILAR OCCURRENCES WERE IDENTIFIED. THE AFFECTED EQUIPMENT FOR THESE EIGHT OCCURRENCES WAS EVALUATED AND THE INFORMATION IN THE BINDERS SUPPORTED QUALIFICATION.

[190] SUSQUEHANNA 1 DOCKET 50-387 LER 90-015
TWO OUT OF FOUR EMERGENCY DIESELS DECLARED INOPERABLE.
EVENT DATE: 072690 REPORT DATE: 082490 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)
VENDOR: COOPER-BESSEMER CO.

(NSIC 219367) ON JULY 26, 1990 AT 0035 HOURS, WITH BOTH UNIT 1 AND UNIT 2 OPERATING AT 100% POWER, THE 'A' EMERGENCY DIESEL GENERATOR (EDG) WAS DECLARED INOPERABLE WHEN IT FAILED TO REACH REQUIRED FREQUENCY WITHIN 10 SECONDS OF A MANUAL INITIATION, WHICH WAS BEING PERFORMED PER TECHNICAL SPECIFICATION ACTION 3.8.1.1.B DUE TO THE 'E' EDG BEING INOPERABLE. THE 'E' EDG, WHICH WAS SUBSTITUTING FOR THE 'B' EDG, HAD BEEN DECLARED INOPERABLE AT 1300 HOURS ON 7/25/90 DUE TO THE FAILURE OF A FUEL SAMPLE FROM ITS FUEL OIL STORAGE TANK TO MEET THE INSOLUBLES LIMIT OF TECHNICAL SPECIFICATION 4.8.1.1.2.C. TWO OUT OF FOUR EDG'S INOPERABLE CONSTITUTES A REPORTING REQUIREMENT PURSUANT TO 10CFR50.73(A)(2)(V) AND (VI). THE CAUSE OF THE 'A' START TIME FAILURE WAS THE SHEARING OF A ONE-INCH DOUBLE-THREADED REDUCER ON THE RIGHT BANK STARTING AIR HEADER FILTER. THE FAILURE TO MEET THE FUEL OIL INSOLUBLES LIMIT WAS ATTRIBUTED TO LOSS OF STABILITY OF THE FUEL OIL IN THE 'E' STORAGE TANK. THE REDUCER WAS REPLACED AND THE 'A' EDG WAS SUCCESSFULLY RETESTED AND DECLARED OPERABLE. THE SHEARING OF THIS REDUCER IS CONSIDERED TO BE AN ISOLATED EVENT REQUIRING NO ADDITIONAL CORRECTIVE ACTIONS. THE 'E' EDG DAY AND STORAGE TANKS WERE EMPTIED AND CLEANED AND AN INSPECTION OF THE STORAGE TANK INTERIOR AND ITS SEALS WAS PERFORMED. NO ANOMALIES WERE FOUND AND THE SYSTEM WAS REFILLED WITH NEW, SAMPLED FUEL OIL AND RETURNED TO SERVICE.

[191] SUSQUEHANNA 1 DOCKET 50-387 LER 90-017
SECONDARY CONTAINMENT ISOLATION DIVISION I AUTOMATIC AND MANUAL INITIATION FUNCTIONS LOST DUE TO FAILED RELAY.
EVENT DATE: 080690 REPORT DATE: 090590 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219522) ON 8/6/90 AT APPROXIMATELY 0815, WITH UNIT 1 AND UNIT 2 OPERATING IN CONDITION 1 AT 60% AND 100% POWER RESPECTIVELY, ONE DIVISION OF SECONDARY CONTAINMENT ISOLATION WAS RENDERED INOPERABLE WHEN ITS LOGIC POWER SUPPLY FUSE BLEW DUE TO A FAILED RELAY IN THE CIRCUIT. THE FAILED RELAY WAS DETERMINED FROM THE CIRCUIT AND A NEW FUSE WAS INSTALLED. UPON RE-ENERGIZING THE LOGIC, RELAY

INTERACTION CAUSED, AS EXPECTED, THE ISOLATION LOGIC TO TRIP. AT 2255 HOURS THE LOGIC WAS RESET AND REACTOR BUILDING VENTILATION WAS RESTORED TO NORMAL. ON 8/08/90 THE RELAY WAS REPLACED. THE CAUSE OF THE FAILURE WAS A SHORT IN THE RELAY COIL. THIS FAILURE RESULTED IN HIGH CURRENT WHICH IN TURN CAUSED THE LOGIC POWER SUPPLY FUSE TO BLOW. THIS EVENT WAS DETERMINED TO BE REPORTABLE PER 10CFR50.73(A)(2)(IV), IN THAT AN UNPLANNED ESF ACTUATION OCCURRED. AN EVENT REVIEW TEAM HAS BEEN ESTABLISHED TO REVIEW AND ANALYZE THE EVENT. A SUPPLEMENT WILL BE PROVIDED REFLECTING THE RESULTS OF THE INVESTIGATION.

[192] TROJAN DOCKET 50-344 LER 90-020 REV 01
UPDATE ON PARTICULATE CHANNEL OF CONTAINMENT RADIATION MONITOR INOPERABLE DUE TO LOSS OF FILTER PAPER AND FAILURE OF FILTER PAPER ALARM.
EVENT DATE: 013090 REPORT DATE: 090590 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 219454) ON 1/30/90, THE TROJAN NUCLEAR PLANT WAS OPERATING AT 100% POWER. AT APPROXIMATELY 1855, DURING THE PERFORMANCE OF A PERIODIC OPERATING TEST ON PROCESS RADIATION MONITOR (PRM) 2, A CHECK OF PRM 1 WAS PERFORMED TO OBSERVE THE FILTER PAPER INSTALLATION. IT WAS DISCOVERED THAT THE FILTER PAPER IN THE RADIOACTIVE AIRBORNE PARTICULATE CHANNEL OF PRM 1 WAS DEPLETED. NO ALARMS HAD BEEN RECEIVED AT THE CONTROLLER FOR THIS CONDITION. THE MONITOR WAS DECLARED INOPERABLE AND A NEW SUPPLY OF FILTER PAPER WAS INSTALLED. AN ALARM WAS NOT RECEIVED DUE TO A BROKEN MICRO-SWITCH ON THE PAPER OUT ALARM AND A BROKEN WIRE ON THE PAPER TEAR ALARM. NO PROCEDURES EXISTED TO DETERMINE OPERABILITY OF THE ALARMS ON A PERIODIC BASIS. THE BROKEN MICRO-SWITCH HAS BEEN BYPASSED AND THE PAPER TEAR ALARM HAS BEEN RESTORED. A PERIODIC CHECK HAS BEEN IMPLEMENTED TO MONITOR THE FILTER PAPER SUPPLY. ON 2/16/90, DURING AN EVALUATION OF THE ABOVE EVENT IT WAS DETERMINED THAT THE FILTER PAPER SPEED WAS SET AT 2.5 INCHES PER HOUR INSTEAD OF THE REQUIRED 1 INCH PER HOUR. IMMEDIATE ACTIONS WERE TAKEN TO RESTORE THE FILTER PAPER SPEED TO 1 INCH PER HOUR. THE CAUSE WAS DUE TO INADEQUATE PROCEDURES FOR RESTORING POWER TO THE MONITOR CONTROLLER. THE OPERATING INSTRUCTION HAS BEEN REVISED TO INCLUDE PROVISIONS FOR RESETTING FILTER PAPER SPEED WHEN REENERGIZING THE MONITOR.

[193] TROJAN DOCKET 50-344 LER 90-006 REV 01
UPDATE ON BOTH TRAINS OF EMERGENCY CORE COOLING SYSTEM WERE INOPERABLE DURING MODE 3 SURVEILLANCE TESTING DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 022190 REPORT DATE: 082790 NSSS: WE TYPE: PWR

(NSIC 219433) ON FEBRUARY 21, 1990 THE PLANT WAS IN MODE 1, POWER OPERATION, WITH THE REACTOR COOLANT SYSTEM (RCS) AT 584 DEGREES F, AND 2239 PSIG. DURING A REVIEW OF CORRECTIVE ACTION FOR A NUCLEAR REGULATORY COMMISSION (NRC) OPEN ITEM, IT WAS NOTED THAT A REVISION OF A PERIODIC OPERATING TEST (POT), CAUSED BOTH TRAINS OF EMERGENCY CORE COOLING SYSTEM (ECCS) TO BE INOPERABLE WHEN SECTION 7.6 OF POT 2-4 WAS PERFORMED IN MODE 3, HOT STANDBY, ON JULY 24, 1989. THIS CONDITION VIOLATED TROJAN TECHNICAL SPECIFICATION (TTS) 3.5.2, ECCS SUBSYSTEM - TAVG 350 DEGREES F, AND CONSTITUTED AN UNINTENTIONAL ENTRY INTO TTS 3.0.3. THE INITIAL CONDITION FOR PERFORMANCE OF THE POT WAS RCS PRESSURE OF AT LEAST 1800 PSIG, WHICH CORRESPONDS TO MODE 3 OR MODE 4 ABOVE 260 DEGREES F. THE POT DID NOT DIFFERENTIATE BETWEEN THE TWO MODES WITH RESPECT TO TTS REQUIREMENTS FOR ECCS OPERABILITY. THE CAUSE OF THE OCCURRENCE WAS AN INADEQUATE PROCEDURE DUE TO INADEQUATE TECHNICAL AND SAFETY REVIEW OF THE REVISION. CORRECTIVE ACTIONS INCLUDE REVIEW AND REVISION OF THE POT TO PREVENT UNINTENTIONAL TTS 3.0.3 ENTRY. ALSO, A TASK FORCE WAS ASSEMBLED TO REVIEW OTHER POTS TO ENSURE ADDITIONAL SIMILAR CONDITIONS DID NOT EXIST WHICH WOULD ALLOW TESTING OF ECCS SYSTEMS TO CAUSE INADVERTENT ENTRIES INTO TTS 3.0.3.

[194] TROJAN DOCKET 50-344 LER 90-012 REV 01
UPDATE ON ESF ELECTRICAL SWITCHGEAR COULD EXPERIENCE COMMON MODE FAILURE FROM ELEVATED TEMPERATURES AS A RESULT OF ESF ROOM COOLER FAN DESIGN ERROR.
EVENT DATE: 040990 REPORT DATE: 082490 NSSS: WE TYPE: PWR

(NSIC 219359) DURING AN ANALYSIS OF EMERGENCY DIESEL GENERATOR LOADING SEQUENCE,

THE DESIGN ENGINEERING GROUP IDENTIFIED THAT MANUAL ACTIONS WERE NECESSARY TO START FANS THAT PROVIDE COOLING AIR FLOW IN THE ESF ELECTRICAL SWITCHGEAR ROOMS. AN EVALUATION DETERMINED THAT MANUAL ACTION WOULD BE REQUIRED IN A SHORT PERIOD OF TIME TO AVOID EXCEEDING THE ESF ELECTRICAL SWITCHGEAR ROOM DESIGN TEMPERATURE OF 104F, IF A LOSS OF OFF SITE POWER AND/OR SAFETY INJECTION SIGNAL OCCURRED. PERFORMANCE OF THIS MANUAL ACTION IS NOT EXPLICITLY SPECIFIED IN PLANT PROCEDURES. THIS CONDITION WAS APPARENTLY DUE TO A DESIGN ERROR, BY THE ARCHITECT-ENGINEER, IN THE ORIGINAL PLANT DESIGN. CORRECTIVE ACTION WAS TO CHANGE THE DESIGN OF THE ROOM COOLING FANS TO AUTOMATICALLY RESTART WHEN POWER IS RESTORED TO THE ESF BUSES, OR IF A SAFETY INJECTION SIGNAL OCCURS. THE CONTROL CIRCUITS FOR OTHER ESF ROOM COOLER FANS WERE REVIEWED TO ENSURE THAT THEY RESTART AUTOMATICALLY AFTER A LOSS OF OFF SITE POWER, OR UPON A SAFETY INJECTION SIGNAL. FAILURE OF ESF ELECTRICAL SWITCHGEAR ROOM COOLERS TO RESTART AFTER A LOSS OF OFF SITE POWER WOULD HAVE CREATED AN UNANALYZED CONDITION. ONLY IF EXPECTED OPERATOR ACTION WAS NOT TAKEN TO RESTORE COOLING COULD THIS UNANALYZED CONDITION HAVE RESULTED IN A COMMON FAILURE MECHANISM FOR THE ESF SWITCHGEAR.

[195] TROJAN DOCKET 50-344 LER 90-018 REV 01
UPDATE ON LACK OF PERIODIC COOLER INSPECTION AND CLEANING PROGRAM RESULTS IN EXCESSIVE COOLER BLOCKAGE AND OPERATION IN AN UNANALYZED CONDITION.
EVENT DATE: 052590 REPORT DATE: 082990 NSSS: WE TYPE: PWR

(NSIC 219434) ON APRIL 27, 1990 THE TROJAN NUCLEAR PLANT WAS IN A REFUELING OUTAGE. INSPECTIONS OF THE "A" TRAIN CONTAINMENT SPRAY AND CENTRIFUGAL CHARGING PUMP ROOM COOLERS WERE IN PROGRESS. THE INSPECTIONS REVEALED THAT UP TO 13 PERCENT OF THE CONTAINMENT SPRAY PUMP ROOM COOLER'S TUBES WERE BLOCKED BY SILT AND CLAM SHELLS. THE SIZE OF THIS COOLER PROVIDED SUFFICIENT MARGIN FOR IT TO BE CONSIDERED OPERABLE. AS A PRECAUTIONARY MEASURE, IT WAS DECIDED TO INSPECT 23 ADDITIONAL ROOM COOLERS WITH LESS AVAILABLE MARGIN. ON MAY 25, 1990, EVALUATION OF THE ADDITIONAL INSPECTIONS SHOWED THAT TRAIN ELECTRICAL SWITCHGEAR ROOM COOLERS DID NOT HAVE SUFFICIENT CAPACITY TO MAINTAIN ROOM TEMPERATURES WITHIN DESIGN LIMITS UNDER ORIGINALLY ANALYZED DESIGN CONDITIONS. THE CAUSE OF THE TUBE BLOCKAGE WAS THE LACK OF A PERIODIC ROOM COOLER INSPECTION AND CLEANING PROGRAM AT TROJAN. ALL SERVICE WATER SUPPLIED ENGINEERED SAFETY FEATURES EQUIPMENT ROOM COOLERS, EXCEPT THREE WHICH HAVE A LARGE DESIGN MARGIN AND TWO IN AN AREA WHERE ALTERNATE COOLING IS AVAILABLE, WERE INSPECTED AND CLEANED DURING THE 1990 REFUELING OUTAGE. THE REMAINING FIVE COOLERS WILL BE INSPECTED PRIOR TO RESTART FROM THE 1991 REFUELING OUTAGE. A COOLER PREVENTATIVE MAINTENANCE AND INSPECTION PLAN IS BEING DEVELOPED.

[196] TROJAN DOCKET 50-344 LER 90-027
INADEQUATE IMPLEMENTATION OF A PROGRAMMATIC CHANGE IN HOW A TECHNICAL SPECIFICATION SURVEILLANCE WAS TO BE MET RESULTS IN A MISSED SURVEILLANCE DUE TO AN INADEQUATE PROCEDURE.
EVENT DATE: 071790 REPORT DATE: 081690 NSSS: WE TYPE: PWR

(NSIC 219360) ON JULY 17, 1990, THE PLANT WAS IN MODE 1 (POWER OPERATION) WITH A GENERATOR LOAD OF 1140 MWE. DURING A REVIEW OF A PLANT PROCEDURE, PLANT SYSTEMS ENGINEERING PERSONNEL IDENTIFIED THAT THE MONTHLY SURVEILLANCE REQUIRED BY TROJAN TECHNICAL SPECIFICATION (TTS) 4.6.1.1, "PRIMARY CONTAINMENT - CONTAINMENT INTEGRITY" DID NOT INCLUDE TWELVE VALVES ASSOCIATED WITH THE CONTAINMENT PENETRATION BOUNDARY FOR THE STEAM GENERATOR BLOWDOWN SYSTEM. TWO DRAIN VALVES HAD NOT BEEN INCLUDED IN THE PLANT PROCEDURES USED TO PERFORM THE REQUIRED SURVEILLANCE DUE TO INADEQUATE IMPLEMENTATION OF A 1988 PROGRAMMATIC CHANGE WHICH ADDED ALL VENT, TEST, AND DRAIN VALVES WITHIN THE CONTAINMENT PENETRATION BOUNDARY TO THE LIST OF VALVES REQUIRING TTS 4.6.1.1 SURVEILLANCE. THE OTHER TEN VALVES WERE NOT INCLUDED AS THEY ARE LOCATED OUTSIDE OF THE FIRST CONTAINMENT ISOLATION VALVE FOR A CLOSED SYSTEM INSIDE CONTAINMENT. THE ISOLATION DESIGN BASIS FOR THIS PENETRATION IS TWO VALVES OUTSIDE CONTAINMENT. CORRECTIVE ACTION WAS TO PLACE THESE 12 VALVES IN THE PROCEDURE WHICH IS USED TO PERFORM THE TTS 4.6.1.1 REQUIRED SURVEILLANCE. ADDITIONAL CORRECTIVE ACTIONS WILL BE TO VERIFY THAT VALVES THAT ARE PART OF THE CONTAINMENT PENETRATION BOUNDARY ARE INCLUDED IN APPROPRIATE SURVEILLANCE PROCEDURES.

[197] TROJAN DOCKET 50-344 LER 90-032
PERSONNEL ERROR IN DEVELOPMENT OF SURVEILLANCE PROCEDURE RESULTS IN INADEQUATE SURVEILLANCE OF POST ACCIDENT EFFLUENT IODINE SAMPLERS.
EVENT DATE: 080390 REPORT DATE: 090490 NSSS: WE TYPE: PWR

(NSIC 219476) ON 8/3/90, THE TROJAN NUCLEAR PLANT WAS OPERATING IN MODE 1 (POWER OPERATION) AT 100% RATED THERMAL POWER. DURING REVIEW OF A PLANT SURVEILLANCE PROCEDURE, IT WAS DISCOVERED THAT THE CHANNEL CHECK FOR THE HIGH RANGE EFFLUENT IODINE SAMPLERS WAS NOT BEING CONDUCTED IN ACCORDANCE WITH THE TROJAN TECH SPEC 3/4.3.3.11, RADIOACTIVE GAS PROCESS AND EFFLUENT MONITORING INSTRUMENTATION, WHICH REQUIRES THE CONTAINMENT, AUXILIARY BUILDING AND CONDENSER AIR EJECTOR HIGH RANGE EFFLUENT IODINE SAMPLERS TO BE DEMONSTRATED OPERABLE BY PERFORMANCE OF A CHANNEL CHECK AT LEAST ONCE PER 12 HOURS. THE REQUIRED CHANNEL CHECK CONSISTS OF VERIFYING FLOW THROUGH THE SAMPLERS. THE IMPLEMENTING PROCEDURE FOR THIS SURVEILLANCE REQUIREMENT REQUIRED VERIFICATION THAT THE PROCESS EFFLUENT AND RADIATION MONITORS WHICH CONTAIN THE HIGH RANGE IODINE SAMPLERS WERE FUNCTIONING NORMALLY, AND THAT THEY WERE AVAILABLE TO TRANSFER TO THE ACCIDENT SAMPLING MODE, BUT IT DID NOT REQUIRE VERIFICATION OF FLOW THROUGH THE SAMPLER. THIS EVENT WAS CAUSED BY PERSONNEL ERROR IN THE ORIGINAL DEVELOPMENT OF THE PROCEDURE TO IMPLEMENT THE SURVEILLANCE REQUIREMENTS FOR THESE SAMPLERS. THE SURVEILLANCE PROCEDURE WAS REVISED ON 8/9/90. SURVEILLANCE WAS PERFORMED AND THE SAMPLERS WERE RETURNED TO OPERABLE STATUS ON 8/10/90.

[198] TROJAN DOCKET 50-344 LER 90-034
INCORRECT FEEDWATER PUMP PROTECTIVE INSTRUMENT ADJUSTMENT LEADS TO PUMP TRIP FOLLOWED BY REACTOR TRIP AND AUXILIARY FEEDWATER ACTUATION.
EVENT DATE: 080990 REPORT DATE: 091090 NSSS: WE TYPE: PWR

(NSIC 219455) ON AUGUST 9, 1990, THE TROJAN NUCLEAR PLANT WAS OPERATING IN MODE 1 (POWER OPERATION) AT 100 PERCENT RATED THERMAL POWER. THE GENERATOR LOAD WAS APPROXIMATELY 1135 MW. AT 1648 THE "B" FEEDWATER PUMP TRIPPED ON INDICATION OF HIGH THRUST BEARING WEAR. IN RESPONSE TO THE LOSS OF THE FEEDWATER PUMP, THE MAIN TURBINE BEGAN AN AUTOMATIC RUNBACK AND THE CONTROL RODS BEGAN AUTOMATICALLY STEPPING IN TO REDUCE REACTOR POWER. THE TURBINE RUNBACK CONTINUED FOR ONE MINUTE, TWENTY SECONDS; THEN THE REACTOR AUTOMATICALLY TRIPPED ON LOW-LOW LEVEL IN THE "D" STEAM GENERATOR. FOLLOWING THE REACTOR TRIP, AUTOMATIC FEEDWATER ISOLATION AND AUXILIARY FEEDWATER INITIATION OCCURRED. THE "B" FEEDWATER PUMP TRIP WAS INITIATED BY PROTECTIVE INSTRUMENTATION WHICH RECEIVED INDICATION OF EXCESSIVE THRUST BEARING WEAR IN THE PUMP'S TURBINE DRIVER. THE EXCESSIVE BEARING WEAR INDICATION WAS THE RESULT OF INCORRECTLY SET INSTRUMENTATION, NOT ACTUAL WEAR. THE THRUST BEARING WEAR INDICATING INSTRUMENTS WERE RECALIBRATED AND THE PUMP WAS REASSEMBLED. THE PLANT WAS RETURNED TO POWER OPERATION ON AUGUST 11, 1990. THIS EVENT HAD NO EFFECT UPON PUBLIC HEALTH AND SAFETY. THE REACTOR PROTECTIVE SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM FUNCTIONED AS REQUIRED. THE PLANT RESPONDED AS EXPECTED TO THE TRIP.

[199] TURKEY POINT 3 DOCKET 50-250 LER 90-011 REV 01
UPDATE HI-HI STEAM GENERATOR WATER LEVEL TURBINE TRIP AND SUBSEQUENT REACTOR TRIP DUE TO FAILURE OF A SWITCH IN A FEEDWATER VALVE CONTROLLER HAND/AUTO STATION.
EVENT DATE: 060990 REPORT DATE: 082790 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 219385) ON JUNE 9, 1990, AT 0648 EDT, WITH UNIT 3 IN MODE 1 (POWER OPERATION) AT 26 PERCENT POWER AND UNIT 4 IN MODE 1 AT 100 PERCENT POWER, UNIT 3 EXPERIENCED A HI-HI STEAM GENERATOR WATER LEVEL TURBINE TRIP AND SUBSEQUENT REACTOR TRIP. ALL SAFETY SYSTEMS PERFORMED AS DESIGNED. AFTER THE TRIP, THE OPERATORS STABILIZED THE UNIT IN MODE 3 (HOT STANDBY) BY USING APPLICABLE PROCEDURES. THE CAUSE OF THIS EVENT WAS A MALFUNCTION OF THE 3C FEEDWATER REGULATOR HAND/AUTO STATION OPEN PUSHBUTTON SWITCH FOR VALVE CONTROLLER FC-3-498F. THE SWITCH IS A MOMENTARY ACTION SWITCH DESIGNED TO SPRING BACK TO THE "NO CONTACT" POSITION UPON RELEASE. THE SWITCH WAS FOUND SPRUNG BACK TO THE "NO CONTACT" POSITION, BUT THE SWITCH CONTACTS WERE STILL CLOSED. THIS RESULTED IN A FULL OPEN DEMAND SIGNAL CAUSING THE 3C FEEDWATER REGULATING VALVE TO FULLY OPEN. THE FAILED 3C FEEDWATER REGULATING VALVE HAND/AUTO STATION AND THE

HAND/AUTO STATION FOR THE 3B FEEDWATER REGULATING VALVE WERE REPLACED WITH HAND/AUTO STATIONS HAVING NEW STYLE SWITCHES. THE HAND/AUTO STATION FOR THE 3A FEEDWATER REGULATING VALVE HAD BEEN REPLACED IN JUNE, 1989. ON JUNE 9, 1990, AT 0716 EDT, THE NRC WAS NOTIFIED OF THIS EVENT IN ACCORDANCE WITH 10 CFR 50.72(B)(2)(II).

[200] TURKEY POINT 3 DOCKET 50-250 LER 90-016
TECHNICAL SPECIFICIATION ENTERED TO REPAIR BORIC ACID FILTER DISCHARGE ISOLATION VALVE.
EVENT DATE: 080190 REPORT DATE: 083090 NSSS: WE TYPE: PWR
VENDOR: GRINNELL INDUSTRIAL PIPING, INC.

(NSIC 219405) ON 7/29/90, WITH UNIT 3 IN MODE 1 AT 100% POWER, A LEAK OF APPROXIMATELY 1 DROP/MINUTE WAS NOTICED IN THE AREA OF BORIC ACID FILTER DISCHARGE ISOLATION VALVE 3-348. A FLOW PATH FROM THE BORIC ACID TANKS TO THE REACTOR COOLANT SYSTEM (RCS) WAS ESTABLISHED VIA THE BORIC ACID FILTER BYPASS LINE AND VALVE 3-348 WAS CLOSED TO REDUCE VALVE LEAKAGE. VALVE 3-348 IS A DIAPHRAGM VALVE. THE DIAPHRAGM ALSO SERVES AS THE GASKET BETWEEN THE VALVE BODY AND BONNET MATING SURFACES. UPON INSPECTION, THE DIAPHRAGM EXHIBITED WAVED EDGES ON TWO SIDES WHICH APPEARED SIGNIFICANT ENOUGH TO DEGRADE THE SEAL BETWEEN THE VALVE BODY AND BONNET. THIS DEFORMATION WAS CAUSED BY EXPOSURE TO HIGHER THAN NORMAL TEMPERATURES WHICH REDUCED THE NORMAL LIFE EXPECTANCY OF THE DIAPHRAGM. IN ORDER TO REPAIR VALVE 3-348, THE BORIC ACID FILTER BYPASS LINE MUST BE ISOLATED. THIS CONFIGURATION RESULTS IN NO FLOW PATH FROM THE BORIC ACID TANKS TO THE RCS. TECHNICAL SPECIFICATION (TS) 3.6 REQUIRES SYSTEM PIPING, INTERLOCKS AND VALVES TO BE OPERABLE TO THE EXTENT OF ESTABLISHING ONE FLOW PATH FROM THE BORIC ACID TANKS, AND ONE FLOW PATH FROM THE REFUELING WATER STORAGE TANK, TO THE RCS WHEN THE REACTOR IS CRITICAL. AT 0930, ON 8/1/90, WITH UNIT 3 IN MODE 1 AT 100% POWER, UNIT 3 ENTERED TS 3.0.1 TO REPAIR VALVE 3-348. THE VALVE WAS REPAIRED AND RETURNED TO SERVICE AT 1038. UNIT 3 EXITED TS 3.0.1 AT THIS TIME.

[201] TURKEY POINT 3 DOCKET 50-250 LER 90-017
MISSED FIRE PROTECTION SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 081790 REPORT DATE: 091790 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)

(NSIC 219553) ON 3/10/90, WITH UNITS 3 & 4 IN MODE 1 (POWER OPERATION) AT 100% POWER, THE 18 MONTH PERIODICITY OF TECH SPEC (TS) 4.15.5.A.1, INCLUDING THE 25% EXTENSION PERMITTED BY TS 4.0.1, WAS EXCEEDED FOR THE INSPECTION OF 14 DAMPERS COVERED BY 10 CFR 50, APPENDIX A AND APPENDIX R. THEREFORE THE DAMPERS AND THEIR ASSOCIATED FIRE BARRIERS WERE TECHNICALLY INOPERABLE FROM 3/10/90, UNTIL THE DAMPERS WERE INSPECTED USING PROCEDURE 0-SME-016.4, "FIRE DAMPER INSPECTION." THE MISSED SURVEILLANCE OF 10 OF THE DAMPERS WAS DISCOVERED DURING A QUALITY ASSURANCE (QA) INSPECTION ON 8/17/90. THE OTHER FOUR DAMPERS WERE DISCOVERED ON 9/12/90, DURING A HVAC REVIEW TO HAVE ALSO MISSED THE MARCH 10 DUE DATE. TS 3.14.5.B.2 REQUIRES THAT A SPECIAL REPORT BE SUBMITTED IF NON-FUNCTIONAL FIRE BARRIERS ARE NOT RESTORED TO OPERABLE STATUS WITHIN 7 DAYS. THIS EVENT WAS CAUSED BY INADEQUATE PROCEDURAL GUIDANCE AND COGNITIVE PERSONNEL ERROR. PROCEDURE 0-SME-016.4 HAS BEEN REVISED TO INCLUDE THE 14 APPENDIX A DAMPERS. THE FIRST 10 DAMPERS WERE INSPECTED AND DETERMINED TO BE FUNCTIONAL ON 8/20/90. THE OTHER FOUR DAMPERS WERE INSPECTED AND DETERMINED TO BE FUNCTIONAL ON 9/14/90. THE RESULTS OF THE INVESTIGATION CONCERNING THE FOUR DAMPERS, WHICH WERE NOT IN THE FIRE DAMPER INSPECTION PROCEDURE, WILL BE REPORTED IN A SUPPLEMENTAL LER.

[202] TURKEY POINT 4 DOCKET 50-251 LER 90-008
AUTOMATIC REACTOR TRIP ON LOW-LOW STEAM GENERATOR LEVEL DUE TO LOSS OF 4A STEAM GENERATOR FEEDWATER PUMP.
EVENT DATE: 081290 REPORT DATE: 091090 NSSS: WE TYPE: PWR
VENDOR: AGASTAT RELAY CO.
ALLIS, LOUIS

(NSIC 219487) AT 1625, ON AUGUST 12, 1990, WITH UNIT 4 IN MODE 1 AT 100 PERCENT POWER, THE 4B CONDENSATE PUMP MOTOR AUTOMATICALLY TRIPPED ON OVERCURRENT. THE 4A

CONDENSATE PUMP AUTOMATICALLY STARTED AS DESIGNED. THE 4A STEAM GENERATOR FEEDWATER PUMP TRIPPED UNEXPECTEDLY AND INITIATED A TURBINE RUNBACK. THE TURBINE RUNBACK AND THE REDUCED FEEDWATER FLOW CAUSED THE STEAM GENERATOR LEVELS TO DECREASE. OPERATOR ACTIONS WERE TAKEN IN AN ATTEMPT TO RESTORE STEAM GENERATOR WATER LEVELS. AT 1628, A TWO-OUT-OF-THREE STEAM GENERATOR "A" LOW-LOW LEVEL TRIP SIGNAL INITIATED AN AUTOMATIC REACTOR TRIP AND SUBSEQUENT AUTOMATIC TURBINE TRIP. THE UNIT WAS STABILIZED IN MODE 3 (HOT STANDBY) USING EXISTING SITE PROCEDURES. THE 4B CONDENSATE PUMP MOTOR TRIP WAS CAUSED BY A PHASE-TO-PHASE SHORT. FPL POSTULATES THAT A WEAK SPOT IN THE INSULATION WAS CREATED WHEN THE MOTOR WAS REWOUND IN 1980. THE WEAK SPOT DEGRADED TO THE POINT THAT MOISTURE COULD PENETRATE THE INSULATION. THE MOISTURE PROVIDED AN ELECTRICAL SHORT ACROSS THE COILS. THE UNEXPECTED 4A STEAM GENERATOR FEEDWATER PUMP (SGFP) TRIP WAS CAUSED BY AN INCORRECT SETPOINT ON AN AGASTAT TIME DELAY BREAKER TRIP RELAY. THE 4A AND 4B SGFP BREAKER TRIP RELAYS HAVE BEEN RECALIBRATED. THE CONDENSATE PUMP IS BEING REPAIRED.

[203] VERMONT YANKEE DOCKET 50-271 LER 90-010
FAILURE TO MEET TECHNICAL SPECIFICATIONS FOR DIESEL GENERATOR OPERATIONAL READINESS TEST.
EVENT DATE: 081690 REPORT DATE: 091390 NSSS: GE TYPE: BWR

(NSIC 219495) ON 8/16/90, AT APPROXIMATELY 1630 HOURS, WITH THE REACTOR OPERATING AT 89.1% POWER, IT WAS IDENTIFIED THAT THE REQUIRED MONTHLY OPERATIONAL READINESS TESTS FOR THE A AND B EMERGENCY DIESEL GENERATORS HAVE NOT BEEN PERFORMED IN ACCORDANCE WITH TECH SPEC SECTION 4.10.A.1.A. IT WAS DETERMINED THAT THE EXPECTED MAXIMUM EMERGENCY LOAD USED IN THE SURVEILLANCE PROCEDURE WAS NOT THE TRUE MAXIMUM BASED UPON THE VALUE STATED IN THE FSAR (AFTER POWER FACTORS WERE TAKEN INTO ACCOUNT). THE FSAR PROVIDED A VALUE OF 2467.3 KW FOR THE MAXIMUM EMERGENCY LOAD FOR THE DIESEL GENERATORS. THE SURVEILLANCE PROCEDURE WAS TESTING THE DIESELS AT 2500 KW. IT APPEARED THAT THE PROCEDURE WAS ADEQUATELY TESTING THE DIESELS. HOWEVER THE FSAR VALUE WAS NOT AT UNITY POWER FACTOR AS ASSUMED. IT IS DESIRABLE TO RUN THE DIESEL SURVEILLANCE TEST AT A POWER FACTOR OF UNITY. A CALCULATION PROVIDED A CONSERVATIVE VALUE OF 2751.2 KW ASSUMING A POWER FACTOR OF 0.85 FOR MOTORS LARGER THAN 50 HP. TO CORRECT THIS SITUATION, THE CALCULATED LOAD WAS CONVERTED TO A POWER FACTOR OF UNITY AND THE VALUE OF 3175 KVA WAS DERIVED. THE PROCEDURE WAS THEN CHANGED TO INCORPORATE THE VALUE OF 3200 KW AT UNITY POWER FACTOR FOR TESTING THE DIESELS. A SURVEILLANCE TEST WAS PERFORMED ON BOTH DIESELS AND CONCLUDED SATISFACTORILY. THIS WAS ATTRIBUTED TO THE FSAR NOT BEING CLEAR IF ITS VALUE WAS AT A UNITY POWER FACTOR.

[204] VOGTLE 1 DOCKET 50-424 LER 87-083
USE OF ALTERNATE INSTRUMENT RESULTS IN INADEQUATE VERIFICATION OF REFUELING WATER STORAGE TANK TEMPERATURE.
EVENT DATE: 022787 REPORT DATE: 091290 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: VOGTLE 2 (PWR)
VENDOR: INTERNATIONAL INSTRUMENTS, INC.

(NSIC 219534) ON 8-13-90, IT WAS DISCOVERED THAT INSTRUMENT 2TIS-10980, USED FOR VERIFYING REFUELING WATER STORAGE TANK (RWST) TEMPERATURE, WAS NOT INDICATING. A READING WAS THEN ATTEMPTED FROM INSTRUMENT 2TI-10982 WHICH MEASURES RWST SLUDGE MIXING RECIRCULATION TEMPERATURE. HOWEVER, IT WAS FOUND TO NOT BE INSTALLED. THE VALUES FOR RWST TEMPERATURE ASSUMED IN THE SAFETY ANALYSIS ARE 50F (MINIMUM) AND 120F (MAXIMUM). THE TECH SPEC (TS) LIMITS OF 54F TO 116F ACCOUNT FOR A LOOP TOLERANCE OF 4F FOR TI-10982. THE LOOP TOLERANCE OF TIS-10980 IS 10.792F. THEREFORE, USE OF TIS-10980 WOULD REQUIRE ADMINISTRATIVE LIMITS OF 61F TO 109F TO ENSURE COMPLIANCE WITH THE TS. HOWEVER, SINCE NO SUCH LIMITS HAD BEEN IMPOSED, A REVIEW WAS CONDUCTED OF PREVIOUS RWST TEMPERATURES RECORDED FROM TIS-10980. WHILE NO UNIT 2 RWST TEMPERATURES WERE FOUND TO HAVE BEEN OUTSIDE OF THESE ADMINISTRATIVE LIMITS, UNIT 1 RWST TEMPERATURE HAD BEEN LESS THAN 61F ON SEVERAL OCCASIONS DURING 1987 AND 1988. THEREFORE, VERIFICATION OF RWST TEMPERATURE ON THOSE OCCASIONS WAS INADEQUATE TO ENSURE COMPLIANCE WITH THE TS. THE ROOT CAUSE FOR THIS EVENT WAS PROCEDURAL INADEQUACY. THE APPROPRIATE PROCEDURES HAVE BEEN REVISED TO IMPOSE THE NECESSARY ADMINISTRATIVE LIMITS TO ALLOW CONTINUED USE OF TIS-10980.

[205] VOGTLE 1 DOCKET 50-424 LER 90-016
FAILED TRANSFORMER LEADS TO MAIN FEEDPUMP TRIP AND REACTOR TRIP.
EVENT DATE: 072390 REPORT DATE: 082190 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219379) ON 7-23-90 AT 0600 CDT, UNIT 1 WAS OPERATING IN MODE 1 AT 100% POWER WHEN A 4160/480 VOLT NON-1E TRANSFORMER (1NB01) EXPERIENCED AN INTERNAL FAULT, CAUSING A TRIP OF THE ASSOCIATED FEEDER BREAKER. THIS TRANSFORMER WAS SUPPLYING POWER TO THE SPEED CONTROL CIRCUITRY FOR BOTH OF THE MAIN FEEDWATER PUMP (MFP) TURBINES. THE MFPS TRIPPED, THE STEAM GENERATOR (SG) WATER LEVELS DECREASED TO 24%, NARROW RANGE LEVEL, AND THE REACTOR OPERATOR INITIATED A MANUAL REACTOR TRIP AT 0602 CDT. THE MAIN FEEDWATER SYSTEM ISOLATED AND THE MOTOR-DRIVEN AUXILIARY FEEDWATER (AFW) PUMPS STARTED, AS EXPECTED, WHEN THE REACTOR TRIP OCCURRED. THE TURBINE DRIVEN AFW PUMP STARTED WHEN TWO OF THE FOUR SG'S REACHED THEIR LOW-LOW LEVEL SETPOINT. NORMAL PLANT CONDITIONS WERE ESTABLISHED IN MODE 3 (HOT STANDBY) AT 0656 CDT. THE CAUSE OF THE EVENT WAS AN INTERNAL FAULT IN THE 1NB01 TRANSFORMER. FURTHER INVESTIGATION OF THE CAUSE OF THE FAULT IS IN PROGRESS.

[206] WATERFORD 3 DOCKET 50-382 LER 90-011
ESF CONTROL ROOM VENTILATION ACTUATIONS DUE TO EQUIPMENT MALFUNCTION.
EVENT DATE: 072190 REPORT DATE: 082090 NSSS: CE TYPE: PWR

(NSIC 219388) BETWEEN 7/21 AND 8/12/90, WATERFORD STEAM ELECTRIC STATION UNIT 3 EXPERIENCED 4 UNPLANNED ACTUATIONS OF THE ESF PORTION OF THE CONTROL ROOM VENTILATION SYSTEM. EACH 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. IN EACH EVENT, 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. EACH EVENT IS REPORTABLE AS AN UNPLANNED ESF ACTUATION. THE ROOT CAUSE FOR TWO OF THE FOUR ACTUATIONS WAS EQUIPMENT MALFUNCTION OF THE CROAI RADIATION MONITOR CAUSED BY A PERFORATION IN THE ALUMINUM FOIL BETA SHIELD. THE ROOT CAUSE FOR THE OTHER TWO ACTUATIONS IS CURRENTLY UNDER INVESTIGATION AND WILL BE REPORTED AS A REVISION TO THIS LER ON POSITIVE DETERMINATION OF THE CAUSE. IN EACH EVENT, THE CONTROL ROOM EMERGENCY FILTRATION SYSTEM FUNCTIONED AS DESIGNED AND THERE WAS NO ACTUAL RELEASE OF RADIOACTIVE MATERIAL; THEREFORE THESE EVENTS DID NOT RESULT IN AN INCREASED RISK TO THE HEALTH AND SAFETY OF THE PUBLIC OR PLANT PERSONNEL.

[207] WOLF CREEK 1 DOCKET 50-482 LER 90-017
PARTIAL WASTE GAS DECAY TANK RELEASE CAUSED BY INOPERABLE DRAIN TRAPS.
EVENT DATE: 072990 REPORT DATE: 082890 NSSS: WE TYPE: PWR
VENDOR: ARMSTRONG MACHINE WORKS

(NSIC 219443) ON 7/29/90 AT 1000 CDT, A RADWASTE OPERATOR REPORTED THAT WASTE GAS DECAY TANK #3 (WGDT) HAD UNDERGONE A PRESSURE DROP WHEN ITS DRAIN VALVE HAD BEEN CYCLED FOLLOWING MAINTENANCE. IT WAS DETERMINED THAT A PORTION OF THE CONTENTS OF WGDT #3 HAD BEEN DISCHARGED TO THE RADWASTE BUILDING HEATING, VENTILATION AND AIR CONDITIONING SYSTEM. THIS SITUATION CONSTITUTES A VIOLATION OF TECH SPEC 3.11.2.1, WHICH REQUIRES, IN PART, SAMPLING OF A WGDT PRIOR TO ITS RELEASE. IT WAS DETERMINED THAT THE CONNECTING DRAIN TRAPS WERE NOT FUNCTIONING PROPERLY BECAUSE OF INADEQUATE WATER IN THE TRAPS. A WORK REQUEST HAS BEEN INITIATED TO INSPECT THE TRAPS FOR PROPER OPERATION. A PROCEDURAL CHANGE HAS BEEN COMPLETED TO PROVIDE INSTRUCTIONS FOR FILLING THE DRAIN TRAPS. A LETTER HAS BEEN PROVIDED TO ALL RADWASTE OPERATORS RE-EMPHASIZING WORKING WITH MECHANICAL MAINTENANCE DURING VALVE MANIPULATIONS. CALCULATIONS PERFORMED SUBSEQUENT TO THIS EVENT VERIFIED THAT NO RELEASE LIMITS WERE EXCEEDED DURING THIS EVENT.

[208] WOLF CREEK 1 DOCKET 50-482 LER 90-018
IMPROPER SCAFFOLD INSTALLATION CAUSES INOPERABILITY OF TURBINE DRIVEN AUXILIARY FEEDWATER PUMP.
EVENT DATE: 080390 REPORT DATE: 090490 NSSS: WE TYPE: PWR

(NSIC 219527) ON 8/3/90 AT 0911 CDT, THE TURBINE BUILDING OPERATOR REPORTED TO THE SHIFT SUPERVISOR THAT SCAFFOLDING WAS DISCOVERED INTERFERING WITH THE OPERATION OF CHECK VALVE ALV001 IN THE SUCTION LINE TO THE TURBINE DRIVEN AUXILIARY FEEDWATER PUMP PAL02 FROM THE CONDENSATE STORAGE TANK. THE INTERFERENCE WOULD NOT ALLOW VALVE ALV001 TO FULLY OPEN, THUS MAKING PUMP PAL02 INOPERABLE. THE SCAFFOLDING WAS ADJUSTED SUCH THAT MOVEMENT OF ALV001 WAS NO LONGER RESTRICTED. A REVIEW OF THIS EVENT CONCLUDED THAT THE SCAFFOLDING WAS IN SUCH A POSITION THAT IT MADE PUMP PAL02 INOPERABLE BETWEEN JULY 25 AND AUGUST 3. THIS IS CONTRARY TO TECH SPEC 3.7.1.2, WHICH REQUIRES, IN PART, THAT THREE INDEPENDENT AUXILIARY FEEDWATER PUMPS BE OPERABLE. THIS EVENT OCCURRED BECAUSE OF A FAILURE TO COMPLY WITH THE SCAFFOLDING PROCEDURE. THE PROCEDURE REQUIREMENTS WERE REVIEWED WITH ALL SCAFFOLD BUILDERS. THE PROCEDURE HAS ALSO BEEN REVIEWED FOR POSSIBLE ENHANCEMENTS, BUT NONE WERE DEEMED NECESSARY. ALL SCAFFOLDING IN SAFETY-RELATED AREAS WERE INSPECTED TO ENSURE NO SIMILAR SITUATIONS EXISTED. TWO MINOR SCAFFOLD INTERFERENCES WERE FOUND WHICH HAD NO ADVERSE EFFECT ON SYSTEM FUNCTION AND WERE CORRECTED.

[209] WOLF CREEK 1 DOCKET 50-482 LER 90-019
A LATE PERFORMANCE OF REQUIRED SURVEILLANCE OF OFFSITE A.C. POWER SOURCES CAUSED BY PERSONNEL ERROR.
EVENT DATE: 082090 REPORT DATE: 091890 NSSS: WE TYPE: PWR

(NSIC 219560) ON 8/20/90 AT 0230 CDT, EMERGENCY DIESEL GENERATOR "B" WAS REMOVED FROM SERVICE FOR PLANNED MAINTENANCE. THE CORRECT BREAKER ALIGNMENTS AND INDICATED POWER AVAILABILITY OF THE OFFSITE A.C. POWER SOURCES WERE VERIFIED AT 10345 CDT. TECH SPEC 3.8.1.1, ACTION STATEMENT 'B', REQUIRES THAT OFFSITE A.C. POWER SOURCES MUST BE VERIFIED OPERABLE WITHIN ONE HOUR AFTER MAKING AN EMERGENCY DIESEL GENERATOR INOPERABLE. THIS LATE PERFORMANCE OF AN ACTION STATEMENT CONSTITUTES A VIOLATION OF TECH SPEC 3.8.1.1 SINCE THE OFFSITE A.C. POWER SOURCES WERE NOT VERIFIED PRIOR TO 0330 CDT. THE ROOT CAUSE OF THIS EVENT WAS COGNITIVE PERSONNEL ERROR BY LICENSED OPERATORS, WHO FAILED TO PROPERLY PRIORITIZE WORK ACTIVITIES TO ASSURE COMPLETION OF THE REQUIRED SURVEILLANCE WITHIN THE ALLOWABLE ONE HOUR TIME PERIOD. ALL LICENSED OPERATORS WILL BE COUNSELED ON THE NEED TO PROPERLY PRIORITIZE ALL SURVEILLANCE REQUIREMENTS. A COPY OF THIS REPORT WILL BE ADDED TO OPERATIONS REQUIRED READING AND A COPY WILL BE SENT TO TRAINING TO REITERATE THAT SURVEILLANCE REQUIREMENTS MUST BE COMPLETED WITHIN THE ALLOWABLE TIME FRAME. IN ADDITION, A PROCEDURE WILL BE WRITTEN TO GOVERN THE ACTIONS THAT MUST BE TAKEN WHEN AN EMERGENCY DIESEL GENERATOR IS REMOVED FROM SERVICE FOR PLANNED MAINTENANCE ACTIVITIES.

[210] WPPSS 2 DOCKET 50-397 LER 89-011 REV 01
UPDATE ON MISSING LIMITORQUE MOTOR OPERATOR TORQUE SWITCH BYPASS JUMPERS DUE TO PLANT DESIGN BASIS DOCUMENTATION.
EVENT DATE: 050289 REPORT DATE: 090690 NSSS: GE TYPE: BWR
VENDOR: LIMITORQUE CORP.

(NSIC 219480) ON MAY 2, 1989, PLANT DESIGN ENGINEERS DETERMINED THAT EIGHT VALVE MOTORS DID NOT MEET THE WNP-2 COMMITMENT MADE IN RESPONSE TO IE CIRCULAR NUMBER 81-13, "TORQUE SWITCH ELECTRICAL BYPASS CIRCUIT FOR SAFEGUARD SERVICE VALVE MOTORS." THE SUPPLY SYSTEM PREVIOUSLY SUBMITTED LER 87-024-00 WHICH REPORTED A SIMILAR CONDITION, AND THE EIGHT ADDITIONAL VALVE SAFETY FUNCTIONS DESCRIBED IN THIS LER WERE IDENTIFIED AS A RESULT OF CORRECTIVE ACTION COMMITTED TO IN LER 87-024. ONE MAIN STEAMLINE DRAIN ISOLATION VALVE, MS-V-20, WAS POWERED BY A NON CLASS 1E SUPPLY AND, AS A RESULT, WOULD BE LOST DURING A LOSS OF OFFSITE POWER EVENT. NO IMMEDIATE CORRECTIVE ACTIONS WERE REQUIRED SINCE ALL THE VALVE SAFETY FUNCTIONS DESCRIBED IN THIS LER ARE NOT APPLICABLE TO THE CURRENT PLANT OPERATIONAL MODES 4 (COLD SHUTDOWN) AND 5 (REFUELING). ALL VALVE MOTOR TORQUE SWITCH BYPASS CIRCUITS WILL BE INSTALLED PRIOR TO PLANT RESTART. THE POWER SUPPLY FOR MS-V-20 WILL BE UPGRADED TO CLASS 1E PRIOR TO PLANT RESTART. THE ROOT CAUSE OF THIS EVENT IS LESS THAN ADEQUATE DESIGN BASIS DOCUMENTATION IN THAT THE PLANT ARCHITECT/ENGINEER (BURNS AND ROE, INC.) DID NOT FORMALIZE THE ORIGINAL SAFETY FUNCTION MOTOR OPERATED VALVE LIST. IN JANUARY 1989, THE SUPPLY SYSTEM ISSUED A CALCULATION WHICH FORMALIZED THE SAFETY VALVE SELECTION CRITERIA AND THE SAFETY FUNCTION OF EACH MOTOR OPERATED VALVE.

[211] WPPSS 2 DOCKET 50-397 LER 90-015
DIESEL FUEL OIL ANALYSIS TECHNICAL SPECIFICATION SURVEILLANCE VERIFICATION TIME REQUIREMENT EXCEEDED DUE TO INADEQUATE PROCEDURE AND PERSONNEL ERROR.
EVENT DATE: 080690 REPORT DATE: 091090 NSSS: GE TYPE: BWR

(NSIC 219511) ON AUGUST 9, 1990, IT WAS DISCOVERED THAT A VIOLATION OF WNP-2 TECHNICAL SPECIFICATIONS HAD OCCURRED WHEN THE MONTHLY DIESEL GENERATOR FUEL OIL PARTICULATE CONTAMINATION ANALYSIS FOR DIESEL FUEL OIL TANK 1B HAD NOT BEEN VERIFIED WITHIN THE REQUIRED SEVEN DAY TIME FRAME. THE ANALYSIS RESULTS WHICH WERE WITHIN TECHNICAL SPECIFICATION REQUIREMENTS WERE RECEIVED AT 1250 HOURS ON AUGUST 9, 1990, TEN DAYS AFTER THE SAMPLE WAS TAKEN. THE ROOT CAUSES OF THIS EVENT WERE EVALUATED AS BEING: WORK PRACTICES LESS THAN ADEQUATE - MANUAL TRACKING SYSTEMS USED BY THE PLANT CHEMISTRY GROUP AND MATERIAL REQUIREMENTS GROUP FAILED TO IDENTIFY THE SURVEILLANCE PRIOR TO EXCEEDING THE SEVEN DAY TIME FRAME, THEREFORE A PERSONNEL ERROR WENT UNDETECTED UNTIL IT WAS TOO LATE; PROCEDURES LESS THAT ADEQUATE - THREE SEPARATE PROCEDURES HAD TO BE USED TO COMPLETE THE FUEL OIL SAMPLING AND ANALYSIS, NONE OF THEM WAS ADEQUATE TO ENSURE THAT THE SEVEN DAY TIME REQUIREMENT WAS MET. A SINGLE KEY STEP IN ONE OF THE PROCEDURES CONTAINED TWO SEPARATE ACTIONS. ONE OF THE PROCEDURES WAS RECENTLY DEVIATED, THE DEVIATION OMITTED A KEY STEP WHICH REQUIRED MEETING THE SEVEN DAY TIME RESTRICTION.

[212] YANKEE ROWE DOCKET 50-029 LER 90-006
EMERGENCY DIESEL GENERATORS FAILED TECHNICAL SPECIFICATION SURVEILLANCE TEST.
EVENT DATE: 081190 REPORT DATE: 091390 NSSS: WE TYPE: PWR

(NSIC 219503) ON 8/10/90 AT 1700 HOURS, WHILE IN MODE 5 DURING A REFUELING SHUTDOWN, ALL THREE PLANT EMERGENCY DIESEL GENERATORS (EDGS) WERE CONSERVATIVELY CALLED INOPERABLE, BUT AVAILABLE FOR SERVICE, WHILE AWAITING RESULTS OF POST MAINTENANCE TESTING BEING CONDUCTED ON EDGS NO.S 1 AND 2. THE ACTION STATEMENTS FOR TS 3.8.1.2 AND 3.8.2.2 WERE ENTERED AND ACTIONS TO SUSPEND CORE ALTERATIONS OR POSITIVE REACTIVITY CHANGES, AND ESTABLISH REFUELING CONTAINMENT INTEGRITY WERE TAKEN. SURVEILLANCE TESTING CONDUCTED ON 8/11/90 AT 0200 HOURS, CONFIRMED THAT EDG NO. 2 COULD NOT MEET THE TS ACCEPTANCE CRITERIA. TESTING OF EDG NO. 3 ALSO VERIFED ITS INABILITY TO MEET TS ACCEPTANCE CRITERIA. AN INDEPENDENT INVESTIGATION TEAM HAS BEEN FORMED TO REVIEW THE PLANT'S EDG TESTING HISTORY (INCLUDING THE ACCEPTANCE CRITERIA), DETERMINE ROOT CAUSE AND SAFETY CONSEQUENCES, AND RECOMMEND SHORT/LONG TERM CORRECTIVE ACTION RECOMMENDATIONS. THIS EVALUATION IS PRESENTLY ONGOING; THE RESULTS OF THE INVESTIGATION WILL BE FACTORED INTO A SUPPLEMENT TO THIS LER. A DESIGN CHANGE HAS BEEN INITIATED TO REPLACE ALL THREE EDGS WITH NEW UNITS HAVING A DESIGN RATING OF 600 KW STANDBY, AND 450 KW CONTINUOUS. A PRELIMINARY RE-ANALYSIS OF THE EVENT INDICATES THAT AT ALL TIMES THE EDGS WERE CAPABLE OF PERFORMING THEIR INTENDED EMERGENCY FUNCTION.

[213] ZION 1 DOCKET 50-295 LER 90-015
INCORRECT CONTAINMENT FLOOD LEVEL IS SPECIFIED IN UPDATED FINAL SAFETY ANALYSIS REPORT.
EVENT DATE: 072690 REPORT DATE: 082790 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 2 (PWR)

(NSIC 219415) ON JULY 3, 1990 PRELIMINARY ENGINEERING CALCULATIONS INDICATED THAT THE CONTAINMENT FLOOD LEVELS UNDER CERTAIN ACCIDENT CONDITIONS, WOULD BE HIGHER THAN ANTICIPATED. ON JULY 26, 1990, AN INDEPENDENT REVIEW CONFIRMED THESE CALCULATIONS. THE CURRENT FLOOD LEVEL IS 3.5 FEET, WHILE THE NEW LEVELS WOULD BE 3.75, 5.06 OR 5.84 FEET DEPENDING ON WHETHER OR NOT THE REFUELING WATER STORAGE TANK (RWST) IS TOTALLY EMPTIED DURING AN ACCIDENT INVOLVING CONTAINMENT SPRAY. THIS HIGHER FLOOD LEVEL WILL RESULT IN SOME EQ EQUIPMENT BEING SUBMERGED. THE EQUIPMENT INVOLVED IN THIS ISSUE INCLUDES: TERMINAL BLOCKS IN JUNCTION BOXES, LEVEL/PRESSURE TRANSMITTERS, AND RTD'S. THIS EVENT WAS CAUSED BY TWO THINGS. (1) THE INCORE SHAFT AREA WAS CLOSED OFF FROM THE REST OF THE CONTAINMENT BY THE INSTALLATION OF A LEAK TIGHT ENCLOSURE. THIS INSTALLATION WAS MADE DURING PLANT CONSTRUCTION AND WAS NEVER REVIEWED FOR IMPACT ON CONTAINMENT FLOODING. (2) THE EMERGENCY OPERATING PROCEDURES (EOP'S) CALL FOR THE ENTIRE VOLUME OF THE RWST TO BE DUMPED INTO THE CONTAINMENT. HOWEVER, ZION FSAR CALLS FOR CONTAINMENT SPRAY

TO STOP AT THE RWST LO-LO ALARM SETPOINT. CORRECTIVE ACTIONS TAKEN OR TO BE PERFORMED INCLUDE REPLACING THE TERMINAL BLOCKS WITH SPLICES QUALIFIED FOR SUBMERGENCE, AND OPENING THE LEAK TIGHT ENCLOSURE AROUND THE INCORE SHAFT AREA, TO ALLOW A PORTION OF THE RWST INVENTORY TO ENTER THIS AREA.

[214] ZION 1 DOCKET 50-295 LER 90-016
DISCHARGE TANK FAILURE TO ISOLATE ON HI RAD SIGNAL.
EVENT DATE: 072790 REPORT DATE: 082790 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 2 (PWR)

(NSIC 219416) AT 1007 ON JULY 27, 1990, WATER WAS BEING DISCHARGED TO LAKE MICHIGAN FROM THE OA LAKE DISCHARGE TANK VIA THE NUMBER 2 DISCHARGE CANAL. A HIGH BACKGROUND IN THE LIQUID CANNISTER CAUSED ORT-PR04 RADIATION MONITOR TO SPURIOUSLY ALARM. THE INTERLOCK FUNCTION OF ORT-PR04 FLOW CONTROL VALVE (OFCV-WD08) FAILED TO ISOLATED THE DISCHARGE. THE RELEASE WAS TERMINATED BY MANUALLY CLOSING OFCV-WD08. THE CAUSE OF THE HIGH ALARM WAS DUE TO A HIGH BACKGROUND READING IN THE LIQUID CANNISTER OF ORT-PR04. THE ROOT CAUSE OF OFCV-WD08 FAILING TO ISOLATE COULD NOT BE DETERMINED. CORRECTIVE ACTIONS INVOLVED CLEANING THE LIQUID CANNISTER AND INSPECTING OFCV-WD08 FOR ANY SIGNS THAT COULD CAUSE IT NOT TO FUNCTION PROPERLY. THE VALVE WAS TESTED BEFORE THE RELEASE AND AFTER THE INSPECTION TO CONFIRM THE OPERABILITY OF THE VALVE. THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AT RISK. THE HIGH RADIATION ALARM WAS DUE TO HIGH BACKGROUND IN THE LIQUID CANNISTER ORT-PR04 AND NOT TO THE DISCHARGE WATER EXCEEDING RELEASE LIMITS. SAMPLES WERE TAKEN PRIOR TO THE DISCHARGE AND AFTER THE RELEASE WAS TERMINATED TO ASSURE THE SAFETY OF THE PUBLIC WAS NOT COMPROMISED.

[215] ZION 1 DOCKET 50-295 LER 90-017
TURBINE TRIP/REACTOR TRIP DUE TO PERSONNEL ERROR.
EVENT DATE: 081390 REPORT DATE: 091290 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 2 (PWR)

(NSIC 219465) UNIT 1 WAS AT STEADY STATE AT 1030 MWE. UNIT 2 WAS IN HOT SHUTDOWN, WITH PERIODIC TEST (PT)-5A, "REACTOR PROTECTION LOGIC REACTOR AT HOT SHUTDOWN" IN PROGRESS. DUE TO DIFFICULTIES ENCOUNTERED IN TESTING, THE NEED AROSE TO TRIP UNIT 2 LOCALLY. AN EXTRA NUCLEAR STATION OPERATOR RUSHED OUT OF THE CONTROL ROOM, WENT TO THE WRONG UNIT, AND TRIPPED UNIT 1. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR. ALL SAFETY SYSTEMS OPERATED AS DESIGNED. THERE WAS THEREFORE NO SAFETY SIGNIFICANCE TO THIS EVENT. CORRECTIVE ACTIONS INCLUDE TAILGATE SESSIONS, BETTER LABELING AT THE TURBINE PEDESTALS, REPAIRING THE TURBINE STOP VALVE BYPASS VALVES, AND AN INVESTIGATION TO DETERMINE THE FEASIBILITY OF BYPASSING THE TURBINE BEARING LIFT OIL PUMP SPEED SWITCH TRIP AT LOW TURBINE SPEED.

[216] ZION 2 DOCKET 50-304 LER 90-008
INADVERTENT START OF 2C CONTAINMENT SPRAY PUMP DUE TO PERSONNEL ERROR.
EVENT DATE: 072890 REPORT DATE: 082790 NSSS: WE TYPE: PWR

(NSIC 219417) THE ELECTRICAL MAINTENANCE (EM) DEPARTMENT WAS PERFORMING TROUBLESHOOTING ON THE 2C CONTAINMENT SPRAY (BE) PUMP DUE TO A FAILURE OF THE ENGINE TO START DURING THE PERFORMANCE OF PERIODIC TEST (PT)-6. THE 'A' ELECTRICIAN WANTED THE ENGINE TO BE CRANKED OVER TO MONITOR BATTERY VOLTAGE WHILE CRANKING. THE INTENT WAS TO LIFT A LEAD TO THE FUEL SHUTOFF SOLENOID, AND HAVE THE OPERATOR TURN THE START SWITCH. THIS SHOULD HAVE CAUSED THE ENGINE TO SIMPLY TURN OVER WITHOUT STARTING. INSTEAD, THE WRONG LEAD WAS LIFTED BY THE 'A' ELECTRICIAN, AND THE ENGINE STARTED WHEN THE NSO TURNED THE CONTROL SWITCH. THE 'A' ELECTRICIAN HAD THE NSO SECURE THE PUMP AFTER RUNNING APPROXIMATELY 15 SECONDS. THE CAUSE OF THE EVENT IS PERSONNEL ERROR, IN THAT THE 'A' ELECTRICIAN LIFTED THE WRONG LEAD, WHICH CAUSED THE DIESEL ENGINE TO START INSTEAD OF CRANKING OVER AS DESIRED. UNIT 2 WAS IN MODE 5 AND THE CONTAINMENT SPRAY SYSTEM IS NOT REQUIRED IN THIS MODE. THE PUMP WAS LINED UP FOR PT-6, WHICH PREVENTED ANY SPRAY DOWN OF CONTAINMENT. THE PUMP WAS PARTIAL CLEARED FOR SERVICE TO ALLOW FOR MAINTENANCE RUNS, THUS NO EQUIPMENT DAMAGE OCCURRED. THERE WAS THEREFORE NO

SAFETY SIGNIFICANCE TO THIS EVENT. THE IMPORTANCE OF COMPLETE AND ACCURATE COMMUNICATIONS WITH THE CONTROL ROOM WILL BE EMPHASIZED TO ALL MAINTENANCE GROUPS.

[217] ZION 2 DOCKET 50-304 LER 90-009
INADVERTENT AUTO-CLOSURE OF THE 2B DIESEL GENERATOR OUTPUT BREAKER AND BUS 249 MAIN FEED BREAKER.
EVENT DATE: 080990 REPORT DATE: 091090 NSSS: WE TYPE: PWR

(NSIC 219464) AT 0900 ON 08-09-90, THE 2B DIESEL GENERATOR (D/G) OUTPUT BREAKER AUTO-CLOSED WHILE ATTEMPTING TO SYNCHRONIZE THE 2B D/G TO BUS 249 DURING THE PERFORMANCE OF TECHNICAL STAFF SURVEILLANCE (TSS) 15.6.43-2. INITIAL INVESTIGATION CONCLUDED THAT A FAULTY BREAKER CONTROL SWITCH MAY HAVE CAUSED THE BREAKER TO AUTO-CLOSE. THE BREAKER CONTROL SWITCH WAS REPLACED AND TESTED TO VERIFY THE EVENT WOULD NOT REPEAT ITSELF. AT 0735 ON 08-10-90, THE 2B D/G OUTPUT BREAKER AND THE BUS 249 MAIN FEED BREAKER AUTO-CLOSED WHILE TESTING CONTINUED PER TSS 15.6.43-2. FURTHER INVESTIGATION IDENTIFIED A SPARE CABLE CONNECTING THE BUS 249 SYNCHRO-CHECK RELAY TO THE SAFEGUARDS ACTUATION CIRCUIT. WITH TEST JUMPERS INSTALLED PER TSS 15.6.43-2, A CIRCUIT PATH WAS COMPLETED WHICH PROVIDED DC VOLTAGE TO THE SYNCHRO-CHECK RELAY, THUS ALLOWING THE RELAY TO CLOSE IN THE BREAKER. THE CABLE WAS SUBSEQUENTLY DISCONNECTED AT EACH END PER THE DESIGN DRAWINGS. THE WIRING ERROR HAD NO ADVERSE AFFECT ON THE BUS EMERGENCY MODE OF OPERATION, ONLY THE BREAKER MANUAL FUNCTIONS WERE AFFECTED BY THE PRESENCE OF THE INSTALLED TEST JUMPERS IN CONJUNCTION WITH THE SPARE CABLE. CORRECTIVE ACTIONS INCLUDED A WALKDOWN TO IDENTIFY OTHER SIMILAR WIRING DISCREPANCIES ON ALL ESF BUSES ON BOTH UNITS. ONE OTHER DISCREPANCY WAS IDENTIFIED AND CORRECTED.

COMPONENT INDEX

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

COMPONENT INDEX

SYSTEM INDEX

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

SYSTEM INDEX

SYSTEM INDEX

KEYWORD INDEX

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

KEYWORD INDEX

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KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

VENDOR CODE INDEX

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SEE INSTRUCTIONS ON THE REVERSE

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11a. TYPE OF REPORT
Monthly Report

b. PERIOD COVERED (Inclusive dates)
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12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one month period identified on the cover of the document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting for revisions to those events occurring prior to 1984 are described in NRC Regulatory Guide 1.16 and NUREG-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.

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