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Licensee Event Report (LER) Compilation

For month of February 1992

Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

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NOTE TO
RECIPIENTS:

CANCELLATION OF NUREG/CR-2000

The LICENSEE EVENT REPORT (LER) COMPILATION will no longer be issued. The last report will include data for the month of March 1992, Vol. 11, No. 3.

Licensee Event Report information is maintained for the Nuclear Regulatory Commission at the Oak Ridge National Laboratory by the Nuclear Operations Analysis Center (NOAC). This information is maintained in the Sequence Coding and Search System database which can be searched in a number of ways to obtain nuclear power plant operating experience data.

Contact W.P. Poore of NOAC to obtain information from this database:

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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 Operations Analysis Center (NOAC) 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, its contents, or LER searches should be directed to:

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[1] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 88-024 REV 01
UPDATE ON INADVERTENT JARRING OF RELAY SENSITIVE TO MECHANICAL SHOCK RESULTS IN CLOSURE OF DECAY HEAT REMOVAL SUCTION VALVE AND LOSS OF DECAY HEAT REMOVAL SYSTEM FLOW.
EVENT DATE: 121988 REPORT DATE: 020792 NSSS: BW TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223917) ON DECEMBER 19, 1988 THE DECAY HEAT REMOVAL (DHR) SYSTEM INBOARD SUCTION VALVE (CV-1050) CLOSED RESULTING IN A LOSS OF DHR SYSTEM FLOW. FOLLOWING INDICATION THAT THE DHR SUCTION VALVE WAS CLOSING, THE PLANT OPERATOR FOLLOWED THE APPROPRIATE PROCEDURES TO SECURE THE OPERATING DHR PUMP. ACTIONS WERE THEN TAKEN WHICH RETURNED THE DHR SYSTEM TO OPERATION IN APPROXIMATELY 12 MINUTES. AT THE TIME OF THE EVENT, A CONTRACT ELECTRICIAN WAS PERFORMING EQUIPMENT INSPECTIONS IN THE ROOM WHICH CONTAINS A PANEL HOUSING THE CONTROL RELAYS FOR CV-1050. THIS INDIVIDUAL INADVERTENTLY JARRED THE PANEL HOUSING THE CONTROL RELAYS FOR CV-1050 AT APPROXIMATELY THE TIME OF THIS EVENT. THE CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE INADVERTENT OPENING OF THE NORMALLY CLOSED PERMISSIVE CONTACTS OF A CONTROL RELAY FOR CV-1050. AS DETERMINED DURING THE INVESTIGATION OF THIS EVENT, THE PERMISSIVE CONTACTS OF THIS RELAY ARE SENSITIVE TO MECHANICAL SHOCK. AS A RESULT OF THIS EVENT, A CAUTION LABEL HAS BEEN PLACED AT THIS CONTROL PANEL TO CAUTION AGAINST MECHANICAL AGITATION OF THE PANEL. A PLANT MODIFICATION HAS BEEN IMPLEMENTED TO REPLACE THIS RELAY WITH A MODEL LESS SENSITIVE TO MECHANICAL SHOCK. ADDITIONALLY, SAFETY-RELATED RELAYS OF THIS TYPE HAVE BEEN REVIEWED FOR POSSIBLE SAFETY OR OPERATIONAL PROBLEMS DUE TO SUSCEPTIB

[2] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 91-008 REV 01
UPDATE ON SAFETY RELATED INSTRUMENTATION NOT INSTALLED IN A SEISMICALLY QUALIFIED CONFIGURATION DUE TO AN INADEQUATE DESIGN CHANGE DOCUMENTATION REVIEW.
EVENT DATE: 082791 REPORT DATE: 010892 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 2 (PWR)
VENDOR: FOXBORO CO., THE

(NSIC 223717) ON AUGUST 28, 1991 AT 1523, ANO-1 ENTERED TECHNICAL SPECIFICATION LIMITING CONDITION FOR OPERATION (LCO) 3.0.3 BASED ON AN ENGINEERING DETERMINATION THAT CERTAIN SAFETY RELATED ANO-1 FOXBORO SPECIFICATION 200 INSTRUMENTS WERE INOPERABLE DUE TO INADEQUATE SEISMIC QUALIFICATION. ALTHOUGH THE CONDITION WAS DETERMINED TO BE APPLICABLE TO ANO-2 ALSO, THE EXTENT OF INVOLVEMENT WAS LESS AND REPAIRS WERE ABLE TO BE PERFORMED WITHIN THE FRAMEWORK OF SPECIFIC SYSTEM LCOS. A 7 DAY 'TEMPORARY WAIVER OF COMPLIANCE' FROM THE SHUTDOWN REQUIREMENTS OF THE APPLICABLE LCOS WAS GRANTED TO ANO-1 BY THE NRC TO ALLOW SUFFICIENT TIME TO MAKE REPAIRS WITHOUT SUBJECTING THE PLANT TO AN UNNECESSARY TRANSIENT. REPAIRS WERE COMPLETED FOR ANO-1 ON AUGUST 31, 1991 AND FOR ANO-2 ON SEPTEMBER 10, 1991. THE CAUSES OF THIS CONDITION WERE DETERMINED TO BE OBSCURE VENDOR DOCUMENTATION AND AN INADEQUATE PLANT MODIFICATION PROCESS REVIEW OF VENDOR DOCUMENTATION. TECHNICAL MANUALS FOR SEISMICALLY QUALIFIED FOXBORO SPECIFICATION 200 COMPONENTS WERE REVISED TO INCLUDE APPROPRIATE INSTRUCTIONS FOR MAINTAINING SEISMIC CONFIGURATION. DESIGN REVIEW REQUIREMENTS FOR DESIGN CHANGE PACKAGE DEVELOPMENT WILL BE CLARIFIED TO ENSURE AND SPECIFIC INSTRUCTIONS ARE PROVIDED CONCERNING THE QUALIFIED CONFIGURATION OF COMPONENTS OR SYSTEMS.

[3] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 91-011
INADEQUATE PROCEDURES FOR SETTING LUBE OIL PRESSURE CONTROL VALVES RESULTS IN OIL LEAKAGE AND POTENTIAL COMMON MODE FAILURE OF HIGH PRESSURE INJECTION PUMPS.
EVENT DATE: 101091 REPORT DATE: 111191 NSSS: BW TYPE: PWR

(NSIC 223784) ON OCTOBER 9, 1991, FOLLOWING AN OVERHAUL OF HIGH PRESSURE INJECTION (HPI) PUMP P36A, SIMULTANEOUS OPERATION OF P36A AND ITS AUXILIARY OIL PUMP P64A WAS REQUESTED FOR POST-MAINTENANCE TESTING OF REPAIRS TO THE LUBE OIL SYSTEM PRESSURE CONTROL VALVE. DURING THIS TEST RUN, SYSTEM ENGINEERING PERSONNEL OBSERVED THAT APPROXIMATELY ONE CUP OF OIL PER MINUTE WAS FLOWING FROM THE OUTBOARD PUMP BEARING AREA. A SIMILAR CONDITION WAS LATER FOUND TO EXIST ON PUMP P36C. ON OCTOBER 10, 1991, IT WAS DETERMINED THAT THIS CONDITION REPRESENTED A POTENTIAL PROBLEM THAT COULD HAVE RESULTED IN THE HPI PUMPS BECOMING INOPERABLE

DURING AN ESAS ACTUATION. THE CAUSE OF THIS CONDITION WAS LACK OF PROCEDURES FOR ADJUSTMENT OF THE LUBE OIL CONTROL VALVES ON THE HPI PUMPS. ADJUSTMENTS WERE MADE ON THE LUBE OIL SYSTEM FOR P36A AND P36C TO STOP THE OIL LEAKAGE. A NIGHT ORDER WAS ISSUED TO OPERATIONS PERSONNEL ADVISING THEM TO BE ALERT TO ANY OIL LEAKAGE AND TO INITIATE PROMPT CORRECTIVE ACTIONS IF LEAKAGE IS DETECTED. IN ADDITION, THE LUBE OIL SYSTEM DESIGN IS BEING EVALUATED TO DETERMINE IF LONG TERM USE OF THE CURRENTLY INSTALLED TYPE OF PRESSURE CONTROL VALVES IS APPROPRIATE.

[4] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 91-012
AUTOMATIC ACTUATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM DURING VENTILATION SYSTEM MAINTENANCE CAUSED BY HIGH AIRBORNE ACTIVITY WHICH RESULTED FROM AN INADEQUATE PRE-JOB EVALUATION.
EVENT DATE: 111191 REPORT DATE: 121191 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 2 (PWR)

(NSIC 223606) ON NOVEMBER 11, 1991 AN ACTUATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) OCCURRED DURING THE PERFORMANCE OF MAINTENANCE TO CHANGE THE FILTERS ON THE RADWASTE HEAT EXHAUST VENTILATION SYSTEM. ON NOVEMBER 11, 1991 AT 2035, WORKERS ENTERED THE VENTILATION SYSTEM AIR HANDLER TO BEGIN CHANGING THE PREFILTERS. APPROXIMATELY 20 SECONDS AFTER OPENING THE DOOR TO THE AIR HANDLER, THE CONTINUOUS AIR MONITOR BEGAN TO ALARM. AT 2059, REPLACEMENT OF THE REFILTERS WAS STOPPED AND THE DOOR TO THE AIR HANDLER WAS CLOSED. THE CONTROL ROOM ISOLATION OCCURRED AT 2100 HOURS WHEN THE COUNT RATE ON THE CONTROL ROOM VENTILATION RADIATION MONITOR EXCEEDED ITS ALARM SETPOINT OF 160 COUNTS PER MINUTE (CPM). THE CONTROL ROOM RADIATION MONITOR INDICATION QUICKLY RETURNED TO APPROXIMATELY 60 CPM AFTER THE ISOLATION. THE HIGH ACTIVITY IN THE AIR HANDLER WAS MOST LIKELY CAUSED BY BACK LEAKAGE THROUGH THE AIR HANDLER INLET DAMPER OF FISSION GASES INTRODUCED INTO THE SYSTEM BY THE CONDENSER VACUUM SYSTEM. THE FISSION GASES WERE THEN DRAWN INTO THE CONTROL ROOM DUCTWORK THROUGH EXISTING PILOT TUBE TRAVERSE ACCESS PORTS, SEAMS OR JOINTS IN DUCTWORK AND/OR LEAKING GASKETS IN EQUIPMENT, THUS CAUSING THE CONTROL ROOM ISOLATION. PROCEDURES WERE REVISED.

[5] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 91-013
AUTOMATIC ACTUATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM DUE TO A VALID RADIATION SIGNAL WHICH RESULTED FROM LEAKAGE FROM A LETDOWN FILTER DRAIN VALVE.
EVENT DATE: 111391 REPORT DATE: 121391 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 2 (PWR)

(NSIC 223607) ON NOVEMBER 13, 1991 AT 1307, AN AUTOMATIC ACTUATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM WAS INITIATED BY 2RE-8750-1, WHICH IS A RADIATION MONITOR LOCATED IN THE CONTROL ROOM VENTILATION INTAKE DUCTWORK. AT THE TIME OF THE ACTUATION, THE CONTROL ROOM VENTILATION SYSTEM WAS LINED UP IN THE RECIRCULATION MODE WITH THE SUPPLY ISOLATION DAMPER MANUALLY CLOSED. THE SUPPLY AND EXHAUST FANS, WHICH ARE NORMALLY SECURED WHEN THE CONTROL ROOM IS AUTOMATICALLY ISOLATED, WERE IN SERVICE SUPPLYING VENTILATION TO THE COMPUTER ROOM. AT 1022, THE REACTOR COOLANT SYSTEM (RCS) DEVELOPED A SMALL LEAK FROM A FILTER DRAIN VALVE. THIS LEAK WAS STOPPED AT 1034. AT 1253, THE PENETRATION ROOM EXHAUST SYSTEM, WHICH DRAWS FROM THE FILTER ROOM WHERE THE LEAK WAS LOCATED, WAS PLACED IN SERVICE. THIS SYSTEM EXHAUSTS DOWNWARD THROUGH A U-SHAPED VENTILATION EXHAUST PIPE (GOOSENECK) LOCATED ON THE SIDE OF THE CONTAINMENT BUILDING THAT IS ORIENTED TOWARD THE CONTROL ROOM NORMAL VENTILATION INTAKE DUCT. THE MOST LIKELY CAUSE OF THIS EVENT WAS THE ENTRY OF HIGH AIRBORNE ACTIVITY FROM THE LETDOWN FILTER DRAIN VALVE THROUGH THE GOOSENECK INTO THE INTAKE DUCTWORK. THE BACK PRESSURE ON THE CONTROL ROOM VENTILATION SUPPLY DAMPER CREATED BY THE VENTILATION SYSTEM LINEUP CAUSED THE DAMPER TO LEAK.

[6] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 90-024 REV 01
UPDATE ON INADEQUATE PREVENTIVE MAINTENANCE PROGRAM FOR STEAM TURBINE DRIVEN EMERGENCY FEEDWATER PUMP RESULTS IN DEGRADED TURBINE GOVERNOR SYSTEM AND SUBSEQUENT OVERSPEED TRIPS OF TURBINE.
EVENT DATE: 120590 REPORT DATE: 121091 NSSS: CE TYPE: PWR

VENDOR: WOODWARD GOVERNOR COMPANY

(NSIC 223620) ON DECEMBER 5, 1990 BASED ON EVALUATIONS OF TWO PREVIOUS EVENTS INVOLVING OVERSPEED TRIPS OF THE STEAM TURBINE DRIVEN EMERGENCY FEEDWATER PUMP, IT WAS CONCLUDED THAT THE CAUSE OF THE TRIPS HAD BEEN WATER SLUGGING OF THE TURBINE ON STARTUP DUE TO CONDENSATE ACCUMULATION IN THE STEAM SUPPLY LINE TO THE TURBINE. FOLLOWING ANOTHER OVERSPEED TRIP ON DECEMBEP 6, 1990, THE ACTUAL CAUSE FOR THE TURBINE TRIPS WAS FOUND TO BE SLUGGISH RESPONSE OF THE TURBINE GOVERNOR VALVE DUE TO A CONTAMINATED CONTROL OIL SYSTEM. THE ROOT CAUSE WAS CONSIDERED TO BE INADEQUACIES IN THE PREVENTIVE MAINTENANCE PROGRAM. THE PROGRAM DID NOT APPROPRIATELY ADDRESS AND MINIMIZE THE POTENTIAL EFFECTS OF OIL CONTAMINATION AND DEGRADATION OF GOVERNOR COMPONENTS OVER TIME. FOLLOWING THE LAST OVERSPEED TRIP, THE OIL AND OIL FILTER ASSEMBLY WERE CHANGED, A HYDRAULIC ACTUATOR WAS REPLACED AND A REMOTE SERVO VALVE AND CONTROL OIL TUBING WERE CLEANED. THE TURBINE IS BEING TESTED ON AN INCREASED FREQUENCY AND THE OIL QUALITY IS BEING MONITORED TO ENSURE IT IS NOT DEGRADING. LONG TERM ACTIONS INCLUDE PROCEDURE REVISIONS TO INCLUDE PERIODIC CLEANING AND/OR REPLACEMENT OF CONTROL OIL SYSTEM COMPONENTS. ADDITIONALLY, THE TURBINE OIL SYSTEM WILL BE CLEANED TO REMOVE VARNISH AND HARDENED OIL DEPOSITS DURING REFUELING.

[7] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 91-018
SERVICE WATER VALVE STUCK IN INTERMEDIATE POSITION CAUSES SERVICE WATER LOOP TO BE INOPERABLE RESULTING IN ENTRY INTO TECHNICAL SPECIFICATION 3.0.3.
EVENT DATE: 123091 REPORT DATE: 012492 NSSS: CE TYPE: PWR
VENDOR: CLOW CORP.

(NSIC 223859) ON DECEMBER 30, 1991, A SERVICE WATER MOTOR-OPERATED VALVE STUCK IN AN INTERMEDIATE POSITION DURING SYSTEM REALIGNMENT FROM SUPPLYING COOLING TOWER BASIN MAKEUP. THIS VALVE IS A BOUNDARY BETWEEN THE SEISMIC LOOP 1 SERVICE WATER RETURN HEADER AND THE NON-SEISMIC AUXILIARY COOLING WATER (ACW) SYSTEM. SERVICE WATER LOOP 1 WAS INOPERABLE BECAUSE ISOLATION FROM A PIPING BREAK IN ACW COULD NOT BE ASSURED IF A SEISMIC EVENT WERE TO HAVE OCCURRED. SINCE ROUTINE MAINTENANCE WAS IN PROGRESS ON ONE EMERGENCY FEEDWATER PUMP AND ONE CONTAINMENT HYDROGEN ANALYZER WHOSE OPERABILITY DEPENDED UPON SERVICE WATER SUPPLIED FROM LOOP 2, LOOP 1 BEING INOPERABLE RESULTED IN BOTH EMERGENCY FEEDWATER PUMPS AND BOTH HYDROGEN ANALYZERS BEING INOPERABLE AND THEREFORE REQUIRED ENTRY INTO TECHNICAL SPECIFICATION 3.0.3. ONE HYDROGEN ANALYZER WAS RESTORED TO OPERABILITY WITHIN TWENTY FIVE MINUTES. ONE EMERGENCY FEEDWATER PUMP WAS OPERABLE WITHIN FORTY TWO MINUTES TO ALLOW EXIT FROM TECHNICAL SPECIFICATION 3.0.3. THE STUCK VALVE WAS MANUALLY SHUT AFTER APPROXIMATELY SIXTEEN HOURS TO ALLOW EXIT FROM OTHER TECHNICAL SPECIFICATION ACTION STATEMENTS WITHOUT ANY TIME LIMITS HAVING BEEN EXCEEDED. THE ROOT CAUSE OF THE VALVE HAVING STUCK WILL BE DETERMINED AFTER PLANT CONDITIONS ALLOW REMOVAL, INSPECTION, AND REPAIR.

[8] ARNOLD DOCKET 50-331 LER 91-013
LACK OF BACKDRAFT DAMPERS IN STANDBY FILTER UNITS.
EVENT DATE: 110791 REPORT DATE: 122691 NSSS: GE TYPE: BWR

(NSIC 223726) ON NOVEMBER 7, 1991 DURING SYSTEM MAINTENANCE, IT WAS NOTED THAT BACKDRAFT DAMPERS DESIGNED TO BE IN PLACE IN THE CONTROL BUILDING STANDBY FILTER UNIT SYSTEM WERE NOT INSTALLED. MANUAL ADJUSTABLE DAMPERS WERE FOUND INSTALLED IN THEIR PLACE. SPECIAL TESTING PROVED THE SYSTEM REMAINED OPERABLE UNDER THE FAILURE MODE CONDITIONS. THE CAUSE FOR THE LACK OF INSTALLATION OF PROPER EQUIPMENT WAS A FAILURE BY THE CONTRACTOR TO FOLLOW DESIGN DOCUMENTS DURING CONSTRUCTION. THIS REPORT IS BEING VOLUNTARILY SUBMITTED FOR INFORMATIONAL PURPOSES.

[9] ARNOLD DOCKET 50-331 LER 91-012
REACTOR WATER CLEANUP SYSTEM ISOLATION DUE TO DEFORMED CONNECTOR CONTACTS.
EVENT DATE: 120791 REPORT DATE: 122691 NSSS: GE TYPE: BWR
VENDOR: AMPHENOL

(NSIC 223725) ON DECEMBER 7, 1991, AT 100% REACTOR POWER, A PRIMARY CONTAINMENT

ISOLATION SYSTEM (REACTOR WATER CLEANUP SYSTEM) ISOLATION OCCURRED. WHILE TAKING LOGS, A CONTROL ROOM OPERATOR NOTICED THE SYSTEMS FLOW SUMMER MODULE NOT FULLY FLUSH WITH THE FRONT OF THE PANEL. WHEN THE OPERATOR ATTEMPTED TO FULLY INSERT THE MODULE, A REACTOR WATER CLEANUP DIFFERENTIAL FLOW ALARM WAS RECEIVED. APPROXIMATELY 45 SECONDS LATER A REACTOR WATER CLEANUP SYSTEM ISOLATION WAS RECEIVED. ALL AUTOMATIC ACTIONS ASSOCIATED WITH THE ISOLATION OCCURRED AS DESIGNED AND WERE VERIFIED. THE CAUSE OF THE ISOLATION WAS DEFORMED CONTACTS ON THE CONNECTOR AT THE REAR OF THE SUMMER MODULE. THE CAUSE OF THE DEFORMED CONTACTS IS NOT KNOWN ALTHOUGH A BENT ALIGNMENT PLATE CAUSING MISALIGNMENT OF THE CONNECTOR IS SUSPECTED. THE CONTACTS WERE RETURNED TO THEIR NORMAL SHAPE AND CONTINUITY WAS VERIFIED DURING MOVEMENT OF THE MODULE.

[10] ARNOLD DOCKET 50-331 LER 92-001
POTENTIAL LOSS OF VESSEL LEVEL INSTRUMENTATION FUNCTIONS DUE TO INACCURACIES UNDER HIGH DRYWELL TEMPERATURE CONDITIONS.
EVENT DATE: 011592 REPORT DATE: 021492 NSSS: GE TYPE: BWR
VENDOR: ITT-BARTON
YARWAY CORP.

(NSIC 224037) ON 1/15/92, WITH THE PLANT OPERATING AT 86% POWER COASTING DOWN FOR A REFUEL OUTAGE, IT WAS DETERMINED THAT THE TRIPLE LOW LEVEL TRIP FUNCTION ASSOCIATED WITH YARWAY LEVEL SWITCHES MAY NOT BE FUNCTIONAL AT HIGH DRYWELL TEMPERATURES. ON 1/29/92, DURING REVIEW OF THE EFFECTS OF HIGH DRYWELL TEMPERATURES ON OTHER VESSEL LEVEL INSTRUMENTATION, IT WAS DETERMINED THAT THE HIGH PRESSURE COOLANT INJECTION (HPCI) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM AUTOMATIC HIGH LEVEL TRIPS MAY NOT FUNCTION AT HIGH DRYWELL TEMPERATURES. FOLLOWING VERIFICATION OF THE PROBLEMS ON 1/15 AND 1/29, THE ASSOCIATED INSTRUMENTATION WAS SATISFACTORILY RE-CALIBRATED TO ENSURE TRIPS WOULD OCCUR, AS REQUIRED, UNDER HIGH DRYWELL TEMPERATURE CONDITIONS. THE CAUSE OF THE PROBLEM IDENTIFIED ON 1/15 WAS INCORRECT INFORMATION CONTAINED IN GE SIL 299 COMPOUNDED BY AN INCORRECT STATEMENT IN INFORMATION PROVIDED BY THE INSTRUMENT VENDOR AND THE VERIFICATION CALCULATION PERFORMED BY GENERAL ELECTRIC. THE CAUSE OF THE PROBLEM IDENTIFIED ON 1/29 WAS FAILURE TO ACCOUNT FOR THE EFFECTS OF HIGH DRYWELL TEMPERATURES ON THE HIGH LEVEL TRIP INSTRUMENTATION WHEN THE INSTRUMENT LINE REFERENCE LEGS WERE RE-ROUTED DURING THE 1988 REFUEL OUTAGE. NEITHER OF THE INSTRUMENTATION PROBLEMS HAD AN EFFECT ON THE SAFE OPERATION OF THE PLANT.

[11] BEAVER VALLEY 1 DOCKET 50-334 LER 91-023 REV 01
UPDATE ON REACTOR TRIP DUE TO ELECTRO-HYDRAULIC CONTROL SYSTEM MALFUNCTION.
EVENT DATE: 072791 REPORT DATE: 121891 NSSS: WE TYPE: PWR

(NSIC 223677) ON 7/26/91 AT 1930 HOURS IN OPERATING MODE 1 AT 30 PERCENT POWER, THE UNIT BEGAN TO EXPERIENCE TURBINE GOVERNOR CONTROL OSCILLATIONS FROM THE TURBINE ELECTRO-HYDRAULIC CONTROL (EHC) SYSTEM. BETWEEN 1930 HOURS 7/26/91 AND 0655 HOURS 7/27/91, THREE LOAD REJECTIONS OF BETWEEN 100 AND 150 MWE WERE EXPERIENCED. OPERATORS STABILIZED THE PLANT AFTER EACH LOAD REJECTION. EHC TROUBLESHOOTING BY PLANT PERSONNEL AND CONTRACTOR TECHNICIANS BEGAN AFTER THE FIRST OCCURRENCE. AFTER THE THIRD OCCURRENCE, STATION MANAGEMENT DECIDED TO REMOVE THE TURBINE FROM SERVICE IN ACCORDANCE WITH THE ABNORMAL OPERATING PROCEDURE FOR A LOAD REJECTION. DURING PREPARATIONS FOR SHUTDOWN, A FOURTH LOAD REJECTION, ACCOMPANIED BY ERRATIC TURBINE GOVERNOR VALVE CONTROL, OCCURRED. THE REACTOR TRIPPED AT 0713 HOURS, DUE TO A LOW LEVEL COINCIDENT WITH A LOW FEEDWATER FLOW IN THE "B" STEAM GENERATOR. CONTROL ROOM PERSONNEL STABILIZED THE PLANT IN OPERATING MODE 3. THE SPURIOUS GOVERNOR VALVE SIGNALS WERE DUE TO A FAULTY CIRCUIT CARD IN THE EHC SYSTEM. THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC AS A RESULT OF THIS EVENT. ALL SHUTDOWN AND CONTROL RODS INSERTED, AND ALL PLANT SYSTEMS RESPONDED AS DESIGNED.

[12] BEAVER VALLEY 1 DOCKET 50-334 LER 91-030
INADEQUATE FILTER BANK SURVEILLANCE TESTING.
EVENT DATE: 110891 REPORT DATE: 120991 NSSS: WE TYPE: PWR

(NSIC 223611) ON 11/8/91 DURING A MANAGEMENT REEVALUATION OF SAFETY SYSTEM

FUNCTIONAL EXAMINATION (SSFE) AUDIT FINDINGS, A REPORTABLE FINDING, CONCERNING AN UNCONSERVATIVE TEST METHOD FOR DETERMINING THE SUPPLEMENTARY LEAKAGE COLLECTION AND RELEASE SYSTEM (SLCRS) FILTER REMOVAL EFFICIENCY WAS IDENTIFIED. THE TEST METHOD IN PLACE AT THAT TIME ONLY DETERMINED FILTER BANK REMOVAL EFFICIENCY AND EXCLUDED BYPASS DAMPER LEAKAGE EFFECTS WHICH WOULD LOWER THE SYSTEMS OVERALL EFFICIENCY. THEREFORE, THE SYSTEMS' OVERALL REMOVAL EFFICIENCY MAY NOT HAVE MET TECHNICAL SPECIFICATION REQUIREMENTS. UPON ISSUANCE OF THE SSFE FINDING, IT WAS DETERMINED THAT IF BYPASS DAMPER LEAKAGE WAS LESS THAN 500 CFM THEN REQUIRED LIMITS WOULD NOT HAVE BEEN EXCEEDED. ACTUAL LEAKAGE MEASUREMENTS VERIFIED THAT THIS LEAK RATE WAS NOT EXCEEDED. AFTER REVISING THE TEST METHOD TO ACCOUNT FOR BYPASS DAMPER LEAKAGE, BOTH TRAINS OF SLCRS FILTER BANKS WERE SATISFACTORILY TESTED. THE SSFE IS AN ONGOING INTERNAL SELF-ASSESSMENT PROCESS.

[13] BEAVER VALLEY 1 DOCKET 50-334 LER 91-031
AUTO START OF 1B RIVER WATER PUMP DURING MAINTENANCE ACTIVITIES.
EVENT DATE: 111491 REPORT DATE: 121291 NSSS: WE TYPE: PWR

(NSIC 223612) ON 11/14/91, WITH THE UNIT IN OPERATING MODE 5 (COLD SHUTDOWN), ELECTRICIANS BEGAN PERFORMANCE OF A PREVENTIVE MAINTENANCE PROCEDURE (PMP) ON CUBICLE E14 ON THE E1A 4160 VOLT EMERGENCY BUS. THIS CUBICLE IS ONE OF THE BREAKERS USED FOR THE 1C RIVER WATER (RW) PUMP. DURING THE PERFORMANCE OF THE PMP, A SPRING CLIP WAS OBSERVED LYING ON THE CUBICLE FLOOR NEAR THE CELL SWITCH LINKAGE. WHILE REPLACING THE CLIP, THE ELECTRICIANS INADVERTENTLY ACTUATED THE CELL SWITCH. BECAUSE OF A MECHANICAL INTERLOCK, THIS CAUSED THE RUNNING 1A RW PUMP TO TRIP AND THE STANDBY 1B RW PUMP TO START ON LOW HEADER PRESSURE. THE CAUSE FOR THIS EVENT WAS PERSONNEL ERROR. THE 1A RW PUMP WAS RESTARTED AND THE 1B RW PUMP WAS SHUTDOWN. THIS EVENT WILL BE REVIEWED BY MAINTENANCE PERSONNEL. THERE WERE NO SAFETY IMPLICATIONS AS A RESULT OF THIS EVENT. THE STANDBY RW PUMP IMMEDIATELY STARTED UPON LOSS OF THE RUNNING RW PUMP. RIVER WATER COOLING FLOW WAS MAINTAINED AT ALL TIMES.

[14] BEAVER VALLEY 1 DOCKET 50-334 LER 91-032
INADEQUATE VENTILATION FLOW FROM HIGH HEAD SAFETY INJECTION PUMP CUBICLES.
EVENT DATE: 120391 REPORT DATE: 010292 NSSS: WE TYPE: PWR
VENDOR: AMERICAN WARMING & VENTILATING INC.

(NSIC 223848) ON 12/3/91 WITH THE UNIT IN POWER OPERATION AT 100% POWER, VENTILATION CONCERNS INVOLVING THE HIGH HEAD SAFETY INJECTION (HHSI) PUMPS LED TO THE DETERMINATION THAT THE PLANT'S SAFE SHUTDOWN CAPABILITY WAS POTENTIALLY DEGRADED. THE ACTUAL MEASURED VENTILATION FLOW FROM EACH HHSI PUMP CUBICLE WAS FOUND TO BE LESS THAN THE FLOW-RATE REQUIRED BY THE CALCULATIONS TO MAINTAIN MOTOR TEMPERATURES WITHIN REQUIRED ENVIRONMENTAL QUALIFICATION LIMITS. AN INVESTIGATION DETERMINED THAT A MANUAL ISOLATION DAMPER, COMMON TO ALL THREE HHSI PUMPS, WAS FAILED IN A PARTIALLY CLOSED POSITION. THE DAMPER WAS REPAIRED AND PLACED IN THE PROPER POSITION TO MEET FLOW REQUIREMENTS. THE NUCLEAR REGULATORY COMMISSION WAS NOTIFIED IN ACCORDANCE WITH 10 CFR 50.72.B.2.III.A AND THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10 CFR 73.A.2.V.A AS A CONDITION THAT POTENTIALLY COULD ALONE HAVE PREVENTED SAFE REACTOR SHUTDOWN. THERE WAS NO ADVERSE IMPACT TO THE SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT.

[15] BEAVER VALLEY 1 DOCKET 50-334 LER 92-001
FAILURE TO DETERMINE STROKE TIMES FOR CONTAINMENT ISOLATION VALVES.
EVENT DATE: 010792 REPORT DATE: 020692 NSSS: WE TYPE: PWR

(NSIC 223939) ON 12/8/91, DURING THE PERFORMANCE OF CONTAINMENT ISOLATION VALVE SURVEILLANCE TESTING, OPERATIONS PERSONNEL NOTED THAT THE PROCEDURE ONLY REQUIRED TIMING VALVE TV-SV-100A, MAIN CONDENSER AIR EJECTOR DISCHARGE TO CONTAINMENT, TO THE OPEN POSITION. THIS VALVE HAS TWO FUNCTIONS: 1) OPEN ON A HIGH RADIATION CONDITION IN THE MAIN CONDENSER AND DIVERT AIR EJECTOR EXHAUST INTO THE REACTOR CONTAINMENT BUILDING; 2) CLOSE ON A CONTAINMENT ISOLATION PHASE "B" (CIB) SIGNAL. THE OPERATORS BELIEVED THE VALVE SHOULD ALSO BE TIMED CLOSED. THEY IMMEDIATELY TIMED IT CLOSED AND DETERMINED COMPLIANCE WITH THE TECHNICAL SPECIFICATION ISOLATION TIME. THE ASME INSERVICE TEST PROGRAM COORDINATOR WAS REQUESTED TO

DETERMINE IF TIMING THE VALVE BOTH OPEN AND CLOSED WAS REQUIRED. ON 1/7/92, THE ASME INSERVICE TEST PROGRAM COORDINATOR DETERMINED THAT THE PROCEDURE SHOULD HAVE REQUIRED TIMING THE VALVE IN BOTH DIRECTIONS. A REVIEW OF OTHER CONTAINMENT ISOLATION VALVE TESTING THUS FAR HAS IDENTIFIED ONE ADDITIONAL DISCREPANCY. ONE VALVE DID NOT HAVE THE CLOSING STROKE TIME RECORDED DURING THE LAST OUTAGE DUE TO A PROCEDURE ERROR. THE TEST PROCEDURES FOR BOTH THESE VALVES WILL BE REVISED. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. THE SERIES ISOLATION VALVE FOR EACH OF THE INVOLVED VALVES WERE VERIFIED TO HAVE BEEN OPERABLE DURING THE TIME OF THE EVENT.

[16]   BEAVER VALLEY 2   DOCKET 50-412   LER 91-005
REACTOR TRIP DUE TO SPURIOUS COMPONENT ACTUATION.
EVENT DATE: 112691   REPORT DATE: 122391   NSSS: WE   TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 223683) ON 11/26/91, WITH THE UNIT IN POWER OPERATION AT 100 PERCENT REACTOR POWER, A LOAD REJECTION (CLOSURE OF THE GOVERNOR AND INTERCEPTOR VALVES) OCCURRED AT 1143 HOURS. THE CLOSURE OF THE GOVERNOR AND INTERCEPTOR VALVES WAS THE RESULT OF A SIGNAL FROM THE OVERSPEED PROTECTION CONTROL (OPC) CIRCUIT. THE LOAD REJECTION CAUSED A RAPID REACTOR COOLANT SYSTEM (RCS) HEATUP OF THREE DEGREES. THE CONTROL RODS, IN AUTOMATIC, BEGAN TO STEP INTO THE CORE, HOWEVER, A REACTOR TRIP ON POWER RANGE NEUTRON FLUX NEGATIVE RATE OCCURRED AT 1143 HOURS. THIS TRIP WAS THE RESULT OF THE COMBINED EFFECTS OF THE REACTOR FUEL (DOPPLER) AND MODERATOR TEMPERATURE COEFFICIENTS DUE TO THE RCS HEATUP RESULTING FROM THE LOAD REJECTION. OPERATIONS PERSONNEL STABILIZED THE PLANT IN HOT SHUTDOWN. AN INVESTIGATION REVEALED THAT A SOLID STATE RELAY IN THE OPC CIRCUIT INADVERTENTLY ACTUATED CAUSING THE LOAD REJECTION. THE FAILED RELAY WAS REPLACED. THERE WERE NO SAFETY IMPLICATIONS AS RESULT OF THIS EVENT. THE REACTOR CONTROL SYSTEM AND THE AUXILIARY FEEDWATER SYSTEM ACTUATED AS DESIGNED UPON RECEIPT OF THE REACTOR TRIP SIGNAL AND LOW-LOW STEAM GENERATOR LEVEL SIGNALS.

[17]   BIG ROCK POINT   DOCKET 50-155   LER 91-007
FAILURE OF SOLENOID TO CLOSE CONTAINMENT ISOLATION VALVE ON LOSS OF POWER.
EVENT DATE: 112391   REPORT DATE: 121891   NSSS: GE   TYPE: BWR
VENDOR: ASCO VALVES

(NSIC 223665) BIG ROCK POINT TECHNICAL SPECIFICATION 3.4.3(B) REQUIRES THAT "NORMALLY OPEN LINES, CARRYING FLUIDS INTO THE CONTAINMENT SPHERE, SHALL BE EQUIPPED WITH CHECK VALVES TO PREVENT BACKFLOW UPON LOSS OF INWARD PROPELLENT FORCE. IN ADDITION, THESE LINES SHALL BE CAPABLE OF BEING SECURED BY MANUALLY-OPERATED GATE VALVES OR BY AIR-OPERATED CONTROL VALVES. THE LATER SHALL CLOSE UPON AIR OR POWER FAILURE....". CONTRARY TO THE ABOVE, AT 0415 ON NOVEMBER 23, 1991, DEMINERALIZED WATER VALVE CV-4105 FAILED TO CLOSE VIA A HAND SWITCH IN THE CONTROL ROOM DURING THE ROUTINE PERFORMANCE OF A 90 DAY SURVEILLANCE PROCEDURE. THE SHIFT SUPERVISOR WAS NOTIFIED, AND VERIFIED BY VISUAL INSPECTION THAT THE VALVE WAS STILL OPEN. AIR PRESSURE ON THE VALVE OPERATOR WAS ALSO OBSERVED, SUGGESTING A LACK OF RESPONSE AT THE SOLENOID VALVE, SV-4897, WHEN THE CONTROL ROOM SWITCH WAS ACTUATED. CV-4105 WAS CLOSED BY VENTING THE AIR PRESSURE FROM THE CV-4105 OPERATOR. THE CAUSE OF THE FAILURE HAS BEEN ATTRIBUTED TO THE SOLENOID VALVE, SV-4897. SV-4897 WAS REPAIRED, TESTED AND DECLARED OPERABLE ON NOVEMBER 27, 1991. TO PREVENT RECURRENCE, AN ENGINEERING PROJECT WILL BE INITIATED THAT INCLUDES TESTING, INSPECTION AND/OR THE REPLACEMENT OF ALL SIMILAR SOLENOID VALVES IN USE AT BIG ROCK POINT.

[18]   BIG ROCK POINT   DOCKET 50-155   LER 91-009
RETURN TO CRITICALITY DURING A REACTOR SHUTDOWN EVOLUTION.
EVENT DATE: 113091   REPORT DATE: 011392   NSSS: GE   TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.
        WORCESTER CONTROLS CORP.

(NSIC 223785) IN PREPARATION FOR THE UPCOMING REFUELING OUTAGE, AN ORDERLY REACTOR SHUTDOWN COMMENCED AT 2157 ON NOVEMBER 29, 1991. POWER WAS REDUCED BY INSERTING CONTROL RODS, AND AT 2333 THE TURBINE GENERATOR WAS REMOVED FROM

SERVICE. AROUND MIDNIGHT, A SHIFT TURNOVER WAS CONDUCTED BETWEEN THE CONTROL ROOM OPERATORS. DURING THIS PERIOD OF TIME, THE CONTROL RODS WERE NOT BEING INSERTED. THE REACTOR POWER BEGAN TO TURN AT 0011, AND AT 0013 THE OPERATOR RESPONSIBLE FOR REACTIVITY CONTROL RESPONDED BY INSERTING CONTROL RODS. POWER WAS REDUCED, AND ALL-RODS-ALL-IN WAS REACHED AT 0330, NOVEMBER 30, 1991. TO PREVENT RECURRENCE, A REACTIVITY MANAGEMENT PROGRAM HAS BEEN FORMED. MANAGEMENT WILL ALSO UNDERTAKE NEW INITIATIVES THAT WILL PROVIDE THEIR EXPECTATIONS TO THE OPERATING CREWS.

[19] BIG ROCK POINT DOCKET 50-155 LER 91-008
CONTROL ROD DRIVE REPLACEMENT DELAY.
EVENT DATE: 121091 REPORT DATE: 121691 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223666) BIG ROCK POINT TECHNICAL SPECIFICATION 7.5.7 REQUIRES THAT "....THE CONTROL ROD DRIVE THAT WAS REMOVED SHALL WITHOUT DELAY BE REPLACED BY A SPARE CONTROL ROD DRIVE OR THE ORIGINAL CONTROL ROD DRIVE SHALL BE REINSTALLED". CONTRARY TO THE ABOVE, ON DECEMBER 10, 1991 AT 0330, A CONTROL ROD DRIVE WAS REMOVED FROM THE REACTOR, AND A SPARE DRIVE NOR THE ORIGINAL DRIVE COULD BE REINSTALLED. PROBLEMS WITH ALIGNMENT OF THE DRIVES CAUSED THE DELAY TO OCCUR. SINCE THE ALIGNMENT PROBLEM REQUIRED ADDITIONAL TIME TO INVESTIGATE, A BLANK FLANGE WAS INSTALLED AS A TEMPORARY SAFEGUARD TO PREVENT A PATH FOR POSSIBLE DRAINAGE OF THE REACTOR. LATER, ON DECEMBER 10, 1991, FOLLOWING REVIEW OF THE INSTALLATION DIFFICULTIES, PERSONNEL RE-INSTALLED THE SPARE DRIVE. THE FACILITY IS PRESENTLY IN A MAJOR REFUELING OUTAGE, AND THE REACTOR WAS DEFUELED AT THE TIME OF OCCURRENCE. THE CAUSE OF THE ALIGNMENT TROUBLE WAS WITH THE HOIST AND PULLEY ASSEMBLY USED TO INSTALL THE CONTROL ROD DRIVE TO THE REACTOR VESSEL. THE LOCATION OF THE WORK AND FIRST TIME EXPERIENCE FOR SOME OF THE PERSONNEL PERFORMING THE INSTALLATION ALSO CONTRIBUTED TO THE DELAY. TO PREVENT RECURRENCE, THE EVENT WAS REVIEWED WITH CRD CREW PERSONNEL STRESSING PROPER INSTALLATION TECHNIQUE, PRIOR TO RESUMING FURTHER DRIVE CHANGE-OUTS. PROCEDURE IMPROVEMENTS ARE ALSO BEING INVESTIGATED.

[20] BIG ROCK POINT DOCKET 50-155 LER 91-011
FAILURE TO ESTABLISH AIR LOCK INTERLOCK REQUIRED FOR CONTAINMENT INTEGRITY.
EVENT DATE: 122391 REPORT DATE: 012092 NSSS: GE TYPE: BWR

(NSIC 223819) BIG ROCK POINT TECHNICAL SPECIFICATION 3.4.2(A) REQUIRES THAT: "(CONTAINMENT) ACCESS AIR LOCKS SHALL HAVE TWO IN-SERIES GASKETED DOORS, WHICH SHALL BE INTERLOCKED TO INSURE THAT AT LEAST ONE DOOR IS LOCKED CLOSED AT ALL TIMES WHEN CONTAINMENT INTEGRITY IS REQUIRED". ON DECEMBER 22, 1991 AT 2255, THE INNER AND OUTER DOORS OF THE ESCAPE LOCK WERE CLOSED AFTER INSTRUMENT CABLES USED FOR REACTOR VESSEL EXAMINATIONS WERE REMOVED. ON DECEMBER 23, 1991 AT 0925 THE OPERATIONS DEPARTMENT RESTORED CONTAINMENT INTEGRITY AFTER A VISUAL INSPECTION THAT BOTH DOORS WERE CLOSED. CONTRARY TO THE TECHNICAL SPECIFICATION, ON DECEMBER 25, 1991 AT 09.0 DURING A DAILY CHECK PERFORMED BY OPERATIONS, THE INTERLOCK WAS DISCOVERED TO BE DISABLED. THE INTERLOCK WAS IMMEDIATELY ENABLED AND REPORTED TO THE SHIFT SUPERVISOR. THERE WERE NO OPEN DOOR ALARMS (EXCEPT ROUTINE TESTING) REPORTED BY SECURITY DURING THE TIME THE INTERLOCK WAS DISABLED. THE CAUSE OF THE INTERLOCK NOT BEING REESTABLISHED IS PERSONNEL ERROR. THIS IS CONSIDERED TO BE AN ISOLATED INCIDENT. TO PREVENT RECURRENCE, PROCEDURES AND SHIFT ROUTINES ARE BEING EVALUATED.

[21] BIG ROCK POINT DOCKET 50-155 LER 92-001
BRITTLE MOTOR LEAD WIRES DISCOVERED IN MAIN STEAM ISOLATION VALVE.
EVENT DATE: 010492 REPORT DATE: 020692 NSSS: GE TYPE: BWR
VENDOR: LIMITORQUE CORP.

(NSIC 223924) ON JANUARY 4, 1992 AT 0800, BRITTLE AND CRACKED MOTOR WIRE LEADS WERE DISCOVERED IN THE MAIN STEAM ISOLATION VALVE OPERATOR (VOP-7050) AS A RESULT OF PLANNED MAINTENANCE DURING A SCHEDULED REFUELING OUTAGE. FOLLOWING EVALUATION AND A CORRECTIVE ACTION REVIEW BOARD MEETING ON THE SUBJECT, THE NRC OPERATING CENTER WAS NOTIFIED OF THE EVENT ON JANUARY 8, 1992 AT 0945 PURSUANT TO 10 CFR 50.72(B)(2)(III). THE CAUSE OF THE DEGRADATION HAS BEEN ATTRIBUTED TO EXPOSURE

OF THE MOTOR WIRE LEADS IN THE LIMIT SWITCH HOUSING. ALTHOUGH NOT CONFIRMED, THIS TEMPERATURE RISE IS BELIEVED TO BE DUE TO CONDUCTION OF HEAT FROM THE MAIN STEAM PIPING TO THE VALVE OPERATOR WHICH ACCELERATED THE AGING OF THE WIRE INSULATION. IMMEDIATE CORRECTIVE ACTIONS WILL INCLUDE AGE SENSITIVITY COMPARISONS OF MOTOR LEADS TO OTHER COMPONENTS IN THE VALVE OPERATOR, DIAGNOSTIC TESTING OF THE VALVE OPERATOR TO ACCESS PERFORMANCE, INSPECTION OF SIMILAR VALVE OPERATORS IN HIGH TEMPERATURE ENVIRONMENTS, AND REPLACEMENT OF THE MOTOR, LIMIT SWITCH AND TORQUE SWITCH IN THE MSIV OPERATOR. SUBSEQUENT ACTIONS TO PREVENT RECURRENCE INCLUDE INSTALLATION OF DEVICES TO MONITOR TEMPERATURE INSIDE SWITCH HOUSINGS OF THE MSIV AND SIMILAR OPERATORS, AND EVALUATION OF EEQ ITEMS IN SIMILAR ENVIRONMENT.

[22] BIG ROCK POINT DOCKET 50-155 LER 92-003
MAXIMUM STAY TIME NOT PROVIDED ON RWP IN LIEU OF DIRECT, CONTINUOUS SURVEILLANCE BY RADIATION PROTECTION QUALIFIED PERSONNEL.
EVENT DATE: 010892 REPORT DATE: 011092 NSSS: GE TYPE: BWR

(NSIC 223925) BIG ROCK POINT TECHNICAL SPECIFICATION 6.12.2 REQUIRES THAT "...DOORS SHALL REMAIN LOCKED EXCEPT DURING PERIODS OF ACCESS BY PERSONNEL UNDER AN APPROVED RWP WHICH SHALL SPECIFY THE DOSE RATE LEVELS IN THE IMMEDIATE WORK AREAS AND THE MAXIMUM ALLOWABLE STAY TIME FOR INDIVIDUALS IN THAT AREA. IN LIEU OF THE STAY TIME SPECIFICATION OF THE RWP, DIRECT OR REMOTE (SUCH AS CLOSED CIRCUIT TV CAMERAS) CONTINUOUS SURVEILLANCE MAY BE MADE BY PERSONNEL QUALIFIED IN RADIATION PROTECTION PROCEDURES. CONTRARY TO THE TECHNICAL SPECIFICATION, OPERATIONS PERSONNEL NOT QUALIFIED IN RADIATION PROTECTION PROCEDURES ENTERED A HIGH RADIATION AREA (> 1000 MR/HR) WITHOUT CONTINUOUS SURVEILLANCE AND THE MAXIMUM STAY TIME WAS NOT SPECIFIED ON THE RWP. THIS EVENT OCCURRED ON JANUARY 8, 1992 AT 0535 DURING A PLANNED REFUELING OUTAGE. AFTER INTERVIEWING THE HEALTH PHYSICS PERSONNEL INVOLVED, IT WAS APPARENT THAT THE INDIVIDUALS WERE UNAWARE OF THE REQUIREMENT. INADEQUATE EMPHASIS DURING TRAINING AND THE INFREQUENT SITUATIONS ENCOUNTERED WHERE THIS REQUIREMENT APPLIES CONTRIBUTED TO THE EVENT. CORRECTIVE ACTION WILL BE DIRECTED AT INFORMING HIGH RADIATION AREA QUALIFIED INDIVIDUALS OF THE REQUIREMENTS AND REMOVING AMBIGUITIES THAT EXIST IN THE PROGRAM.

[23] BIG ROCK POINT DOCKET 50-155 LER 92-002
AS-FOUND SETPOINT FAILURE (LOW) OR LIQUID POISON RELIEF VALVE RV-5049.
EVENT DATE: 010992 REPORT DATE: 020792 NSSS: GE TYPE: BWR
VENDOR: LONERGAN, J.E., CO.

(NSIC 223909) ON JANUARY 9, 1992 @ 2300, TESTING RESULTS SHOWED THAT A LIQUID POISON SYSTEM (LPS) EQUALIZATION LINE RELIEF VALVE DID NOT MEET THE AS-FOUND RELIEF PRESSURE ACCEPTANCE CRITERIA (1655 PSIG V9 A REQUIRED 1950-2050 PSIG). DUE TO EXCESSIVE SEAT LEAKAGE, THE PRESSURE TEST WAS NOT ABLE TO ACCURATELY DETERMINE THE LIFT PRESSURE. THIS CONDITION COULD HAVE AFFECTED THE VOLUME OF POISON SOLUTION DELIVERED TO THE CORE AND THE ESTABLISHMENT OF A SIPHON IN THE EQUALIZING LINE THAT ASSISTS IN DISCHARGING THE CONTENTS OF THE POISON TANK. THE PLANT IS IN COLD SHUTDOWN AND THE FUEL HAS BEEN REMOVED FROM THE REACTOR AND PLACED IN THE SPENT FUEL POOL. TO CORRECT THE DEFICIENCY, THE FAILED VALVE WAS REPLACED WITH ANOTHER LIKE RELIEF VALVE AND WAS ACCEPTED AFTER TESTING. CONSERVATIVELY ASSUMING THE AS-FOUND VALUE WAS THE LIFT PRESSURE OF THE VALVE, AN ANALYSIS WAS PERFORMED TO ESTIMATE THE ADEQUACY OF THE NITROGEN SYSTEM TO SUPPLY SUFFICIENT LIQUID DISPLACEMENT FROM THE LPS TANK TO INITIATE A SIPHON. ASSUMING THE NITROGEN SYSTEM PRESSURE WAS REDUCED TO 1500 PSIA, THE ANALYSIS CONCLUDED THAT SUFFICIENT NITROGEN WAS AVAILABLE TO ESTABLISH THE REQUIRED SIPHON FOR LIQUID POISON INJECTION.

[24] BRAIDWOOD 1 DOCKET 50-456 LER 91-013
UNTESTED RELAY CONTACTS CAUSE DIESEL GENERATOR INOPERABILITY.
EVENT DATE: 111491 REPORT DATE: 121291 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BRAIDWOOD 2 (PWR)

(NSIC 223653) DURING A NRC ELECTRICAL DISTRIBUTION SYSTEM FUNCTIONAL INSPECTION (EDSFI) AT LA SALLE STATION, A TESTING DEFICIENCY WAS IDENTIFIED ON SOME RELAYS

ASSOCIATED WITH THE 4160 VOLT SAFETY-RELATED BUSSES. ON NOVEMBER 14, 1991 IT WAS RECOGNIZED THAT SOME RELAY CONTACTS ASSOCIATED WITH THE ESF 4160 VOLT BUSSES HAD ALSO NOT BEEN TESTED AT BRAIDWOOD STATION. SURVEILLANCE REQUIREMENT 4.8.1.1.2.F.4.A REQUIRES VERIFICATION OF DE-ENERGIZATION OF THE ENGINEERED SAFEGUARD FEATURES (ESF) BUSSES AND LOAD SHEDDING FROM THE ESF BUSSES. THE EXISTING SURVEILLANCE PROCEDURE DID NOT VERIFY THAT ALL BREAKERS ON THE ESF BUSS WOULD OPEN. A FAILURE OF THESE BREAKERS TO OPEN DURING A UNDERVOLTAGE CONDITION WOULD PREVENT AUTOMATIC CLOSURE OF THE DIESEL GENERATOR (DG) OUTPUT BREAKER. AT 1510 THIS CONDITION WAS CONSIDERED UNACCEPTABLE AND ALL DG'S WERE DECLARED INOPERABLE. A TEMPORARY WAIVER OF COMPLIANCE (TWC) WAS REQUESTED TO DELAY, FOR 48 HOURS, THE ACTIONS REQUIRED BY THE APPLICABLE TECHNICAL SPECIFICATIONS. AT 1845 THE NRC APPROVED THE TWC REQUEST. ON NOVEMBER 15, 1991 THE 2A DG AND 1A DG WERE DECLARED OPERABLE AFTER TESTING WAS SATISFACTORILY COMPLETED. ON NOVEMBER 16, 1991 TESTING OF THE 1B AND 2B DG WAS SATISFACTORILY COMPLETED. THE CAUSE OF THE EVENT WAS A SURVEILLANCE PROCEDURAL DEFICIENCY. THE APPLICABLE SURVEILLANCE TESTING PROCEDURES WILL BE REVISED.

[25]   BRAIDWOOD 1   DOCKET 50-456   LER 92-001
TECHNICAL SPECIFICATION VIOLATION CAUSED BY PROCEDURE DEFICIENCY.
EVENT DATE: 011492   REPORT DATE: 021192   NSSS: WE   TYPE: PWR

(NSIC 224048) ON 1/12/92 THE 0B HYDROGEN RECOMBINER WAS TAKEN OUT-OF-SERVICE (OOS) FOR INSPECTION AND PLANNED MAINTENANCE. ON 1/14/92 THE 1A DIESEL GENERATOR (DG) WAS SCHEDULED FOR A MAINTENANCE OUTAGE. THE ACTION STATEMENTS, APPLICABLE TO AN INOPERABLE DIESEL GENERATOR, OF TECH SPEC 3.8.1.1 WERE REVIEWED. ACTION STATEMENT C.1 WAS APPLICABLE AND REQUIRED THAT, "WITH ONE DIESEL INOPERABLE, VERIFY THAT: ALL REQUIRED SYSTEMS, SUBSYSTEMS, TRAINS, COMPONENTS AND DEVICES THAT DEPEND ON THE REMAINING OPERABLE DIESEL GENERATOR AS A SOURCE OF EMERGENCY POWER ARE ALSO OPERABLE." TO ASSIST IN COMPLYING WITH THE REQUIREMENTS CONTAINED IN THIS ACTION STATEMENT, LIMITING CONDITION FOR OPERATION ACTION REQUIREMENT (LCOAR) PROCEDURE 1BWOS 8.1.1-1A WAS USED. THE GUIDANCE FOR REQUIRED SUBSYSTEMS IN THE LCOAR PROCEDURE DID NOT SPECIFICALLY INLCUDE THE 0B HYDROGEN RECOMBINER. BASED ON THIS INFORMATION, MAINTENANCE WAS ALLOWED TO PROCEED ON THE 1A DG. AT 0440, THE 1A DG WAS DECLARED INOPERABLE. AT 1234 ON 1/15/92 THE 0B HYDROGEN RECOMBINER WAS RETURNED TO SERVICE. ON 1/21/92 AFTER RE VIEWING THE REQUIREMENTS FOR SYSTEM OPERATION FOLLOWING A LOSS-OF-COOLANT ACCIDENT, IT WAS DETERMINED THAT ACTION STATEMENT C.1 OF TECH SPEC 3.8.1.1 WAS APPLICABLE TO THE HYDROGEN RECOMBINERS. AT THIS TIME, THE EVENT WAS DETERMINED TO BE REPORTABLE IN ACCORDANCE WITH 10CFR50.73(A)(2)(I)(B).

[26]   BRAIDWOOD 2   DOCKET 50-457   LER 91-006
GENERATOR TRIP CAUSED BY SPURIOUS ACTUATION OF NEUTRAL GROUND RELAY.
EVENT DATE: 120191   REPORT DATE: 122391   NSSS: WE   TYPE: PWR

(NSIC 223754) AT 1738 ON 12/1/91 A GENERATOR NEUTRAL GROUND OVERCURRENT BACKUP PROTECTIVE RELAY ACTUATED AND TRIPPED THE UNIT 2 MAIN GENERATOR. A TURBINE AND REACTOR TRIP FOLLOWED AS DESIGNED. OPERATORS PERFORMED APPLICABLE STEPS OF THE REACTOR TRIP RESPONSE PROCEDURES AND STABILIZED THE PLANT. THE GENERATOR TRIP WAS CAUSED BY THE NEUTRAL GROUND BACKUP RELAY. THE DESIGN PURPOSE OF THIS RELAY IS TO ACTUATE WHEN AN UNBALANCED VOLTAGE CONDITION EXISTS BETWEEN THE THREE PHASES OF THE GENERATOR. AN OSCILLOGRAPH, USED TO RECORD VOLTAGE APPLIED ACROSS THE NEUTRAL RESISTOR, INDICATED THAT NO FAULT HAD EXISTED ON THE GENERATOR. ADDITIONALLY, A MEGGER READING OF THE GENERATOR CONFIRMED THAT NO INTERNAL FAULT EXISTED. WITH NO GENERATOR FAULT, THE CAUSE OF THE TRIP WAS IN THE POTENTIAL TRANSFORMER CIRCUITRY. UPON COMPLETION OF TROUBLESHOOTING, THE CAUSE COULD NOT BE DETERMINED. PRECAUTIONARY MEASURES WERE TAKEN TO PREVENT AN INTERMITTENT POTENTIAL FUSE FAILURE FROM CAUSING ANOTHER GENERATOR TRIP. THE POTENTIAL TRANSFORMER FUSES WERE REPLACED AND THE SUSPECT CIRCUITRY WILL BE MONITORED

[27]   BROWNS FERRY 2   DOCKET 50-260   LER 91-019
AUTOMATIC REACTOR SCRAM FOLLOWING A TURBINE TRIP WHICH WAS A RESULT OF AN UNEXPECTED FUSE FAILURE.
EVENT DATE: 120891   REPORT DATE: 010792   NSSS: GE   TYPE: BWR

VENDOR: GOULD INC. (POWER SYS DIV)

(NSIC 223703) ON DECEMBER 8, 1991 AT 2320 CST, BROWNS FERRY UNIT 2 AUTOMATICALLY SCRAMMED FROM APPROXIMATELY 80 PERCENT POWER FOLLOWING A TURBINE TRIP. IN ADDITION, DURING THIS EVENT, AN UNEXPECTED LOSS OF ONE OF THE OFFSITE POWER SUPPLIES OCCURRED. THE ROOT CAUSE OF THIS EVENT WAS AN UNEXPECTED AND UNFORESEEN FUSE (GOULD-SHAWMUT FAILURE). TVA HAS REPLACED THE EXISTING 500KV BUS POTENTIAL TRANSFORMER SECONDARY FUSES TO MINIMIZE THE POSSIBILITY OF SPURIOUS RELAY OPERATIONS. TVA WILL EVALUATE RECONFIGURING THE EXISTING PLANT DESIGN TO ELIMINATE THE POSSIBILITY THAT THE LOSS OF THE RELAYING POTENTIAL FROM A SINGLE BUS WILL RESULT IN THE LOSS OF THE GENERATING UNIT. IN ADDITION, TVA WILL INVESTIGATE THE SPURIOUS TRIP OF TRINITY LINE 1 LINE DURING THIS EVENT AND CORRECT ANY IDENTIFIED DEFICIENCIES.

[28] BROWNS FERRY 3 DOCKET 50-296 LER 92-001
ENGINEERED SAFETY FEATURE ACTUATION CAUSED BY A FAILED RELAY COIL.
EVENT DATE: 012592 REPORT DATE: 021492 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 1 (BWR)
BROWNS FERRY 2 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 224033) ON JANUARY 25, 1992 AT 0720 CST, A UNIT 3 RELAY COIL, GENERAL ELECTRIC (GE) TYPE CR-120 ASSOCIATED WITH PRIMARY CONTAINMENT ATMOSPHERE CONTROL VALVE LOGIC CHANNEL A FAILED RESULTING IN ACTUATION OF ENGINEERED SAFETY FEATURES. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO AN END-OF-LIFE FAILURE CAUSED BY THE THERMAL AGING OF A GE TYPE CR-120 RELAY COIL. IMMEDIATE CORRECTIVE ACTIONS INCLUDED REPLACING THE FAILED RELAY COIL. ADDITIONAL CORRECTIVE ACTIONS INCLUDE REPLACING THE GE TYPE CR-120 RELAY COILS IN NORMALLY ENERGIZED, SAFETY-RELATED CIRCUITS PRIOR TO THE RESTART OF UNITS 1 AND 3 AND PRIOR TO STARTUP OF UNIT 2 FOLLOWING THE CYCLE 6 REFUELING OUTAGE. IN ADDITION, RELAY COILS IN THIS APPLICATION WHOSE FAILURE COULD POTENTIALLY LEAD TO A UNIT 2 PLANT SHUTDOWN WERE COMMITTED TO BE REPLACED BY MARCH 30, 1992.

[29] BRUNSWICK 1 DOCKET 50-325 LER 91-003 REV 03
UPDATE ON FAIL AS-IS POSITION OF CBEAF SYSTEM INLET AND OUTLET DAMPERS NOT EVALUATED WITH RESPECT TO A CHLORINE EVENT.
EVENT DATE: 013191 REPORT DATE: 020592 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)

(NSIC 223938) A CONDITION THAT WAS DISCOVERED TO BE OUTSIDE THE DESIGN BASIS WAS REPORTED ON 1/31/91 AT 4:02 P.M. THIS CONDITION INVOLVES THE INLET AND OUTLET DAMPERS FOR THE CONTROL BUILDING EMERGENCY AIR FILTRATION (CBW) TRAINS WHICH FAIL AS-IS ON LOSS OF POWER. WHEN THE CBEAF SYSTEM IS OPERATING IN THE RECIRCULATING MODE, A SINGLE ELECTRICAL FAILURE WILL CAUSE THE DAMPERS TO REMAIN OPEN. THIS WILL RESULT IN AN AIRFLOW PATH TO THE CONTROL ROOM ENVIRONMENT THAT HAS NOT BEEN PREVIOUSLY EVALUATED FOR ITS EFFECT ON CONTROL ROOM HABITABILITY RELATIVE TO CHLORINE PROTECTION. THIS CONDITION RESULTED FROM A FAILURE TO EVALUATE THE FAIL AS-IS POSITION (IN COMBINATION WITH A SINGLE FAILURE) OF THE CBEAF DAMPERS WITH RESPECT TO A CHLORINE EVENT. ON 1/31/91, AT APPROXIMATELY 6:11 P.M., THE CHLORINE TANK CAR WAS REMOVED FROM THE SITE PENDING AN ENGINEERING EVALUATION OF CBEAF SINGLE FAILURES. THE EVALUATION DETERMINED THE CHLORINE TANK CAR COULD BE RETURNED TO THE SITE PROVIDED THE MOST LIMITING CONDITION OF OPERATION (LCO) IS ENTERED WHEN THE CBEAF IS IN OPERATION. A TRACKING LCO WAS ESTABLISHED TO ENSURE IMPLEMENTATION OF THE LCO REQUIREMENTS AND THE CHLORINE TANK CAR WAS RETURNED TO THE SITE ON FEBRUARY 10, 1991. A PROJECT (PCN G0187A) HAS BEEN INITIATED TO MAKE THE INLET AND OUTLET DAMPERS AUTOMATICALLY CLOSE ON LOSS OF POWER.

[30] BRUNSWICK 1 DOCKET 50-325 LER 91-026 REV 01
UPDATE ON REACTOR SHUTDOWN REQUIRED WHEN EMERGENCY DIESEL GENERATOR MAINTENANCE WAS DETERMINED TO REQUIRE MORE TIME THAN THE TECHNICAL SPECIFICATION LIMITNG CONDITION OF OPERATION WOULD ALLOW.
EVENT DATE: 101591 REPORT DATE: 011392 NSSS: GE TYPE: BWR
VENDOR: WISCONSIN BRIDGE & IRON

(NSIC 223797) AT 0700 ON OCTOBER 1, 1991, #3 EMERGENCY DIESEL GENERATOR (EDG) WAS DECLARED INOPERABLE TO ALLOW FOR SCHEDULED MAINTENANCE AS PART OF THE UNIT 2 REFUELING OUTAGE. BY 0600 ON 10/15/91, IT WAS APPARENT THAT THE #3 EDG WOULD NOT BE RETURNED TO SERVICE IN TIME TO AVOID THE REACTOR SHUTDOWN REQUIRED BY THE LIMITING CONDITION FOR OPERATION (LCO) ACTION STATEMENT OF THE ELECTRICAL POWER SYSTEMS, A. C. SOURCES, TECHNICAL SPECIFICATION. THE TIME REQUIRED TO COMPLETE THE ORIGINAL WORK SCOPE HAD INCREASED DUE TO EMERGENT WORK ITEMS, SCHEDULING AND COORDINATION ISSUES, AND A PROCEDURAL COMPLIANCE ISSUE. THIS MADE IT IMPOSSIBLE TO RETURN THE #3 EDG TO SERVICE PRIOR TO THE LCO EXPIRING. ON 10/15/91 AT 0600, UNIT 1 WAS AT 100% REACTOR POWER WITH THE EMERGENCY CORE COOLING SYSTEMS (ECCS) OPERABLE. UNIT 2 WAS DEFUELED IN A REFUELING OUTAGE. THE DECISION WAS MADE TO COMMENCE A REACTOR SHUTDOWN BEGINNING AT 0630 ON 10/15/91. UNIT 1 ENTERED HOT SHUTDOWN AT 1803 ON 10/15/91, AND COLD SHUTDOWN AT 1740 ON 10/16/91. THE #3 EDG WAS DECLARED OPERABLE AT 1850 ON 10/20/91, UPON COMPLETION OF MAINTENANCE AND TESTING. UNIT 1 REACTOR STARTUP WAS COMMENCED AT 0048 ON 10/21/91, AND THE UNIT WAS SYNCHRONIZED TO THE SYSTEM GRID AT 1724 ON 10/21/91. A PREVIOUS EVENT IS LER 1-91-009.

[31] BRUNSWICK 1 DOCKET 50-325 LER 92-001
INADEQUATELY TESTING SEALED SOURCES RESULTS IN TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 010792 REPORT DATE: 020592 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)

(NSIC 223915) CP&L'S SHEARON HARRIS NUCLEAR POWER PLANT (SHNPP) ISSUED LER 1-91-020 WHICH IDENTIFIED DEFICIENCIES IN A RADIATION SURVEILLANCE TEST WHICH IS USED TO LEAK TEST SEALED SOURCES TO SATISFY A TECHNICAL SPECIFICATION REQUIREMENT. THIS PROCESS INCLUDES SMEARING THESE SOURCES AND ANALYZING THE SMEARS WITH AN INSTRUMENT CAPABLE OF DETECTING 0.005 MICROCURIES OF CONTAMINATION PER TEST SAMPLE. DURING A ROUTINE REVIEW OF A COMPLETED TEST, AN ENVIRONMENTAL & RADIATION CONTROL (E&RC) PERS ON NOTED A DEFICIENCY IN THE EQUIPMENT/PROCEDURE USED TO DETECT THE LOW ENERGY EMISSIONS PRODUCED BY NICKEL-63 (NI-63) AND IRON-55 (FE-55) ISOTOPES. BRUNSWICK E&RC PERSONNEL REVIEWED THIS LER. UPON FURTHER INVESTIGATION, BRUNSWICK E&RC PERSONNEL DETERMINED THAT THEIR SITE PROCEDURE CONTAINED THE SAME DEFICIENCY AS DID SHNPP'S AND THAT THE SAME REPORTABILITY REQUIREMENTS APPLIED TO AN FE -55 SOURCE. (BRUNSWICK WAS NOT AFFECTED BY NI-63). THE BRUNSWICK FACILITY HAD BEEN UTILIZING A TENNELEC GAS FLOW PROPORTIONAL COUNTER TO PERFORM AN ANALYSIS ON SMEARS OF AN FE-55 SOURCE BEING USED INSIDE AN ALLOY ANALYZER. A SMEAR OF THE FE-55 SOURCE WAS OBTAINED AND SENT TO THE HE&EC FOR ANALYSIS. THE TEST WAS SATISFACTORY AND THE SOURCE WAS DETERMINED NOT TO BE LEAKING.

[32] BRUNSWICK 2 DOCKET 50-324 LER 91-019
LOCAL LEAK RATE TEST FAILURE OF TWO MAIN STEAM LINES' ISOLATION VALVES RESULTING IN A CONDITION POTENTIALLY OUTSIDE THE PLANT DESIGN BASIS.
EVENT DATE: 111291 REPORT DATE: 121291 NSSS: GE TYPE: BWR
VENDOR: ROCKWELL-INTERNATIONAL

(NSIC 223639) ON NOVEMBER 12, 1991, THE UNIT 2 REACTOR WAS SHUTDOWN IN A REFUEL/MAINTENANCE OUTAGE. TYPE C LOCAL LEAK RATE TESTING (LLRT) OF THE MAIN STEAM ISOLATION VALVES (MSIVS) HAD BEEN PERFORMED. THE RESULTS OF TESTING INDICATED THAT MSIV LEAKAGE ON MAIN STEAM LINES (MSLS) C AND D EXCEEDED THE TECHNICAL SPECIFICATION (TS) LIMIT OF 11.5 SCFH. A ROOT CAUSE ANALYSIS IS IN PROGRESS AND THE RESULTS WILL BE PROVIDED IN A SUPPLEMENT TO THIS REPORT. THE MSLS C AND D MSIVS REPAIRS WERE COMPLETED AND A SUBSEQUENT REPEAT OF THE TESTING REVEALED ZERO LEAKAGE IN BOTH MSLS (IE; MSLC=0.173 SCFH AND MSLD=0.266 SCFH). ADDITIONAL CORRECTIVE ACTIONS RESULTING FROM THE ROOT CAUSE DETERMINATION WILL BE REPORTED IN THE SUPPLEMENT TO THIS REPORT. A SAFETY ASSESSMENT OF THIS EVENT HAS NOT BEEN COMPLETED AT THIS TIME. THE ASSESSMENT OF THIS EVENT WILL CONTINUE AND WILL BE REPORTED IN THE SUPPLEMENT TO THIS REPORT. A PREVIOUS FAILURE OF TWO MSIVS TO MEET THE TS REQUIRED 11.5 SCFH LEAKAGE REQUIREMENT HAS BEEN REPORTED IN LER 1-88-025. THAT LER INVOLVED THE OUTBOARD MSIVS ON TWO SEPARATE MSLS.

[33] BRUNSWICK 2 DOCKET 50-324 LER 91-020
HIGH PRESSURE COOLANT INJECTION SYSTEM STARTUP ROUTINE SURVEILLANCE TESTING FINDS LOOSE COMPONENTS AFTER A REFUELING OUTAGE REBUILD.
EVENT DATE: 121491 REPORT DATE: 011392 NSSS: GE TYPE: BWR
VENDOR: WOODWARD GOVERNOR COMPANY

(NSIC 223796) AT 1400 ON 12/14/91, UNIT 2 REACTOR WAS AT 2% REACTOR POWER DURING THE STARTUP FOLLOWING A FUELING OUTAGE. THE HIGH PRESSURE COOLANT INJECTION (HPCI) TURBINE WAS UNCOUPLED FOR TURBINE OVERSPEED TRIP MECHANISM TESTING. THE OVERSPEED TRIP MECHANISM HAD BEEN ADJUSTED AFTER THE INITIAL TEST RUN SHOWED THAT THE TURBINE DID NOT TRIP WITHIN THE REQUIRED SPEED RANGE. DURING THE SECOND TEST RUN, THE HPCI TURBINE STEAM SUPPLY VALVE (E41-F001) WOULD NOT OPEN FROM THE CONTROL ROOM. THE E41-F001 VALVE ACTUATOR HAD BEEN REBUILT DURING THE OUTAGE BUT THE LIMIT SWITCH ASSEMBLY FINGER BASE WHICH HOLDS THE INDIVIDUAL LIMIT SWITCHES WAS FOUND LOOSE. THE E41-F001 LIMIT SWITCH ASSEMBLY WAS TIGHTENED AND ADJUSTED. THE OTHER HPCI VALVE ACTUATORS THAT HAD BEEN REBUILT BY THE SAME SERVICE VENDOR WERE CHECKED AND NO PROBLEMS WERE FOUND WITH THESE LIMIT SWITCH ASSEMBLIES. BASED ON DISCUSSIONS WITH THE SERVICE VENDOR AND THE INSPECTION RESULTS, THIS IS CONSIDERED AN ISOLATED EVENT. AT 1436 ON 12/15/91, THE OVERSPEED TESTING WAS RECOMMENCED BUT AN ERRATIC TURBINE SPEED FEEDBACK SIGNAL STOPPED THE TEST. THE ERRATIC SIGNAL WAS DUE TO THE SPEED SENSING GEAR HAVING COME LOOSE FROM ITS SHAFT. NO DAMAGE WAS FOUND AND THE SPEED SENSING GEAR SET SCREWS RETIGHTENED. THE SPEED SENSING GEAR HAD NOT BEEN REMOVED FROM THE SHAFT THAT OUTAGE.

[34] BRUNSWICK 2 DOCKET 50-324 LER 91-021
VOLTMETER INTERNAL FAILURE DURING TESTING RESULTS IN AN INADVERTENT HPCI INJECTION AND REACTOR SCRAM.
EVENT DATE: 121791 REPORT DATE: 011392 NSSS: GE TYPE: BWR
VENDOR: FLUKE, JOHN MANUFACTURING COMPANY

(NSIC 223795) AT 09:42, ON DECEMBER 17, 1991, UNIT 2 WAS AT 5% REACTOR POWER. A STARTUP WAS IN PROGRESS ON COMPLETION OF A REFUELING OUTAGE. THE EMERGENCY CORE COOLING SYSTEMS WERE OPERABLE. THE REACTOR SCRAMMED WHEN THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM INADVERTENTLY STARTED AND INJECTED RELATIVELY COLD WATER INTO THE REACTOR VESSEL. INSTRUMENTATION AND CONTROLS (I&C) TECHNICIANS WERE PERFORMING A ROUTINE SURVEILLANCE (2MST-IR23M) ON THE HPCI REACTOR VESSEL LOW LEVEL 2 (118") INITIATION LOGIC. PER THE SURVEILLANCE, AN "A" CHANNEL LEVEL INSTRUMENT WAS ACTUATED AND ITS RELAY CONTACTS WERE VERIFIED CLOSED BY CHECKING THE VOLTAGE ACROSS THE OPEN "B" CHANNEL CONTACTS. THIS VOLTAGE LACK WAS PERFORMED WITH A FLUKE 860 CA-01 DIGITAL VOLTMETER (DVM), WHICH HAS SINCE BEEN VERIFIED TO HAVE AN INTERNAL FAULT. THE FAULT ALLOWED SUFFICIENT CURRENT FLOW TO ACTUATE AN INITIATION OF THE HPCI SYSTEM. THE RELATIVELY COLD WATER INJECTED BY THE HPCI SYSTEM CAUSED THE REACTOR POWER TO EXCEED THE 15% AVERAGE POWER RANGE MONITOR (APRM) SCRAM SETPOINTS. ONCE THE HPCI INITIATION WAS VERIFIED AS INVALID, THE HPCI SYSTEM WAS MANUALLY SHUT DOWN. AFTER THE CAUSE OF THE UNIT 2 REACTOR SCRAM HAD BEEN IDENTIFIED, STARTUP WAS RECOMMENCED AT 02:23 ON 12/18/91. THIS ISOLATED EVENT POSED MINIMAL SAFETY SIGNIFICANCE IN THAT PLANT SYSTEMS RESPONDED AS DESIGNED.

[35] BYRON 1 DOCKET 50-454 LER 90-007 REV 02
UPDATE ON MAIN STEAMLINE ISOLATION SYSTEM INOPERABLE DUE TO FAILURE TO TEST MANUAL INITIATION HANDSWITCH.
EVENT DATE: 061290 REPORT DATE: 121791 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 2 (PWR)

(NSIC 223652) ON 06/12/90, DURING A REVIEW OF THE BYRON UNIT 1 MAIN STEAM ISOLATION VALVE (MSIV) FULL STROKE TEST SURVEILLANCES, IT WAS FOUND THAT THE STEAM LINE ISOLATION HANDSWITCH ON MAIN CONTROL BOARD PANEL 1PM06J HAD NOT BEEN TESTED DURING THE PAST REFUELING OUTAGES FOR BOTH UNITS. TECHNICAL SPECIFICATION 3.3.2, TABLE 4.3-2 ITEM 4 A.2, REQUIRES TESTING OF THE STEAM LINE ISOLATION HANDSWITCH ON A REFUELING OUTAGE INTERVAL AND REQUIRES THAT TWO TRAINS BE OPERABLE IN MODES 1, 2 AND 3. AT 1100 ON 6-12-90, THE MSIV MANUAL ISOLATION SYSTEM WAS DECLARED INOPERABLE FOR BOTH UNITS. DUE TO AN UNRELATED PROBLEM, UNIT 1 WAS SHUTDOWN (MODE 4) ON 06/13/90. THEREFORE, IT WAS DECIDED TO PERFORM THE

TEST ON UNIT 1 FIRST. THE HANDSWITCHES WERE TESTED AND THE SYSTEM WAS DECLARED OPERABLE AT 1545. ON 06/13/90, AT 2345, WITH UNIT 2 OPERATING IN MODE 1, THE UNIT ENTERED TECHNICAL SPECIFICATION 3.0.3 BECAUSE BOTH TRAINS OF MANUAL ISOLATION WERE INOPERABLE DURING THE TEST. ON 06/14/90 AT 0001, THE TEST WAS COMPLETED AND TECHNICAL SPECIFICATION 3.0.3 WAS EXITED. THE SYSTEM WAS DECLARED OPERABLE AT 0730. AS PREVENTIVE ACTION, BYRON PLANS TO HAVE A REVIEW PERFORMED ON ALL SIMILAR EQUIPMENT TO ENSURE THAT PROPER TESTING IS BEING DONE. IN ADDITION, THE STATION WILL REVIEW RELATED PROCEDURES AND THE PROCEDURE REVIEW PROCESS.

[36] BYRON 1 DOCKET 50-454 LER 91-006
EMERGENCY DIESEL GENERATOR INOPERABILITY DUE TO TESTING PROCEDURE INADEQUACY.
EVENT DATE: 111491 REPORT DATE: 121191 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 2 (PWR)

(NSIC 223627) DURING THE NRC ELECTRICAL DISTRIBUTION SYSTEM FUNCTIONAL INSPECTION AT LASALLE COUNTY NUCLEAR STATION, THE INSPECTION TEAM DETERMINED THAT THE TESTING OF THE 4160 VOLT ENGINEERED SAFETY FEATURE (ESF) BUS UNDERVOLTAGE RELAY CIRCUITS WAS INADEQUATE. FURTHER EVALUATION OF THIS CONDITION FOR BYRON STATION LED TO THE CONCLUSION THAT A TRIP CIRCUIT FOR AN ESF BUS UNDERVOLTAGE AND DIESEL GENERATOR SPEED AND VOLTAGE PERMISSIVE HAD NOT BEEN TESTED. BOTH DIESEL GENERATORS ON BOTH UNITS WERE DECLARED INOPERABLE ON NOVEMBER 14, 1991 AT 1505. A TEMPORARY WAIVER OF COMPLIANCE WAS OBTAINED. TESTING OF THE AFFECTED CIRCUITS WAS SUCCESSFULLY PERFORMED ON NOVEMBER 15, 1991. SURVEILLANCE PROCEDURES WILL BE REVISED TO INCLUDE THIS TESTING. THIS EVENT IS REPORTABLE IN ACCORDANCE WITH 10CFR50.73(A)(2)(VII). ANY EVENT WHERE A SINGLE CAUSE OR CONDITION CAUSED TWO INDEPENDENT TRAINS TO BECOME INOPERABLE IN A SINGLE SYSTEM DESIGNED TO; (A) SHUTDOWN THE REACTOR AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION; (B) REMOVE RESIDUAL HEAT; (C) CONTROL THE RELEASE OF RADIOACTIVE MATERIAL; OR (D) MITIGATE THE CONSEQUENCES OF AN ACCIDENT.

[37] BYRON 2 DOCKET 50-455 LER 91-005
REACTOR TRIP ON LOW-2 STEAM GENERATOR LEVEL DURING STARTUP DUE TO D-5 STEAM GENERATOR LEVEL CONTROL.
EVENT DATE: 110791 REPORT DATE: 120991 NSSS: WE TYPE: PWR

(NSIC 223626) AT 0357, ON 11/07/91, AFTER UNIT 2 STARTUP AND DURING PREPARATION FOR THE TURBINE ROLL, A LOW-2 STEAM GENERATOR (S/G) LEVEL ON THE 2A S/G WAS RECEIVED WHICH INITIATED A REACTOR TRIP. ALL SAFEGUARD ACTUATION FEATURES FUNCTIONED AS DESIGNED. THE LOW-2 LEVEL IN THE STEAM GENERATOR WAS THE RESULT OF DIFFICULTY IN CONTROLLING THE WESTINGHOUSE D-5 S/G LEVEL AT LOW POWER AND DUE TO LEAK BY OF THE MAIN FEEDWATER REGULATING VALVE FOR THE 2A STEAM GENERATOR. THE LEVEL INSTRUMENT TAPS IN THE STEAM GENERATOR ARE BEING MOVED DURING THE NEXT REFUELING OUTAGE TO REDUCE THIS LEVEL INSTABILITY. THE MAIN FEEDWATER REGULATORY VALVE WAS ADJUSTED BEFORE THE NEXT STARTUP. A REVISION WILL BE MADE TO THE STARTUP PROCEDURE TO VERIFY POSITION OF THIS VALVE FOR EACH STEAM GENERATOR. THIS EVENT IS REPORTABLE IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV), ANY EVENT OR CONDITION THAT RESULTS IN A MANUAL OR AUTOMATIC ACTUATION OF ANY ENGINEERED SAFETY FEATURE .

[38] CALLAWAY 1 DOCKET 50-483 LER 91-007
FAILURE TO VERIFY LOAD REJECTION SURVEILLANCE VALUE OF 1352 KW FOR EMERGENCY DIESEL GENERATORS SINCE THIS VALUE DOES NOT REFLECT ACTUAL CONFIGURATION.
EVENT DATE: 111491 REPORT DATE: 121291 NSSS: WE TYPE: PWR

(NSIC 223628) DATE OF EVENT - 111491. POWER LEVEL - 100%. ON 11/13/91, UTILITY MANAGEMENT WAS NOTIFIED BY A SIMILAR PLANT THAT A POTENTIAL CONCERN EXISTED WITH THE ABILITY TO SATISFY THE 1352 KILOWATT (KW) ESSENTIAL SERVICE WATER (ESW) PUMP LOAD REJECTION REQUIREMENT OF TECH SPEC SURVEILLANCE 4.8.1.1.2.F(2). UTILITY MANAGEMENT IMMEDIATELY INITIATED AN INVESTIGATION INTO THE APPLICABILITY TO CALLAWAY PLANT. ON 11/14/91 AT 1515 CST, IT WAS DETERMINED THAT THE PLANT HAD NOT DEMONSTRATED VERBATIM COMPLIANCE WITH T/S SURVEILLANCE 4.8.1.1.2.F(2) AND T/S 3.0.3 WAS ENTERED. THE PLANT WAS IN MODE 1 - POWER OPERATIONS AT 100% REACTOR POWER. THE UTILITY SHIFT SUPERVISOR IMMEDIATELY SCHEDULED A PLANT SHUTDOWN TO

BEGIN AT 1715 CST. ON 11/14/91 AT 1620 CST, A TEMPORARY WAIVER OF COMPLIANCE WAS GRANTED TO WAIVE THE REQUIREMENT TO INITATE ACTION TO SHUTDOWN THE PLANT PER T/S 3.0.3 UNTIL AN EMERGENCY T/S AMENDMENT TO CHANGE THE 1352 KW LOAD REJECTION REQUIREMENT WAS APPROVED. THE VALUE OF 1352 KW IS NOT REPRESENTATIVE OF THE ACTUAL INSTALLED CONFIGURATION. THE VALUE WAS BASED ON ASSUMED LOAD VALUES USED TO SIZE THE DIESEL GENERATORS. THE PROPOSED CHANGE WILL CLARIFY THAT THE ESW PUMP IS THE LARGEST SINGLE EMERGENCY LOAD WITHOUT SPECIFYING A KW RATING.

[39] CALLAWAY 1 DOCKET 50-483 LER 91-008
FAILURE TO VERIFY THAT CONTAINMENT PENETRATION VENT VALVE WAS LOCKED CLOSED PER TECHNICAL SPECIFICATION 4.6.1.1.A PRIOR TO 1989 DUE TO INCORRECT LOCKED VALVE LIST.
EVENT DATE: 121091 REPORT DATE: 123191 NSSS: WE TYPE: PWR

(NSIC 223750) ON 12/10/91, WHILE IN MODE 1 AT 100 PERCENT POWER, A REVIEW OF SURVEILLANCE PROCEDURE OSP-GP-00001, CONTAINMENT INTEGRITY VERIFICATION, REVEALED THAT THE RESIDUAL HEAT REMOVAL (RHR) PUMP 'A' SUCTION HEADER VENT VALVE (EJV154) HAD NOT BEEN VERIFIED AS LOCKED CLOSED PRIOR TO 3/14/89. TECHNICAL SPECIFICATION (T/S) 4.6.1.1.A REQUIRES THAT CONTAINMENT PENETRATIONS WITH VALVES LOCATED INSIDE CONTAINMENT BE VERIFIED DURING COLD SHUTDOWN AS LOCKED CLOSED, EJV154 WAS NOT INCLUDED IN A 3/12/82 LISTING OF LOCKED CONTAINMENT PENETRATION VALVES. THIS CAUSED A RHR SYSTEM DRAWING REVISION DELETING THE LOCKED REQUIREMENT. AS THIS DRAWING WAS USED TO DEVELOP PROCEDURE OSP-GP-00001, EJV154 WAS NOT INCLUDED. ON 10/21/87, A UTILITY LICENSED OPERATOR REQUESTED THAT EJV154 BE DESIGNATED LOCKED CLOSED. ON PLANT DRAWINGS FOR CONTAINMENT INTEGRITY PURPOSES. WHEN EJV154 WAS DETERMINED TO REQUIRE LOCKED CLOSED VERIFICATION ON 11/27/87, UTILITY PERSONNEL REVIEWING THIS DID NOT EVALUATE THE APPLICABILITY OF T/S 4.6.1.1.A. THE DRAWING, VALVE LINE-UP, AND SURVEILLANCE PROCEDURES HAVE BEEN REVISED TO ENSURE THAT EJV154 IS VERIFIED LOCKED CLOSED. NO OTHER T/S PROHIBITED CONDITIONS WERE IDENTIFIED BY THE MOST RECENT UTILITY REVIEW OF CONTAINMENT PENETRATIONS. THIS IS AN ISOLATED OCCURRENCE. NO ADDITIONAL CORRECTIVE ACTIONS ARE REQUIRED.

[40] CALVERT CLIFFS 1 DOCKET 50-317 LER 91-009
LOSS OF OPERABLE HIGH PRESSURE SAFETY INJECTION TRAIN DUE TO INOPERABLE PUMP BREAKER.
EVENT DATE: 112691 REPORT DATE: 122691 NSSS: CE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223773) ON NOVEMBER 26, 1991 WHILE OPERATING AT 100 PERCENT RATED THERMAL POWER, CALVERT CLIFFS UNIT 1 ENTERED PLANT TECHNICAL SPECIFICATION (TS) LIMITING CONDITION FOR OPERATION (LCO) 3.0.3 WHEN ALL HIGH PRESSURE SAFETY INJECTION (HPSI) TRAINS BECAME INOPERABLE. THE PLANT REMAINED IN LCO 3.0.3 FOR APPROXIMATELY 35 MINUTES. WHILE OPERATING WITH ONE OPERABLE HPSI TRAIN, THAT TRAIN'S PUMP BREAKER FAILED TO ALIGN FOR STARTING. THE CAUSE OF THIS EVENT WAS A MISPOSITIONED FIBER SPACER THAT INTERFERED WITH AN OPERATING-MECHANISM-SOLENOID PLUNGER. INVESTIGATIVE EFFORTS ARE CONTINUING WITH THE POSSIBILITY OF MANUFACTURING OR DESIGN DEFECT BEING EVALUATED. IF SHOWN TO HAVE EXISTED, BALTIMORE GAS AND ELECTRIC SHALL SUPPLEMENT THIS REPORT, DOCUMENTING THE CAUSE AND ADDITIONAL CORRECTIVE ACTION. CORRECTIVE ACTION INCLUDED REPLACING THE BREAKER WITH A SUITABLE REPLACEMENT TO RESTORE OPERABILITY AND REPAIRING THE NON-ALIGNING BREAKER. CORRECTIVE ACTION THAT CONTINUES INCLUDES INSPECTING 16 OF 60 SAFETY-RELATED BREAKERS IN THE FIELD AND INCLUDING AN OPERATING MECHANISM INSPECTION IN SCHEDULED PMS TO ACCOMMODATE INSPECTION OF ALL REMAINING SAFETY-RELATED BREAKERS.

[41] CALVERT CLIFFS 1 DOCKET 50-317 LER 91-007
EMERGENCY AIR LOCK INNER AND OUTER DOORS OPEN SIMULTANEOUSLY DUE TO MECHANICAL MALFUNCTION.
EVENT DATE: 122491 REPORT DATE: 012392 NSSS: CE TYPE: PWR

(NSIC 223829) ON DECEMBER 24, 1991, MAINTENANCE PERSONNEL, UNAWARE THAT THE UNIT 1 CONTAINMENT EMERGENCY AIR LOCK INNER DOOR WAS OPEN, OPENED THE EXTERIOR DOOR, BRIEFLY BREACHING CONTAINMENT. CONTAINMENT INTEGRITY WAS RESTORED WITHIN ONE

MINUTE. THE INTERIOR DOOR REMOTE POSITION INDICATION DID NOT INDICATE THAT THE DOOR WAS OPEN AND AN INTERLOCK MECHANISM THAT NORMALLY PREVENTS BOTH DOORS FROM BEING OPEN SIMULTANEOUSLY, DID NOT FUNCTION PROPERLY. THE CAUSE IS ATTRIBUTED TO MECHANICAL MALFUNCTION WHICH INCLUDES A SLIPPED CHAIN AND MISALIGNED CAM IN THE INTERLOCK MECHANISM AND INACCURATE INTERIOR DOOR POSITION INDICATION. THE CHAIN AND CAM WERE REALIGNED AND THE MECHANISM VERIFIED OPERABLE. A SIGN HAS BEEN PLACED NEXT TO THE EXTERIOR DOOR WARNING PERSONNEL OF THE LOOSE CHAIN AND CAUTIONING THEM TO OPEN THE DOOR CAREFULLY. APPROPRIATE PERSONNEL ARE BEING BRIEFED ON THE DETAILS OF THIS INCIDENT AND CAUTIONED AGAINST OPENING THE DOOR BEFORE PRESSURE IS EQUALIZED ACROSS IT. ADDITIONAL TESTING TO CONFIRM THE CAUSE OF THE MECHANICAL MALFUNCTION WILL BE CONDUCTED WHEN ALARA CONDITIONS AND TECHNICAL SPECIFICATION OPERABILITY REQUIREMENTS PERMIT. SHOULD IT BECOME NECESSARY TO USE THE EMERGENCY AIR LOCK PRIOR TO IMPLEMENTATION OF FINAL CORRECTIVE ACTIONS, PERSONNEL WILL IMPLEMENT ADDITIONAL CONTROLS TO ENSURE PERSONNEL ENTERING THE AIR LOCK ARE AWARE OF THE STATUS OF BOTH AIR LOCK DOORS.

[42] CALVERT CLIFFS 1 DOCKET 50-317 LER 91-008
INADVERTENT ACTUATION OF THE AUXILIARY FEEDWATER ACTUATION SYSTEM DUE TO CIRCUIT DESIGN.
EVENT DATE: 122991 REPORT DATE: 012892 NSSS: CE TYPE: PWR

(NSIC 223855) ON DECEMBER 29, 1991, AN INADVERTENT ACTUATION OF THE AUXILIARY FEEDWATER ACTUATION SYSTEM (AFAS) OCCURRED AT CALVERT CLIFFS UNIT 1. AT THE TIME OF THE EVENT, UNIT 1 WAS IN MODE 3 (HOT STANDBY) WITH THE REACTOR COOLANT SYSTEM AT 532 DEGREES FAHRENHEIT AND 2250 PSIA. PREPARATIONS WERE UNDERWAY TO ENTER MODE 2 (STARTUP). THE CAUSE OF THE EVENT IS THE SENSITIVITY OF THE ORIGINAL AFAS START LOGIC CIRCUIT DESIGN TO INDUCED NOISE PULSES. THIS SENSITIVITY WAS MASKED BY THE TIME DELAY IN THE CIRCUIT. THE MODIFICATION TO THE LOGIC CIRCUIT RELOCATED THE TIME DELAY AND REVEALED THE UNEXPECTED SUSCEPTIBILITY OF THIS CIRCUIT TO NOISE INDUCED PULSES SUCH AS ELECTRO STATIC DISCHARGE. THE ACTUATION IS ATTRIBUTED TO AN ELECTRO STATIC DISCHARGE ON THE AFAS CABINET WHILE A LICENSED UTILITY OPERATOR WAS SHUTTING AN ADJACENT AFAS SENSOR CABINET DOOR. THE VENDOR HAS BEEN REQUESTED TO DESIGN AND PROVIDE HARDWARE FOR A CORRECTIVE MODIFICATION. TO PREVENT RECURRENCE THE AFAS CABINET AREA IS ROPED OFF AND ACCESS TO THE AREA REQUIRES SHIFT SUPERVISOR PERMISSION. CAUTION TAGS ARE IN PLACE TO WARN PLANT PERSONNEL OF THE POTENTIAL FOR SPURIOUS ACTUATION OF THE AFAS CABINETS DUE TO STATIC CHARGE. PREVIOUSLY ESTABLISHED ELECTRO STATIC DISCHARGE COUNTERMEASURES WERE EXPANDED FOR WORK IN AND AROUND ALL AFAS CABINETS.

[43] CALVERT CLIFFS 2 DOCKET 50-318 LER 92-001
MANUAL TRIP FOLLOWING MINOR FEEDWATER LEAK AND SUBSEQUENT ELECTRICAL GROUNDS.
EVENT DATE: 010292 REPORT DATE: 013192 NSSS: CE TYPE: PWR
VENDOR: CONSOLIDATED CONTROLS CORP.

(NSIC 223937) ON JANUARY 2, 1992 AT 2219 HOURS, CALVERT CLIFFS UNIT 2 WAS MANUALLY TRIPPED FROM 92 PERCENT POWER FOLLOWING A FEEDWATER LEAK FROM A LIFTED 3/4 INCH FEEDWATER HEATER RELIEF VALVE. A MANUAL TRIP WAS INITIATED AFTER LEAKING FEEDWATER CAUSED ELECTRICAL GROUNDS IN THE SECONDARY SYSTEM AND A CHARGING PUMP TRIPPED WITHOUT APPARENT CAUSE. THE FAILED RELIEF VALVE HAS BEEN REPLACED AND AN ENGINEERING EVALUATION WILL RECOMMEND CORRECTIVE ACTIONS TO PREVENT FUTURE FAILURES OF SIMILAR VALVES.

[44] CATAWBA 1 DOCKET 50-413 LER 91-020
TECHNICAL SPECIFICATION 3.0.3 ENTRY DUE TO TWO INOPERABLE TRAINS OF THE CONTROL ROOM VENTILATION SYSTEM.
EVENT DATE: 091591 REPORT DATE: 121291 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 223647) ON 9/15/91, AT 1300 HOURS, WITH UNIT 1 IN MODE 1, POWER OPERATION, AT 98% POWER AND UNIT 2 IN MODE 5, COLD SHUTDOWN, OPERATIONS ATTEMPTED TO SWAP TO TRAIN A OF THE CONTROL ROOM AREA VENTILATION AND CHILLED WATER (VC/YC) SYSTEM. DURING THIS ACTIVITY, IT WAS DISCOVERED THAT BREAKER 1EKPG #22, TRAIN A VC/YC SYSTEM CONTROLS, WAS OPEN. THE BREAKER WAS LAST FUNCTIONALLY DEMONSTRATED TO BE

CLOSED ON SEPTEMBER 11 AT 2210 HOURS. A REVIEW OF ALL ACTIVITIES BETWEEN SEPTEMBER 11 AT 2210 HOURS AND SEPTEMBER 15 AT 1300 HOURS DID NOT CLEARLY REVEAL HOW/WHEN 1EKPG #22 WAS OPENED. HOWEVER, IT IS CONSIDERED MOST PROBABLE THAT 1EKPG #22 WAS OPENED BETWEEN 0930 AND 1520 HOURS ON SEPTEMBER 13. TRAIN A VC/YC HAD BEEN DECLARED INOPERABLE ON SEPTEMBER 13 FROM 0455 HOURS TO 2340 HOURS FOR WORK ON VARIOUS TRAIN A COMPONENTS. PRIOR TO THIS WORK, A TAGOUT WAS PLACED TO OPEN 1EKPG #21 INSTEAD OF #22. IT IS CONSIDERED PROBABLE THAT TECH SPEC 3.0.3 WAS ENTERED ON SEPTEMBER 13 FROM 0810 TO 0930 HOURS BECAUSE NEITHER TRAIN OF VC/YC WOULD HAVE BEEN CAPABLE OF ADEQUATELY PRESSURIZING THE CONTROL ROOM. THIS INCIDENT IS ATTRIBUTED TO INAPPROPRIATE ACTIONS. CORRECTIVE ACTIONS INCLUDE FORMATION OF A TASK FORCE TO STUDY COMPONENT MISPOSITIONING EVENTS AND DISCUSSION OF THIS INCIDENT WITH OPERATIONS PERSONNEL.

[45] CATAWBA 1 DOCKET 50-413 LER 91-027
VITAL BATTERY 1EBB INOPERABLE DUE TO RESISTANCE READING EXCEEDING TECHNICAL SPECIFICATION.
EVENT DATE: 101891 REPORT DATE: 112091 NSSS: WE TYPE: PWR

(NSIC 223551) ON OCTOBER 18, 1991, UNIT 1 WAS OPERATING IN MODE 1, POWER OPERATION, AT 100% POWER. MAINTENANCE ENGINEERING SERVICES (MES) PERSONNEL DETERMINED THAT BATTERY BANK 1EBB IN THE 125VDC VITAL INSTRUMENTATION AND CONTROL POWER (EPL) SYSTEM WAS INOPERABLE. DURING THE FINAL REVIEW OF STANDING WORK REQUEST (SWR) 007877SWR, ORIGINATED TO MEASURE CELL TO CELL AND TERMINAL CONNECTION RESISTANCE, MES IDENTIFIED TWO BATTERY TERMINAL CONNECTIONS THAT EXCEEDED THE MAXIMUM TECHNICAL SPECIFICATION RESISTANCE VALUES. THE SWR HAD BEEN PERFORMED ON AUGUST 27, 1991. THIS INCIDENT IS ATTRIBUTED TO INAPPROPRIATE ACTION DUE TO LACK OF ATTENTION TO DETAIL. CORRECTIVE ACTIONS INCLUDED REPLACEMENT OF CABLE CONNECTORS, PROCEDURE ENHANCEMENT, AND MANAGEMENT DISCUSSION OF THE INCIDENT WITH PERSONNEL INVOLVED.

[46] CATAWBA 1 DOCKET 50-413 LER 91-023
BOTH TRAINS OF CONTROL ROOM AREA VENTILATION SYSTEM BEING INOPERABLE DUE TO EQUIPMENT FAILURE.
EVENT DATE: 111791 REPORT DATE: 121791 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 223695) ON NOVEMBER 17, 1991, AT 1040 HOURS, UNIT 1 AT 100% POWER IN MODE 1, POWER OPERATION, AND UNIT 2 IN NO MODE (DEFUELED), OPERATIONS PERSONNEL DISCOVERED THAT DAMPER 1-CR-D-10, SUCTION ISOLATION DAMPER FOR A TRAIN CONTROL ROOM AIR HANDLING UNIT, IN THE CONTROL ROOM VENTILATION (VC) AND CHILLED WATER (YC) SYSTEM HAD FAILED CLOSED. WITH 1-CR-D-10 CLOSED, A TRAIN VC/YC WAS INOPERABLE. TRAIN B OF THE VC/YC SYSTEM HAD BEEN DECLARED INOPERABLE ON NOVEMBER 11, 1991 AT 0145 HOURS, DUE TO B TRAIN NUCLEAR SERVICE WATER (RN) SYSTEM WORK. WITH BOTH TRAINS OF VC/YC INOPERABLE, UNIT 1 ENTERED TECHNICAL SPECIFICATION (T/S) 3.0.3, AT 1040 HOURS. DAMPER 1-CR-D-10 WAS MANUALLY POSITIONED AND BLOCKED OPEN TO RESTORE OPERABILITY TO A TRAIN VC/YC AT 1045 HOURS AND T/S 3.0.3 WAS EXITED. WORK REQUEST 566920PS WAS INITIATED TO INVESTIGATE AND REPAIR THE ACTUATOR ON DAMPER 1-CR-D-10. THE ACTUATOR WAS REPAIRED AND THE DAMPER WAS DECLARED OPERABLE AT 1610 HOURS ON NOVEMBER 19, 1991. THIS INCIDENT IS ATTRIBUTED TO EQUIPMENT FAILURE/MALFUNCTION DUE TO A FAILED HYDROMETER. THE VENTILATION TASK FORCE IS EVALUATING REPLACEMENT OF HYDROMETERS WITH A MORE RELIABLE ACTUATOR.

[47] CATAWBA 1 DOCKET 50-413 LER 91-030
TECHNICAL SPECIFICATION VIOLATION DUE TO A MISSED GRAB SAMPLE ON RADIATION MONITOR EMF-33 AS A RESULT OF INAPPROPRIATE ACTION.
EVENT DATE: 112191 REPORT DATE: 121891 NSSS: WE TYPE: PWR

(NSIC 223685) ON NOVEMBER 21, 1991, AT 2200 HOURS, WITH UNIT 1 IN MODE 1, POWER OPERATION, AT 100% POWER, A TECHNICAL SPECIFICATION (T/S) VIOLATION OCCURRED DUE TO A MISSED GRAB SAMPLE FOR THE INOPERABLE CONDENSER AIR EJECTOR EXHAUST MONITOR 1EMF33. RADIATION PROTECTION (RP) SPECIALIST A WAS INVOLVED WITH OTHER JOB DUTIES AND FAILED TO OBTAIN THE GRAB SAMPLE AT 2200 HOURS. THE GRAB SAMPLE WAS OBTAINED AT 2250 HOURS AND RP MANAGEMENT WAS NOTIFIED OF THE TECHNICAL SPECIFICATION

VIOLATION DUE TO THE INOPERABLE 1EMF33 NOT BEING SAMPLED WITHIN THE REQUIRED 12 HOUR TIME FRAME. THIS INCIDENT IS ATTRIBUTED TO INAPPROPRIATE ACTION DUE TO RP SPECIALIST A IDENTIFYING THE PROPER RESPONSE, BUT NOT IN TIME. AS A CORRECTIVE ACTION, EMF'S WHICH REQUIRE A GRAB SAMPLE EVERY 12 HOURS WHEN INOPERABLE ARE NOW BEING SAMPLED TWICE PER SHIFT. RP PROCEDURE HP/0/B/1009/11, EMF LOSS, WILL BE UPDATED TO REFLECT THIS CHANGE. RP ALSO PLANS TO PURSUE THE POSSIBILITY OF OBTAINING GREATER FLEXIBILITY FOR SCHEDULING EMF GRAB SAMPLES WHEN REQUIRED BY T/S ACTION STATEMENTS. THIS WILL BE PURSUED WHEN THIS T/S IS CONVERTED INTO A SELECTED LICENSEE COMMITMENT (SLC).

[48] CATAWBA 1 DOCKET 50-413 LER 91-029
TECHNICAL SPECIFICATION VIOLATION FROM FAILURE TO PERFORM REACTOR TRIP SYSTEM SURVEILLANCE DUE TO INAPPROPRIATE ACTION.
EVENT DATE: 112691 REPORT DATE: 121891 NSSS: WE TYPE: PWR

(NSIC 223684) ON NOVEMBER 26, 1991, UNIT 1 WAS IN MODE 1, POWER OPERATION, AT APPROXIMATELY 100% POWER. DURING PREPARATION OF A RETYPE OF UNIT 2 PROCEDURES, IT WAS DISCOVERED THAT THE SURVEILLANCE REQUIREMENT FOR THE TURBINE STARTUP PRESSURE SWITCH ALARM (4.3-1.1, TABLE 4.3-1, ITEM 16A) HAD NOT BEEN PERFORMED ON UNIT 1 DURING 1EOC5 AND SUBSEQUENT TURBINE STARTUPS. THIS SURVEILLANCE REQUIREMENT WAS INADVERTENTLY LEFT OUT OF RETYPE #8 OF PT/1/A/4250/02B, WEEKLY MAIN TURBINE VALVE MOVEMENT TEST. OPERATIONS SUPPORT ENGINEER DISCOVERED THE ERROR AND IMMEDIATELY ISSUED RETYPE 9 TO PT/1/A/4250/02B, WHICH INCLUDED THE SURVEILLANCE REQUIREMENT. THIS INAPPROPRIATE ACTION WAS DISCUSSED WITH THE RESPONSIBLE OPERATION SUPPORT ENGINEER. THE PROCEDURE WAS PLACED ON SHIFT AND PERFORMED TO SATISFY TECHNICAL SPECIFICATION REQUIREMENT. A CONTINUING IMPROVEMENT ACTION TEAM HAS BEEN FORMED TO IMPROVE OVERALL QUALITY OF THE OPERATIONS DEPARTMENT.

[49] CATAWBA 1 DOCKET 50-413 LER 91-031
TECHNICAL SPECIFICATION VIOLATION AS A RESULT OF INAPPROPRIATE ACTION DUE TO A MISSED IWV INSERVICE INSPECTION STROKE TIME TEST OF VALVE 1CA64.
EVENT DATE: 120191 REPORT DATE: 123091 NSSS: WE TYPE: PWR

(NSIC 223778) ON NOVEMBER 5, 1991, UNIT 1 WAS IN MODE 1, POWER OPERATION. PERFORMANCE (PRF) TECHNICIANS WERE ATTEMPTING TO PERFORM THE QUARTERLY IWV SURVEILLANCE STROKE TIMING TEST ON VALVE 1CA64, BASED ON TECHNICAL SPECIFICATION 4.0.5 SURVEILLANCE REQUIREMENTS. DUE TO A PROBLEM WITH THE OPERATOR AID COMPUTER (OAC) VALVE INDICATION, PRF WAS UNABLE TO COMPLETE THE TEST. A WORK REQUEST (W/R) WAS WRITTEN TO REPAIR THE PROBLEM. THE W/R WAS ASSIGNED AN APPROPRIATE PRIORITY AND WAS "TO BE COMPLETED" BY NOVEMBER 25, 1991, WITHIN THE 25% GRACE PERIOD. UPON DISCOVERY ON DECEMBER 2, 1991, THAT THE W/R HAD NOT BEEN WORKED, A 72 HOUR ACTION STATEMENT WAS ENTERED FOR CA OPERABILITY AND THE VALVE WAS TESTED LOCALLY. INSTRUMENT AND ELECTRICAL (IAE) PERSONNEL FOUND THE LIMIT SWITCHES MISADJUSTED ON 1CA64. IAE ADJUSTED THE LIMIT SWITCHES FOR PROPER INDICATION. THE FAILURE TO RETEST VALVE 1CA64 WITHIN THE REQUIRED TIME IS ATTRIBUTED TO INAPPROPRIATE ACTION. CORRECTIVE ACTIONS INCLUDED PERFORMING A NEW STROKE TIME TEST FOR VALVE 1CA64 TO VERIFY ACCEPTANCE CRITERIA WERE MET.

[50] CATAWBA 1 DOCKET 50-413 LER 92-002
TWO INOPERABLE TRAINS OF THE CONTROL ROOM VENTILATION SYSTEM.
EVENT DATE: 011692 REPORT DATE: 021392 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 224051) ON 1/16/92, AT 2000 HOURS, WITH UNITS 1 AND 2 IN MODE 1, POWER OPERATION, OPERATIONS PERSONNEL WERE REVIEWING ACTIVITIES ASSOCIATED WITH STARTUP OF THE TRAIN B CONTROL ROOM AREA VENTILATION AND CHILLED WATER (VC/YC) SYSTEM FOLLOWING MAINTENANCE. IT WAS DISCOVERED THAT BREAKER 2EKPH #22, TRAIN B VC/YC SYSTEM CONTROLS, AHD NOT BEEN OPENED DURING PRECEDING MAINTENANCE ACTIVITIES AS REQUIRED BY OPERATIONS TECHNICAL MEMORANDUMS. VC/YC TRAIN B HAD BEEN INOPERABLE ON JANUARY 16 FROM 0353 TO 1725 HOURS. DURING THIS TIME, THE TRAIN B CONTROL ROOM AIR HANDLING UNIT WAS REMOVED FROM SERVICE AND ASSOCIATED ACCESS DOORS WERE OPENED TO ALLOW FOR EQUIPMENT INSPECTIONS. WITH THESE ACCESS PANELS OPEN AND 2EKPH #22 CLOSED, VC/YC TRAIN B DAMPERS WOULD HAVE REPOSITIONED UPON RECEIPT OF A

SAFETY SIGNAL, THUS ALLOWING AIR FLOW TO ESCAPE THROUGH THE OPENINGS IN THE SYSTEM. VC/YC TRAIN A, WHICH WAS OPERABLE AND IN SERVICE, WOULD NOT HAVE BEEN CAPABLE OF ADEQUATELY PRESSURIZING THE CONTROL ROOM. THEREFORE, BOTH VC/YC TRAINS WERE INOPERABLE WHILE THE ACCESS PANELS WERE OPEN AND TECHNICAL SPECIFICATION 3.0.3 WAS UNKNOWINGLY ENTERED. THIS INCIDENT IS ATTRIBUTED TO INAPPROPRIATE ACTIONS, CONTROL ROOM OPERATORS DID NOT RECOGNIZE THE NEED TO OPEN 2EKPH #22. CORRECTIVE ACTIONS INLCUDE RED TAG COMPUTER PROGRAM ENHANCEMENTS AND A TECHNICAL MEMORANDUM PROGRAM REVIEW.

[51] CATAWBA 2 DOCKET 50-414 LER 91-014
ESF ACTUATION DUE TO ESSENTIAL BUS UNDERVOLTAGE CONDITION.
EVENT DATE: 110591 REPORT DATE: 120491 NSSS: WE TYPE: PWR

(NSIC 223625) ON NOVEMBER 5, 1991, AT 2300 HOURS, UNIT 2 WAS IN NO MODE (DEFUELED). AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION OCCURRED AS A RESULT OF AN UNDERVOLTAGE SIGNAL GENERATED ON A TRAIN 4160V ESSENTIAL BUS (2ETA). THE UNDERVOLTAGE CONDITION WAS CAUSED WHEN ONE OF THE FUSEHOLDER CONNECTION STABS OF THE UNDERVOLTAGE RELAY POTENTIAL TRANSFORMER (PT) DID NOT MAKE CONTACT WITH THE ESSENTIAL BUS. SUBSEQUENT INVESTIGATION REVEALED THAT THE CONNECTION STAB HAD BEEN BENT AS A RESULT OF A LOOSE PT FUSEHOLDER. THE LOOSE FUSEHOLDER WAS DETERMINED TO BE CAUSED BY IMPROPER REMOVAL OF FUSES FROM THE PT FUSEHOLDER. THIS INCIDENT IS ATTRIBUTED TO A MANAGEMENT DEFICIENCY, INADEQUATE TRAINING PROVIDED FOR PERSONNEL PERFORMING HIGH VOLTAGE FUSE REMOVAL. MANAGEMENT WILL RE-EVALUATE THE ISSUE CONCERNING OPERATIONS' RESPONSIBILITY OF HIGH VOLTAGE EQUIPMENT MANIPULATION AT CATAWBA. CORRECTIVE ACTION WILL INCLUDE RESOLUTION OF THE RESPONSIBILITY ISSUE, PROCEDURE ENHANCEMENT AND DEVELOPMENT, AND ADDITIONAL TRAINING.

[52] CATAWBA 2 DOCKET 50-414 LER 91-015
TECHNICAL SPECIFICATION VIOLATION DUE TO LACK OF A BORATION FLOW PATH DURING REFUELING ACTIVITIES DUE TO DEFICIENT COMMUNICATION.
EVENT DATE: 111891 REPORT DATE: 121891 NSSS: WE TYPE: PWR

(NSIC 223686) ON NOVEMBER 18, 1991 UNIT 2 WAS OPERATING IN NO MODE WITH THE CORE DEFUELED. AN ADDITION TO THE REFUELING WATER STORAGE TANK (FWST) WAS MADE FROM THE BORIC ACID TANK (BAT). FWST MAKEUP WAS COMPLETED AT 1022 HOURS ON NOVEMBER 19. CORE RELOADING HAD BEGUN AT 2335 HOURS AND UNIT 2 ENTERED MODE 6, REFUELING. REQUIRED TANK RECIRCULATION WAS PERFORMED. AT 0355 HOURS ON NOVEMBER 20, A CHEMISTRY SAMPLE INDICATED THAT THE FWST BORON CONCENTRATION WAS 1975 PPM, LESS THAN THE TECHNICAL SPECIFICATION REQUIREMENT OF 2000 PPM. THE CONTROL ROOM WAS NOTIFIED UPON RECEIPT OF CONFIRMATORY SAMPLE RESULTS AT 0422 HOURS AND CORE LOADING WAS STOPPED AT 0435 HOURS. A TECHNICAL SPECIFICATION VIOLATION HAD OCCURRED BECAUSE THE MINIMUM REQUIRED FWST BORON CONCENTRATION WAS NOT MET, RESULTING IN LACK OF AN OPERABLE BORATION FLOW PATH. THE FWST WAS DECLARED OPERABLE AT 1430 HOURS AFTER BORON ADDITION AND ACCEPTABLE SAMPLES WERE OBTAINED. FUEL LOADING RESUMED AT 1500 HOURS. THIS INCIDENT IS ATTRIBUTED TO DEFICIENT COMMUNICATION BETWEEN CHEMISTRY AND OPERATIONS REGARDING THE PROPER BORON CONCENTRATION TO USE IN FWST MAKEUP CALCULATIONS. CORRECTIVE ACTIONS INCLUDED TERMINATION OF FUEL LOADING, FWST MAKEUP TO ACHIEVE REQUIRED CONCENTRATION, AND COMMUNICATION ENHANCEMENTS TO PREVENT RECURRENCE.

[53] CATAWBA 2 DOCKET 50-414 LER 91-016
TECHNICAL SPECIFICATION VIOLATION DUE TO FAILURE OF PRESSURIZER POWER OPERATED RELIEF BLOCK VALVE 2NC-31B.
EVENT DATE: 120991 REPORT DATE: 010892 NSSS: WE TYPE: PWR
VENDOR: ROCKWELL MANUFACTURING COMPANY

(NSIC 223838) ON DECEMBER 9, 1991, AT 0230 HOURS, UNIT 2 WAS IN MODE 5, COLD SHUTDOWN. THE PRESSURIZER (PZR) FAILED TO DEPRESSURIZE PRIOR TO THE PZR NITROGEN FILL. A PZR POWER OPERATED RELIEF VALVE (PORV) BLOCK VALVE, 2NC-31B, WAS NOT OPEN AS INDICATED. 2NC-31B REMAINED CLOSED BECAUSE THE STEM SEPARATED FROM THE WEDGE ASSEMBLY. THE PZR PORVS HAD BEEN CLOSED IN THE AUTO OPEN MODE SINCE DECEMBER 2 AT 2000 HOURS. WITH THE FINAL PZR SAFETY RELIEF VALVE IN PLACE, T/S REQUIRE THAT AN

INOPERABLE PZR PORV BE RETURNED TO SERVICE WITHIN 7 DAYS OR DEPRESSURIZE AND VENT THE NC SYSTEM WITHIN THE NEXT 8 HOURS. DUE TO THE 2NC-31B FAILURE, ONE OF THE TWO PORVS REQUIRED TO BE OPERABLE WAS NOT AVAILABLE AND THE T/S ACTION REQUIREMENT WAS NOT SATISFIED. THEREFORE, THE T/S VIOLATION OCCURRED. THIS EVENT IS ATTRIBUTED TO EQUIPMENT FAILURE. THE UNIT 1 AND 2 PZR PORV ISOLATION VALVES WERE RADIOGRAPHED TO VERIFY THAT THE VALVES ARE OPEN. 2NC-31B WAS REPLACED AND THE INITIAL STEM ANALYSIS WAS PERFORMED. PLANNED CORRECTIVE ACTIONS INCLUDE VERIFICATION OF BLOCK VALVE POSITION FOLLOWING STROKE TESTS AND TO CONTINUE EVALUATION OF STEM FAILURE ANALYSIS DATA.

[54] CLINTON 1 DOCKET 50-461 LER 91-007
UNDETECTED FAILURE OF AN AIR HANDLING UNIT AND OPERATIONS LACK OF UNDERSTANDING OF THE AIR HANDLING UNIT DESIGN AND OPERATIONS RESULTED IN INOPERABLE LEAK DETECTION INSTRUMENTS.
EVENT DATE: 121291 REPORT DATE: 011392 NSSS: GE TYPE: BWR
VENDOR: TRANE COMPANY

(NSIC 223808) ON 12/12/91, WITH THE PLANT IN POWER OPERATION, THE 'B' REACTOR WATER CLEANUP (RWCU) HEAT EXCHANGER (HX) ROOM AIR HANDLING UNIT (AHU) FAN FAILED RESULTING IN INOPERABLE LEAK DETECTION (LD) DIFFERENTIAL TEMPERATURE (DELTA T) INSTRUMENTATION. THE SHIFT SUPERVISOR NOTED THE 'B' RWCU HX ROOM TEMPERATURE INCREASING AND THE DELTA T RECORDER INDICATING ZERO, INDICATING A POSSIBLE PROBLEM WITH THE LD SYSTEM. DURING THE EVENT, THE LD INSTRUMENTS WERE INITIALLY DECLARED INOPERABLE AND THEN INCORRECTLY DECLARED OPERABLE DUE TO OPERATIONS LACK OF UNDERSTANDING OF THE AHU DESIGN AND OPERATION. SUBSEQUENT TO THIS EVENT, A REVIEW OF THE DELTA T RECORDER STRIP CHART SHOWED THAT THE FAN FAILED APPROX. 2 HOURS AND 25 MINUTES BEFORE THE SHIFT SUPERVISOR HAD INDICATION OF THE FAILURE. THEREFORE, THE REQUIREMENTS OF TECH SPEC 3.3.2 TO ISOLATE THE HX WITHIN 1 HOUR WERE NOT MET. NO REMOTE INDICATION IS AVAILABLE TO IDENTIFY THE FAILURE OF THE AHU WITHIN 1 HOUR. THE CAUSES OF THIS EVENT ARE AN UNDETECTED FAILURE OF A FAN RESULTING IN INOPERABLE LD INSTRUMENTS AND OPERATIONS LACK OF UNDERSTANDING OF THE EFFECT OF THE FAILED FAN ON THE LD INSTRUMENTS. CORRECTIVE ACTIONS INCLUDE EVALUATING THE NEED FOR A TECHNICAL SPECIFICATION CHANGE, REWORKING THE FAILED FAN, AND TRAINING FOR OPERATIONS PERSONNEL ON THIS EVENT.

[55] CLINTON 1 DOCKET 50-461 LER 91-008
FAILURE OF THE REACTOR RECIRCULATION FLOW CONTROL VALVE POSITION FEEDBACK LOOP AND LEAKING HYDRAULIC VALVES RESULTED IN ENTRANCE INTO RESTRICTED OPERATING REGION AND MANUAL SCRAM.
EVENT DATE: 122291 REPORT DATE: 012092 NSSS: GE TYPE: BWR
VENDOR: GREER HYDRAULICS, INC.
SCHAEVITZ ENGINEERING

(NSIC 223843) ON DECEMBER 22, 1991, WITH THE PLANT IN POWER OPERATION, A MANUAL SCRAM WAS INITIATED DUE TO OPERATION IN THE RESTRICTED ZONE OF THE THERMAL POWER VERSUS CORE FLOW MAP OF TECHNICAL SPECIFICATION 3.4.1.3. WHILE THE CONTROL ROOM OPERATORS CONDUCTED A SCHEDULED POWER REDUCTION, THE "B" REACTOR RECIRCULATION (RR) FLOW CONTROL VALVE (FCV) STARTED TO OPERATE ERRATICALLY. CONTROL ROOM OPERATORS, ATTEMPTING TO STOP THE ERRATIC OPERATION, HYDRAULICALLY LOCKED OUT THE "B" RR FCV. THE "B" RR FCV CONTINUED TO OPERATE ERRATICALLY AND THEN SUDDENLY CLOSED ENOUGH TO CAUSE CORE FLOW TO DROP INTO THE RESTRICTED ZONE. THE CONTROL ROOM OPERATOR IMMEDIATELY INITIATED A REACTOR SCRAM IN ACCORDANCE WITH PLANT PROCEDURE 3005.01, "UNIT POWER CHANGES". NO REACTOR POWER OSCILLATIONS WERE OBSERVED. THE CAUSES OF THIS EVENT ARE ATTRIBUTED TO THE FAILURE OF THE "B" RR FCV POSITION FEEDBACK LOOP AND TO INTERNAL LEAKAGE OF THE HYDRAULIC VALVES ON THE HYDRAULIC POWER UNIT. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDE REPLACING THE RR FCV POSITION AND VELOCITY TRANSDUCERS, REPLACING THE LEAKING HYDRAULIC VALVES, EXAMINING AND TESTING THE FEEDBACK SENSOR AND ASSOCIATED LEAD, AND DETERMINING IF ANY LONG-TERM CORRECTIVE ACTIONS ARE APPROPRIATE.

[56] CLINTON 1 DOCKET 50-461 LER 92-001
INTERNAL FAULT IN THE "B" PHASE MAIN POWER TRANSFORMER RESULTED IN A TURBINE GENERATOR TRIP AND AN AUTOMATIC REACTOR SCRAM.
EVENT DATE: 010492 REPORT DATE: 013192 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223946) ON JANUARY 4, 1992, WITH THE PLANT IN POWER OPERATION AT 99 PERCENT REACTOR POWER, THE "B" PHASE MAIN POWER TRANSFORMER (MPT 1B) FAILED DUE TO AN INTERNAL FAULT. THE TRANSFORMER FAILURE RESULTED IN A TURBINE GENERATOR TRIP AND AN AUTOMATIC REACTOR SCRAM. THE AUTOMATIC REACTOR SCRAM OCCURRED DUE TO THE TURBINE CONTROL VALVE FAST CLOSURE. WITHIN SECONDS OF THE SCRAM, THE TURBINE DRIVEN REACTOR FEED PUMP (TDRFP) 1B TRIPPED. ADDITIONALLY, AFTER THE REACTOR SCRAM WAS RESET, A SCRAM DISCHARGE VOLUME (SDV) DRAIN VALVE FAILED TO REOPEN AND A SDV VENT VALVE ONLY OPENED TO AN INTERMEDIATE POSITION. THE CAUSE OF THE SCRAM WAS AN INTERNAL FAULT IN MPT 1B. THE CAUSE OF THE TDRFP 1B TRIP WAS ATTRIBUTED TO A WORN THRUST BEARING, AND THE CAUSE OF THE SDV VENT AND DRAIN VALVES FAILING TO REOPEN WAS ATTRIBUTED TO AIR LEAKAGE PAST THE SEAT OF A THREEWAY SOLENOID VALVE. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDE REPLACING THE FAILED MPT WITH A SPARE MPT, RESTORING THE TDRFP 1B THRUST BEARING CLEARANCE TO ORIGINAL MANUFACTURER SPECIFICATIONS, AND REBUILDING THE THREE-WAY SOLENOID VALVE.

[57] COMANCHE 1 DOCKET 50-445 LER 90-024 REV 01
UPDATE ON PERSONNEL ERROR RESULTING IN FAILURE TO SATISFY TECHNICAL SPECIFICATION STAGGERED TEST BASIS REQUIREMENT.
EVENT DATE: 082490 REPORT DATE: 020792 NSSS: WE TYPE: PWR

(NSIC 223944) ON AUGUST 24, 1990, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 WAS IN MODE 1, POWER OPERATIONS, WITH REACTOR POWER AT 100 PERCENT. WHILE PREPARING TO PERFORM SURVEILLANCE TESTING IN CONTAINMENT PURGE AND HYDROGEN PURGE ISOLATION VALVES, TEST DEPARTMENT PERSONNEL DISCOVERED THAT TESTING ACTIVITIES WERE NOT BEING PERFORMED ON A STAGGERED TEST BASIS AS SPECIFIED BY THE ASSOCIATED TECHNICAL SPECIFICATION. THE EVENT WAS CAUSED BY PERSONNEL ERROR DURING INITIAL SURVEILLANCE PROGRAM DEVELOPMENT. THE INDIVIDUAL RESPONSIBLE FOR INPUTTING DATA TO THE SCHEDULING DATABASE OVERLOOKED THE REQUIREMENT. CORRECTIVE ACTIONS INCLUDED TESTING AND PROGRAM REVIEW.

[58] COMANCHE 1 DOCKET 50-445 LER 90-037 REV 01
UPDATE ON BLACKOUT SEQUENCER ACTUATION DUE TO PERSONNEL ERROR.
EVENT DATE: 110590 REPORT DATE: 123091 NSSS: WE TYPE: PWR

(NSIC 223743) AT APPROXIMATELY 0300 ON NOVEMBER 5, 1990, ELECTRICAL MAINTENANCE PERSONNEL WERE CONDUCTING A TEST OF THE NUCLEAR SAFETY RELATED TRAIN B 6.9 KV SWITCHGEAR UNDER VOLTAGE (UV) RELAYS. AT 0443, WHILE ATTEMPTING TO RE-LAND A WIRE, THE ELECTRICIAN INADVERTENTLY MADE CONTACT WITH AN ENERGIZED POINT ON THE UV RELAY, RESULTING IN A UV ACTUATION. AS A RESULT OF THE UV ACTUATION, THE REACTOR OPERATOR (RO) OBSERVED THE TRANSFER OF TRAIN B 6.9 KV SWITCHGEAR TO THE ALTERNATE POWER SUPPLY, AND ACTUATION OF THE TRAIN B BLACKOUT SEQUENCER (BOS). AT 0454, THE RO RESET THE BOS, AND RESTORED ACTUATED COMPONENTS TO THEIR ORIGINAL CONFIGURATION. AT 0652 ON NOVEMBER 5, 1990, WITH RESTORATION COMPLETE, THE RO RESTORED THE NORMAL POWER SUPPLY TO TRAIN B 6.9 KV SWITCHGEAR. THE ROOT CAUSE WAS DETERMINED TO BE PERSONNEL ERROR. CORRECTIVE ACTIONS INCLUDE A MEMO TO ELECTRICAL MAINTENANCE PERSONNEL DISCUSSING THIS EVENT.

[59] COMANCHE 1 DOCKET 50-445 LER 91-022
ENGINEERED SAFETY FEATURE ACTUATION DUE TO BIRD IMPACTION OF A INSULATOR AND SUBSEQUENT PERSONNEL ERROR.
EVENT DATE: 090491 REPORT DATE: 100491 NSSS: WE TYPE: PWR

(NSIC 223813) AT 0657 ON 9/4/91, POWER FROM THE PREFERRED 345KV TRANSMISSION LINE WAS MOMENTARILY LOST WHEN THE TRANSMISSION LINE BREAKERS OPENED AND IMMEDIATELY RECLOSED, DUE TO BIRD IMPACTION OF AN INSULATOR ON THE LINE. BOTH SAFEGUARDS BUSSES TRANSFERRED TO THE ALTERNATE POWER SUPPLY AND BOTH BLACKOUT SEQUENCERS (BOS) OPERATED. ALL SAFETY SYSTEMS RESPONDED AS EXPECTED, EXCEPT FOR THE TRAIN A

DIESEL GENERATOR WHICH WAS OUT-OF-SERVICE FOR PRE-PLANNED MAINTENANCE. AT 0754, DURING THE FINAL STAGES OF RECOVERY, THE TRAIN A SAFETY-RELATED BUS WAS INADVERTENTLY DEENERGIZED DUE TO PERSONNEL ERROR. THE TRAIN A BOS OPERATED, AND TRAIN A EQUIPMENT ACTUATED. AT 0800, THE TRAIN A BOS WAS RESET AND THE AFFECTED TRAIN A COMPONENTS RETURNED TO THEIR ORIGINAL CONFIGURATION. AT 0840, RESTORATION PROCEDURES WERE COMPLETE AND THE PLANT WAS STABLE AT 100% POWER. THE IMMEDIATE CAUSE OF THIS EVENT WAS BIRD IMPACTION OF AN INSULATOR ON THE 345KV TRANSMISSION LINE. THE ROOT CAUSE OF THE SECOND EVENT WAS LESS THAN ADEQUATE PERSONNEL PERFORMANCE. CORRECTIVE ACTIONS INCLUDE REPLACEMENT OF THE INSULATOR; A DESIGN MODIFICATION TO REDUCE SUSCEPTIBILITY TO THIS TYPE OF EVENT; AND COUNSELING AND TRAINING OF THE REACTOR OPERATOR INVOLVED.

[60] COMANCHE 1 DOCKET 50-445 LER 91-028
MISSED REFUELING MACHINE AUXILIARY MONORAIL HOIST SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 111291 REPORT DATE: 121291 NSSS: WE TYPE: PWR

(NSIC 223651) ON NOVEMBER 12, 1991, WITH THE REACTOR REFUELED AND THE UPPER REACTOR INTERNALS INSTALLED, THE NEXT TASK WAS TO RELATCH THE CONTROL RODS. THE REFUELING REACTOR OPERATOR (RO) WAS TO ENSURE THAT A LOAD TEST OF THE REFUELING MACHINE AUXILIARY MONORAIL HOIST HAD BEEN PERFORMED PER SURVEILLANCE REQUIREMENTS. AT 0130 ON NOVEMBER 12, 1991, THE SHIFT SUPERVISOR (SS) AND THE REFUELING RO CONFIRMED THAT A LOAD TEST HAD BEEN PREVIOUSLY PERFORMED. AT 0300 ON NOVEMBER 12, 1991, LATCHING OF THE CONTROL RODS BEGAN, AND WAS COMPLETED AT 0615. AT 0645 ON NOVEMBER 12, 1991, THE REFUELING COORDINATOR WAS REVIEWING PAPERWORK WHEN HE DISCOVERED THAT A LOAD TEST HAD NOT BEEN PERFORMED. AFTER FURTHER REVIEW IT WAS DETERMINED THAT THE WRONG LOAD TEST DOCUMENTATION HAD BEEN REVIEWED, AND THAT THE SURVEILLANCE REQUIREMENT HAD NOT BEEN MET. AT 0730 ON NOVEMBER 12, 1991, A LOAD TEST WAS SATISFACTORILY PERFORMED ON THE REFUELING MACHINE AUXILIARY MONORAIL HOIST. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. CORRECTIVE ACTIONS INCLUDED COUNSELING OF THE PERSONNEL INVOLVED.

[61] COMANCHE 1 DOCKET 50-445 LER 91-029
TECHNICAL SPECIFICATION VIOLATION DUE TO STEAM SUPPLY VALVES TO THE TURBINE DRIVEN AUXILIARY FEEDWATER PUMP BEING ISOLATED IN MODE 3.
EVENT DATE: 120491 REPORT DATE: 010392 NSSS: WE TYPE: PWR

(NSIC 223744) DURING RESTORATION FROM COMANCHE PEAK UNIT 1'S FIRST REFUELING OUTAGE, THE HANDSWITCH POSITIONS FOR THE STEAM SUPPLY VALVES TO THE TURBINE DRIVEN AUXILIARY FEED WATER (TDAFW) PUMP WERE LEFT IN PULL-TO-LOCK AFTER ENTERING MODE 3, THUS DEFEATING THE AUTOMATIC FUNCTION OF THESE VALVES REQUIRED BY THE TECHNICAL SPECIFICATIONS (T/S). OPERATORS INCORRECTLY ASSUMED THAT THE T/S 4.0.4 EXCEPTION FOR TESTING THE TDAFW PUMP UNTIL SUFFICIENT STEAM PRESSURE COULD BE PRODUCED MEANT THAT THE PROPER VALVE LINEUP FOR THE TDAFW PUMP WAS NOT REQUIRED UNTIL THE SURVEILLANCE TEST WAS RUN. THE SHIFT SUPERVISOR DISCOVERED THE DISCREPANCY APPROXIMATELY TWO HOURS AFTER ENTERING MODE 3 WHILE MAKING A MAIN CONTROL BOARD WALKDOWN. THE SWITCHES WERE IMMEDIATELY PLACED IN THE AUTO POSITION. THE ROOT CAUSE WAS A COGNITIVE ERROR BY OPERATORS ASSUMING THAT THERE WAS A T/S 3.0.4 EXCEPTION TO THE LIMITING CONDITION FOR OPERATION REQUIRING THE TDAFW PUMP AND ASSOCIATED FLOW PATH TO BE OPERABLE WHEN ENTERING MODE 3 FROM MODE 4. CORRECTIVE ACTIONS INCLUDE COUNSELLING OF PERSONNEL BY THE OPERATIONS MANAGER AND TRAINING OF OPERATORS ON THE REQUIREMENTS OF T/S SECTIONS 3.0 AND 4.0.

[62] COMANCHE 1 DOCKET 50-445 LER 91-030 REV 01
UPDATE ON PERSONNEL ERROR LEADING TO MISPOSITIONED RESIDUAL HEAT REMOVAL SYSTEM CROSSTIE VALVES.
EVENT DATE: 120491 REPORT DATE: 012492 NSSS: WE TYPE: PWR

(NSIC 223841) ON DECEMBER 4, 1991, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 ENTERED MODE 3 WITH TWO MISPOSITIONED VALVES IN THE EMERGENCY CORE COOLING SYSTEM. THE EVENT IS CONSIDERED TO BE A FAILURE TO SATISFY A LIMITING CONDITION FOR OPERATION AND A SURVEILLANCE REQUIREMENT OF THE PLANT'S TECHNICAL SPECIFICATION. THE CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE PERSONNEL ERROR

LEADING TO THE FAILURE TO PROPERLY POSITION THE CROSSTIE VALVES IN THE RESIDUAL HEAT REMOVAL SYSTEM FOLLOWING FILLING OF A PORTION OF THE SYSTEM. CORRECTIVE ACTIONS INCLUDE TRAINING AND PROCEDURE ENHANCEMENT.

[63]   COMANCHE 1   DOCKET 50-445   LER 91-031
CONTAINMENT VENTILATION ISOLATION DUE TO OVERCONSERVATIVE RADIATION MONITOR SETPOINT.
EVENT DATE: 120691   REPORT DATE: 010692   NSSS: WE   TYPE: PWR

(NSIC 223746) AT 0909 ON DECEMBER 6, 1991, THE CONTAINMENT WAS BEING VENTED TO DECREASE CONTAINMENT PRESSURE. AT 0918 THE CONTAINMENT PARTICULATE, IODINE, GASEOUS (PIG) MONITOR GAS CHANNEL SPIKED INTO AN ALARM CONDITION. AS A RESULT, A CONTAINMENT VENTILATION ISOLATION AUTOMATICALLY OCCURRED. AT 0956 ON DECEMBER 6, 1991, THE CONTAINMENT VENTILATION ISOLATION SIGNAL WAS RESET. CONTAINMENT ATMOSPHERIC CONDITIONS WERE NORMAL. THE ROOT CAUSE OF THIS EVENT WAS THAT THE PIG GAS CHANNEL ALARM SETPOINT WAS SET TOO LOW. FURTHERMORE, THE ADMINISTRATIVE PROCEDURE ESTABLISHING THE ALARM SETPOINT CALCULATION REQUIREMENTS DID NOT CONSIDER ALARM SETPOINT CALCULATION WITH NO DETECTABLE NOBLE GAS ACTIVITY IN THE CONTAINMENT ATMOSPHERE. CORRECTIVE ACTION INCLUDED RECALCULATION OF THE ALARM SETPOINT AND A PROCEDURE CHANGE.

[64]   COMANCHE 1   DOCKET 50-445   LER 91-032
PROCEDURE ERROR LEADING TO NONCONSERVATIVE MISCALIBRATION OF POWER RANGE NUCLEAR INSTRUMENTATION CHANNELS.
EVENT DATE: 120691   REPORT DATE: 012492   NSSS: WE   TYPE: PWR

(NSIC 223842) ON DECEMBER 6, 1991, AN INSTRUMENT AND CONTROL TECHNICIAN PERFORMED CALIBRATION ACTIVITIES ON THE FOUR POWER RANGE CHANNELS OF THE NUCLEAR INSTRUMENTATION SYSTEM. AN ERROR IN THE WORK INSTRUCTIONS RESULTED IN INCOMPLETE CALIBRATION OF THE FOUR CHANNELS AND NONCONSERVATIVE INDICATION OF REACTOR POWER. ON DECEMBER 11, FOLLOWING SYNCHRONIZATION OF THE TURBINE-GENERATOR WITH THE GRID, THE SHIFT SUPERVISOR OBSERVED A DISCREPANCY BETWEEN INDICATED AND ACTUAL PLANT CONDITIONS BASED ON PREVIOUS EXPERIENCE WITH THE EVOLUTION. PERFORMANCE OF A UNIT CALORIMETRIC CONFIRMED THE INCONSISTENCY BETWEEN THE INDICATED REACTOR POWER OF 9% AND ACTUAL POWER OF 17%. THE CAUSE OF THE EVENT WAS A PROCEDURE ERROR, AND CORRECTIVE ACTIONS INCLUDE PROCEDURE REVISION AND TRAINING.

[65]   COMANCHE 1   DOCKET 50-445   LER 92-001
PERSONNEL ERROR WHILE TAKING MANUAL CONTROL OF THE GENERATOR PRIMARY WATER SYSTEM LEADS TO A REACTOR TRIP.
EVENT DATE: 010892   REPORT DATE: 020792   NSSS: WE   TYPE: PWR

(NSIC 223945) ON JANUARY 8, 1992, THE REACTOR OPERATOR WAS ATTEMPTING TO REESTABLISH THE REQUIRED DIFFERENTIAL TEMPERATURE BETWEEN THE MAIN GENERATOR PRIMARY WATER SYSTEM AND THE HYDROGEN GAS SYSTEM BY TAKING MANUAL CONTROL OF PRIMARY WATER FLOW. WHILE IN MANUAL CONTROL, PRIMARY WATER TEMPERATURE STARTED TO RISE AND COULD NOT BE CHECKED, WHICH RESULTED IN A TURBINE TRIP FOLLOWED BY A REACTOR TRIP OCCURRED DUE TO HIGH PRIMARY WATER TEMPERATURE. ROOT CAUSES WERE DETERMINED TO BE FAILURE TO UNDERSTAND THE POTENTIAL CONSEQUENCES OF CONTROLLING PRIMARY WATER FLOW IN MANUAL AND FAILURE OF THE SHIFT SUPERVISOR TO ADEQUATELY MONITOR THE EVOLUTION. CONTRIBUTING FACTORS WERE FAILURE TO USE AVAILABLE PROCEDURES, LACK OF SPECIFIC INFORMATION IN PROCEDURES REGARDING THIS RISK, AND THE MALFUNCTION OF THE PRIMARY WATER HIGH TEMPERATURE ALARM. CORRECTIVE ACTIONS INCLUDE COUNSELLING THE SPECIFIC INDIVIDUALS, INTENSIVE TRAINING FOR THE CREW INVOLVED, ENHANCING REQUALIFICATION TRAINING, REVISING PROCEDURES, REPAIRING THE ALARM, AND ESTABLISHING A TASK TEAM TO EVALUATE THE TECHNICAL INFORMATION AVAILABLE TO OPERATE AND MAINTAIN THE MAIN GENERATOR.

[66]   COMANCHE 1   DOCKET 50-445   LER 92-002
PERSONNEL ERROR LEADING TO FAILURE TO SATISFY TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT FOR AREA TEMPERATURE MONITORING.
EVENT DATE: 011592   REPORT DATE: 021492   NSSS: WE   TYPE: PWR

(NSIC 224047) ON JANUARY 15, 1992, AT APPROXIMATELY 0615 CST, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 CONTROL ROOM SUPERVISORY PERSONNEL MADE TASK ASSIGNMENTS BASED ON PLANNED ACTIVITIES FOR THE DAY. THE UNIT SUPERVISOR BECAME DISTRACTED BY OTHER ACTIVITIES AND NEGLECTED TO ASSIGN AN AUXILIARY OPERATOR RESPONSIBILITY FOR TAKING SHIFTLY LOCAL SURVEILLANCE LOGS. THE LOGS SATISFY IN PART THE REQUIREMENT OF TECHNICAL SPECIFICATION 4.7.10 TO DETERMINE AT LEAST ONCE PER 12 HOURS THAT TEMPERATURES IN SPECIFIED AREAS ARE WITHIN LIMITS. THE OVERSIGHT WAS DISCOVERED AT APPROXIMATELY 1300, AND THE SURVEILLANCE WAS SUCCESSFULLY COMPLETED. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR. CORRECTIVE ACTIONS INCLUDE IMPROVED CONTROL OVER AS SIGNMENT OF DUTIES AND DISPOSITION OF COMPLETED LOGS.

[67]   CONNECTICUT YANKEE   DOCKET 50-213   LER 89-006 REV 02
UPDATE ON HEATING STEAM CONTAINMENT ISOLATION VALVES FAILED SURVEILLANCE TEST.
EVENT DATE: 041489   REPORT DATE: 020792   NSSS: WE   TYPE: PWR
VENDOR: CONTROMATICS CORP.

(NSIC 223955) ON 4/14/89 AT 0805, WITH THE PLANT IN MODE 1 AT 100% POWER, THE TWO CONTAINMENT ISOLATION VALVES FOR HEATING STEAM TO CONTAINMENT (HS-TV-380 AND 381) FAILED TO OPERATE DURING QUARTERLY SURVEILLANCE TESTING. THESE FAILURES CONSTITUTED A LOSS OF CONTAINMENT INTEGRITY. THE OPERATORS IMMEDIATELY CLOSED MANUAL ISOLATION VALVES FOR PENETRATION AND COMMENCED A LOAD REDUCTION AT 0903. ONE VALVE WAS VERIFIED OPERABLE AT 0905. THE LOAD REDUCTION WAS TERMINATED AT 0911. AN ENGINEERING EVALUATION DETERMINED THE CAUSE OF THE EVENT TO BE A COMBINATION OF AN UNDERSIZED AIR OPERATOR AND USE OF A SEAT MATERIAL THAT REQUIRES INCREASED OPERATING TORQUE AT HIGHER TEMPERATURES. SHORT TERM CORRECTIVE ACTION CONSISTED OF LOCKING THE TWO CONTAINMENT TRIP VALVES IN THE CLOSED POSITION. THIS SUPPLEMENTAL REPORT IS BEING SUBMITTED TO PROVIDE FOLLOW-UP INFORMATION ON THE ROOT CAUSE OF THE EVENT AND CORRECTIVE ACTIONS TAKEN TO PROVIDE INFORMATION REGARDING SIGNIFICANT DESIGN DEFICIENCY. THE EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B) AS IT INVOLVED A CONDITION PROHIBITED BY TECH SPECS, UNDER 10CFR50.73(A)92)(II) SINCE IT REPRESENTED DEGRADATION OF A CONTAINMENT BOUNDARY, AND UNDER 10CFR50.73(A)(2)(VII)(C) DUE TO A DESIGN DEFICIENCY THAT INVOLVES A COMMON MODE FAILURE MECHANISM FOR REDUNDANT COMPONENTS.

[68]   CONNECTICUT YANKEE   DOCKET 50-213   LER 90-024 REV 01
UPDATE ON CONTAINMENT ISOLATION VALVE CC-TV-920 DECLARED INOPERABLE.
EVENT DATE: 103190   REPORT DATE: 020792   NSSS: WE   TYPE: PWR
VENDOR: CONTROMATICS CORP.

(NSIC 223923) ON 10/31/90, AT 1039 HOURS, WITH THE PLANT IN MODE 5 (COLD SHUTDOWN) CONTAINMENT ISOLATION VALVE CC-TV-920 (COMPONENT COOLING WATER SUPPLY TO THE NEUTRON SHIELD TANK COOLER) FAILED TO FULLY CLOSE DURING SURVEILLANCE TESTING AND WAS DECLARED INOPERABLE. THE ROOT CAUSE WAS DETERMINED TO BE EXCESSIVE FRICTION IN THE STEM AREA OF THE VALVE DUE TO LACK OF LUBRICATION AND POSSIBLE METAL-TO-METAL CONTACT DUE TO STEM/ACTUATOR MISALIGNMENT. THE VALVE WAS DISASSEMBLED, LUBRICATED, REASSEMBLED AND SUCCESSFULLY RETESTED. SHORT TERM CORRECTIVE ACTION CONSISTED OF INCREASING THE TEST FREQUENCY OF THE VALVE FOR AN UNDETERMINED PERIOD. LONG TERM CORRECTIVE ACTION CONSISTS OF EVALUATING IF THE VALVE SHOULD BE MODIFIED OR REPLACED. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B) SINCE IT INVOLVED A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS. THIS SUPPLEMENTAL REPORT IS BEING SUBMITTED TO PROVIDE FOLLOW-UP INFORMATION ON ROOT CAUSE AND CORRECTIVE ACTIONS AS WELL AS TO PROVIDE INFORMATION REGARDING A SIGNIFICANT VALVE DESIGN DEFICIENCY. SINCE THIS DESIGN DEFICIENCY INVOLVES A COMMON MODE FAILURE MECHANISM FOR REDUNDANT COMPONENTS, THIS EVENT IS ALSO REPORTABLE UNDER 10CFR50.73(A)92)(VII)(C).

[69]   CONNECTICUT YANKEE   DOCKET 50-213   LER 91-023
STEAM GENERATOR EDDY CURRENT TESTING RESULTS CLASSIFIED AS CATEGORY C-3.
EVENT DATE: 111091   REPORT DATE: 120691   NSSS: WE   TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 223593) SCHEDULED STEAM GENERATOR (SG) EDDY CURRENT TESTING (ECT) IS BEING CONDUCTED DURING THE CURRENT REFUELING OUTAGE IN ACCORDANCE WITH ASME SECTION XI

AND TECHNICAL SPECIFICATION 4.4.5. ON NOVEMBER 9 AT APPROXIMATELY 1600 HOURS, WITH THE PLANT IN MODE 6 (REFUELING), IT WAS DETERMINED THAT ANALYZED ECT DATA PLACED SG NO. 2 INTO THE C-3 CATEGORY. A PROMPT REPORT OF THIS EVENT WAS MADE UNDER 10CFR50.72(B)(2)(I). SUBSEQUENTLY, ON NOVEMBER 10, 1991, ANALYZED ECT RESULTS FOR SG NO. 1 PLACED IT IN CATEGORY C-3, THUS REQUIRING EXPANSION OF THE ECT PROGRAM TO 100 PERCENT OF THE TUBES IN ALL FOUR SG'S. THE CAUSE OF THE TUBE DEGRADATION HAS NOT BEEN DETERMINED. ADDITIONAL INFORMATION ON ALL FOUR SG'S WILL BE PROVIDED IN A SUPPLEMENTAL REPORT AFTER ALL TESTING AND EVALUATIONS ARE COMPLETED. ALL TUBES WITH DEGRADATION GREATER THAN OR EQUAL TO THE TECHNICAL SPECIFICATIONS LIMITS WILL BE PLUGGED. THIS EVENT IS REPORTABLE PER 10CFR50.73(A)(2)(I)(B) SINCE IT INVOLVES A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[70] CONNECTICUT YANKEE DOCKET 50-213 LER 91-024 REV 01
UPDATE ON INOPERABLE CONTAINMENT ISOLATION VALVE DUE TO A DESIGN DEFICIENCY.
EVENT DATE: 111491 REPORT DATE: 121191 NSSS: WE TYPE: PWR

(NSIC 223594) ON 10/16/91, AT 1600, WITH THE PLANT IN MODE 1 AT 100% POWER, A DESIGN DEFICIENCY WAS IDENTIFIED THAT COULD POSSIBLY PREVENT THE CONTAINMENT ISOLATION VALVE INSIDE OF CONTAINMENT FOR THE REACTOR COOLANT LETDOWN LINE (LD-TV-230) FROM CLOSING FOLLOWING RECEIPT OF A CONTAINMENT ISOLATION SIGNAL. SPECIFICALLY, A FAILURE OF THE AIR REGULATOR THAT SUPPLIES AIR TO LD-TV-230 VIA A THREE WAY SOLENOID VALVE COULD RESULT IN AIR PRESSURE GREATER THAN THE DESIGN MAXIMUM OPERATING PRESSURE DIFFERENTIAL (MOPD) BEING SUPPLIED TO THE SOLENOID VALVE. THE RESULTING AIR PRESSURE COULD HAVE PREVENTED THE SOLENOID VALVE FROM VENTING AIR FROM LD-TV-230 AFTER BEING DEENERGIZED, WHICH IN TURN COULD HAVE PREVENTED LD-TV-230 FROM GOING CLOSED WHEN REQUIRED TO ISOLATE THE CONTAINMENT. IMMEDIATE CORRECTIVE ACTIONS CONSISTED OF INITIATING ALTERNATE LETDOWN AND ISOLATING THE NORMAL LETDOWN LINE IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.6.1.1. IN ADDITION, A DETAILED OPERABILITY EVALUATION WAS INITIATED. ON NOVEMBER 14, 1991, THE DETAILED OPERABILITY EVALUATION CONCLUDED THAT LD-TV-230 WAS INOPERABLE WITH THIS CONDITION EXISTING. THIS EVENT IS REPORTABLE UNDER 10 CFR 50.73 (A)(2)(I)(B), SINCE THE PLANT WAS IN OPERATION WITH A CONDITION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. THE ROOT CAUSE OF THIS EVENT WAS A DESIGN DEFICIENCY.

[71] CONNECTICUT YANKEE DOCKET 50-213 LER 91-025
FAILURE TO MAINTAIN CONTAINMENT PENETRATION ALIGNMENT DURING CORE ALTERATIONS.
EVENT DATE: 111491 REPORT DATE: 121191 NSSS: WE TYPE: PWR

(NSIC 223595) ON NOVEMBER 14, 1991, WITH THE PLANT IN MODE 6 AND CORE ALTERATIONS IN PROGRESS, OPERATORS DISCOVERED THAT ALL CONTAINMENT PENETRATIONS HAD NOT BEEN PROPERLY ALIGNED AS REQUIRED BY PLANT TECHNICAL SPECIFICATION 3.9.4. CORE ALTERATIONS WERE IMMEDIATELY HALTED AND CONTAINMENT PENETRATIONS WERE VERIFIED TO BE PROPERLY ALIGNED. A PROMPT REPORT OF THIS EVENT WAS MADE UNDER 10CFR50.72(B)(2)(III). THE CAUSE OF THIS EVENT WAS PROCEDURAL INADEQUACY. CORRECTIVE ACTIONS INVOLVE REVISION OF PROCEDURES TO PROVIDE A SEPARATE CLEARANCE FOR CONTAINMENT PENETRATIONS, WALKDOWN VERIFICATION OF CONTAINMENT PENETRATIONS, AND ENHANCED MEASURES FOR CONTROLLING THE CONTAINMENT BOUNDARY ONCE IT IS ESTABLISHED. LESSONS LEARNED WILL BE INCORPORATED INTO OPERATOR TRAINING PROGRAMS. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(V)(C) AS A CONDITION THAT ALONE COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF A SYSTEM NEEDED TO CONTROL THE RELEASE OF RADIOACTIVE MATERIAL.

[72] CONNECTICUT YANKEE DOCKET 50-213 LER 91-026
MISSED SURVEILLANCE ON EMERGENCY DIESEL GENERATOR DUE TO PERSONNEL ERROR.
EVENT DATE: 111991 REPORT DATE: 121891 NSSS: WE TYPE: PWR

(NSIC 223688) ON NOVEMBER 19, 1991, AT 0800 HOURS, WITH THE PLANT IN A REFUELING OUTAGE AND THE CORE OFFLOADED TO THE SPENT FUEL POOL, OPERATIONS PERSONNEL DISCOVERED THE 31 DAY SURVEILLANCE FOR EMERGENCY DIESEL GENERATOR EG-2B HAD NOT BEEN PERFORMED AS REQUIRED FOR THE MONTH OF OCTOBER. THE REDUNDANT DIESEL GENERATOR (EG-2A) WAS INOPERABLE HAVING BEEN REMOVED FROM SERVICE ON OCTOBER 21

FOR MAINTENANCE PURPOSES. UPON DISCOVERY, THE SURVEILLANCE WAS IMMEDIATELY AND SUCCESSFULLY PERFORMED. THE ROOT CAUSE OF THE EVENT WAS A PERSONNEL ERROR IN UPDATING THE STATUS BOARD UTILIZED FOR TRACKING AND SCHEDULING OPERATIONS DEPARTMENT SURVEILLANCES. THE ERRONEOUS INFORMATION FROM THE STATUS BOARD WAS THEN TRANSFERRED TO THE CONTROL ROOM SURVEILLANCE TRACKING PROCEDURE. CORRECTIVE ACTIONS TO ASSURE THE TWO TRACKING SYSTEMS REMAIN INDEPENDENT OF EACH OTHER INCLUDE REQUIRING INDEPENDENT VERIFICATION OF ALL STATUS BOARD ENTRIES AND REQUIRING CONTROL ROOM PERSONNEL TO PERSONALLY REVIEW EACH COMPLETED SURVEILLANCE PROCEDURE PRIOR TO UPDATING THE CONTROL ROOM SURVEILLANCE TRACKING PROCEDURE. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B) SINCE IT IS A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[73]   CONNECTICUT YANKEE   DOCKET 50-213   LER 91-027
MISSED SURVEILLANCE ON MAIN STACK GAS SAMPLER FLOW RATE MONITOR DUE TO A PERSONNEL ERROR.
EVENT DATE: 112191   REPORT DATE: 122091   NSSS: WE   TYPE: PWR

(NSIC 223667) ON NOVEMBER 21, 1991, WITH THE PLANT IN A REFUELING OUTAGE AND THE CORE OFFLOADED TO THE SPENT FUEL POOL, IT WAS DETERMINED THAT THE REQUIRED TEST INTERVAL FOR SURVEILLANCE PROCEDURE SUR 5.2-84, "REPLACEMENT OF THE CONTAINMENT GAS AND MAIN STACK GAS SAMPLER FLOW RATE MONITOR", HAD BEEN EXCEEDED BY APPROXIMATELY TEN WEEKS. THE HADDAM NECK TECHNICAL SPECIFICATIONS REQUIRE THAT THE MAIN STACK GAS SAMPLER FLOW RATE MONITOR BE CALIBRATED ONCE EVERY EIGHTEEN MONTHS. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR IN MISINTERPRETATION OF THE PLANT TECHNICAL SPECIFICATION SURVEILLANCE FREQUENCY REQUIREMENTS. THE SURVEILLANCE WAS SUBSEQUENTLY PERFORMED ON NOVEMBER 21, 1991 WITH SATISFACTORY RESULTS. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B) SINCE IT INVOLVES A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[74]   CONNECTICUT YANKEE   DOCKET 50-213   LER 91-029
POTENTIAL STEAM LINE BREAK PROTECTION SYSTEM INOPERABILITY DUE TO POSTULATED HIGH ENERGY LINE BREAKS.
EVENT DATE: 120591   REPORT DATE: 010392   NSSS: WE   TYPE: PWR

(NSIC 223761) ON DECEMBER 5, 1991 AT 1550 HOURS, WITH THE PLANT IN A REFUELING OUTAGE AND THE CORE OFF-LOADED TO THE SPENT FUEL POOL, THE RESULTS OF AN ENGINEERING ANALYSIS INDICATED THAT THE MAXIMUM POSTULATED TEMPERATURE WHICH COULD BE EXPECTED IN THE TERRY TURBINE BUILDING DURING A HIGH ENERGY LINE BREAK (HELB) WOULD BE APPROXIMATELY 320 DEGREES (F). THE ENGINEERING ANALYSIS FURTHER CONCLUDED THAT THE ENVIRONMENTAL CONDITIONS PRESENTED BY THE HELB WOULD RENDER THE STEAM LINE BREAK (SLB) DETECTION PORTION OF THE REACTOR PROTECTION SYSTEM (RPS) INOPERABLE. BASED ON THESE FINDINGS, A PROMPT REPORT WAS ISSUED PER 10CFR50.72(B)(2)(III)(B) AND (D) ON DECEMBER 5, 1991. THIS EVENT IS BEING CONSERVATIVELY REPORTED UNDER 10CFR50.73(A)(2)(VII)(A) AS THE POSTULATED ENVIRONMENTAL CONDITIONS IN THE TERRY TURBINE BUILDING WOULD HAVE RESULTED IN ALL FOUR CHANNELS OF SLB DETECTION BEING INOPERABLE. CORRECTIVE ACTION WILL CONSIST OF RELOCATING THE SLB DETECTION TRANSMITTERS. THIS ACTION WILL BE COMPLETE PRIOR TO STARTUP FROM THE CURRENT REFUELING OUTAGE.

[75]   CONNECTICUT YANKEE   DOCKET 50-213   LER 91-030
POTENTIAL VALVE INOPERABILITY AND EXCESSIVE LEAKAGE DUE TO A DESIGN INADEQUACY.
EVENT DATE: 122391   REPORT DATE: 012292   NSSS: WE   TYPE: PWR
VENDOR: CONTROMATICS CORP.
        JENKINS VALVE CORP.

(NSIC 223820) ON DECEMBER 23, 1991 AT 1130 HOURS, WITH THE PLANT IN A REFUELING OUTAGE AND THE CORE OFFLOAD TO THE SPENT FUEL POOL, AN ENGINEERING REVIEW DETERMINED THAT THE TEFLON SEAT MATERIAL IN SEVERAL BALL VALVES IN THE CONTAINMENT SUMP RECIRCULATION FLOWPATH COULD DEGRADE DURING POST LOCA CONTAINMENT SUMP RECIRCULATION, CAUSING UNACCEPTABLE SYSTEM LEAKAGE. ON JANUARY 13, 1992, FURTHER REVIEW DETERMINED THAT ADDITIONAL VALVES IN THE SUMP RECIRCULATION FLOW PATH AND IN ISOLATION OR CONTAINMENT BOUNDARY SERVICE WERE SUBJECT TO THE SAME PHENOMENON. THIS REVIEW RESULTED FROM FOLLOWUP TO OTHER BALL

VALVE PROBLEMS AS DISCUSSED IN LERS 89-006 AND 90-024. THE ROOT CAUSE OF THE EVENT IS A DESIGN INADEQUACY, IN THAT, AT THE TIME THESE VALVES WERE SPECIFIED AND INSTALLED, THE ABILITY OF TEFLON SEAT MATERIAL TO FUNCTION UNDER SPECIFIED RADIATION AND TEMPERATURE ENVIRONMENTS WAS NOT WELL UNDERSTOOD. CORRECTIVE ACTION CONSISTS OF BLANKING OR REPLACING THE AFFECTED VALVES. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(V)(C) FOR THOSE VALVES THAT COULD RESULT IN UNACCEPTABLE POST LOCA RELEASE OF RADIOACTIVE MATERIALS AND UNDER 10CFR50.73(A)(2)(I)(B) FOR THOSE CONTAINMENT ISOLATION OR BOUNDARY VALVES THAT COULD BECOME INOPERABLE UNDER POST LOCA CONDITIONS.

[76] COOK 2 DOCKET 50-316 LER 91-009
INOPERABLE ICE CONDENSER DUE TO INCORRECT FLOW PASSAGE INSPECTIONS.
EVENT DATE: 111291 REPORT DATE: 121291 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 1 (PWR)

(NSIC 222659) AN ENGINEERING EVALUATION REVEALED THAT THE ICE CONDENSER FLOW PASSAGE INSPECTION, TO SATISFY ICE CONDENSER TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT, DID NOT INCLUDE THE ENTIRE FLOW AREA ASSUMED IN THE SHORT TERM CONTAINMENT INTEGRITY ANALYSES PRESENTED IN THE UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR). SINCE THE FLOW AREA ASSUMED IN THE CONTAINMENT INTEGRITY ANALYSIS WAS NOT FULLY INSPECTED, THE POTENTIAL EXISTED FOR AN UNACCEPTABLE DEGRADATION OF THE FLOW AREA. THE DEFICIENCY WAS DETERMINED TO BE REPORTABLE ON NOVEMBER 12, 1991, AND BOTH UNIT 1 AND 2 ICE CONDENSERS WERE DECLARED INOPERABLE. THE SURVEILLANCE TEST PROCEDURE WAS REVISED TO INCLUDE THE NEW INSPECTION AREAS. THE ICE CONDENSER FLOW PASSAGES IN BOTH UNITS WERE INSPECTED, IN ACCORDANCE WITH THE REVISED PROCEDURE, AND FOUND TO BE OPERABLE PRIOR TO EXPIRATION OF THE TECHNICAL SPECIFICATION LIMITING CONDITION FOR OPERATION TIME LIMIT.

[77] COOK 2 DOCKET 50-316 LER 91-010
REACTOR PROTECTION SYSTEM ACTUATION DUE TO LOW-LOW STEAM GENERATOR LEVEL WHEN STEAM PRESSURE INCREASED FROM MAIN TURBINE CONTROL VALVE CLOSURE.
EVENT DATE: 111591 REPORT DATE: 121691 NSSS: WE TYPE: PWR
VENDOR: HANCOCK CO.

(NSIC 223675) ON 11/15/91 AT 1113 HOURS, THE UNIT 2 REACTOR TRIPPED AS A RESULT OF A LOW-LOW STEAM GENERATOR (SG) WATER LEVEL IN SG 21. PRIOR TO THE REACTOR TRIP, INSTRUMENTATION WAS BEING INSTALLED ON THE MAIN TURBINE CONTROL FLUID CIRCUIT. THE PURPOSE OF THIS INSTRUMENT WAS TO INVESTIGATE OSCILLATIONS OF CONTROL VALVE A WHICH WAS SCHEDULED TO BE TESTED LATER THAT EVENING. A PRESSURE INDICATOR WHICH MEASURES THE OUTPUT OF THE TURBINE OF OPERATING DEVICE WAS TO BE ISOLATED AND ANOTHER TRANSMITTER WAS TO BE INSTALLED IN PARALLEL TO IT. HOWEVER, UNKNOWN TO PERSONNEL INVOLVED, WORN THREADS ON THE STEM OF THE MANUAL ISOLATION VALVE PREVENTED THE PRESSURE INDICATOR FROM BEING COMPLETELY ISOLATED. AS A DRAIN PLUG WAS REMOVED TO VENT PRESSURE FROM THE INSTRUMENT ISOLATION VALVE, PRESSURE ON THE OUTPUT OF THE OPERATING DEVICE WAS ALSO DECREASED. THIS EFFECTIVELY LOWERED THE SETPOINT SUPPLIED TO THE SPEED GOVERNOR, WHICH COMPARES ACTUAL TURBINE SPEED TO THE OPERATING DEVICE SETPOINT, AND FULLY CLOSED THE HIGH PRESSURE AND LOW PRESSURE TURBINE CONTROL VALVES. WITH THE STEAM SUPPLY ISOLATED, SG PRESSURE RAPIDLY INCREASED AND CAUSED SG LEVEL TO DECREASE BELOW THE REACTOR TRIP SETPOINT. THE ROOT VALVE'S BONNET WAS REPLACED AND DAMAGED STEM THREADS WERE REPAIRED. DURING THE OUTAGE, REPAIRS WERE ALSO COMPLETED TO REDUCE THE TURBINE LOAD SWINGS EXPERIENCED.

[78] COOPER DOCKET 50-298 LER 91-018
FAILURE OF A 4160V CIRCUIT BREAKER TO TRIP DUE TO STICKING TRIP LATCH ROLLER MECHANISM CAUSED BY HARDENED LUBRICANT.
EVENT DATE: 032391 REPORT DATE: 010292 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223708) ON MARCH 23, 1991, 4160V CIRCUIT BREAKER 1CN, WHICH FEEDS THE 'A' REACTOR RECIRC MOTOR GENERATOR (RRMG) MOTOR, FAILED TO OPEN UPON ACTUATION OF THE BREAKER CONTROL SWITCH. A SECOND ATTEMPT TO TRIP THE BREAKER WAS SUCCESSFUL, ALTHOUGH IT TOOK APPROXIMATELY TEN SECONDS FOR THE BREAKER TO OPEN. INSPECTION OF

THE BREAKER IMMEDIATELY AFTER THE EVENT REVEALED THAT THE TRIP LATCH ROLLER WOULD NOT ROTATE FREELY, AND THEREFORE INHIBITED THE BREAKER FROM TRIPPING ON DEMAND. THIS PROBLEM WAS CONSIDERED TO BE SIMILAR TO THREE PREVIOUS 4160V BREAKER DEFICIENCIES THAT HAD BEEN DOCUMENTED IN 1987. THESE EVENTS INVOLVED A FAILURE OF TWO RESIDUAL HEAT REMOVAL (RHR) PUMP MOTOR BREAKERS TO INITIALLY CLOSE. THE ROOT CAUSE OF THIS CONDITION WAS INCOMPLETE MAINTENANCE OF THE 4160V BREAK IN THAT HARDENED GREASE DEPOSITS EXISTED WHICH INHIBITED BREAKER OPERATION. THE 4160V BREAKERS ARE IDENTICAL AND WERE SUPPLIED AS ORIGINAL PLANT EQUIPMENT. WITH THE EXCEPTION OF THE REFURBISHMENT OF TEN SAFETY RELATED BREAKERS TO DATE, THE MAINTENANCE PRACTICES FOR THESE BREAKERS HAVE BEEN IDENTICAL. CONSEQUENTLY, A NUMBER OF 4160V BREAKERS WERE POTENTIALLY SUBJECT TO THE SAME FAILURE MODE. THIS CONDITION WAS DETERMINED TO BE REPORTABLE DURING DOCUMENT CLOSE OUT ACTIVITIES FOR THIS EVENT. AT THE TIME, THE 1991 REFUELING OUTAGE WAS IN PROGRESS.

[79] COOPER DOCKET 50-298 LER 91-013
FAILURE TO ESTABLISH A CONTINUOUS FIRE WATCH FOR AN OBSTRUCTED FIRE DOOR AS REQUIRED BY TECHNICAL SPECIFICATIONS DUE TO PERSONNEL ERROR AND PROCEDURAL DEFICIENCIES.
EVENT DATE: 101891 REPORT DATE: 111891 NSSS: GE TYPE: BWR

(NSIC 223494) ON OCTOBER 25, 1991, DURING A REVIEW OF SURVEILLANCE PROCEDURE (SP) 6.4.5.1, FIRE PROTECTION SYSTEM MONTHLY EXAMINATION, THE FIRE PROTECTION ENGINEER NOTED THAT THE DOOR TO THE STEAM TUNNEL HAD BEEN RECORDED AS BEING OPEN AND OBSTRUCTED FOR WORK IN THE STEAM TUNNEL. SP 6.4.5.1 HAD BEEN CONDUCTED ON OCTOBER 18, 1991. THE STEAM TUNNEL DOOR, A FIRE DOOR WHICH IS A REQUIRED FIRE BARRIER AS SPECIFIED IN TECHNICAL SPECIFICATIONS, IS SUBJECT TO SPECIFIC PROCEDURAL CONTROLS PRESCRIBED IN CNS PROCEDURE 0.16, CONTROL OF FIRE DOORS. THE FIRE PROTECTION ENGINEER CONDUCTED A FOLLOW-UP INSPECTION TO DETERMINE IF THE DOOR WAS STILL OBSTRUCTED, AND IF SO, TO ENSURE A FIRE WATCH WAS POSTED. THE DOOR WAS FOUND TO STILL BE OBSTRUCTED. HOWEVER, NO FIRE WATCH WAS POSTED, CONTRARY TO THE REQUIREMENTS OF CNS PROCEDURE 0.39, FIRE WATCH/FIRE WATCH PATROL ACTIVITIES. AT THE TIME, THE PLANT WAS IN COLD SHUTDOWN FOR THE 1991 REFUELING OUTAGE. TWO ROOT CAUSES CONTRIBUTED TO THIS SITUATION. PERSONNEL ERROR HAS BEEN ASSIGNED DUE TO FAILURE TO FOLLOW PROCEDURAL REQUIREMENTS FOR FIRE WATCHES. PROCEDURE DEFICIENCY HAS BEEN ASSIGNED BECAUSE PROCEDURE 0.39 WAS NOT REFERENCED IN PROCEDURE 0.16, CAUSING CONFUSION AS TO THE CORRECT REQUIREMENTS.

[80] COOPER DOCKET 50-298 LER 91-016
SPURIOUS REACTOR PROTECTION TRIP WHILE SHUTDOWN DUE TO DECONTAMINATION ACTIVITIES UNDER THE REACTOR VESSEL.
EVENT DATE: 111091 REPORT DATE: 121091 NSSS: GE TYPE: BWR

(NSIC 223604) ON 11/10/91, AT 10:41 P.M., A SPURIOUS RPS TRIP OCCURRED WHEN A CONTRACTOR ASSIGNED TO DECONTAMINATION WORK IN THE DRYWELL BUMPED HIS HEAD ON NUCLEAR INSTRUMENTATION (NI) ASSEMBLIES WHILE ON THE EQUIPMENT PLATFORM UNDER THE REACTOR VESSEL. THIS INCIDENT CAUSED AN INTERMEDIATE RANGE MONITOR (IRM) TO SPIKE TO THE HI-HI TRIP SETPOINT. AT THE TIME OF THE EVENT THE PLANT WAS SHUT DOWN FOR THE 1991 REFUELING OUTAGE, WITH ALL FUEL OFF LOADED INTO THE SPENT FUEL STORAGE POOL. FINAL PREPARATIONS FOR RELOADING FUEL INTO THE REACTOR VESSEL WERE IN PROGRESS. THESE PREPARATIONS INCLUDED ACTIVATING THE RPS NON-COINCIDENCE TRIP FUNCTION, SUCH THAT ACTUATION OF A SINGLE NI CHANNEL WOULD RESULT IN A FULL SCRAM. THE ROOT CAUSE OF THIS EVENT IS CONSIDERED TO BE A PROGRAMMATIC DEFICIENCY, IN THAT WHILE THE POTENTIAL FOR AN RPS TRIP WAS RECOGNIZED, SUFFICIENT PRECAUTIONS WERE NOT TAKEN TO PRECLUDE ITS OCCURRENCE. UPON ACTIVATING THE NON-COINCIDENCE TRIP FUNCTION, DECONTAMINATION WORK (AND ANY OTHER NON-CRITICAL ACTIVITY) THAT COULD POTENTIALLY DISTURB THE RESPONSE OF THE NI SYSTEM AND CAUSE A TRIP SHOULD NOT HAVE BEEN PERMITTED. DECONTAMINATION ACTIVITIES UNDER THE REACTOR VESSEL WERE SUSPENDED UNTIL FUEL LOADING WAS COMPLETED AND THE RPS NON-COINCIDENCE TRIP FUNCTION WAS DEACTIVATED.

[81] COOPER DOCKET 50-298 LER 91-017
PARTIAL REACTOR COOLANT SYSTEM, PRIMARY AND SECONDARY CONTAINMENT ISOLATIONS RECEIVED DURING DESIGN CHANGE ACCEPTANCE TESTING DUE TO A BLOWN FUSE.
EVENT DATE: 112391 REPORT DATE: 122391 NSSS: GE TYPE: BWR

(NSIC 223690) ON NOVEMBER 23, 1991 AT 12:06 AM, A SPURIOUS ACTUATION OF SEVERAL REACTOR COOLANT SYSTEM (RCS) PRIMARY CONTAINMENT AND SECONDARY CONTAINMENT ISOLATION VALVES OCCURRED WHEN FUSE 16A-F21 BLEW UPON OR SHORTLY FOLLOWING REMOVAL OF A CIRCUIT LEAD. THE FUSE SUPPLIES 120 VOLT AC CONTROL POWER TO A PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) CIRCUIT. AT THE TIME, THE CIRCUIT WAS UNDERGOING ACCEPTANCE TESTING FOR A DESIGN CHANGE ASSOCIATED WITH INSTALLATION OF TEST JACKS. AT THE TIME OF THE EVENT, THE PLANT WAS SHUTDOWN FOR THE 1991 REFUELING OUTAGE. FUEL HAD BEEN LOADED IN THE VESSEL, WHICH WAS AWAITING REASSEMBLY. THE MOST IMMEDIATE EFFECTS OF THESE ACTUATIONS WERE THE TEMPORARY LOSSES OF SHUTDOWN COOLING AND NORMAL REACTOR BUILDING VENTILATION. AT 12:45 AM, APPROXIMATELY 40 MINUTES FOLLOWING THE SPURIOUS ACTUATION, SHUTDOWN COOLING WAS REESTABLISHED AND NORMAL REACTOR BUILDING VENTILATION WAS RESTORED. DURING THE PERIOD OF TIME THAT SHUTDOWN COOLING WAS OFF, THE RCS TEMPERATURE RISE WAS INSIGNIFICANT. FOR A VERY SHORT TIME PERIOD, THE LEAD WAS NOT UNDER POSITIVE CONTROL OF THE LICENSED OPERATOR. IT IS SUSPECTED THAT THE LEAD SHORTED TO GROUND SUBSEQUENT TO ITS BEING RELEASED. THE ACCEPTANCE TESTING ACTIVITIES PERFORMED BY THE LICENSED OPERATOR IN THE CONTROL ROOM WERE REVIEWED, AND FOUND ACCEPTABLE.

[82] COOPER DOCKET 50-298 LER 91-020
FAILURE OF THE PRIMARY CONTAINMENT INTEGRATED LEAK RATE TEST DUE TO DRYWELL VENT MONITOR SYSTEM AND CONTAINMENT PENETRATION LEAKAGE.
EVENT DATE: 121091 REPORT DATE: 010992 NSSS: GE TYPE: BWR
VENDOR: ANCHOR VALVE CO.
NUCLEAR MEASUREMENTS CORP.

(NSIC 223770) ON 12/16/91, THE CONTAINMENT INTEGRATED LEAK RATE TEST (ILRT) LEAK RATE WAS DETERMINED TO BE 437.5 STANDARD CUBIC FEET PER HOUR (SCFH) OR 1.38 LA, WHERE LA IS THE ALLOWABLE CONTAINMENT LEAK RATE. THIS LEAK RATE INCLUDED AN ILRT PENALTY EQUAL TO 192 SCFH AND A TYPE B AND C LOCAL LEAK RATE TEST (LLRT) ADJUSTMENT OF 143 SCFH. THE PENALTY WAS INCURRED DUE TO LEAKAGE FROM THE DRYWELL VENT RADIATION MONITOR WHEN THE GASEOUS DETECTOR BREACHED ITS SHIELD CHAMBER AT A CONTAINMENT PRESSURE OF 51.6 PSIG DURING PRESSURIZATION FOR THE ILRT. THE 143 SCFH ADJUSTMENT, ACCOUNTING FOR PENETRATION REPAIRS MADE PRIOR TO CONDUCT OF THE ILRT, WAS PRINCIPALLY DUE TO TWO LEAKING CONTAINMENT ISOLATION VALVE PENETRATIONS FOR THE REACTOR WATER CLEANUP (RWCU) SYSTEM AND THE REACTOR FEEDWATER (RF) SYSTEM. THE CAUSE OF THE DRYWELL VENT RADIATION MONITOR FAILURE IS ATTRIBUTED TO DESIGN. CONSIDERABLE PERIODIC MAINTENANCE AND TESTING, INVOLVING REMOVAL OF THE DETECTORS FROM THEIR SHIELD CHAMBERS, HAS BEEN PERFORMED OVER THE YEARS THAT THE MONITOR HAS BEEN IN SERVICE. THE NUMEROUS DISASSEMBLY AND REASSEMBLY EVOLUTIONS RESULTED IN THE SHIELD CHAMBER THREADED CONNECTIONS BECOMING DEGRADED, SUCH THAT WHEN THE UNIT WAS PRESSURIZED, THE THREADED ENGAGEMENT OF THE DETECTOR RETAINING BOLTS IN THE SHIELD CHAMBER FAILED.

[83] COOPER DOCKET 50-298 LER 91-022
ACTUATION OF THE PRIMARY CONTAINMENT GROUP 3 ISOLATION FUNCTION WHEN PLACING A PARTIALLY FILLED REACTOR WATER CLEANUP FILTER DEMINERALIZER IN SERVICE.
EVENT DATE: 122191 REPORT DATE: 011792 NSSS: GE TYPE: BWR

(NSIC 223793) ON DECEMBER 21, 1991, AT 6:52 P.M., UPON OPENING THE INLET VALVE TO THE 'B' REACTOR WATER CLEANUP (RWCU) FILTER DEMINERALIZER, A PARTIAL GROUP 3 ISOLATION WAS RECEIVED, CAUSING ONE OF THE TWO INLET ISOLATION VALVES, RWCU-MOV-M018 TO CLOSE. THE GROUP 3 ISOLATION OCCURRED DUE TO ACTUATION OF DIFFERENTIAL PRESSURE FLOW SWITCH RWCU-DPIS-170B. THE SETPOINT WAS REACHED AS A RESULT OF AN APPARENT FLOW SURGE DUE TO THE FILTER DEMINERALIZER NOT BEING COMPLETELY FULL OF WATER. AT THE TIME, THE PLANT WAS OPERATING AT APPROXIMATELY 55 PERCENT POWER (450 MWE). POWER WAS BEING INCREASED WITH CONTROL RODS AND REACTOR RECIRCULATION FLOW DURING THE RETURN TO FULL POWER OPERATION FOLLOWING THE 1991 REFUELING OUTAGE. AN INVESTIGATION REVEALED THAT THE NORMALLY OPEN WASTE SAMPLE PUMP DISCHARGE VALVE TO THE CONDENSATE SUPPLY (CM) SYSTEM,

CM-AOV-643AV, WAS CLOSED, PREVENTING PROPER FILLING OF THE FILTER DEMINERALIZER. IT IS POSTULATED THAT DURING OTHER PROCESSING ACTIVITIES DURING THE 00-0800 SHIFT ON DECEMBER 21, THE CONTROL SWITCH FOR CM-AOV-643AV WAS MIS-POSITIONED TO CLOSE. THE FILTER DEMINERALIZER WAS FILLED AND VENTED. THE GROUP 3 ISOLATION WAS RESET, AND AT 7:01 P.M., THE 'B' FILTER DEMINERALIZER WAS PLACED IN SERVICE. IN ORDER TO PRECLUDE A SIMILAR ERROR IN THE FUTURE, A SWITCH GUARD HAS BEEN INSTALLED AROUND THE CONTROL SWITCH.

[84] COOPER DOCKET 50-298 LER 92-001
GROUP 3 HIGH FLOW ISOLATION WHILE SHIFTING REACTOR WATER CLEANUP PUMPS AS A RESULT OF AN INSUFFICIENTLY THROTTLED VALVE.
EVENT DATE: 011192 REPORT DATE: 021092 NSSS: GE TYPE: BWR

(NSIC 224034) ON JANUARY 11, 1992, WITH THE PLANT IN NORMAL POWER OPERATION, THE B REACTOR WATER CLEAN UP (RWCU) PUMP WAS STARTED TO TRANSFER OPERATION FROM THE A TO THE B RWCU PUMP. SHORTLY AFTER STARTING THE PUMP, THE LICENSED OPERATOR OBSERVED THE RWCU HIGH FLOW ALARM, AND THEN THE CLOSURE OF THE RWCU PRIMARY CONTAINMENT ISOLATION VALVES AND ACTUATION OF THE PRIMARY CONTAINMENT GROUP 3 INDICATION LIGHTS. THE RWCU FILTER/DEMINERALIZERS HAD BEEN PREVIOUSLY REMOVED FROM SERVICE, AND THE BYPASS VALVE, RWCU-MOV-M074, WAS THROTTLED TO MAINTAIN NORMAL PUMP DISCHARGE PRESSURE. THE ISOLATION RESULTED FROM INSUFFICIENT THROTTLING OF THE RWCU-MOV-M074 VALVE. THIS ALLOWED THE COMBINED FLOW FROM BOTH PUMPS TO EXCEED THE SETPOINT OF THE HIGH FLOW ISOLATION SWITCH. WITH THE FILTER/DEMINERALIZERS ISOLATED, SYSTEM FLOW INDICATION IS NOT AVAILABLE IN THE CONTROL ROOM. THE SYSTEM OPERATING PROCEDURE WILL BE REVISED TO DIRECT THE USE OF LOCAL FLOW INDICATIONS WHEN SHIFTING PUMPS WITH THE FILTER/DEMINERALIZERS ISOLATED.

[85] CRYSTAL RIVER 3 DOCKET 50-302 LER 91-012
PROCEDURE DEFICIENCY LEADS TO INADVERTENT EMERGENCY DIESEL GENERATOR ACTUATION DURING ENGINEERED SAFEGUARDS TESTING.
EVENT DATE: 111491 REPORT DATE: 121691 NSSS: BW TYPE: PWR

(NSIC 223673) ON NOVEMBER 14, 1991, AT 2349 HOURS, CRYSTAL RIVER UNIT 3 WAS IN MODE 5 (COLD SHUTDOWN). DURING THE PERFORMANCE OF ENGINEERED SAFEGUARDS (ES) SURVEILLANCE TESTING, THE "A" EMERGENCY DIESEL GENERATOR (EDG) WAS STARTED WHEN AN ES SIGNAL WAS INADVERTENTLY APPLIED TO THE DIESEL START CIRCUITRY. THE START SIGNAL WAS INITIATED WHEN THE DIESEL START CIRCUITRY WAS PLACED BACK INTO AUTOMATIC AFTER SURVEILLANCE TESTING WHILE THE ES "A" HIGH PRESSURE INJECTION AUTO TEST GROUP 2 SWITCH WAS STILL IN THE "TEST" POSITION. THE SURVEILLANCE PROCEDURE (SP) IMPLIED THAT THIS TEST SWITCH WOULD SPRING RETURN OUT OF THE TEST POSITION. BY DESIGN, THE SWITCH DOES NOT HAVE A RETURN SPRING. THE SP WILL BE REVISED TO CORRECT THE DEFICIENCIES DISCUSSED IN THIS REPORT. ALL EQUIPMENT FUNCTIONED AS DESIGNED. THIS EVENT IS BEING REPORTED UNDER 10CFR50.73(A)(2)(IV).

[86] CRYSTAL RIVER 3 DOCKET 50-302 LER 91-013
TEST SWITCH INSTALLATION ERROR CAUSES ENTRY INTO TECHNICAL SPECIFICATION 3.0.3.
EVENT DATE: 111991 REPORT DATE: 122091 NSSS: BW TYPE: PWR

(NSIC 223636) ON NOVEMBER 19, 1991 CRYSTAL RIVER UNIT 3 (CR-3) WAS IN MODE 3 PERFORMING START-UP SURVEILLANCE TESTING. THE STEAM DRIVEN EMERGENCY FEEDWATER PUMP (SDEFP) WAS CONSERVATIVELY DECLARED INOPERABLE (INOP) SINCE A SURVEILLANCE TEST HAD NOT BEEN PERFORMED WITHIN 24 HOURS OF ENTERING MODE 3. WHEN A DIFFICULTY WITH A SYSTEM TEST ASSOCIATED WITH THE MOTOR DRIVEN EMERGENCY FEEDWATER PUMP (MDEFP) REQUIRED ITS CONTROL SWITCH TO BE PLACED IN PULL-TO-LOCK, PRECLUDING AN AUTOMATIC START, IT TOO WAS DECLARED INOP. TECHNICAL SPECIFICATION (TS) 3.0.3 WAS ENTERED DUE TO BOTH EMERGENCY FEEDWATER PUMPS BEING DECLARED INOP. THIS CONDITION EXISTED FOR 24 MINUTES. THE PLANT EXITED TS 3.0.3 AS CONDITIONS ALLOWED THE MDFWP CONTROL SWITCH TO BE RETURNED TO NORMAL. ALSO, CR-3 DID NOT COMPLY WITH TS 3.0.5 WHEN THE EMERGENCY DIESEL GENERATOR WAS LATER DECLARED INOP. THE PROBLEM WITH THE TEST WAS DETERMINED TO BE AN INCORRECTLY INSTALLED TEST SWITCH DUE TO A DRAWING ERROR. THE NEED TO REVIEW SIMILAR DRAWINGS IS BEING EVALUATED. THE TEST WAS REPERFORMED ACCEPTABLY. THE SDEFP SURVEILLANCE TEST, WHICH WAS LATER PERFORMED

SUCCESSFULLY, WAS NOT TESTED WITHIN THE TIME INTERVAL DUE TO THE PLANT NOT ESTABLISHING CONDITIONS FOR PUMP OPERATION. FAILURE TO COMPLY WITH TS 3.0.5 WAS CAUSED BY A MISINTERPRETATION. THIS EVENT HAS BEEN DISCUSSED WITH SHIFT SUPERVISORS TO PROVIDE CLARIFICATION OF TS 3.0.5 REQUIREMENTS.

[87] CRYSTAL RIVER 3 DOCKET 50-302 LER 91-014
GASKET FAILURES CAUSE FEEDWATER BOOSTER PUMP AND MAIN FEEDWATER PUMP TRIP.
EVENT DATE: 112591 REPORT DATE: 122691 NSSS: BW TYPE: PWR

(NSIC 223709) ON THE EVENING OF 11/25/91, CRYSTAL RIVER UNIT 3 WAS OPERATING IN MODE 1, POWER OPERATION, AT APPROXIMATELY 20 PERCENT POWER. PLANT STARTUP WAS IN PROGRESS. THE OUTPUT BREAKERS WERE CLOSED AT 1716. JUST AFTER BREAKER CLOSURE OPERATORS NOTICED THE DEAERATOR LEVEL WAS HIGH AND INCREASING. EFFORTS TO RESTORE PROPER DEAREATOR LEVEL WERE UNSUCCESSFUL. AT 1720 THE 'B' MAIN FEEDWATER BOOSTER PUMP TRIPPED, CAUSING THE MAIN FEEDWATER PUMPS TO TRIP. THIS CAUSED AN ANTICIPATORY REACTOR TRIP AND INITIATION OF EMERGENCY FEEDWATER. THE FEEDWATER BOOSTER PUMP TRIP WAS CAUSED BY A FALSE LOW LEVEL SIGNAL FROM THE DEAREATOR LEVEL SWITCH. THE FALSE SIGNAL RESULTED FROM AGE RELATED FAILURE OF TWO GASKETS ON TWO OTHER LEVEL SWITCHES SHARING COMMON SENSING LINES. TECHNICIANS REPLACED THE GASKETS WITH A DIFFERENT TYPE GASKET. TECHNICIANS INVESTIGATED THE RESPONSE OF A DEAREATOR DUMP VALVE WHICH HAD NOT OPERATED PROPERLY AS PART OF THE LEVEL CONTROL SCHEME. INSTRUMENTS WERE INSTALLED TO MONITOR OPERATION OF CONDENSATE PUMPS AND ASSOCIATED CONTROLS DURING SUBSEQUENT STARTUP TO ASSURE THEY CONTROLLED PROPERLY. GUIDANCE AND TRAINING IS BEING PROVIDED TO THE OPERATIONS CREW ON STARTUP EVOLUTIONS TO ASSURE CONSISTENCY.

[88] CRYSTAL RIVER 3 DOCKET 50-302 LER 91-015
FAILURE TO COMPLETE REQUIRED TECHNICAL SPECIFICATION SURVEILLANCE PRIOR TO MODE CHANGE DUE TO PERSONNEL ERROR.
EVENT DATE: 112591 REPORT DATE: 122691 NSSS: BW TYPE: PWR

(NSIC 223710) ON NOVEMBER 24, 1991, CRYSTAL RIVER UNIT 3 (CR-3) BEGAN MODE ASCENSION FROM MODE 3 TO MODE 1. THE SHIFT SUPERVISOR BELIEVED THE MONTHLY REACTOR PROTECTION SYSTEM SURVEILLANCES FOR THE ANTICIPATORY REACTOR TRIPS ON LOSS OF MAIN FEEDWATER OR MAIN TURBINE WERE UP TO DATE AND MODE ASCENSION WAS ALLOWED. HOWEVER, THESE SURVEILLANCES HAD NOT BEEN PERFORMED ON THE REQUIRED SCHEDULE. THUS, AT 1700, THE PLANT ENTERED MODE 1 IN VIOLATION OF TECHNICAL SPECIFICATION REQUIREMENTS 4.3.1.1.1 AND 4.0.4. THIS CONDITION WAS DISCOVERED DURING REVIEW OF THE SURVEILLANCE PROCEDURE ON NOVEMBER 26, 1991. THIS EVENT WAS CAUSED BY PERSONNEL ERROR. THE RESPONSIBLE SHOP SUPERVISOR INCORRECTLY THOUGHT THE SURVEILLANCE WAS CURRENT AND SO INFORMED THE SHIFT SUPERVISOR. FLORIDA POWER IS CONTINUING TO INVESTIGATE CORRECTIVE ACTIONS.

[89] CRYSTAL RIVER 3 DOCKET 50-302 LER 91-016
LOSS OF INTEGRATED CONTROL SYSTEM POWER LEADS TO EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM ACTUATION.
EVENT DATE: 112591 REPORT DATE: 122691 NSSS: BW TYPE: PWR

(NSIC 223711) ON NOVEMBER 25, 1991 AT 1108, CRYSTAL RIVER UNIT 3 (CR-3) EXPERIENCED A LOSS OF INTEGRATED CONTROL SYSTEM (ICS) POWER. DURING THE EVENT, RECOVERY, EMERGENCY FEEDWATER ACTUATED DUE TO LOW STEAM GENERATOR LEVELS. THE TRANSIENT ALSO CAUSED POWER TO INCREASE FROM 13 PERCENT TO 19 PERCENT RATED THERMAL POWER (RTP) WHICH RESULTED IN ENTERING THE REGION OF APPLICABILITY OF TECHNICAL SPECIFICATION 3.2.4, WHILE THE LIMITING CONDITION FOR OPERATION (LCO) WAS NOT MET. THE EMERGENCY FEEDWATER ACTUATION IS BEING REPORTED PER 10CFR50.73(A)(2)(IV). ENTERING THE APPLICABILITY OF TECHNICAL SPECIFICATION 3.2.4 WHILE THE LCO IS NOT MET IS A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS AND IS BEING REPORTED PER 10CFR50.73(A)(2)(I)(B). THE CORRECTIVE ACTION FOR THIS EVENT WAS TO REGAIN CONTROL OF THE FEEDWATER SYSTEM, SECURE THE EMERGENCY FEEDWATER PUMPS, RESET EMERGENCY FEEDWATER INITIATION AND CONTROL, AND RETURN POWER TO LESS THAN 15 PERCENT RTP. ALL OF THESE ITEMS WERE DONE IMMEDIATELY AFTER THE EVENT BEGAN. ALL SYSTEMS ACTED AS EXPECTED. THERE ARE NO FURTHER ACTIONS PLANNED.

[90] CRYSTAL RIVER 3 DOCKET 50-302 LER 91-017
REACTOR TRIP CAUSED BY FEEDWATER REDUCTION DUE TO NUCLEAR POWER INSTRUMENTATION CHANNEL BEING SELECTED FOR CONTROL WHICH CONTAINED A FAILED DETECTOR.
EVENT DATE: 120291 REPORT DATE: 010692 NSSS: BW TYPE: PWR

(NSIC 223712) ON DECEMBER 2, 1991 AT 2000, CRYSTAL RIVER UNIT 3 (CR-3) WAS COMMENCING A SHUTDOWN FROM 100 PERCENT RATED THERMAL POWER (RTP) TO ALLOW TROUBLE SHOOTING POWER RANGE NUCLEAR INSTRUMENTATION (NI) CHANNEL NI-8, WHICH HAD PREVIOUSLY GONE TO ZERO INDICATED POWER AND TRIPPED REACTOR PROTECTION SYSTEM (RPS) CHANNEL "D". AS POWER WAS REDUCED, A PRE-EXISTING QUADRANT POWER TILT (QPT) BEGAN TO INCREASE. AT 2315 QPT EXCEEDED STEADY STATE LIMIT OF TECHNICAL SPECIFICATIONS AND REDUCTION OF THE NUCLEAR OVERPOWER TRIP SETPOINT WAS REQUIRED. AT 0200 ON DECEMBER 3, POWER WAS STABILIZED AT 50 PERCENT RTP TO ALLOW ADJUSTMENT OF THE NUCLEAR OVERPOWER TRIP SETPOINT. TECHNICIANS PLACED CHANNEL "A" OF RPS IN BYPASS AND BEGAN REDUCING TO OVERPOWER TRIP SETPOINT. DURING ADJUSTMENT OF THE OVERPOWER TRIP SETPOINT, AN ERRONEOUS REACTOR POWER SIGNAL WAS SENT TO THE INTEGRATED CONTROL SYSTEM (ICS), DUE TO NI-8 INDICATING ZERO, RESULTING IN FEEDWATER FLOW DECREASE TO THE STEAM GENERATORS AND CONTROL ROD WITHDRAWAL. THE REACTOR TRIPPED ON HIGH PRESSURE IN APPROXIMATELY 30 SECONDS. SEVERAL FACTORS CONTRIBUTED TO THIS EVENT INCLUDING INCORRECT OPERATOR TRAINING ON THE NI POWER AVERAGER/AUCTIONEER CIRCUIT, AND LACK OF CAUTIONS IN THE PROCEDURE REGARDING SETPOINT ADJUSTMENT WITH A FAILED NI. PLANT PROCEDURES ARE BEING REVISED TO PROVIDE CAUTIONS.

[91] CRYSTAL RIVER 3 DOCKET 50-302 LER 91-018
REDUCTION IN REACTOR COOLANT SYSTEM PRESSURE DUE TO FAILURE OF PRESSURIZER SPRAY VALVE AND ASSOCIATED POSITION INDICATION RESULTS IN ACTUATION OF REACTOR PROTECTION SYSTEM AND ENGINEERED SAFEGUARDS.
EVENT DATE: 120891 REPORT DATE: 010792 NSSS: BW TYPE: PWR
VENDOR: LIMITORQUE CORP.
WALWORTH COMPANY

(NSIC 223713) ON DECEMBER 8, 1991, CRYSTAL RIVER UNIT 3 WAS BEING RETURNED TO POWER OPERATION. AS REACTOR POWER WAS BEING INCREASED FROM 11% RATED THERMAL POWER (RTP) TO 15% RTP, REACTOR COOLANT SYSTEM (RCS) PRESSURE INCREASED TO THE OPEN SETPOINT FOR THE PRESSURIZER SPRAY VALVE, RCV-14. RCV-14 OPENED; HOWEVER, THE "CLOSED" INDICATING LAMP DID NOT EXTINGUISH. ON DECREASING RCS PRESSURE, RCV-14 DID NOT CLOSE, RESULTING IN A CONTINUED SLOW DECREASE IN RCS PRESSURE. PRIOR TO RCS PRESSURE REACHING THE ENGINEERED SAFEGUARDS (ES) ACTUATION SETPOINT, AN OPERATOR INAPPROPRIATELY BYPASSED ES. SHIFT SUPERVISION DIRECTED ES OUT OF BYPASS AND ES ACTUATION WAS INITIATED. AFTER ES WAS RESET, A PLAN WAS IMPLEMENTED WHICH BYPASSED ES AND USED HIGH PRESSURE INJECTION TO RAISE RCS PRESSURE. RCV-14 WAS MANUALLY ISOLATED, TERMINATING THE EVENT. THE PLANT WAS PLACED IN MODE 5 (COLD SHUTDOWN) AND RCV-14 WAS REPAIRED AND TESTED SATISFACTORILY. PLANT MAINTENANCE PROCEDURES ARE BEING REVISED TO PRECLUDE RECURRENCE OF THIS TYPE OF MOTOR OPERATED VALVE FAILURE. ADMINISTRATIVE GUIDANCE HAS BEEN DEVELOPED ON THE BYPASSING OF ES ACTUATION SIGNALS.

[92] DAVIS-BESSE 1 DOCKET 50-346 LER 91-007
SHUTDOWN REQUIRED BY TECHNICAL SPECIFICATIONS DUE TO EMERGENCY DIESEL GENERATOR PROBLEMS.
EVENT DATE: 120691 REPORT DATE: 010692 NSSS: BW TYPE: PWR
VENDOR: SYNCHRO START PRODUCTS

(NSIC 223734) ON DECEMBER 3, 1991, AT 0500 HOURS, EMERGENCY DIESEL GENERATOR (EDG) 1-2 WAS TAKEN OUT OF SERVICE FOR ROUTINE MAINTENANCE. POST MAINTENANCE TESTING WAS UNSUCCESSFUL. THE SYMPTOMS OF THE FAILURE INDICATED A DEFECTIVE SPEED SWITCH. THE 72 HOUR ACTION STATEMENT OF TS 3.8.1.1(A) EXPIRED ON DECEMBER 6, 1991, AT 0500 HOURS AND A PLANT SHUTDOWN WAS COMMENCED. HOT STANDBY (MODE 3) WAS REACHED AT 1054 HOURS ON DECEMBER 6, 1991. TOLEDO EDISON'S REQUEST FOR A TEMPORARY WAIVER OF COMPLIANCE TO ALLOW THE PLANT TO REMAIN IN HOT STANDBY FOR UP TO SEVEN DAYS WHILE PERFORMING TROUBLESHOOTING WAS GRANTED ON DECEMBER 6, 1991. TROUBLESHOOTING IDENTIFIED A DEFECTIVE SPEED SWITCH. UPON REPLACEMENT OF THE SPEED SWITCH, EDG 1-2 WAS SUCCESSFULLY TESTED AND RESTORED TO OPERABLE STATUS AT

0205 HOURS ON DECEMBER 11, 1991. THE PLANT COMMENCED STARTUP AND MODE 1 WAS ACHIEVED AT 1117 HOURS ON DECEMBER 11, 1991. THE INITIATION OF THE PLANT SHUTDOWN REQUIRED BY TS WAS REPORTED TO THE NRC VIA EMERGENCY NOTIFICATION SYSTEM (ENS) AT 0834 HOURS ON DECEMBER 6, 1991 UNDER 10 CFR 50.72 (B)(1)(I)(A). THE COMPLETION OF A PLANT SHUTDOWN REQUIRED BY TS IS BEING REPORTED AS AN LER UNDER 10 CFR 50.73(A)(2)(I)(A).

[93] DAVIS-BESSE 1 DOCKET 50-346 LER 91-009
SEAT TEST NOT PERFORMED ON EMERGENCY AIR LOCK.
EVENT DATE: 120891 REPORT DATE: 011592 NSSS: BW TYPE: PWR

(NSIC 223776) ON 12/5/91, WITH THE PLANT IN MODE 1, THE CONTAINMENT EMERGENCY AIR LOCK (CEAL) INNER DOOR WAS CYCLED BY RADIOLOGICAL CONTROL (RC) PERSONNEL DURING THE PERFORMANCE OF DB-HP-01101, CONTAINMENT ENTRY, TO VERIFY THAT THE STRONG BACKS WERE REMOVED FROM THE CEAL INNER DOOR. ON 12/10/91, AT 1130, WITH THE PLANT IN MODE 3, THE SHIFT SUPERVISOR DISCOVERED THE TIME LIMIT FOR PERFORMANCE OF TECHNICAL SPECIFICATION (TS) SURVEILLANCE REQUIREMENT 4.6.1.3.A HAD EXPIRED AND DECLARED THE CEAL INOPERABLE. DUE TO A MISCOMMUNICATION AND CONFLICTING INFORMATION REGARDING CEAL ALARM FUNCTION, THE CEAL OUTER DOOR WAS TESTED AND THE CEAL WAS DECLARED OPERABLE AT 2358, ON 12/10/91. PLANT STARTUP COMMENCED AND PROCEEDED TO MODE 1. SUBSEQUENT TO MODE 1 ENTRY, THE SHIFT SUPERVISOR WAS INFORMED THAT THE CEAL INNER DOOR HAD BEEN CYCLED ON 12/5/91. THE CEAL WAS AGAIN DECLARED INOPERABLE. BOTH THE INNER AND OUTER CEAL DOORS WERE SATISFACTORILY TESTED AND THE CEAL WAS DECLARED OPERABLE AT 1604, ON 12/11/91. THIS EVENT WAS CAUSED BY A DEFICIENCY IN DB-HP-01101 AND A MISCOMMUNICATION BETWEEN RC AND OPERATIONS PERSONNEL. THE STRONG BACKS ARE CURRENTLY BEING STORED OUTSIDE THE CEAL OUTER DOOR IN THE CEAL BLOCK HOUSE. DB-HP-01101 HAS BEEN CHANGED TO REFLECT THE NEW STORAGE LOCATION AND TO REQUIRE NOTIFICATION OF THE SHIFT SUPERVISOR IN THE EVENT EITHER CEAL DOOR IS OPENED.

[94] DAVIS-BESSE 1 DOCKET 50-346 LER 91-008
REACTOR TRIP DUE TO BLOWN FUSE DURING MAINTENANCE ON NON-ESSENTIAL 4160V AC BUS D2.
EVENT DATE: 121091 REPORT DATE: 010992 NSSS: BW TYPE: PWR

(NSIC 223775) ON DECEMBER 10, 1991 AT 0906, WITH THE PLANT IN MODE 3, WORK WAS BEING PERFORMED ON NON-ESSENTIAL 4160V AC BUS D2 WHICH RESULTED IN A BLOWN FUSE. THIS CAUSED THE REVERSE PHASE SEQUENCE UNDERVOLTAGE RELAY FOR BUS D2 (RELAY 47/D2) TO ACTUATE. ALTHOUGH NO ACTUAL UNDERVOLTAGE CONDITION EXISTED, ACTUATION OF RELAY 47/D2 CAUSED SELECT BUS D2 LOAD BREAKERS TO TRIP INCLUDING THE MOTOR DRIVEN FEEDWATER PUMP (MDFP) FEEDER BREAKER. THE LOSS OF THE MDFP RESULTED IN A LOSS OF FEEDWATER TO THE ONCE THROUGH STEAM GENERATORS (OTSGS), AND THE SUBSEQUENT ACTUATION OF THE STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM (SFRCS). THE SFRCS INITIATED AUXILIARY FEEDWATER, ISOLATED MAIN FEEDWATER, AND ISOLATED MAIN STEAM, AS DESIGNED. IN ADDITION, THE SFRCS ACTUATION INITIATED THE ANTICIPATORY REACTOR TRIP SYSTEM (ARTS) WHICH TRIPPED OPEN THE CONTROL ROD DRIVE (CRD) TRIP BREAKERS. THE GROUP ONE RODS, WHICH WERE 100% WITHDRAWN, PROPERLY INSERTED INTO THE REACTOR CORE. AUTOMATIC PLANT RESPONSE WAS SATISFACTORY AND OPERATOR ACTIONS WERE APPROPRIATE. THE BLOWN FUSE WAS REPLACED AND FEEDWATER WAS RESTORED BY RESTARTING THE MDFP APPROXIMATELY 15 MINUTES AFTER IT TRIPPED. A REVIEW TEAM HAS BEEN ASSEMBLED TO DETERMINE WHAT ADDITIONAL PRECAUTIONS CAN BE TAKEN TO PREVENT SIMILAR EVENTS. THE TEAM IS EXPECTED TO MAKE ITS RECOMMENDATIONS BY MARCH 4, 1992.

[95] DIABLO CANYON 1 DOCKET 50-275 LER 83-039
CONTROL ROOM VENTILATION SYSTEM OUTSIDE DESIGN BASIS.
EVENT DATE: 111583 REPORT DATE: 122391 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: DIABLO CANYON 2 (PWR)

(NSIC 223706) ON NOVEMBER 21, 1991, WITH UNITS 1 AND 2 IN MODE 1 AT 100 PERCENT POWER, DETERMINED THAT A POSTULATED SINGLE ACTIVE FAILURE OF ONE OF THE REDUNDANT BOOSTER FANS OR ITS DAMPER IN THE CONTROL ROOM VENTILATION SYSTEM (CRVS) WOULD CAUSE THE CRVS TO BE OUTSIDE ITS DESIGN BASIS. A 1-HOUR, NON-EMERGENCY REPORT WAS

MADE FOR UNITS 1 AND 2 IN ACCORDANCE WITH 10 CFR 50.72(B)(1)(II)(B) ON NOVEMBER 21, AT 1335 PST. AS A RESULT OF INVESTIGATIONS STEMMING FROM PG&E'S CONFIGURATION MANAGEMENT PROGRAM PG&E DETERMINED THAT A POSTULATED SINGLE ACTIVE FAILURE OF ONE OF THE REDUNDANT BOOSTER FANS OR BOOSTER FAN DAMPERS IN THE CRVS COULD POTENTIALLY CAUSE THE CRVS TO BE OUTSIDE ITS DESIGN BASIS. THIS CONCLUSION RESULTED FROM THE DETERMINATION THAT THERE WAS NEITHER AN ALARM TO NOTIFY CONTROL ROOM OPERATORS OF THE FAILURE NOR AN AUTOMATIC SWITCHOVER TO THE UNAFFECTED REDUNDANT CRVS TRAIN. THE CRVS IN PART IS DESIGNED TO LIMIT RADIATION EXPOSURE TO PERSONNEL OCCUPYING THE CONTROL ROOM CONSISTENT WITH THE REQUIREMENTS OF GENERAL DESIGN CRITERION 19 OF APPENDIX A TO 10 CFR 50. WITH THE ABOVE CONDITION, THE POTENTIAL EXISTED FOR AN UNDETECTED FAILURE OF A BOOSTER FAN OR DAMPER DURING THE PRESSURIZATION MODE (MODE 4 FOR THE CRVS), RESULTING IN INFILTRATION UNFILTERED AIRBORNE RADIOACTIVITY INTO THE CONTROL ROOM.

[96] DIABLO CANYON 1 DOCKET 50-275 LER 91-017
MISSED SURVEILLANCES OF ROD POSITION INDICATIONS BECAUSE OF AN INADEQUATE PLANT PROCESS COMPUTER PROGRAM.
EVENT DATE: 111491 REPORT DATE: 121691 NSSS: WE TYPE: PWR

(NSIC 223765) FROM APPROXIMATELY 1615 PST ON NOVEMBER 14, 1991, TO 2305 PST ON NOVEMBER 16, 1991, CONDITIONAL SURVEILLANCES OF THE CONTROL ROD POSITIONS REQUIRED BY TECHNICAL SPECIFICATION (TS) 4.1.3.1.1 AND 4.1.3.2, INCLUDING THE ALLOWED EXTENSION OF TS 4.0.2, WERE MISSED. CONTROL ROOM OPERATORS WERE UNAWARE THAT THE ROD POSITION DEVIATION MONITOR (RPDM) OF THE PLANT PROCESS COMPUTER (PPC) WAS INOPERABLE. ON NOVEMBER 17, 1991, LICENSED OPERATORS COMPLETED SURVEILLANCE TEST PROCEDURE (STP) I-42, "ROD POSITION DEVIATION MONITOR FUNCTIONAL TEST," AND DISCOVERED THAT THE RPDM FAILED TO INITIATE AN ALARM ON THE MAIN ANNUNCIATOR PANEL, THE PPC TYPEWRITER, AND THE PPC ALARM VIDEO SCREEN AT THE APPROPRIATE PROCEDURAL STEP. OPERATIONS DECLARED THE RPD INOPERABLE AND INITIATED FOUR-HOUR CONDITIONAL SURVEILLANCES IN ACCORDANCE WITH TS 4.1.3.1.1 AND 4.1.3.2. THE FOUR-HOUR SURVEILLANCES WERE MISSED FROM APPROXIMATELY 1615 PST ON NOVEMBER 14, 1991, TO 2305 PST ON NOVEMBER 16, 1991. DURING THAT TIME, ROD POSITIONS WERE CHECKED EVERY 12 HOURS AS REQUIRED BY TS WHEN THE RPDM IS OPERABLE. THE IMMEDIATE CAUSE OF THIS EVENT WAS THAT AFTER THE PPC WAS SHUT DOWN AND RESTARTED ON NOVEMBER 14, THE RPDM LOGIC INHIBITED THE ALARM FUNCTION BECAUSE THE PPC INCORRECTLY STATUTED THE REACTOR TRIP BREAKERS AS OPEN.

[97] DIABLO CANYON 1 DOCKET 50-275 LER 91-018
COMPONENT COOLING WATER SYSTEM OUTSIDE DESIGN BASIS.
EVENT DATE: 121991 REPORT DATE: 011792 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: DIABLO CANYON 2 (PWR)

(NSIC 223830) ON DECEMBER 19, 1991, WITH UNITS 1 AND 2 IN MODE 1 AT 100 PERCENT POWER, PG&E DETERMINED THAT UNDER CERTAIN CONDITIONS AND EQUIPMENT CONFIGURATIONS, THE CCW WATER TEMPERATURE DESIGN BASIS LIMITS MAY BE EXCEEDED. A ONE-HOUR, NON-EMERGENCY REPORT WAS MADE FOR UNITS 1 AND 2 IN ACCORDANCE WITH 10 CFR 50.72(B)(1)(II)(B) ON DECEMBER 19, 1991, AT 1021 PST. DURING AN INVESTIGATION TO DETERMINE THE VIABILITY OF A POTENTIAL CHANGE TO THE PLANT TECHNICAL SPECIFICATIONS (TS), THE IMPACT ON THE CCW SYSTEM TEMPERATURE UNDER VARIOUS CONDITIONS WAS EVALUATED. EARLY IN THE INVESTIGATION, PG&E JUDGED THAT OPERATION WITH TS LIMITS ASSURED THAT CCW SYSTEM TEMPERATURE LIMITS WOULD NOT BE EXCEEDED. HOWEVER, FURTHER REVIEW DETERMINED THAT THE MAGNITUDE OF THE HEAT LOAD FROM THE RESIDUAL HEAT REMOVAL SYSTEM HEAT EXCHANGERS AND THE CONTAINMENT FAN COOLER UNITS TO THE CCW SYSTEM WAS SIGNIFICANTLY GREATER THAN ANTICIPATED AND THAT THE HEAT LOAD DURING THE COLD-LEG RECIRCULATION PHASE MAY EXCEED THE CCW SYSTEM DESIGN BASIS TEMPERATURE LIMITS. IMMEDIATE CORRECTIVE ACTIONS INCLUDED REVISION OF EMERGENCY OPERATING PROCEDURE E-1.3 "TRANSFER TO COLD LEG RECIRCULATION," TO INCLUDE DIRECTIONS TO THE OPERATORS REGARDING EQUIPMENT CONFIGURATIONS TO MITIGATE POTENTIAL CCW TEMPERATURE RISE, AND NOTIFICATION TO OPERATIONS PERSONNEL REGARDING THE PROCEDURE REVISION.

[98] DIABLO CANYON 2 DOCKET 50-323 LER 91-002
POTENTIAL MISSETTING OF MAIN STEAM LINE CODE SAFETY VALVE DUE TO FAILURE OF TEST EQUIPMENT.
EVENT DATE: 082691 REPORT DATE: 122091 NSSS: WE TYPE: PWR

(NSIC 223720) THIS VOLUNTARY LER IS SUBMITTED FOR INFORMATIONAL PURPOSES ONLY AS DESCRIBED IN ITEM 19 OF SUPPLEMENT 1 TO NUREG-1022. ON AUGUST 26, 1991, THE SETPOINTS FOR MAIN STEAM SAFETY VALVES (MSSVS) RV-60 AND ON MAIN STEAM LINE 2-4 WERE TESTED USING TREVITEST EQUIPMENT. MSSV RV-60 WAS TESTED AND REQUIRED A NORMAL ADJUSTMENT TO BRING IT INTO TOLERANCE. SUBSEQUENTLY, MSSV RV-225 WAS TESTED AND REQUIRED AN ABNORMALLY LARGE ADJUSTMENT TO BRING IT INTO TOLERANCE. BECAUSE THIS LARGE ADJUSTMENT WAS NEEDED, A PROMPT POST-CALIBRATION VERIFICATION WAS PERFORMED ON THE TREVITEST EQUIPMENT. IT WAS DETERMINED THAT THE TREVITEST EQUIPMENT HAD FAILED DURING THE ADJUSTMENT OF RV-225. RV-225 WAS THEN DECLARED INOPERABLE. RV-60 WAS NOT DECLARED INOPERABLE, SINCE THE TREVITEST FAILURE WAS JUDGED TO HAVE OCCURRED SUBSEQUENT TO ADJUSTING RV-60 (BECAUSE ONLY NORMAL, MINOR ADJUSTMENTS HAD BEEN MADE TO RV-60). ON AUGUST 27, 1991, RV-60 WAS RECHECKED AND LIFTED AT 1105 PSI (4 PSI ABOVE ITS ACCEPTANCE CRITERIA OF 1090 PSI +/-1%). THE TREVITEST EQUIPMENT FAILURE WAS DUE TO A DEFECTIVE CELL IN THE BATTERY OF THE TREVITEST TEST RECORDER. IT COULD NOT BE DETERMINED WHETHER RV-60 LIFTING OUT OF TOLERANCE ON AUGUST 27, 1991 WAS CAUSED BY ADJUSTING THE SETPOINT HIGH DUE TO FAIL THE TREVITEST EQUIPMENT, OR BY NORMAL SETPOINT DRIFT.

[99] DRESDEN 2 DOCKET 50-237 LER 87-012 REV 01
UPDATE ON HIGH PRESSURE COOLANT INJECTION SYSTEM TURBINE TRIPS DUE TO HYDRAULIC CONTROL SYSTEM PROBLEMS.
EVENT DATE: 042287 REPORT DATE: 052187 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)
VENDOR: SQUARE D COMPANY

(NSIC 223951) ON 4/22/87, WHILE CONDUCTING DRESDEN UNIT 2 STARTUP OPERATIONS AT 1% RATED THERMAL POWER, THE REACTOR OPERATOR OBSERVED THAT THE HIGH PRESSURE COOLANT INJECTION (HPCI) TURBINE RESET LIGHT WAS NOT LIT. A UNIT SHUTDOWN TO HOT STANDBY HAS BEEN INITIATED DUE TO A PROBLEM WITH THE MAIN TURBINE SEAL STEAM SYSTEM. INVESTIGATION FOUND THE ROOT CAUSE OF THE TURBINE TRIP RESET INDICATION TO BE A LOOSE HYDRAULIC CONTROL SYSTEM PRESSURE SWITCH CONTACTOR ARM. SUBSEQUENTLY, WHILE PERFORMING HPCI TESTING FOLLOWING REPAIRS, THE HPCI TURBINE WAS OBSERVED TO TRIP WHILE BEING BROUGHT UP TO SPEED. THE TURBINE TRIP, WHICH OCCURRED WHILE THE HPCI EMERGENCY OIL PUMP WAS ON TO SUPPORT TURNING GEAR OPERATION, WAS FOUND TO BE CAUSED BY PREMATURE TRIPPING OF THE HPCI AUXILIARY OIL PUMP. CORRECTIVE ACTIONS INCLUDED INSPECTION, TESTING, AND ADJUSTMENT OF THE HPCI TURBINE HYDRAULIC CONTROL SYSTEM COMPONENTS. SAFETY SIGNIFICANCE WAS MINIMAL BECAUSE THESE PROBLEMS WERE OBSERVED AT LOW REACTOR POWER DURING STARTUP FROM A REFUEL OUTAGE AND REACTOR PRESSURE DID NOT EXCEED THE CAPABILITY OF THE LOW PRESSURE EMERGENCY CORE COOLING SYSTEMS. A RELATED EVENT IS RECORDED BY LER 87-2/050249.

[100] DRESDEN 2 DOCKET 50-237 LER 90-002 REV 01
UPDATE ON REACTOR SCRAM FOLLOWING CONDENSATE/CONDENSATE BOOSTER PUMP FAILURE AND SUBSEQUENT LOSS OF OFFSITE POWER.
EVENT DATE: 011690 REPORT DATE: 021590 NSSS: GE TYPE: BWR
VENDOR: AUTOMATIC VALVE COMPANY
GENERAL ELECTRIC CO.
LIMITORQUE CORP.

(NSIC 223697) AT APPROX. 1724 ON 1/16/90, AN AUTOMATIC UNIT 2 REACTOR SCRAM ON A LOW REACTOR WATER LEVEL (TECH SPEC SETPOINT OF 8 IN. ABOVE INSTRUMENT ZERO) SIGNAL OCCURRED. THE 2D CONDENSATE/CONDENSATE PUMP FAILED DUE TO AN INTERNAL MOTOR FAULT APPROXIMATELY 13 SECONDS PRIOR TO THE SCRAM, PRECIPITATING AUTOMATIC TRIP OF THE RUNNING REACTOR FEED PUMPS ON LOW SUCTION PRESSURE AND THUS RESULTING IN REDUCTION OF REACTOR WATER LEVEL TO THE LOW LEVEL SCRAM SETPOINT. ADDITIONALLY, RESERVE AUXILIARY TRANSFORMER (TR) 22 TRIPPED DURING AUTOMATIC TRANSFER OF HOUSE LOADS, RESULTING IN INTERRUPTION OF NORMAL AC AUXILIARY POWER UNTIL THE EMERGENCY DIESEL GENERATORS AUTOMATICALLY LOADED. COLD SHUTDOWN

CONDITIONS WERE ACHIEVED BY 0210 ON 1/17/90. COMPREHENSIVE ROOT CAUSE INVESTIGATIONS WERE IMMEDIATELY INITIATED FOR ALL COMPONENT PERFORMANCE PROBLEMS, AND CORRECTIVE ACTIONS IMPLEMENTED. THE SAFETY SIGNIFICANCE OF THIS TRANSIENT WAS MITIGATED BY THE FACT THAT REACTOR WATER LEVEL WAS MAINTAINED WELL ABOVE THE AUTOMATIC EMERGENCY CORE COOLING SYSTEM INITIATION SETPOINT AT ALL TIMES, AND MULTIPLE SYSTEMS WERE AVAILABLE FOR REACTOR PRESSURE CONTROL INCLUDING ISOLATION CONDENSER, HIGH PRESSURE COOLANT INJECTION, AND MAIN STEAM RELIEF VALVES. PREVIOUS EVENT INVOLVING A UNIT 3 LOSS OF OFFSITE POWER IS REPORTED BY LER 89-1/050249.

[101] DRESDEN 2 DOCKET 50-237 LER 91-023 REV 01
UPDATE ON 2A RECIRCULATION PUMP DISCHARGE MOTOR-OPERATED VALVE FAILURE TO CLOSE DUE TO TORQUE SWITCH SETTING PROBLEM.
EVENT DATE: 080791 REPORT DATE: 090491 NSSS: GE TYPE: BWR
VENDOR: LIMITORQUE CORP.

(NSIC 223908) ON AUGUST 7, 1991 AT 0215 HOURS, WITH UNIT 2 AT 37% POWER, WHILE ATTEMPTING TO START THE 2A RECIRCULATION PUMP, THE PUMP DISCHARGE MOTOR-OPERATED VALVE (MOV) 2-0202-5A WOULD NOT CLOSE. TROUBLESHOOTING REVEALED THAT THE CLOSING POWER CONTRACTOR FOR THE MOTOR WAS DROPPING OUT SHORTLY AFTER CLOSE SIGNAL INITIATION. THE LOW PRESSURE COOLANT INJECTION (LPCI) SYSTEM WAS DECLARED INOPERABLE BECAUSE THIS MOV IS INCLUDED IN LPCI INITIATION LOGIC. THE ELECTRICAL MAINTENANCE DEPARTMENT ANALYZED STRIP CHART RECORDER MOTOR CURRENT TRACES OF THE MOV AND DETERMINED THAT THE CLOSE TORQUE SWITCH WAS CAUSING THE MOV TO TRIP EARLY. MAINTENANCE PERSONNEL MADE A PRIMARY CONTAINMENT ENTRY TO TEMPORARILY BYPASS THE OPEN-TO-CLOSE TORQUE SWITCH FOR 77%. OF VALVE CLOSING STROKE BY INSTALLING JUMPERS AT THE LIMIT SWITCH ASSEMBLY. ANALYSIS BY THE NUCLEAR ENGINEERING DEPARTMENT (NED) REVEALED AN INCORRECT TORQUE SWITCH SETTING FOR MOV 2-0202-5A. THIS PROBLEM WAS ATTRIBUTED TO INCORRECT DETERMINATION OF THE FORCE-AXIS ZERO COORDINATE DUE TO INVALID TEST DATA OBTAINED DURING A PREVIOUS VALVE DIAGNOSTIC TEST. ON AUGUST 10, 1991, THE TORQUE SWITCH BYPASS LIMIT SWITCH WAS RETURNED TO NORMAL AND THE TORQUE SWITCH SETTING WAS RAISED PER NED INSTRUCTIONS. THE SAFETY SIGNIFICANCE FOR THIS EVENT WAS MITIGATED BY THE AVAILABILITY OF THE CORE SPRAY SUBSYSTEMS.

[102] DRESDEN 2 DOCKET 50-237 LER 91-037
REACTOR SCRAM ON SPURIOUS INTERMEDIATE RANGE MONITOR HI-HI SIGNALS DUE TO ELECTROMAGNETIC INTERFERENCE.
EVENT DATE: 111391 REPORT DATE: 120391 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.
WHITTAKER CORP.

(NSIC 223597) AT 1541 HOURS ON NOVEMBER 13, 1991, AN AUTOMATIC REACTOR SCRAM OCCURRED DUE TO A SPURIOUS HI-HI NEUTRON FLUX SIGNAL ON THE INTERMEDIATE RANGE MONITORS (IRMS). A UNIT STARTUP WAS IN PROGRESS WITH REACTOR PRESSURE AT APPROXIMATELY 101 PSIG AT THE TIME OF THE SCRAM. THE SCRAM FUNCTIONS THAT OCCURRED WERE PROPER UPON RECEIPT OF THE SPURIOUS SPIKES. THE SPURIOUS SPIKES OCCURRED AS A HIGH PRESSURE COOLANT INJECTION (HPCI) STEAM ISOLATION VALVE WAS OPENED. AN EXTENSIVE INVESTIGATION INTO THE SPIKING DETERMINED THE CAUSE TO BE ELECTROMAGNETIC INTERFERENCE (EMI) PRODUCED BY DC VALVE MOTOR OPERATION. THE EMI WAS BEING PICKED UP IN THE DRYWELL BY THE IRM SIGNAL CABLES. DURING THE LAST REFUELING OUTAGE, NEW STAINLESS STEEL JACKETED WHITTAKER SIGNAL CABLES WERE INSTALLED IN THE DRYWELL OUTSIDE OF THE IRM RIGID CONDUIT. CORRECTIVE ACTIONS INVOLVED REROUTING THE IRM SIGNAL CABLES IN THE DRYWELL INSIDE FLEXIBLE CONDUIT FOR ADDITIONAL SHIELDING, AND INSTALLATION OF VARISTORS ON HPCI 250 VDC VALVE MOTOR CIRCUITRY TO REDUCE THE AMOUNT OF EMI PRODUCED BY VALVE OPERATION. IN ADDITION, THE IRM SIGNAL CABLES WILL BE ROUTED INSIDE PERMANENT RIGID CONDUIT DURING THE NEXT REFUELING OUTAGE. PREVIOUS SIMILAR EVENTS ARE DOCUMENTED BY LERS 90-15/050237, 90-17/050237, AND 91-18/050237.

[103] DRESDEN 2 DOCKET 50-237 LER 91-040
ISOLATION CONDENSER GROUP V ISOLATION DUE TO SPURIOUS FLOW SPIKES.
EVENT DATE: 111991 REPORT DATE: 120991 NSSS: GE TYPE: BWR

(NSIC 223630) ON NOVEMBER 19, 1991, AT 2355 HOURS WITH UNIT 2 IN COLD SHUTDOWN WITH ALL CONTROL RODS INSERTED, AN UNPLANNED PRIMARY CONTAINMENT GROUP V ISOLATION OCCURRED, CAUSING CLOSURE OF THE ISOLATION CONDENSER ISOLATION VALVES. THE ISOLATION SIGNAL WAS RESET AFTER VERIFICATION THAT THE SIGNAL WAS SPURIOUS. IN ADDITION, FOLLOWING THE EVENT, ALL OF THE GROUP V INITIATING INSTRUMENTATION SETPOINTS WERE VERIFIED TO BE WITHIN THEIR SPECIFIED TOLERANCES. THE PROBABLE CAUSE IS DIFFERENTIAL PRESSURE FLOW SPIKES IN THE ISOLATION CONDENSER CONDENSATE RETURN LINE WHILE STARTING THE 2B SHUTDOWN COOLING PUMP. ISOLATION CONDENSER OPERABILITY IS NOT REQUIRED WHENEVER REACTOR PRESSURE IS LESS THAN 90 PSIG; ALSO, HAD THIS EVENT OCCURRED UNDER POWER OPERATION, THE HIGH PRESSURE COOLANT INJECTION OR AUTOMATIC DEPRESSURIZATION SYSTEMS COULD HAVE BEEN UTILIZED FOR REACTOR PRESSURE CONTROL. THEREFORE, THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL. A PREVIOUS UNPLANNED GROUP V ISOLATION WAS REPORTED BY LER 91-006/050237.

[104] DRESDEN 2 DOCKET 50-237 LER 92-001
'A' FLOOR DRAIN SAMPLE TANK SURVEILLANCE INTERVAL EXCEEDED DUE TO MANAGEMENT DEFICIENCY.
EVENT DATE: 010992 REPORT DATE: 013092 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 223907) ON JANUARY 6, 1992 AT 0530 WITH UNITS 2 AND 3 SHUT DOWN, THE RADWASTE OPERATOR BEGAN TO FILL THE 'A' RADWASTE FLOOR DRAIN SAMPLE TANK ('A' FDST) WITH EFFLUENT FROM THE 'A' CONCENTRATOR. FILLING OF THE 'A' FDST WAS SECURED BECAUSE OF A STEAM LEAK TO THE DEARATOR TANK. DURING THE NEXT THREE DAYS, DIFFICULTIES WERE ENCOUNTERED WITH THE TEMPORARY HEATING BOILER. PROBLEMS INVOLVING CONDUCTIVITY, LEVEL CONTROL, SOLIDS, AND FOAMING WITH THE 'A' CONCENTRATOR WERE ALSO ENCOUNTERED. DUE TO THE PROBLEMS WITH THE 'A' CONCENTRATOR, RADWASTE SUPERVISION LOST TRACK OF THE ELAPSED TIME SINCE THE 'A' FDST HAD BEEN SAMPLED AS REQUIRED BY TECHNICAL SPECIFICATION 4.8.D, I.E., 72 HOURS AFTER FILLING. RADWASTE SUPERVISION REALIZED THAT A SAMPLE WAS OVERDUE FOR ANALYSIS APPROXIMATELY 30 MINUTES AFTER THE 72 HOUR SURVEILLANCE PERIOD HAD BEEN EXCEEDED. RADWASTE SUPERVISION STOPPED FILLING THE 'A' FDST AND OBTAINED A SAMPLE. THE RESULTS OF THE SAMPLE WERE 2.3 E-5 MICRO-CI/ML (WELL WITHIN THE TECHNICAL SPECIFICATION ACTIVITY LIMITS). THE CAUSE OF THIS EVENT WAS AN INADEQUATE MANAGEMENT CONTROL SYSTEM (THE TIME AND DATE OF STARTING TO FILL THE 'A' FDST AND WHEN ITS SAMPLE WAS REQUIRED WAS NOT COMMUNICATED ADEQUATELY FROM ONE CREW TO THE NEXT). CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED IMPLEMENTATION OF AN IMPROVED RADWASTE ACTIVITY MANAGEMENT CONTROL SYSTEM AND COUNSELING OF THE PERSONNEL INVOLVED.

[105] DRESDEN 2 DOCKET 50-237 LER 92-002
SPURIOUS CLOSURE OF MAIN STEAM ISOLATION VALVE DUE TO FAILURE OF DC PILOT SOLENOID.
EVENT DATE: 011392 REPORT DATE: 020492 NSSS: GE TYPE: BWR
VENDOR: AUTOMATIC VALVE COMPANY

(NSIC 224024) ON 1/13/92 AT 0254 HOURS, WITH UNIT 2 IN COLD SHUTDOWN, OUTBOARD MAIN STEAM ISOLATION VALVE (MSIV) 2-203-2A WAS OBSERVED TO SPURIOUSLY ISOLATE (CLOSE). AT THE TIME OF THIS EVENT, PREPARATIONS WERE BEING MADE TO DE-ENERGIZE MOTOR CONTROL CENTER (MCC) 28-2 PRIOR TO PERFORMING MODIFICATION M12-2-91-027A. AS PART OF THE PREPARATION, THE INSTRUMENT BUS WAS TRANSFERRED FROM ITS NORMAL POWER FEED (MCC 28-2) TO ITS RESERVE POWER FEED (MCC 25-2), WHEN THE MSIV WAS OBSERVED TO SPURIOUSLY ISOLATE. THE APPARENT CAUSE OF THIS EVENT WAS ATTRIBUTED TO FAILURE OF THE MSIV DC PILOT SOLENOID DUE TO AN OPEN COIL. DE-ENERGIZATION OF BOTH AC AND DC PILOT SOLENOIDS IS REQUIRED TO CAUSE MSIV CLOSURE. TRANSFER OF THE INSTRUMENT BUS PROVIDED A NORMAL, MOMENTARY LOSS OF POWER TO THE MSIV AC SOLENOID. THIS ALLOWED FOR CLOSURE OF THE 2-203-2A MSIV DUE TO DE-ENERGIZATION OF BOTH THE AC AND DC PILOT SOLENOIDS. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS CONSIDERED MINIMAL SINCE THE ISOLATION LOGIC PERFORMED AS DESIGNED UPON FAILURE OF THE SOLENOID, AND THE UNPLANNED ESF ACTUATION HAD NO EFFECT ON PLANT STATUS.

THE DC SOLENOID WAS REPLACED UNDER WORK REQUEST 05964 AND 3-203-2A MSIV WAS RETURNED TO SERVICE ON 1/15/92, AT 0115 HOURS. A PREVIOUS SIMILAR EVENT WAS REPORTED BY LER 90-002/050237.

[106] DRESDEN 3 DOCKET 50-249 LER 91-008 REV 01
UPDATE ON UNPLANNED PRIMARY CONTAINMENT GROUP V ISOLATION DUE TO A BLOWN BULB.
EVENT DATE: 083091 REPORT DATE: 092791 NSSS: GE TYPE: BWR
VENDOR: CRANE COMPANY

(NSIC 223860) ON AUGUST 30, 1991, AT 1503 HOURS WITH UNIT 3 IN THE RUN MODE AT 51% 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 3-1501-26A, THE LIGHT BULB CAUSED A SHORT CIRCUIT AND CAUSED 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 BULB AND FUSE WERE IMMEDIATELY REPLACED; SUBSEQUENT REPLACEMENT OF THE SOCKET WAS COMPLETED UNDER WORK REQUEST (WR) 03389. INSPECTION OF THE REMOVED SOCKET INDICATED THAT THE ROOT CAUSE WAS A CRACKED HOUSING. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL SINCE THE ISOLATION SIGNAL WAS RESET AND ALL ACTIVE COMPONENTS OF THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM REMAINED OPERABLE DURING THE EVENT. TO PREVENT RECURRENCE OF THIS EVENT, DURING THE CURRENT UNIT 3 REFUELING OUTAGE (D3R12) AND THE NEXT UNIT 2 REFUELING OUTAGE (D2R13) A MINOR PLANT MODIFICATION WILL ADD TWO FUSES IN SERIES TO ISOLATE THE LPCI INDICATION CIRCUIT FROM THE ISOLATION CONDENSER ISOLATION VALVE CONTROL CIRCUITRY. A PREVIOUS EVENT INVOLVING AN UNPLANNED PRIMARY CONTAINMENT GROUP V ISOLATION WAS REPORTED BY LER 90-005 ON DOCKET 0500237.

[107] DRESDEN 3 DOCKET 50-249 LER 91-015
UNPLANNED STANDBY GAS TREATMENT SYSTEM AUTOMATIC INITIATION SIGNAL DUE TO PERSONNEL ERROR.
EVENT DATE: 110591 REPORT DATE: 120591 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 2 (BWR)

(NSIC 223782) ON 11/5/91, WITH UNIT 2 SHUTDOWN AND UNIT 3 IN A REFUEL OUTAGE, THE ELECTRICAL MAINTENANCE DEPARTMENT (EMD) WAS SCHEDULED TO PERFORM A PREVENTIVE MAINTENANCE SURVEILLANCE OF THE UNIT 3 REACTOR PROTECTION SYSTEM (RPS) MOTOR GENERATOR (MG) SET OUTPUT BREAKERS. WITH THE 'B' TRAIN OF THE STANDBY GAS TREATMENT SYSTEM (SBGTS) ALREADY OPERATING AND THE REACTOR BUILDING VENTILATION SYSTEM (RBVS) ISOLATED, THE EMD INSTALLED A TEMPORARY POWER SUPPLY TO THE CHANNEL 'B' REFUEL FLOOR AND REACTOR BUILDING VENTILATION RADIATION MONITORS TO PREVENT AN AUTOMATIC START OF THE SBGTS AND AN ISOLATION OF THE RBVS UPON THE REMOVAL OF THE 'A' RPS MG SET FROM SERVICE. THE OPRATIONS DEPARTMENT THEN REMOVED THE 'B' RPS MG SET FROM SERVICE, RESULTING IN THE DE-ENERGIZATION OF THE 'A' RPS BUS. THIS PRODUCED AN UNPLANNED SBGTS AUTOMATIC START SIGNAL AND AN UNPLANNED RAVS ISOLATION SIGNAL. THE CONTROL ROOM OPERATORS ANTICIPATED AN AUTOMATIC START OF THE 'A' TRAIN OF THE SBGTS UNDER THESE CONDITIONS; HOWEVER, THIS DID NOT OCCUR. FURTHER REVIEW INDICATED THAT THE 'A' TRAIN SHOULD NOT HAVE STARTED WITH THE 'B' TRAIN RUNNING DUE TO A SEQUENCE OF SBGTS SELECTOR SWITCH MANIPULATIONS THAT TOOK PLACE PRIOR TO THIS EVENT. THIS EVENT HAD MINIMAL SAFETY SIGNIFICANCE. RELATED LER: 91-04.

[108] DRESDEN 3 DOCKET 50-249 LER 91-012
PARTIAL PRIMARY CONTAINMENT GROUP II ISOLATION DURING MAINTENANCE REPAIRS DUE TO PERSONNEL ERROR.
EVENT DATE: 112091 REPORT DATE: 121991 NSSS: GE TYPE: BWR

(NSIC 223670) AT 2213 HOURS, WITH UNIT 3 IN THE SHUTDOWN MODE, 14 OF THE 44 GROUP II PRIMARY CONTAINMENT ISOLATION VALVES WENT CLOSED WHEN AN ELECTRICIAN LIFTED A NEUTRAL LEAD IN PREPARATION FOR REPLACEMENT OF A TERMINAL BLOCK UNDER WORK REQUEST 04380. THIS EVENT HAS BEEN ATTRIBUTED TO PERSONNEL ERROR ON THE PART OF THE ELECTRICAL WORK ANALYST WHO PREPARED THE WORK PACKAGE. THE ANALYST DID NOT FULLY RESEARCH THE WIRING DIAGRAM FOR THE PANEL WHICH CONTAINS THE TERMINAL BLOCKS, AND THUS DID NOT IDENTIFY THE CIRCUITS "DAISY-CHAINED" TO THE NEUTRAL

LEAD CONNECTED AT THE TERMINAL BLOCK TO ESTABLISH THE CORRECT OUT-OF-SERVICE (OOS) BOUNDARIES. ONE OF THESE "DAISY-CHAINED" CIRCUITS CONTAINS THE TRIP CONTACTS FOR THE 14 VALVES WHICH CLOSED. THIS EVENT HAD MINIMAL SAFETY SIGNIFICANCE BECAUSE THE UNPLANNED VALVE CLOSURES HAD NO EFFECT ON THE PLANT STATUS, AND THE GROUP II ISOLATION FUNCTION WAS NOT IMPAIRED. AS A CORRECTIVE ACTION, THIS EVENT WILL BE DISCUSSED WITH THE ELECTRICAL MAINTENANCE DEPARTMENT WORK ANALYSTS. ONE PREVIOUS EVENT INVOLVING IN ADVERTENT GROUP II VALVE CLOSURES DUE TO A LIFTED LEAD IS REPORTED IN LER 90-22/0500237.

[109] DRESDEN 3 DOCKET 50-249 LER 92-001
PRIMARY CONTAINMENT ISOLATION VALVE CLOSURE DUE TO SHUTDOWN COOLING SYSTEM SPURIOUS ISOLATION.
EVENT DATE: 010292 REPORT DATE: 012492 NSSS: GE TYPE: BWR

(NSIC 223852) ON JANUARY 2, 1992, AT 1854 HOURS, WITH UNIT 3 SHUT DOWN FOR A REFUEL OUTAGE AND THE SHUTDOWN COOLING (SDC) SYSTEM IN OPERATION TO COOL THE REACTOR WATER, AN UNPLANNED SDC SYSTEM ISOLATION OCCURRED, CAUSING AUTOMATIC CLOSURE OF THE SDC SYSTEM PRIMARY CONTAINMENT ISOLATION MOTOR OPERATED VALVES (MOVS). THERE WERE NO INDICATIONS AS TO THE CAUSE OF THE ISOLATION. THE SYSTEM WAS PROMPTLY RESET/RESTARTED. A WORK REQUEST WAS INITIATED TO VERIFY THE CALIBRATION OF THE SDC SUCTION TEMPERATURE INSTRUMENT LOOP. DURING COMPLETION OF THIS WORK ON JANUARY 13, 1992, FURTHER DIFFICULTY OCCURRED IN RESETTING AN SDC ISOLATION DUE TO UNRELATED CAUSES (REFER TO LER 92-2/050249). ISOLATION OF THE SDC SYSTEM HAD MINIMAL SAFETY SIGNIFICANCE BECAUSE THE SDC SYSTEM WAS PROMPTLY RESTORED TO NORMAL OPERATION AND THE REACTOR WATER TEMPERATURE WAS MAINTAINED WELL BELOW THE 212 DEGREE F LIMIT REQUIRED FOR PRIMARY CONTAINMENT INTEGRITY. A PREVIOUS SPURIOUS DRESDEN UNIT 2 SDC ISOLATION WAS REPORTED BY LER 91-36/050237.

[110] DRESDEN 3 DOCKET 50-249 LER 92-002
PRIMARY CONTAINMENT ISOLATION VALVE CLOSURE DUE TO A DAMAGED CONTROL RELAY.
EVENT DATE: 011392 REPORT DATE: 013192 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223906) ON JANUARY 13, 1992, AT 1115 HOURS, WITH UNIT 3 SHUTDOWN FOR A REFUEL OUTAGE AND THE SHUTDOWN COOLING (SDC) SYSTEM COOLING THE REACTOR WATER, THE INSTRUMENT MAI NTENANCE DEPARTMENT COMPLETED CALIBRATION CHECKS OF THE SDC TEMPERATURE SWITCHES UNDER WORK REQUEST (WR) 05767. THIS PROCEDURE INCLUDED A PLANNED ISOLATION OF T HE SDC SYSTEM. THE SDC SYSTEM SUCTION HIGH TEMPERATURE INSTRUMENT LOOP WAS FOUND TO BE PROPERLY CALIBRATED. HOWEVER, WHEN OPERATIONS ATTEMPTED TO RESTART THE SD C SYSTEM, THE PRIMARY CONTAINMENT ISOLATION VALVES WOULD NOT OPEN. IT WAS DISCOV ERED THAT THE VALVES' FAILURE TO OPEN WAS DUE TO A DAMAGED ISOLATION CONTROL REL AY. THE RELAY WAS REPLACED AND THE NEW RELAY WAS FUNCTIONALLY TESTED. THE SDC SY STEM WAS THEN RETURNED TO SERVICE AT 1425 HOURS. THE ROOT CAUSE WAS DETERMINED T O BE EXTERNAL DAMAGE TO A CONTROL RELAY (3-595-114) IN THE SDC SYSTEM ISOLATION LOGIC. THE ELECTRICAL MAINTENANCE DEPARTMENT THEN INSPECTED ALL SAFETY RELATED R ELAYS OF THE SAME TYPE, ON BOTH DRESDEN UNITS 2 AND 3, FOR DAMAGE. THIS EVENT HA D MINIMAL SAFETY SIGNIFICANCE DUE TO THE FACT THAT THE SDC SYSTEM WAS PROMPTLY R ESTORED TO OPERATION AND THE REACTOR WATER TEMPERATURE WAS MAINTAINED WELL BELOW THE 212 DEGREE F LIMIT REQUIRED FOR PRIMARY CONTAINMENT INTEGRITY. A PREVIOUS S PURIOUS SDC SYSTEM ISOLATION WAS REPORTED BY LER 92-1/050249.

[111] DRESDEN 3 DOCKET 50-249 LER 92-003
UNEXPECTED PARTIAL GROUP II CONTAINMENT ISOLATION DURING SURVEILLANCE TESTING DUE TO PERSONNEL ERROR.
EVENT DATE: 011592 REPORT DATE: 020692 NSSS: GE TYPE: BWR

(NSIC 223928) AT 1317 HOURS ON JANUARY 15, 1992, WITH UNIT 3 SHUT DOWN FOR A REFUEL OUTAGE, AN UNPLANNED PARTIAL PRIMARY CONTAINMENT GROUP II ISOLATION OCCURRED WHILE ELECTRICIANS WERE PERFORMING A SURVEILLANCE ON MAIN STEAM ISOLATION VALVE (MSIV) ELECTRICAL COMPONENTS. ONE OF THE ELECTRICIANS DETERMINATED A WIRING CONNECTION IN ACCORDANCE WITH THE PROCEDURE; HOWEVER, THIS UNEXPECTEDLY INTERRUPTED CONTROL POWER TO SEVERAL SEAL-IN RELAYS CAUSING CERTAIN

GROUP II ISOLATION VALVES TO CLOSE. NO GROUP II ISOLATION ALARM WAS RECEIVED. THE ELECTRICIANS COMPLETED THE TESTING ON MSIV 3-203-1D AND WERE THEN INFORMED BY THE UNIT 3 OPERATOR THAT A PARTIAL ISOLATION HAD BEEN PRODUCED. THE ROOT CAUSE OF THE EVENT WAS ATTRIBUTED TO PERSONNEL ERROR ON THE PART OF AN INDIVIDUAL PERFORMING A FIELD VERIFICATION OF THE PROCEDURE PRIOR TO ITS USE. CORRECTIVE ACTIONS INCLUDED A FIELD VERIFICATION OF THE SURVEILLANCE, A WORK REQUEST WAS INITIATED TO CORRECT THE WIRING, AND REVISION TO THE SURVEILLANCE PROCEDURE. THIS EVENT HAD MINIMAL SAFETY SIGNIFICANCE BECAUSE THERE WAS NO EFFECT ON THE LOGIC FUNCTION. A PREVIOUS RELATED EVENT WAS REPORTED BY LER 90-22/050237.

[112] FARLEY 1 DOCKET 50-348 LER 91-012
PROCEDURAL INADEQUACIES FOR VERIFYING THE INTERLOCK ACTION OF THE RHR SYSTEM FROM THE RCS.
EVENT DATE: 122991 REPORT DATE: 012792 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: FARLEY 2 (PWR)

(NSIC 223866) ON 12-29-91, DURING A PROCEDURE REVIEW, IT WAS DISCOVERED THAT FNP-2-STP-11.5, "RHR SUCTION VALVE AUTOMATIC ISOLATION TEST", DID NOT INCLUDE A VERIFICATION THAT THE REACTOR COOLANT SYSTEM (RCS) LOOP TO RESIDUAL HEAT REMOVAL (RHR) PUMP SUCTION VALVES Q2E11MOV8701A, Q2E11MOV8701B, Q2E11MOV8702A AND Q2E11MOV8702B COULD NOT BE OPENED FROM THE MAIN CONTROL BOARD (MCB) WHEN RCS PRESSURE IS ABOVE THE AUTOMATIC CLOSURE SETPOINT (700 PSIG). THE VERIFICATION OF THIS INTERLOCK IS PART OF THE REQUIRED SURVEILLANCE TESTING ASSOCIATED WITH TECHNICAL SPECIFICATION 4.5.2. IN ADDITION, PRIOR TO 3-1-88 THIS REQUIREMENT HAD NOT BEEN INCORPORATED INTO FNP-1-STP-11.5 FOR THE CORRESPONDING SYSTEM IN UNIT 1 AND THEREFORE THE SURVEILLANCE HAD NOT BEEN PERFORMED. THIS EVENT WAS CAUSED BY PROCEDURAL INADEQUACY. FNP-2-STP-11.5 HAS BEEN REVISED TO PROVIDE VERIFICATION OF THE INTERLOCK AND TO INCLUDE THIS AS AN ACCEPTANCE CRITERION. UNIT 1 PROCEDURE FNP-1-STP-11.5 PREVIOUSLY CONTAINED A STEP TO VERIFY THE INTERLOCK BUT NOT AS PART OF THE ACCEPTANCE CRITERIA. THE UNIT 1 PROCEDURE HAS ALSO BEEN REVISED TO INCLUDE VERIFICATION OF THE INTERLOCK AS AN ACCEPTANCE CRITERION.

[113] FERMI 2 DOCKET 50-341 LER 91-019
REACTOR WATER CLEANUP SYSTEM ISOLATIONS DUE TO HIGH PUMP ROOM DIFFERENTIAL TEMPERATURE AND PERSONNEL ERROR DURING SYSTEM RESTORATION.
EVENT DATE: 112091 REPORT DATE: 122091 NSSS: GE TYPE: BWR
VENDOR: ORBIT VALVE COMPANY

(NSIC 223640) THIS LER DESCRIBES TWO ENGINEERED SAFETY FEATURES ACTUATIONS OF THE REACTOR WATER CLEANUP SYSTEM (RWCU) ON NOVEMBER 20, 1991. THE FIRST ISOLATION OCCURRED AT 1051 HOURS DUE TO A HIGH DIFFERENTIAL ROOM TEMPERATURE (INSIDE TO OUTSIDE THE ROOM) IN THE B RWCU PUMP ROOM. DETROIT EDISON BELIEVES THE CAUSE OF THE HIGH TEMPERATURE WHICH RESULTED IN THIS EVENT, TO BE FAILURE (LOSS OF CALIBRATION) OF A SWITCH IN THE AUTOMATIC CONTROL CIRCUIT FOR THE B RWCU PUMP ROOM COOLER. THIS SWITCH HAS BEEN RECALIBRATED AND WILL BE MONITORED TO ENSURE PROPER OPERATION. THE SECOND ISOLATION OCCURRED AT 2231 HOURS WHILE REMOVING A JUMPER, UTILIZED DURING RESTORATION OF THE RWCU SYSTEM FROM THE FIRST ISOLATION, IN ACCORDANCE WITH THE SYSTEM OPERATING PROCEDURE. THE CAUSE OF THIS ISOLATION IS BELIEVED TO BE A PERSONNEL ERROR WHEN THE JUMPER WAS INADVERTENTLY GROUNDED DURING REMOVAL, WHICH RESULTED IN A BLOWN FUSE IN THE ISOLATION CIRCUITRY FOR ONE RWCU ISOLATION VALVE. AN ACCOUNTABILITY MEETING HAS BEEN HELD REGARDING THIS ERROR. TESTING OF THE RWCU SYSTEM, AT POWER, WILL BE PERFORMED TO DETERMINE IF AN ALTERNATIVE IS AVAILABLE TO INSTALLATION OF A JUMPER DURING SYSTEM RESTORATION FOLLOWING AN ISOLATION. OPERATING PROCEDURE CHANGES WILL BE MADE AS APPROPRIATE. ADDITIONALLY, THIS LER WILL BE ISSUED AS REQUIRED READING FOR OPERATORS AND APPROPRIATE SYSTEMS ENGINEERING PERSONNEL.

[114] FERMI 2 DOCKET 50-341 LER 91-020
HIGH PRESSURE COOLANT INJECTION SYSTEM START FAILURE DURING QUARTERLY SURVEILLANCE TEST.
EVENT DATE: 112091 REPORT DATE: 122091 NSSS: GE TYPE: BWR
VENDOR: WOODWARD GOVERNOR COMPANY

(NSIC 223641) ON NOVEMBER 20, 1991, AT 1954 HOURS, THE HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI) FAILED TO START DURING PERFORMANCE OF SURVEILLANCE, 24.202.01, "HPCI PUMP TIME RESPONSE AND OPERABILITY TEST AT 1000 PSIG." IN COMPLIANCE WITH THE APPROPRIATE LIMITING CONDITION FOR OPERATION, THE HPCI SYSTEM WAS DECLARED INOPERABLE AND THE TECHNICAL SPECIFICATION ACTION STATEMENT WAS ENTERED. THE SYSTEM FAILED TO START WHEN THE GOVERNOR CONTROL VALVE (E4100F068) FAILED TO OPEN AND ADMIT STEAM TO THE HPCI TURBINE. THE INITIAL INVESTIGATION SHOWED THAT THE HYDRAULIC ACTUATOR (EGR), E41-K203, WAS NOT FUNCTIONING, THUS KEEPING THE GOVERNOR VALVE CLOSED. THE EGR WAS REPLACED. THE FAILED EGR WAS SENT TO WOODWARD GOVERNOR COMPANY FOR FAILURE ANALYSIS. POST MAINTENANCE SURVEILLANCE TESTING DEMONSTRATED THAT THE PCI SYSTEM OPERATED PROPERLY. ON NOVEMBER 22, AT 0612 HOURS, THE HPCI SYSTEM WAS DECLARED OPERABLE. THE HPCI CONTROL/LUBE OIL SYSTEM HAD A HIGHER THAN NORMAL WATER CONTENT FOLLOWING THE SUCCESSFUL SURVEILLANCE RUN ON NOVEMBER 22. THIS WAS DUE TO THE HPCI BAROMETRIC CONDENSER VACUUM PUMP NOT PROVIDING ENOUGH VACUUM TO MAINTAIN PROPER GLAND SEAL LEAKOFF. SUBSEQUENTLY, THE HPCI BAROMETRIC CONDENSER VACUUM PUMP WAS REPLACED.

[115] FERMI 2 DOCKET 50-341 LER 91-021
CONTROL CENTER HEATING VENTILATION AND AIR CONDITIONING SHIFTS TO THE RECIRCULATION MODE DURING SURVEILLANCE 24.413.05.
EVENT DATE: 112791 REPORT DATE: 122791 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223732) ON NOVEMBER 27, 1991, OPERATIONS, AND INSTRUMENTATION AND CONTROL (I&C) PERSONNEL WERE PERFORMING SURVEILLANCE PROCEDURE 24.413.05, "DIVISION II CONTROL ROOM EMERGENCY FILTER AUTO TRANSFER TEST". AT 1315 HOURS, DURING PERFORMANCE OF THE SURVEILLANCE, THE DIVISION II CONTROL CENTER HEATING VENTILATION AND AIR CONDITIONING (CCHVAC) SHIFTED TO THE RECIRCULATION MODE. THIS OCCURRED DURING STEP 5.1.3 OF THE SURVEILLANCE PROCEDURE AS THE I&C REPAIRMAN WAS EXITING PANEL H21-P296D AFTER INSTALLING TEST SWITCH LEADS. DIVISION II CCHVAC WAS RETURNED TO NORMAL OPERATION AT 1325 HOURS. THERE WERE TWO CONTRIBUTING FACTORS TO THE ROOT CAUSE OF THIS EVENT. FIRST, THERE WAS AN INADEQUATE EVALUATION OF TEST METHODS FOR THIS SURVEILLANCE. HOWEVER, IT SHOULD BE NOTED THAT THE SURVEILLANCE HAD BEEN PERFORMED SUCCESSFULLY IN THE PAST. SECOND, THE I&C REPAIRMAN INSTALLING THE SWITCH LEADS ON RELAY T41M105 MADE A PERSONNEL ERROR. PROCEDURES 24.413.05 (DIVISION II) AND 24.413.04 "DIVISION I CONTROL ROOM EMERGENCY FILTER AUTO TRANSFER TEST", WILL BE REVISED SO THAT EXTERNAL TEST EQUIPMENT IS NOT USED TO TIME THE CCHVAC SHIFT TO THE RECIRCULATION MODE. AS A RESULT OF THE PERSONNEL ERROR ELEMENT, THE I&C CONTINUING TRAINING (LESSONS LEARNED) PROGRAM WILL DISCUSS THIS EVENT.

[116] FERMI 2 DOCKET 50-341 LER 91-022
PRIMARY CONTAINMENT NEGATIVE PRESSURE DURING PLANT SHUTDOWN.
EVENT DATE: 121191 REPORT DATE: 011092 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 223733) ON DECEMBER 11, 1991, OPERATIONS PERSONNEL WERE PROCEEDING WITH THE PLANNED PLANT SHUTDOWN FOR REPLACEMENT OF THE 2A MAIN UNIT TRANSFORMER. THE SHUTDOWN PLAN INCLUDED UTILIZING TWO MECHANICAL VACUUM PUMPS TO MAINTAIN MAIN CONDENSER VACUUM FOR LOCATING A SUSPECTED TUBE LEAK IN THE CONDENSER. AT 1212 HOURS, DRYWELL PRESSURE STARTED DECREASING FROM 15.2 PSIA TO 14.4 PSIA. THE TORUS TO DRYWELL AND REACTOR BUILDING TO TORUS VACUUM BREAKERS BEGAN LIFTING AT 1226 HOURS, AS DESIGNED, TO CONTROL THE NEGATIVE PRESSURE. AT 1553 HOURS, THE THIRD MSIVS WERE CLOSED AND THE PRIMARY CONTAINMENT VACUUM BREAKERS STOPPED CYCLING. PRIMARY CONTAINMENT PRESSURE BEGAN INCREASING DUE TO ADDITION OF NITROGEN WHICH HAD BEEN STARTED TWO HOURS EARLIER. DRYWELL PRESSURE WAS RESTORED TO NORMAL AND NITROGEN ADDITION WAS TERMINATED AT 1922 HOURS. INVESTIGATION ESTABLISHED THAT THE CAUSE OF THIS PRESSURE EVENT WAS A REVERSE FLOW THROUGH THE SAFETY RELIEF VALVES (SRVS). THE FLOWPATH WAS FROM THE DRYWELL THROUGH THE SRV VACUUM BREAKERS, PRODUCING REVERSE FLOW THROUGH THE SRVS INTO THE MAIN STEAM LINE DRAINS AND TO THE CONDENSER, WHICH WAS AT A VACUUM CONDITION CREATED BY OPERATING TWO MECHANICAL VACUUM PUMPS. OPERATIONS AND PLANT SUPPORT PERSONNEL WERE NOT FAMILIAR WITH SRV OPERATION IN THE ABNORMAL CONDITION OF A VACUUM ENVIRONMENT.

[117] FITZPATRICK DOCKET 50-333 LER 90-024 REV 01
UPDATE ON PARTIAL ACTIVATION OF PRIMARY CONTAINMENT GROUP II ISOLATION CIRCUIT DUE TO BLOWN FUSE DUE TO HUMAN ERRORS DURING IMPLEMENTATION OF PROTECTIVE TAGGING PROCEDURE.
EVENT DATE: 110190 REPORT DATE: 122791 NSSS: GE TYPE: BWR
VENDOR: AAA PRODUCTS
XOMOX CORP.

(NSIC 223774) WHILE THE REACTOR WAS OPERATING AT 100% POWER AT 233 ON 11/01/90 A HIGH LEVEL ALARM FOR THE PRIMARY CONTAINMENT (DRYWELL) EQUIPMENT DRAIN SUMP WAS RECEIVED. A FUSE OPENED AT 2338 RESULTING IN A PARTIAL ACTUATION OF THE B SIDE OF THE GROUP II PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) (JM). THE REACTOR BUILDING VENTILATION SYSTEM (VA) ISOLATED AND STANDBY GAS TREATMENT B (SBGT) (BH) STARTED AS DESIGNED. THREE VALVES THAT RECEIVE THE ISOLATION SIGNAL WERE ALREADY CLOSED. TWO OTHERS CLOSED AS DESIGNED. THE ISOLATIONS WERE RESET WITHIN 37 MINUTES. A REPAIR TASK ON A SMALL SOLENOID OPERATED AIR ADMISSION VALVE UNEXPECTEDLY EXTENDED OVER THREE SHIFTS DURING WHICH ONE TRAIN OF THE RESIDUAL HEAT REMOVAL SYSTEM (BO) WAS BEING RESTORED TO SERVICE AND RADIOACTIVE RESIN TRANSFER WAS IN PROGRESS. MULTIPLE HUMAN ERRORS AND POOR COMMUNICATION BETWEEN SHIFTS OCCURRED DURING EQUIPMENT PROTECTIVE TAGGING PROCEDURE IMPLEMENTATION. THE SOLENOID WAS REMOVED FROM THE VALVE BUT WAS LEFT ELECTRICALLY CONNECTED DUE TO THE HUMAN ERRORS. AN AUTOMATIC SIGNAL FROM THE SUMP HIGH LEVEL SUBSEQUENTLY ENERGIZED THE SOLENOID DAMAGING THE INSULATION RESULTING IN HIGH CURRENT AND OPENING OF THE FUSE. RELATED LERS: 89-017, 89-013, 87-016, AND 86-019.

[118] FITZPATRICK DOCKET 50-333 LER 91-021
RESIDUAL HEAT REMOVAL, EMERGENCY DIESEL GENERATORS, AND FIRE PUMPS POTENTIALLY MADE INOPERABLE DUE TO INADEQUATE MODIFICATION/INSTALLATION ACTIVITIES.
EVENT DATE: 090491 REPORT DATE: 112791 NSSS: GE TYPE: BWR

(NSIC 223498) THE EXHAUST VENTILATION FAN FOR ONE OF TWO SAFETY-RELATED PUMP ROOMS TRIPPED ON THERMAL OVERLOAD AT 1400 ON 9/4/91 WITH THE PLANT AT FULL POWER. THE FAN TRIP WAS THE RESULT OF A RESTRICTED VENTILATION AIR SUPPLY DUE TO CLOSURE OF 6 FIRE DAMPERS TO ALLOW MODIFICATION OF THE DAMPERS. THE RESTRICTED AIR SUPPLY RESULTED IN OVERLOAD DURING FAN AUTO START WHILE WINDMILLING IN REVERSE OR DUE TO LOW FLOW DURING OPERATION. AN ENGINEERING REVIEW FORMALIZED ON 10/9/91 IDENTIFIED A FIRE SCENARIO WITH POTENTIAL FOR DAMAGE TO THE FAN AND FIRE DAMPER CONTROLS AND RESULTING LOSS OF EXHAUST FANS AND CLOSURE OF VENTILATION FIRE DAMPERS. VENTILATION LOSS DURING PUMP OPERATION WOULD DEGRADE PERFORMANCE OF THE RESIDUAL HEAT REMOVAL (BO), RESIDUAL HEAT REMOVAL SERVICE WATER (BI), EMERGENCY SERVICE WATER (ESW) (BI), AND ELECTRIC AND DIESEL DRIVEN FIRE PUMPS (KP). LOSS OF THE ESW FLOW WOULD DEGRADE THE EMERGENCY DIESEL GENERATORS (EK). INADEQUATE VENTILATION FOLLOWING FIRE DAMPER CLOSURE RESULTS FROM INADEQUATE ANALYSIS OF THE NRC REQUIRED-1980 FIRE DAMPER INSTALLATION. A FIRE WATCH IN PLACE PRIOR TO THE EVENT WILL REMAIN UNTIL ADEQUATE VENTILATION IS ASSURED. INSTALLATION PROCEDURE REVIEW ADEQUACY AND ALTERNATE VENTILATION DESIGNS ARE BEING EVALUATED. LER-91-010 IS RELATED.

[119] FITZPATRICK DOCKET 50-333 LER 91-023
BOTH TRAINS OF SAFE SHUTDOWN EQUIPMENT POTENTIALLY DAMAGED BY A COMMON FIRE DUE TO CABLE CONDUIT NOT BEING PROTECTED FROM FIRE.
EVENT DATE: 100991 REPORT DATE: 121991 NSSS: GE TYPE: BWR

(NSIC 223727) DURING NORMAL FULL POWER OPERATION ON 10/9/91 IT WAS FOUND THAT A SAFETY DIVISION 1 MOTOR CONTROL CENTER (ED) FEEDER CABLE WAS ROUTED THROUGH A FIRE ZONE FOR WHICH DIVISION 1 EQUIPMENT IS RELIED ON TO MEET SAFE SHUTDOWN REQUIREMENTS OF 10 CFR 50, APPENDIX R, SECTION III.G. WHILE THE ORIGINAL APPENDIX R EVALUATION INDICATED THE CABLE WAS WITHIN CONDUIT EMBEDDED IN CONCRETE, IT WAS DISCOVERED THAT A PORTION OF THE CONDUIT IS ACTUALLY NOT EMBEDDED. A FIRE WATCH WAS POSTED IMMEDIATELY. THE CABLE WILL BE REROUTED OR THE CONDUIT WILL BE PROTECTED TO MEET APPENDIX R REQUIREMENTS PRIOR TO PLANT START-UP FOLLOWING THE 1992 REFUEL OUTAGE. LER-91-010 DESCRIBES ADDITIONAL FIRE PROTECTION PROGRAM DEFICIENCIES.

[120] FITZPATRICK DOCKET 50-333 LER 91-025
SUPPRESSION POOL BULK TEMPERATURE MONITOR INOPERABLE DUE TO A MANUFACTURING INADEQUATE SOLDERED TERMINATION AND INCORRECT OPERATION INSTRUCTION PROGRAM INSERTED.
EVENT DATE: 111491 REPORT DATE: 122091 NSSS: GE TYPE: BWR
VENDOR: FOXBORO CO., THE

(NSIC 223692) ON 11/14/91 AT 1920, THE PRIMARY CONTAINMENT (NH) SUPPRESSION POOL WATER TEMPERATURE MONITORING SYSTEM CHANNELS A AND B WERE DECLARED INOPERABLE, PLACING THE PLANT IN A 6-HOUR LIMITING CONDITION FOR OPERATION (LCO) WHEN A POTENTIAL INADEQUATE MANUFACTURER'S SOLDERED WIRING TERMINATION WAS DISCOVERED. ON 11/15/91 AT 1738, AFTER CHANNEL A HAD BEEN RETURNED TO SERVICE AND THE 6-HOUR LCO EXITED AN INCORRECT SENSOR FAILURE DEVIATION DELTA VALUE WAS FOUND TO HAVE BEEN INSERTED INTO ONE OF THE CHANNEL A SUMMATION/AVERAGING COMPUTATION MODULES. THIS ACTION CAUSED THE PLANT NOT TO BE WITHIN THE REQUIREMENT OF TECHNICAL SPECIFICATION (TS) TABLE 3.2-6 LCO ACTION FOR APPROXIMATELY 17 HOURS. THE CAUSE OF THE UNSOLDERED WIRING TERMINATIONS WAS AN INADEQUATE MANUFACTURING ASSEMBLY PROCESS AND THE INCORRECT DEVIATION DELTA VALUE WAS LACK OF ADMINISTRATIVE CONTROLS FOR INSTRUMENTATION COMPUTATION COMPONENT OPERATING INSTRUCTIONS (PROGRAMS). BOTH HAD THE POTENTIAL FOR DEGRADED PLANT OPERATOR RESPONSE TO POSTULATED DESIGN BASIS ACCIDENTS. CORRECTIVE ACTION WAS SOLDERING OF THE UNSOLDERED WIRING TERMINATIONS AND INSERTING THE CORRECT DEVIATION DELTA VALUE INTO THE MODULE. ADMINISTRATIVE CONTROL OF THE PROGRAM WAS STRENGTHENED BY ISSUING ACCESS AND USAGE INSTRUCTION TO STAFF. RELATED LER: 90-029.

[121] FITZPATRICK DOCKET 50-333 LER 91-024
UNSATISFACTORY PENETRATION SEALS FOUND DURING INSPECTION.
EVENT DATE: 111691 REPORT DATE: 121391 NSSS: GE TYPE: BWR

(NSIC 223660) ON NOVEMBER 16, 1991, AT APPROXIMATELY 1730 HOURS WHILE THE PLANT WAS OPERATING AT 100% RATED POWER, SEVEN ELECTRICAL PENETRATION FIRE SEALS WERE DISCOVERED IN UNSATISFACTORY CONDITION DURING THE PERFORMANCE OF AN INSPECTION. THE INSPECTION IS CURRENTLY IN PROGRESS TO ENSURE THAT EACH AS-INSTALLED PENETRATION FIRE SEAL CONFIGURATION IS QUALIFIED AS A THREE HOUR RATED FIRE SEAL BY A QUALIFICATION TEST OR IS ANALYTICALLY EQUIVALENT TO A TESTED CONFIGURATION. THE INSPECTION WILL ALSO SATISFY THE TECHNICAL SPECIFICATION REQUIRED SURVEILLANCE. THIS INSPECTION REPRESENTS AN INTENSIVE EFFORT TO COLLECT AND EVALUATE DATA ON ALL ELECTRICAL AND MECHANICAL PENETRATION FIRE SEALS. THIS LER IS BEING WRITTEN AS AN INTERIM REPORT. A SUPPLEMENTAL LER WILL BE ISSUED, WHICH ADDRESSES THE SAFETY ASSESSMENTS AND ROOT CAUSES, AS WELL AS REWORK AND REPAIRS PERFORMED DURING THE INSPECTION, THAT WILL INCLUDE ALL UNSATISFACTORY PENETRATION SEALS FOUND. UNTIL REPAIRS ARE COMPLETED, FIRE WATCHES WILL BE STATIONED IN ACCORDANCE WITH TECHNICAL SPECIFICATION SECTION 3.12.F.1.B, AS REQUIRED.

[122] FITZPATRICK DOCKET 50-333 LER 91-026
INOPERABLE CONTAINMENT ISOLATION VALVES IN BOTH CORE SPRAY SYSTEMS.
EVENT DATE: 112791 REPORT DATE: 122391 NSSS: GE TYPE: BWR
VENDOR: PACIFIC VALVES, INC.

(NSIC 223728) ON NOVEMBER 27, 1991 AT 10:05 A.M. WITH THE PLANT AT FULL POWER, THE "A" AND "B" TRAIN CORE SPRAY (BM) PUMP MINIMUM FLOW VALVES (14MOV-5A/B) WERE DECLARED INOPERABLE DUE TO THEIR INABILITY TO MEET THEIR PRIMARY CONTAINMENT (NH) ISOLATION FUNCTION. TECHNICAL SERVICES DISCOVERED THAT THE VALVES DID NOT HAVE REMOTE MANUAL OPERATION CAPABILITY AS REQUIRED BY THE PLANT'S FINAL SAFETY ANALYSIS REPORT AND TECHNICAL SPECIFICATIONS.

[123] FITZPATRICK DOCKET 50-333 LER 91-027
LOW PRESSURE COOLANT INJECTION INDEPENDENT POWER SUPPLY POTENTIALLY INOPERABLE DUE TO EQ PROGRAM VALIDATION PROBLEM.
EVENT DATE: 112791 REPORT DATE: 122791 NSSS: GE TYPE: BWR

(NSIC 223755) ON 11/27/91, WITH THE PLANT OPERATING AT 100% POWER, IT WAS DECIDED IN A MANAGEMENT MEETING THAT AN ENVIRONMENTAL QUALIFICATION PROBLEM WITH THE TWO

INDEPENDENT POWER SUPPLIES (ED) FOR THE LOW PRESSURE COOLANT INJECTION (LPCI) (BO) VALVES HAD THE POTENTIAL FOR PLACING THE PLANT IN A CONDITION THAT IS OUTSIDE THE DESIGN BASIS FOR THE PLANT IN THE EVENT OF A LOCA. IT WAS DETERMINED THAT THE INDEPENDENT POWER SUPPLIES, UNDER NORMAL PLANT OPERATING CONDITIONS, WERE OPERATING WITHIN THEIR DESIGN SPECIFICATIONS AND WERE OPERABLE PER THE JAF TECHNICAL SPECIFICATIONS. DUE TO A CONCURRENT PROBLEM WITH THE CORE SPRAY, CONTAINMENT ISOLATION VALVES, THE PLANT WAS SHUT DOWN AND COOLED DOWN (SEE LER 91-026). THE PROBLEM WITH THE INDEPENDENT POWER SUPPLIES WILL BE RESOLVED WITH A PLANT MODIFICATION WHICH WILL BE INSTALLED PRIOR TO RESTART OF THE PLANT.

[124] FITZPATRICK DOCKET 50-333 LER 91-028
REACTOR SHUTDOWN AND COOLDOWN NOT COMPLETE WITHIN TIME REQUIRED DUE TO PERSONNEL ERROR AND PROCEDURE DEFICIENCY.
EVENT DATE: 112891 REPORT DATE: 123091 NSSS: GE TYPE: BWR

(NSIC 223729) ON NOVEMBER 28, 1991 WHILE CONDUCTING A REACTOR SHUTDOWN AND COOLDOWN TO MEET THE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.7.D.3 AS A RESULT OF INOPERABLE PRIMARY CONTAINMENT (NH) ISOLATION VALVES (SEE LER-91-026), THE REACTOR TEMPERATURE WAS NOT REDUCED TO LESS THAN 212F WITHIN 24 HOURS AS REQUIRED. COOLDOWN TO LESS THAT 212F WAS ACHIEVED IN 25 HOURS AND 6 MINUTES. THE EVENT WAS CAUSED BY PERSONNEL ERRORS AND PROCEDURE DEFICIENCIES. OPERATING SHIFT RESOURCE WERE DIVERTED TO TASKS OTHER THAN MEETING THE TECHNICAL SPECIFICATION REQUIREMENT AND THE SHUTDOWN/COOLDOWN PROCESS WAS DELAYED DURING PERFORMANCE OF SURVEILLANCE THAT COULD HAVE BEEN DEFERRED UNTIL AFTER THE TECHNICAL SPECIFICATION ACTIONS WERE MET. CORRECTIVE ACTION INCLUDES PROCEDURE CHANGES AND TRAINING TO REDUCE THE PROBABILITY OF RECURRENCE. NO SIMILAR LERS HAVE BEEN WRITTEN AT THIS FACILITY.

[125] FITZPATRICK DOCKET 50-333 LER 91-029
SPURIOUS ACTIVATION OF PRIMARY CONTAINMENT (PC) VENT AND PURGE ISOLATION SYSTEM DUE TO ELECTRIC-MAGNETIC NOISE INITIATION OF PC HIGH RANGE RADIATION MONITORS.
EVENT DATE: 113091 REPORT DATE: 010292 NSSS: GE TYPE: BWR

(NSIC 223730) FALSE PRIMARY CONTAINMENT (NH) HIGH RADIATION ISOLATION SYSTEM (JM) SIGNALS OCCURRED AT 1431 AND 1938 ON NOVEMBER 30, 1991 AND AGAIN AT 1114 ON DECEMBER 13, 1991, WHILE THE REACTOR WAS SHUTDOWN IN THE COLD CONDITION. THE 12 PRIMARY CONTAINMENT ISOLATION VALVES WHICH ARE ACTIVATED BY THESE SIGNALS WERE ALREADY CLOSED OR WENT CLOSED IN RESPONSE. BEFORE 2 OF THE ACTUATIONS A TWO-WAY RADIO WAS BEING KEYED IN THE VICINITY OF THE RADIATION MONITORS AND AT THE TIME OF THE THIRD, A CABLE PENETRATION FIRE SEAL INSPECTION WAS BEING CONDUCTED WITHIN THE MONITOR CABINET AND A TECHNICAL SPECIFICATION SURVEILLANCE TEST WAS BEING PERFORMED ON MAIN STEAM LINE RADIATION MONITORS LOCATED IN ADJACENT CABINETS. THE PATHWAYS FOR INITIATION OF THE MONITOR TRIPS MAY BE THROUGH A FILTER ASSEMBLY. THESE WERE FOUND ATTACHED TO THE INPUT OF THE MONITORS DURING A ROOT CAUSE INVESTIGATION. THE FILTERS WILL BE REMOVED FROM THE MONITOR INPUTS BEFORE THE PLANT IS RESTARTED IN MARCH 1992. THE ROOT CAUSE INVESTIGATION AND ANALYSIS WILL CONTINUE DURING THE 1992 REFUELING OUTAGE. A SUPPLEMENTAL REPORT WILL BE SUBMITTED FOLLOWING COMPLETION OF THE ANALYSIS. RELATED LER: 90-028, 91-001, 91-018, 91-022, 91-030.

[126] FITZPATRICK DOCKET 50-333 LER 91-030
UNDOCUMENTED ASSEMBLY IN PRIMARY CONTAINMENT HIGH RADIATION MONITOR.
EVENT DATE: 120391 REPORT DATE: 010292 NSSS: GE TYPE: BWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 223847) DURING THE WEEK OF 12/9/91, THE HIGH RANGE PRIMARY CONTAINMENT RADIATION MONITORS (HRCMS) (IL) INSTALLATION AND APPLICATION WAS BEING EVALUATED TO DETERMINE THE REASON FOR THE HISTORY OF ABNORMALLY HIGH SUSCEPTIBILITY TO ELECTROMAGNETIC INTERFERENCE (EMI) (SEE LER 91-001, 91-018, 91-022, AND 91-029). DURING THE COURSE OF THIS EVALUATION, AN UNDOCUMENTED ASSEMBLY WAS FOUND IN THE SIGNAL INPUT PATH TO EACH OF THE TWO RADIATION MONITORS. THE VENDOR FOR THE RADIATION MONITORS WAS CONTACTED TO DETERMINE THE FUNCTION OF THIS ASSEMBLY. THE VENDOR INDICATED THAT THIS ASSEMBLY MAY BE DIAGNOSTIC TEST EQUIPMENT AND THAT THE

ASSEMBLY MAY HAVE AN ADVERSE EFFECT ON THE PERFORMANCE OF THE RADIATION MONITORS UNDER HIGH PRIMARY CONTAINMENT (WH) DRYWELL TEMPERATURE CONDITIONS THAT COULD EXIST DURING A LOSS OF COOLANT ACCIDENT (LOCA). A DETAILED ANALYSIS IS BEING PERFORMED TO QUANTIFY THE EFFECT OF THE UNDOCUMENTED ASSEMBLY ON PRIMARY CONTAINMENT HIGH RANGE RADIATION MONITOR PERFORMANCE. A SUPPLEMENTAL LER WILL BE SUBMITTED WHEN THIS ANALYSIS IS COMPLETE.

[127] FITZPATRICK DOCKET 50-333 LER 91-032
INTAKE DEICING HEATERS POTENTIALLY INOPERABLE DUE TO HYPOTHETICAL FIRE SCENARIO NOT PREVIOUSLY CONSIDERED.
EVENT DATE: 122691 REPORT DATE: 012792 NSSS: GE TYPE: BWR

(NSIC 223914) ON 12/26/91 WITH THE PLANT IN A COLD CONDITION AND UNDERGOING ROUTINE MAINTENANCE, A PLANT OPERATIONS REVIEW COMMITTEE MEETING WAS HELD AND DETERMINED THAT DURING A POSTULATED CONTROL ROOM FIRE, THE EXISTING DESIGN OF THE CIRCULATING WATER SYSTEM INTAKE STRUCTURE (NN) DEICING HEATERS IS OUTSIDE THE DESIGN BASES OF THE PLANT. ENGINEERING EVALUATIONS FOUND THAT THE LOSS OF THESE DEICING HEATERS DUE TO A CONTROL ROOM FIRE IS POSSIBLE AND THIS TYPE OF EVENT HAS NOT BEEN ANALYZED AS PART OF THE APPENDIX R FIRE ANALYSIS. THE DESIGN BASES OF THE DEICING HEATERS IS CURRENTLY BEING REVIEWED AND A DETERMINATION OF THE SAFETY SIGNIFICANCE OF THESE HEATERS IS UNDERWAY. THEREFORE, THE CONCLUSIONS IMPLIED IN THIS DESCRIPTION AND ANALYSES ARE PRELIMINARY AND AN UPDATED REPORT WILL BE SUBMITTED WITHIN 60 DAYS OF THE COMPLETION OF THE NECESSARY INVESTIGATION AND ANALYSES.

[128] FITZPATRICK DOCKET 50-333 LER 92-001
FIRE WATCHES MISSED DUE TO INADEQUATE TRAINING AND SUPERVISION.
EVENT DATE: 010392 REPORT DATE: 020392 NSSS: GE TYPE: BWR

(NSIC 223913) WHILE THE PLANT WAS SHUTDOWN AND IN A COLD CONDITION FOR MAINTENANCE AND REFUELING, THREE SEPARATE FIRE WATCHES WERE NOT COMPLETED AS REQUIRED BY TECHNICAL SPECIFICATION 3.12.F ON JANUARY 3, AND 11, 1992. FIRE WATCHES WERE MISSED FOR A MAXIMUM OF 1 HOUR AND WERE REESTABLISHED WITHIN THE FOLLOWING HOUR. THE EVENTS WERE CAUSED BY INADEQUATE TRAINING AND SUPERVISION OF TEMPORARY PERSONNEL. CORRECTIVE ACTIONS INCLUDE THE HIRING OF SUPERVISORS TO PROVIDE 24-HOUR-A-DAY SUPERVISION. NO PREVIOUS LERS WHICH INVOLVE INADEQUATE TRAINING AND SUPERVISION RESULTING IN A FAILURE TO MEET TECHNICAL SPECIFICATION FIRE WATCH REQUIREMENTS HAVE BEEN SUBMITTED BY THIS FACILITY.

[129] FITZPATRICK DOCKET 50-333 LER 92-002
INOPERABLE MOTOR OPERATED VALVES IN MULTIPLE SYSTEMS DUE TO VARIOUS DEFICIENCIES FOUND DURING GENERIC LETTER 89-10 TESTING.
EVENT DATE: 010792 REPORT DATE: 020492 NSSS: GE TYPE: BWR

(NSIC 223912) THE PLANT WAS SHUTDOWN AND IN THE COLD CONDITION FOR MAINTENANCE AND REFUELING. PLANNED TESTING OF MOTOR OPERATOR VALVES TO MEET GENERIC LETTER 89-10 REQUIREMENTS WAS STARTED AND IS EXPECTED TO CONTINUE FOR AT LEAST TWO MONTHS. ON JANUARY 7, 1992 CALCULATIONS (PRIOR TO TEST) INDICATED THE VALVE OPERATOR DESIGN ON 2 VALVES WOULD NOT PROVIDE ADEQUATE VALVE STEM THRUST FOR VALVE CLOSURE UNDER EVALUATED WORST CASE CONDITIONS. ON JANUARY 15, 17, 18, 22, AND 24, 1992 FIVE ADDITIONAL VALVE OPERATORS WERE FOUND TO BE INCAPABLE OF PRODUCING ADEQUATE THRUST DURING "ASFOUND" TESTING PRIOR TO OVERHAUL OF THE OPERATOR. CAUSES AND CORRECTIVE ACTION HAVE NOT YET BEEN DETERMINED AND WILL BE PROVIDED IN AN UPDATED REPORT ALONG WITH ADDITIONAL DEFICIENCIES FOUND (IF ANY) WITHIN 90 DAYS OF COMPLETION OF THE TESTING AND OPERATOR OVERHAUL.

[130] FITZPATRICK DOCKET 50-333 LER 92-003
SAFETY-RELATED CONTAINMENT ISOLATION & EMERGENCY CORE COOLING VALVES POTENTIALLY OR ADMINISTRATIVELY INOPERABLE DUE TO POTENTIAL OR ACTUAL INSTALLATION OF WRONG PARTS.
EVENT DATE: 010892 REPORT DATE: 020792 NSSS: GE TYPE: BWR
VENDOR: LIMITORQUE CORP.

(NSIC 224038) THE PLANT WAS SHUTDOWN AND IN THE COLD CONDITION FOR MAINTENANCE AND REFUEL. ON JANUARY 8, 1992 IT WAS DETERMINED THAT AN INCORRECT MOTOR PINION KEY HAD BEEN INSTALLED IN CORE SPRAY SYSTEM (BM) INJECTION AND PRIMARY CONTAINMENT (NH) ISOLATION VALVES 14MOV-12A&B FOR A PERIOD OF 5 TO 6 MONTHS BETWEEN JULY 1991 AND LATE DECEMBER 1991 OR EARLY JANUARY 1992. THE WRONG KEYS WERE OBTAINED AS A RESULT OF WRONG INFORMATION PROVIDED BY THE VALVE OPERATOR VENDOR. THE VALVES FUNCTIONED NORMALLY DURING MONTHLY TESTING TO MEET TECHNICAL SPECIFICATION AND INSERVICE TEST PROGRAM REQUIREMENTS. CORE SPRAY SYSTEM LOOP A VALVE 14MOV-12A VALVE OPERATOR WAS REPAIRED BY REPLACING THE MOTOR PINION GEAR KEY AND WAS RETURNED TO SERVICE ON DECEMBER 30, 1991 IN THE STANDBY MODE OF OPERATION FOLLOWING POST-WORK TESTING. CORE SPRAY LOOP B INJECTION VALVE 14MOV-12B VALVE OPERATOR IS SCHEDULED FOR REPAIR LATER DURING THE REFUEL OUTAGE. AN UPDATED REPORT WILL BE SUBMITTED AFTER THE VALVE OPERATOR VENDOR PERFORMS A ROOT CAUSE ANALYSIS OF HOW THE WRONG PART NUMBER WAS SUPPLIED.

[131] FT. CALHOUN 1 DOCKET 50-285 LER 91-025
SAFETY INJECTION PIPE SUPPORTS OUTSIDE DESIGN BASIS.
EVENT DATE: 111491 REPORT DATE: 121691 NSSS: CE TYPE: PWR

(NSIC 223634) WHILE FORT CALHOUN STATION WAS OPERATING AT 100 PERCENT POWER (MODE 1), IT WAS DETERMINED THAT THE UPSET AND FAULTED LOADINGS ON TWO SAFETY INJECTION SYSTEM PIPE SUPPORTS EXCEEDED THE DESIGN CAPACITY OF THE EMBEDDED UNISTRUT TO WHICH THEY ARE ATTACHED. THIS IS A CONDITION OUTSIDE THE PLANT DESIGN BASIS AND WAS REPORTED ON NOVEMBER 14, 1991, PURSUANT TO 10 CFR 50.72(B)(1)(II)(B). THIS REPORT IS BEING SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 50.72(A)(2)(II)(B). THE PRIMARY CAUSE OF THIS EVENT IS ATTRIBUTED TO A DESIGN ANALYSIS DEFICIENCY (I.E., INADEQUATE CONSIDERATION OF ZERO PERIOD ACCELERATION (ZPA) LOADINGS IN AN ANALYSIS PERFORMED BY A CONSULTANT). A CONTRIBUTING FACTOR WAS THE LACK OF EXPERIENCED PERSONNEL AT OMAHA PUBLIC POWER DISTRICT (OPPD) TO REVIEW THE CONSULTANT'S WORK. OPPD HAS SINCE DEVELOPED IN-HOUSE EXPERTISE IN THE AREA OF SEISMIC ANALYSIS AND HAS PURCHASED A COMPUTER PROGRAM WHICH ACCOUNTS FOR ZPA. CORRECTIVE ACTION INVOLVES MODIFYING THESE PIPING SUPPORTS TO MEET DESIGN BASIS REQUIREMENTS DURING THE 1992 REFUELING OUTAGE. ALTHOUGH THE LOADS ON THE UNISTRUT EXCEEDED THE DESIGN BASIS, THESE PIPE SUPPORTS ARE CONSIDERED OPERABLE BASED ON INTERIM OPERABILITY CRITERIA PREVIOUSLY PROVIDED TO THE NRC.

[132] FT. CALHOUN 1 DOCKET 50-285 LER 91-027 REV 01
UPDATE ON VIOLATION OF CONTAINMENT INTEGRITY BY OPENING WD-1060 DURING SAMPLING.
EVENT DATE: 111891 REPORT DATE: 013192 NSSS: CE TYPE: PWR

(NSIC 223864) WHILE OPERATING AT 100 PERCENT POWER, REACTOR COOLANT DRAIN TANK (RCDT) PUMP DISCHARGE TEST VALVE WD-1060 WAS USED TO OBTAIN 20 RCDT SAMPLES BETWEEN OCTOBER 16, 1991, AND NOVEMBER 18, 1991, DURING THE INVESTIGATION OF ABNORMAL INCREASES IN TANK LEVEL. WD-1060 IS A 3/8 INCH SEAL WIRED CLOSED CONTAINMENT ISOLATION VALVE (CIV) WHICH TAPS OFF THE RCDT PUMP DISCHARGE HEADER BETWEEN CIVS HCV-500A AND HCV-500B. OPENING OF WD-1060 VIOLATED CONTAINMENT INTEGRITY AS REQUIRED BY TECHNICAL SPECIFICATION 2.6.(1). THIS REPORT IS BEING SUBMITTED PURSUANT TO 10 CFR 50.73(A)(2)(I)(B). THE ROOT CAUSE OF THIS EVENT WAS THE LACK OF FORMALITY IN ESTABLISHING THE RCDT SAMPLING PROGRAM FOR THE RCDT INCREASED LEAKAGE INVESTIGATION. CONTRIBUTING CAUSES INCLUDE: NO APPROVED PROCEDURE FOR NON-ROUTINE SAMPLING, LACK OF UNDERSTANDING/TRAINING RELATED TO OPENING SEAL WIRED VALVES, AND NO LABELING OF SEAL WIRES. ALTHOUGH CONTAINMENT INTEGRITY WAS VIOLATED, THIS EVENT DID NOT ENDANGER THE HEALTH AND SAFETY OF THE PUBLIC BECAUSE WD-1060 WAS CLOSED AFTER THE SAMPLE WAS OBTAINED; DURING THE VERY SHORT PERIOD FOR SAMPLING, THE SHIFT CHEMIST MAINTAINED CONTROL OF WD-1060; THE GENERAL DESIGN CRITERION WAS MET BY CIV HCV-500B; HCV-500 A/B HAVE BEEN SUCCESSFULLY QUARTERLY STROKE TESTED; AND, THE 1990 REFUELING OUTAGE LOCAL LEAK RATE TEST RESULTS SHOW THAT THERE IS NO LEAKAGE THROUGH THESE VALVES.

[133] FT. CALHOUN 1 DOCKET 50-285 LER 91-028
UNMONITORED RELEASE ON LOSS OF 161KV SYSTEM.
EVENT DATE: 120191 REPORT DATE: 123191 NSSS: CE TYPE: PWR

(NSIC 223767) AT 0205 ON DECEMBER 1, 1991, IT WAS DISCOVERED THAT THE SAMPLE PUMP FOR THE LABORATORY AND RADIOACTIVE WASTE PROCESSING BUILDING (LRWPB) EXHAUST STACK GAS, IODINE AND PARTICULATE MONITORS WAS NOT RUNNING. AT THE TIME OF THE DISCOVERY, THE CHEMISTRY AND RADIATION PROTECTION (CARP) BUILDING EXHAUST FAN WAS RUNNING. OPERATION OF THE EXHAUST FAN WHILE THE SAMPLE PUMP WAS NOT RUNNING WAS DETERMINED TO CONSTITUTE AN UNMONITORED RELEASE IN VIOLATION OF PLANT TECHNICAL SPECIFICATION 2.9.1(2)(H)(I), AND IS REPORTABLE UNDER 10 CFR 50.73(A)(2)(I)(B). THE EVENT IS CONSIDERED TO HAVE BEEN CAUSED BY A POWER EXCURSION THE PREVIOUS DAY DUE TO SEVERE WINTER WEATHER. 161KV OFFSITE ELECTRICAL POWER WAS LOST AND SUBSEQUENTLY RESTORED TWICE ON NOVEMBER 30, 1991. IT IS ASSUMED THAT WHEN THE INITIAL LOSS OF 161KV POWER OCCURRED AT 0430, THE LRWPB STACK RADIATION MONITORS SAMPLE PUMP STOPPED, AND DID NOT RESTART WHEN POWER WAS RESTORED. THE SAMPLE PUMP IS DESIGNED NOT TO REENERGIZE FOLLOWING A POWER INTERRUPTION UNTIL THE START/STOP SWITCH IS MANUALLY PUSHED. THE CARP BUILDING EXHAUST FAN RESTARTED WHEN POWER WAS RESTORED, RESULTING IN THE UNMONITORED RELEASE. DURING THE TIME THAT THE SAMPLE PUMP IS ASSUMED TO HAVE BEEN DEENERGIZED, ROUTINE RELEASES OCCURRED WHICH WERE FOUND TO BE NOT SIGNIFICANT. OPPD IS EVALUATING POTENTIAL CORRECTIVE ACTIONS SUCH AS A DESIGN CHANGE OR TECHNICAL SPECIFICATION REVISION.

[134] FT. CALHOUN 1 DOCKET 50-285 LER 91-029
PERSONNEL AIR LOCK LEAK RATE TEST DEFICIENCY.
EVENT DATE: 120491 REPORT DATE: 010392 NSSS: CE TYPE: PWR

(NSIC 223707) ON DECEMBER 2, 1991, WHILE FORT CALHOUN STATION WAS OPERATING AT 100 PERCENT POWER (MODE 1), A SPECIAL SERVICES ENGINEER (SSE) WAS REVIEWING A PROCEDURE FOR CONTAINMENT LEAK RATE TESTING, WHEN IT WAS REALIZED THAT THIS PROCEDURE DID NOT ADEQUATELY TEST THE INNER PERSONNEL AIR LOCK (PAL) EQUALIZING VALVE AS REQUIRED UNDER TECHNICAL SPECIFICATION 3.5.(3)D. SPECIFICALLY, THIS SPECIFICATION REQUIRES THAT "THE ENTIRE ASSEMBLY BE TESTED TO 60 PSIG" ON A SIX MONTH BASIS. HOWEVER, BECAUSE THE PROCEDURE DID NOT ADEQUATELY TEST THE EQUALIZING VALVE, THE "ENTIRE ASSEMBLY" WAS NOT TESTED. A PROCEED WAS WRITTEN TO TEST THE EQUALIZING VALVE AND THIS TEST WAS SUCCESSFULLY CONDUCTED ON DECEMBER 7, 1991. THIS REPORT IS BEING SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(V). THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL BASED ON PREVIOUS INTEGRATED LEAK RATE TESTS WHICH HAVE PROVEN CONTAINMENT INTEGRITY, AND THE FACT THAT ONCE PROPERLY TESTED THE EQUALIZING VALVE WAS FOUND TO BE OPERABLE. THE CAUSE OF THIS EVENT WAS ATTRIBUTED TO PAST INADEQUATE PROCEDURE CHANGE REVIEWS. CHANGE WAS MADE TO THE TESTING PROCEDURE IN 1974, WHICH REMOVED THE EQUALIZING VALVE FROM THE TEST BOUNDARY. CORRECTIVE ACTIONS INCLUDED DECLARING THE EQUALIZING VALVE INOPERABLE, GENERATING AN PROCEDURE TO TEST THE EQUALIZING VALVE, TESTING THE VALVE, AND REVIEWING ALL TYPE B LEAK RATE TEST PROCEDURES ALONG WITH THE CURRENT CONFIGURATION OF TYPE B PENETRATIONS.

[135] FT. CALHOUN 1 DOCKET 50-285 LER 91-030
RADIATION MONITORS OUT OF SERVICE WITH CONTAINMENT PRESSURE REDUCTION IN PROGRESS.
EVENT DATE: 121091 REPORT DATE: 010992 NSSS: CE TYPE: PWR

(NSIC 223768) ON DECEMBER 10, 1991, FORT CALHOUN STATION WAS OPERATING AT 100 PERCENT POWER (MODE 1) WITH A CONTAINMENT PRESSURE REDUCTION (CPR) IN PROGRESS. A LICENSED OPERATOR WAS DIRECTED TO TERMINATE THE CPR EARLY SO THAT IT WOULD NOT INTERFERE WITH FILTER PLACEMENTS ON AUXILIARY BUILDING STACK RADIATION MONITORS RM-060 (IODINE), RM-061 (PARTICULATE) AND RM-062 (GAS). HOWEVER, CPR WAS NOT TERMINATED WHEN RM-060 WAS REMOVED FROM SERVICE TO REPLACE THE FILTER CARTRIDGE AS PART OF THE WEEKLY SURVEILLANCE TEST (CH-ST-VA-0001). THIS VIOLATED THE REQUIREMENTS OF TECHNICAL SPECIFICATION 2.9.1(2)G WHICH STATES THAT DURING CONTAINMENT VENTING TO THE AUXILIARY BUILDING EXHAUST STACK, THE GAS, IODINE AND PARTICULATE MONITORS SHALL BE MONITORING THE STACK. THEREFORE, THIS REPORT IS BEING SUBMITTED PURSUANT TO 10 CFR 50.73(A)(2)(I)(B). THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL DUE TO THE SHORT TIME THAT THE MONITORS REMOVED FROM SERVICE. ADDITIONALLY, CONTAINMENT RADIATION MONITORS, RM-050 AND RM-5., WERE IN SERVICE AT THE TIME OF THIS EVENT AND NEITHER MONITOR SHOWED ANY INCREASE CONTAINMENT ACTIVITY. THIS EVENT RESULTED FROM A BREAKDOWN IN VERBAL AND WRITTEN COMMUNICATION AND WORK NOTICES. CH-ST-VA-0001 DOES NOT COMPLY WITH THE EXISTING

SURVEILLANCE TEST PROGRAM IN PROPERLY NOTIFYING SHIFT SUPERVISORS PRIOR TO REMOVING RM-060 FROM SERVICE.

[136] FT. CALHOUN 1 DOCKET 50-285 LER 91-031
PERSONNEL AIR LOCK DOOR CONNECTIONS OUTSIDE DESIGN BASIS.
EVENT DATE: 121691 REPORT DATE: 011592 NSSS: CE TYPE: PWR

(NSIC 223790) ON DECEMBER 16, 1991, AT 1305 HOURS WHILE OPERATING AT 100 PERCENT POWER (MODE 1), IT WAS DETERMINED THAT THE CONNECTIONS TO THE PERSONNEL AIR LOCK (PAL) BULKHEADS FOR LEAK RATE TESTING WERE POTENTIALLY OUTSIDE THE ORIGINAL DESIGN REQUIREMENTS. THE CONNECTIONS HAD BEEN INSTALLED IN 1974 TO FACILITATE TESTING OF THE PAL AND ITS DOOR SEALS, HOWEVER, DESIGN DOCUMENTATION COULD NOT BE LOCATED TO PROVE THAT THE COMPONENTS WERE SEISMICALLY QUALIFIED AS A CONTAINMENT ISOLATION BOUNDARY. THIS REPORT IS BEING SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 50.73(A)(2)(II). THE PAL WAS DETERMINED TO BE OPERABLE WITH THE UNQUALIFIED COMPONENTS INSTALLED BASED ON PAST SEAL LEAKAGE TESTS AND BY MAINTAINING THE INNER DOOR CLOSED AND SEALED. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AT RISK. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO LACK OF PROCEDURES TO CONTROL THE PLANT CONFIGURATION CHANGE PROCESS WHEN THESE CONNECTIONS WERE INSTALLED IN 1974. A CONTRIBUTING FACTOR WAS THE LACK OF UNDERSTANDING OF THE DESIGN BASIS OF THE PAL BY THOSE INVOLVED WITH THE 1974 INSTALLATION. IMMEDIATE CORRECTIVE ACTION INCLUDED ESTABLISHING ADMINISTRATIVE CONTROL BY DANGER TAGGING THE OUTER PAL DOOR TO RESTRICT ACCESS, THUS ENSURING THAT CONTAINMENT INTEGRITY WAS MAINTAINED.

[137] FT. ST. VRAIN DOCKET 50-267 LER 92-001
ACM ACTIVITY MONITORS NOT TESTED IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS.
EVENT DATE: 010692 REPORT DATE: 020592 NSSS: GA TYPE: HTGR

(NSIC 223919) ON JANUARY 6, 1992, DURING A ROUTINE REVIEW OF TECHNICAL SPECIFICATION (TS) PROCEDURE ESR-8.1.1A-Q "RADIOACTIVE GASEOUS EFFLUENT SYSTEM FUNCTIONAL TEST" IT WAS DISCOVERED THAT THE ALTERNATE COOLING METHOD (ACM) EXHAUST STACK ACTIVITY MONITORS RT-4801, RT-4802, AND RT-4803 WERE NOT INCLUDED IN THE TEST PROCEDURE. RT-4801, RT-4802, AND RT-4803 ARE PAN OF A SYSTEM OF REDUNDANT ACTIVITY MONITORS THAT SAMPLE AND MONITOR THE GASEOUS EFFLUENT DISCHARGED OUT THE REACTOR BUILDING EXHAUST STACK. TS SECTION ESR-8.1.1 REQUIRES THAT THE EXHAUST STACK MONITORS BE FUNCTIONALLY TESTED QUARTERLY. FAILURE TO FUNCTIONALLY TEST THE ACM EXHAUST STACK ACTIVITY MONITORS CONSTITUTES A VIOLATION OF TS REQUIREMENTS. TS AMENDMENT NO. 71 RECEIVED AUGUST 21, 1989, ALLOWED USE OF RT-4801, RT-4802, AND RT-4803 FOR FULFILLING THE TS REQUIREMENTS TO MONITOR THE REACTOR BUILDING EXHAUST EFFLUENT FOR PARTICULATES, HALOGENS, AND NOBLE GASES. FOLLOWING RECEIPT OF TS AMENDMENT NO. 71, PSC TOOK ACTION TO PERFORM THE VARIOUS DOCUMENT UPDATES REQUIRED TO FULLY IMPLEMENT THE AMENDMENT. HOWEVER, DUE TO AN APPARENT OVERSIGHT THE THREE ACM EXHAUST STACK ACTIVITY MONITORS WERE NOT ADDED TO THE QUARTERLY FUNCTIONAL TEST PROCEDURE, ESR-8.1.1A-Q. TS PROCEDURE ESR-8.1.1A-Q HAS BEEN REVISED TO INCLUDE THE THREE ACM EXHAUST STACK ACTIVITY MONITORS.

[138] GINNA DOCKET 50-244 LER 91-009
AUTOMATIC FEEDWATER CONTROL PERTURBATIONS, DUE TO ELECTROMAGNETIC NOISE SPIKES FROM UNRELATED RELAY ACTUATION, CAUSED STEAM GENERATOR FEEDWATER ISOLATION ON HIGH LEVEL.
EVENT DATE: 111191 REPORT DATE: 121191 NSSS: WE TYPE: PWR

(NSIC 223657) ON NOVEMBER 11, 1991 AT APPROXIMATELY 1214 EST, WITH THE REACTOR AT APPROXIMATELY 98% FULL POWER, STEAM GENERATOR FEEDWATER ISOLATIONS OCCURRED ON BOTH STEAM GENERATORS. THESE FEEDWATER ISOLATIONS WERE CAUSED BY PERTURBATIONS OF THE ADVANCED DIGITAL FEEDWATER CONTROL SYSTEM WHICH INCREASED FEEDWATER FLOW TO THE STEAM GENERATORS. IMMEDIATE OPERATOR ACTION WAS TO MANUALLY CONTROL THE FEEDWATER REGULATING VALVES TO REDUCE STEAM GENERATOR LEVELS AND STABILIZE THE PLANT. THE UNDERLYING CAUSE OF THE EVENT WAS DETERMINED TO BE ELECTROMAGNETIC NOISE SPIKES AFFECTING THE ADVANCED DIGITAL FEEDWATER CONTROL SYSTEM. CORRECTIVE ACTION TAKEN WAS TO MODIFY SPECIFIC RELAY CIRCUITS THAT WERE CAUSING THESE SPIKES.

[139] GINNA DOCKET 50-244 LER 91-010
DURING OPERATOR REVIEW OF PLANT PROCESS COMPUTER LOGS, IT WAS DISCOVERED THAT INVALID DATA HAD BEEN USED FOR CALORIMETRIC CALCULATION.
EVENT DATE: 123091 REPORT DATE: 012992 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 223926) ON DECEMBER 30, 1991 AT APPROXIMATELY 1830 EST, WITH THE REACTOR AT APPROXIMATELY 98% FULL POWER, THE CONTROL ROOM FOREMAN, WHILE PERFORMING A REVIEW OF THE HOURLY PLANT PROCESS COMPUTER SYSTEM (PPCS) LOGS, DETERMINED THAT THE LEADING EDGE FLOW METER (LEFM) DATA USED BY THE PPCS TO CALCULATE CALORIMETRICS, WAS INVALID FROM 1800 EST, DECEMBER 20, 1991 TO APPROXIMATELY 1830 EST, DECEMBER 30, 1991. THIS EVENT IS BEING VOLUNTARILY REPORTED TO ALERT OTHER UTILITIES WHICH MAY HAVE SIMILAR SYSTEMS. THE CONTROL ROOM OPERATORS NOTIFIED THE REACTOR ENGINEER AND REINITIALIZED THE LEFM COMPUTER. AFTER REINITIALIZATION, THE LEFM PRINTOUT VALUES AND PPCS VALUES RETURNED TO EXPECTED/NORMAL INDICATIONS. CORRECTIVE ACTIONS INCLUDE REVISIONS TO THE AFFECTED PROCEDURES TO REQUIRE CHECKING THE OPERATION OF THE LEFM PRIOR TO PERFORMING CALORIMETRICS AND THE ADDITION OF A HIGH ALARM ON PPCS INDICATING A FAILURE MAY HAVE OCCURRED. THE IMMEDIATE CAUSE WAS IDENTIFIED AS A FAILURE OF THE LEFM HP-85B COMPUTER.

[140] GINNA DOCKET 50-244 LER 92-001
FAILURE OF CONTAINMENT RADIATION MONITOR DUE TO UNKNOWN CAUSE CAUSES CONTAINMENT VENTILATION ISOLATION.
EVENT DATE: 010592 REPORT DATE: 020492 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INC

(NSIC 223927) ON JANUARY 5, 1992 AT APPROXIMATELY 0240 EST, WITH THE REACTOR AT APPROXIMATELY 98% FULL POWER, A CONTAINMENT VENTILATION ISOLATION OCCURRED DUE TO AN ACTUATION SIGNAL FROM THE CONTAINMENT PARTICULATE RADIATION MONITOR (R-11). ALL CONTAINMENT ISOLATION VALVES THAT WERE OPEN, CLOSED AS DESIGNED. IMMEDIATE OPERATOR ACTION WAS TO PERFORM THE APPLICABLE ALARM RESPONSE PROCEDURES ACTIONS. THIS INCLUDED VERIFYING AUTOMATIC ACTIONS, DETERMINING THE CAUSE OF THE CONTAINMENT VENTILATION ISOLATION, AND MAKING APPROPRIATE NOTIFICATIONS. THE IMMEDIATE CAUSE OF THE EVENT WAS DETERMINED TO BE THE FAILURE OF R-11. CORRECTIVE ACTION TAKEN WAS TO RETURN THE CONTAINMENT VENTILATION ISOLATION SYSTEM TO PRE-EVENT NORMAL STATUS, SEQUENTIALLY FOLLOWED BY A TROUBLESHOOTING EFFORT BY THE INSTRUMENT AND CONTROL DEPARTMENT, AND THEN CHANGEOUT OF THE R-11 DRAWER WITH A QUALIFIED SPARE. FURTHER INVESTIGATION TO DETERMINE THE ROOT CAUSE IS CONTINUING.

[141] GRAND GULF 1 DOCKET 50-416 LER 91-012
LIGHTNING INDUCED SCRAM.
EVENT DATE: 111991 REPORT DATE: 121891 NSSS: GE TYPE: BWR

(NSIC 223738) ON NOVEMBER 19, 1991, A THUNDERSTORM WITH SEVERE LIGHTNING OCCURRED IN THE SITE VICINITY. AT APPROXIMATELY 2049, DURING THE STORM, THE DIVISION 2 AVERAGE POWER RANGE MONITORS (APRM) (IG) RECEIVED A HIGH NEUTRON FLUX SIGNAL RESULTING IN A TRIP ON THE DIVISION 2 REACTOR PROTECTION SYSTEM (RPS). PRIOR TO THE TRIP OCCURRING ON THE DIVISION 2 RPS, A MAINTENANCE SURVEILLANCE WAS IN PROGRESS WHICH CAUSED THE DIVISION 1 RPS TO BE IN THE TRIPPED CONDITION. BOTH SYSTEMS BEING IN THE TRIPPED CONDITION, RESULTED IN AN AUTOMATIC REACTOR SCRAM. FOLLOWING THE SCRAM, VESSEL WATER LEVEL DECREASED TO -14 INCHES. WATER LEVEL WAS RESTORED BY THE CONDENSATE AND FEEDWATER SYSTEMS. BASED ON A REVIEW OF COMPUTER DATA, IT APPEARS THAT THE OUTPUT OF A FIVE VOLT POWER SUPPLY ASSOCIATED WITH THE APRMS MAY HAVE BEEN DEENERGIZED, THEN REENERGIZED. THIS WOULD HAVE RESULTED IN AN APRM SCRAM. A SUPPLEMENTAL REPORT WILL BE SUBMITTED FOLLOWING THE DETERMINATION OF THE ROOT CAUSE OF THE LIGHTNING INDUCED SCRAMS EXPERIENCED AT GGNS.

[142] GRAND GULF 1 DOCKET 50-416 LER 91-013
RAINWATER ENTERS SECONDARY CONTAINMENT VIA ELECTRICAL CONDUITS.
EVENT DATE: 111991 REPORT DATE: 122091 NSSS: GE TYPE: BWR

(NSIC 223756) A CONSTRUCTION DEFICIENCY WAS DISCOVERED TO HAVE PROVIDED A FLOW

PATH FOR RAINWATER TO ENTER SECONDARY CONTAINMENT (NG) VIA ELECTRICAL CONDUITS. THE DEFICIENCY CONSISTED OF IMPROPERLY INSTALLED CONDUIT SEALS. THE EVENT RESULTED FROM RAINWATER COLLECTING IN THE UNDERGROUND DUCT BANKS IN THE VICINITY OF THE AUXILIARY BUIDLING. CONTRIBUTING FACTORS INCLUDED EXCEEDING CAPACITY OF SUMP PUMPS AND THE COVER BEING OFF A MANHOLE TO THESE BANKS. THE INFLUX OF RAINWATER EXCEEDED THE SUMP PUMP CAPACITY AND PERMITTED THE WATER LEVEL IN THE FIRST MANHOLE TO INCREASE ABOVE THE PUMP'S POWER RECEPTACLE, THEREBY TRIPPING ITS GROUND FAULT CIRCUIT INTERRUPTOR. THE FLOOR DRAIN SYSTEM (WK) CONTROLLED THE INFLUX OF WATER TO PREVENT FLOODING OF THE IMMEDIATE AND SURROUNDING AREAS. WATER ENTERED AN ADJACENT SWITCHGEAR ROOM BUT DID NOT INTRUDE INTO ANY ELECTRICAL EQUIPMENT OR OTHER SAFETY RELATED EQUIPMENT IN THE AUXILIARY BUILDING DUE TO THIS CONDITION. INSPECTION OF CONDUITS WHICH PENETRATE THE BOUNDARY OF SECONDARY CONTAINMENT AND CONTROL BUILDING (NA) WAS PERFORMED TO DETERMINE THE EXTENT OF THE DEFICIENCY. ADDITIONAL CONDUITS WERE FOUND TO HAVE NONCONFORMING SEALS. ALL IDENTIFIED CONDUITS WERE SEALED IN ACCORDANCE WITH M-0800D.

[143] GRAND GULF 1 DOCKET 50-416 LER 91-014
VOLUNTARY REPORT ON DRYWELL AIRLOCK PRESSURIZATION.
EVENT DATE: 111991 REPORT DATE: 122091 NSSS: GE TYPE: BWR

(NSIC 223739) ON NOVEMBER 19, 1991 FOLLOWING AN AUTOMATIC SCRAM, PLANT PERSONNEL ATTEMPTED TO ENTER THE DRYWELL (D/W) PERSONNEL AIRLOCK. ENTRY WAS PREVENTED BY A DIFFERENTIAL PRESSURE ACROSS THE OUTER DOOR. THE OPENING OF THE DOOR WAS PREVENTED BY A SAFETY INTERLOCK. USING CALIBRATED TEST EQUIPMENT, THE INTERNAL PRESSURE OF THE AIRLOCK WAS DETERMINED TO BE APPROXIMATELY 40 PSIG, WHICH IS GREATER THAN THE AIRLOCK'S DESIGN PRESSURE (30 PSIG). FOLLOWING THE DEPRESSURIZATION OF THE AIRLOCK, PERSONNEL WERE ABLE TO ATTAIN ENTRY. UPON INVESTIGATING THE CAUSE OF THE OVERPRESSURIZATION, PERSONNEL DISCOVERED A PLUG INSTALLED IN THE INLET PORT OF THE OUTER EQUALIZING VALVE (EV). A MATERIAL NONCONFORMANCE REPORT WAS WRITTEN TO DOCUMENT THE CONDITION AND INITIATE AN EVALUATION OF THE AIRLOCK. THE EV APPEARS TO HAVE BEEN PLUGGED SINCE CONSTRUCTION. THE PLANT SURVEILLANCE PROCEDURE WHICH GOVERNS LEAK RATE TESTING WILL BE REVISED TO INCLUDE PROVISIONS FOR PLUGGING THE D/W AIRLOCK RELIEF VALVE FOR LEAK RATE TESTING THE AIRLOCK BARREL, AND REMOVING THE PLUG AFTERWARDS. THE WEEKLY OPERATIONS SURVEILLANCE PROCEDURE HAS BEEN REVISED TO REQUIRE INTERNAL BARREL PRESSURE READINGS.

[144] GRAND GULF 1 DOCKET 50-416 LER 91-015
AUTOMATIC ISOLATION OF REACTOR WATER CLEANUP SYSTEM.
EVENT DATE: 112391 REPORT DATE: 121891 NSSS: GE TYPE: BWR
VENDOR: RILEY COMPANY, THE - PANALARM DIVISION

(NSIC 223740) AUTOMATIC ACTUATION OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM OUTBOARD CONTAINMENT ISOLATION VALVES OCCURRED ON NOVEMBER 23, 1991 AT APPROXIMATELY 2312 HOURS. THE RWCU SYSTEM HEAT EXCHANGER ROOM HIGH TEMPERATURE ANNUNCIATOR ALARMED THREE TIMES JUST PRIOR TO THE ISOLATION. NO LEAKAGE OF THE RWCU SYSTEM WAS OBSERVED. NO PLANT CONDITION OR SYSTEM TRANSIENT WHICH PRECEDED OR COINCIDED WITH THE ISOLATION WAS DETERMINED TO HAVE INITIATED THE EVENT. THE RWCU LEAK DETECTION SYSTEM (LDS) CONSISTS OF SENSITIVE RILEY PANALARM TEMPERATURE SWITCHES. THE RWCU LEAK DETECTION LOGIC ACTUATES ON A SINGLE CHANNEL TRIP SIGNAL. IT IS BELIEVED THAT THIS ISOLATION WAS ACTUATED BY A TRIP SIGNAL FROM LDS. PREVIOUS CORRECTIVE ACTIONS TO LDS SYSTEM HAVE REDUCED SUCH TYPE OCCURRENCES. THE SUSPECT TEMPERATURE SWITCHES WERE REPLACED FOLLOWING THIS EVENT. THE ACTUATION OF THE RWCU ISOLATION SYSTEM DID NOT COMPROMISE THE SAFE OPERATION OF GGNS. ALL SAFETY RELATED EQUIPMENT OPERATED AS DESIGNED. THE SAFETY AND HEALTH OF THE GENERAL PUBLIC WAS NOT AFFECTED BY THIS EVENT.

[145] GRAND GULF 1 DOCKET 50-416 LER 91-016
VOLUNTARY REPORT ON RE-CRITICALITY.
EVENT DATE: 123091 REPORT DATE: 021392 NSSS: GE TYPE: BWR

(NSIC 224044) ON DECEMBER 30, 1991 DURING A PLANT SHUTDOWN TO PERFORM MAINTENANCE ON THE 'B' REACTOR RECIRCULATION WATER PUMP, OPERATORS SUSPENDED CONTROL ROD

INSERTION TO PERFORM A CHANNEL FUNCTIONAL TEST ON THE SOURCE RANGE MONITORS (SRMS) AS REQUIRED BY GGNS TECHNICAL SPECIFICATIONS. REACTOR POWER DECREASED TO INTERMEDIATE RANGE MONITOR (IRM) RANGE 1 PRIOR TO THE COMPLETION OF THE TEST. DURING THIS PERIOD, REACTOR POWER BEGAN TO INCREASE DUE TO MODERATOR COOLDOWN. THE POWER INCREASE WAS SELF-LIMITING. THE IRMS INCREASED UNTIL THEY REACHED RANGES 7 AND 8; THEN POWER STABILIZED. AFTER POWER STABILIZED, THE SRM FUNCTIONAL TEST WAS COMPLETED AND THE SHUTDOWN WAS COMPLETED IN ACCORDANCE WITH PLANT PROCEDURES. THE CAUSE OF THE INCREASE IN REACTOR POWER IS ATTRIBUTED TO A DECREASE IN MODERATOR TEMPERATURE WHICH RESULTED IN A POSITIVE REACTIVITY ADDITION. OPERATIONS PERSONNEL UNDERSTOOD, ANTICIPATED AND CONTROLLED THE EVENT IN ACCORDANCE WITH PLANT PROCEDURES. THE OCCURRENCE POSED NO ADVERSE EFFECTS ON PLANT SAFETY OR THE ABILITY OF OPERABLE PLANT SAFETY SYSTEMS TO PERFORM THEIR INTENDED FUNCTIONS. THIS EVENT IS BEING REPORTED AS A VOLUNTARY REPORT.

[146] HATCH 1 DOCKET 50-321 LER 91-014 REV 01
UPDATE ON COMPONENT FAILURE RESULTS IN ENGINEERED SAFETY FEATURE ACTUATIONS.
EVENT DATE: 081091 REPORT DATE: 121791 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 2 (BWR)
VENDOR: ASCO VALVES

(NSIC 223676) ON 8/10/91, AT 1011 CDT, UNIT 1 WAS IN THE HOT SHUTDOWN MODE WITH THE REACTOR VESSEL AT 500 PSIG AND REACTOR COOLANT TEMPERATURE AT 470 DEGREES FAHRENHEIT. UNIT 2 WAS IN THE RUN MODE AT A POWER LEVEL OF 100% RATED THERMAL POWER. AT THAT TIME, THE OUTPUT BREAKERS FOR THE NORMAL POWER SUPPLY TO THE UNIT 1 REACTOR PROTECTION SYSTEM (RPS) BUS "B" TRIPPED WHEN AN OVERVOLTAGE RELAY FAILED. THIS CAUSED A LOSS OF POWER TO THE "B" CHANNEL OF THE RPS, THE PROCESS RADIATION MONITORS, THE NEUTRON MONITORING SYSTEM (NMS), THE PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS), AND THE OFFGAS RADIATION MONITORING SYSTEM. THE "FAIL-SAFE" DESIGN OF THESE SYSTEMS RESULTED IN THEIR ASSUMING THE TRIPPED STATE WHEN POWER WAS INTERRUPTED. THESE TRIPS RESULTED IN A HALF SCRAM SIGNAL ON UNIT 1 RPS CHANNEL "B", CLOSURE OF VARIOUS UNIT 1 PCIS VALVES AND INITIATION OF THE "B" TRAINS OF BOTH UNITS' STANDBY GAS TREATMENT SYSTEMS (SGTS). TWO UNIT 1 SGTS DAMPERS WERE SUBSEQUENTLY FOUND TO HAVE INCORRECTLY RETURNED TO THEIR STANDBY CONFIGURATION FOLLOWING THE RESTORATION OF POWER AND THE RESET OF THE ISOLATION LOGIC. THE CAUSE OF THE LOSS OF POWER TO THE RPS BUS WAS COMPONENT FAILURE. THE CAUSE OF THE DAMPERS REPOSITIONING AFTER RESET OF THE ISOLATION LOGIC WAS A DISCREPANCY IN THE INSTALLATION OF THE DAMPER WIRING AND LESS THAN ADEQUATE VERIFICATION TESTING IN RESPONSE TO IEB 80-06.

[147] HATCH 1 DOCKET 50-321 LER 91-028
PERSONNEL ERROR RESULTS IN AN UNPLANNED ENGINEERED SAFETY FEATURE ACTUATION.
EVENT DATE: 111691 REPORT DATE: 121691 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 2 (BWR)

(NSIC 223637) ON 11/16/91, AT 1630 CST, UNIT 1 WAS IN THE REFUEL MODE WITH THE VESSEL ASSEMBLED AND UNIT 2 WAS IN THE RUN MODE AT 2436 CMWT (100 PERCENT OF RATED THERMAL POWER). AT THAT TIME, THE MAIN CONTROL ROOM ENVIRONMENTAL CONTROL SYSTEM (MCRECS), SHARED BY THE UNIT 1 AND 2 MAIN CONTROL ROOMS (MCR), AUTOMATICALLY TRANSFERRED TO THE PRESSURIZATION MODE. NO PLANT CONDITIONS EXISTED THAT WOULD NECESSITATE THE ACTUATION. COINCIDENT WITH THE MCRECS TRANSFER TO THE PRESSURIZATION MODE, THE MCR LIGHTS FLICKERED, INDICATING A POSSIBLE POWER SUPPLY PROBLEM. THE LICENSED UNIT 1 SHIFT SUPERVISOR, AWARE ELECTRICAL EQUIPMENT LINE-UPS WERE IN PROGRESS PER PROCEDURE 52SV-R43-027-7S, "DIESEL GENERATOR 1C LOCA/LOSP TESTING," CONTACTED THE NONLICENSED PLANT EQUIPMENT OPERATOR (PEO) PERFORMING THE ACTIVITY. THE PEO INFORMED HIM THAT HE HAD INADVERTENTLY TRIPPED THE WRONG BREAKER IN PERFORMING THE PROCEDURE, REALIZED HIS MISTAKE, IMMEDIATELY CLOSED THE BREAKER BACK USING THE LOCAL CONTROL SWITCH, AND WAS ATTEMPTING TO CONTACT HIM. INVESTIGATIONS SHOWED THAT THE MOMENTARY LOSS OF POWER RESULTING FROM THE BREAKER BEING CYCLED CAUSED THE AUTOMATIC TRANSFER OF MCRECS. SUBSEQUENTLY, MCRECS WAS TRANSFERRED BACK TO ITS NORMAL MODE OF OPERATION AND, AT 1735 CST, THE NRC WAS NOTIFIED OF THE EVENT PURSUANT TO 10 CFR 50.72. THE CAUSE OF THE EVENT WAS COGNITIVE PERSONNEL ERROR ON THE PART OF A NONLICENSED INDIVIDUAL.

[148] HATCH 1 DOCKET 50-321 LER 91-029
MALFUNCTIONING MOTOR OPERATED VALVES RESULTS IN GROUP 1 ISOLATION.
EVENT DATE: 112691 REPORT DATE: 121991 NSSS: GE TYPE: BWR
VENDOR: LIMITORQUE CORP.

(NSIC 223691) ON 11/26/91, UNIT 1 WAS IN THE STARTUP MODE AT APPROX. 4% RATED THERMAL POWER AND WAS STARTING UP FROM A REFUELING OUTAGE WHEN SYSTEM LEAKS IN THE DRYWELL LED TO A DECISION TO SHUT DOWN THE UNIT AND PERFORM REPAIRS. AT 1539 CST, THE REACTOR WAS SHUT DOWN BY MANUAL SCRAM PER PROCEDURE. SINCE THE CORE HAD BEEN CRITICAL FOR ONLY 109 HOURS, THE DECAY HEAT LEVEL WAS RELATIVELY LOW, AND THE REACTOR BEGAN TO COOL DOWN RAPIDLY. THEREFORE, LICENSED PERSONNEL CLOSED THE INBOARD MAIN STEAMLINE ISOLATION VALVES, INTENDING TO RETARD COOLDOWN RATE BY THROTTLING A NON-SAFETY RELATED MOTOR OPERATED VALVE IN THE MAIN STEAMLINE DRAIN, 1B21-F020. HOWEVER, THIS VALVE DID NOT RESPOND TO THE CONTROL SWITCH, SO THE MAIN STEAMLINE DRAIN HAD TO BE ISOLATED. THIS ISOLATED ALL STEAM LOADS FROM THE REACTOR AND REQUIRED THE MAIN CONDENSER VACUUM BREAKERS TO BE OPENED SINCE STEAM SEAL PRESSURE WAS NOT AVAILABLE. AT 1620 CST, WHEN CONDENSER VACUUM DECREASED TO THE SET POINT OF THE GROUP 1 PRIMARY CONTAINMENT ISOLATION SYSTEM TRIP ON LOW CONDENSER VACUUM, AN AUTOMATIC ISOLATION SIGNAL WAS RECEIVED. GROUP 1 VALVES WHICH WENT CLOSED WERE THE 3/4-INCH REACTOR WATER SAMPLE VALVES AND REDUNDANT VALVES IN PENETRATIONS WHICH HAD ALREADY BEEN ISOLATED INCLUDING THE OUTBOARD MAIN STEAMLINE ISOLATION VALVES. ALL AFFECTED VALVES FUNCTIONED AS DESIGNED.

[149] HATCH 1 DOCKET 50-321 LER 91-030
INSTRUMENT DRIFT CAUSES AREA RADIATION MONITOR TRIPS RESULTING IN ESF ACTUATIONS.
EVENT DATE: 120291 REPORT DATE: 122391 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 2 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223718) ON 12/02/91 AT 1055 CST AND 1133 CST, AND ON 12/07/91 AT 05?3 CST, UNITS 1 AND 2 WERE IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2436 CMWT (100 PERCENT RATED THERMAL POWER). AT THOSE TIMES, THE UNIT-COMMON MAIN CONTROL ROOM ENVIRONMENTAL CONTROL (MCREC, EIIS CODE VI) SYSTEM AUTOMATICALLY TRANSFERRED FROM THE NORMAL MODE TO THE PRESSURIZATION MODE. THE EVENTS OCCURRED AS DESIGNED WHEN REFUELING FLOOR AREA RADIATION MONITORS (ARMS, EIIS CODE IL) TRIPPED ON FALSE HIGH RADIATION SIGNALS. AFTER EACH ACTUATION, LICENSED OPERATIONS PERSONNEL VERIFIED PROPER OPERATION OF THE MCREC SYSTEM AND VERIFIED THAT AN ACTUAL HIGH RADIATION CONDITION DID NOT EXIST IN THE AFFECTED AREAS. THE ARM TRIP UNITS WERE RESET AND THE MCREC SYSTEM WAS RETURNED TO ITS NORMAL MODE OF OPERATION AFTER EACH EVENT. THE CAUSE OF THESE EVENTS WAS INSTRUMENT SETPOINT DRIFT. AN INVESTIGATION LATER DETERMINED THAT THE EVENTS OCCURRED WHEN THE HIGH RADIATION TRIP SETPOINT FOR THE AFFECTED ARMS DRIFTED, CAUSING THE ARM TO TRIP ON A FALSE HIGH RADIATION SIGNAL. THE CAUSE FOR THE SETPOINT DRIFT WAS INVESTIGATED AND A TRIP UNIT WAS REPLACED. CORRECTIVE ACTIONS FOR THESE EVENTS INCLUDED RETURNING THE MCREC SYSTEM TO ITS NORMAL MODE OF OPERATION, ADJUSTING THE HIGH RADIATION TRIP SETPOINTS TO THEIR PROPER VALUES, FUNCTIONALLY TESTING FOUR OF THE REFUELING FLOOR ARM TRIP UNITS, AND REPLACING A TRIP UNIT.

[150] HATCH 1 DOCKET 50-321 LER 91-031
INADEQUATE PROCEDURE RESULTS IN ESF ACTUATION.
EVENT DATE: 120491 REPORT DATE: 122391 NSSS: GE TYPE: BWR

(NSIC 223719) ON 12/04/91 AT 0455 CST, UNIT 1 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2436 CMWT (100% RATED THERMAL POWER). AT THAT TIME, GROUP 5 PRIMARY CONTAINMENT ISOLATION SYSTEM VALVES 1G31-F001 AND F004 CLOSED ON A REACTOR WATER CLEANUP (RWCU) SYSTEM HIGH DIFFERENTIAL FLOW ISOLATION SIGNAL. THE ISOLATION SIGNAL OCCURRED AS THE "A" RWCU PUMP WAS BEING PLACED INTO SERVICE PER PROCEDURE 42SP-101791-RZ-1-1N, "FT FOR RWCU SEALLESS PUMP." THE SYSTEM HAD BEEN ISOLATED FOR SEVERAL HOURS AND AS A RESULT; SYSTEM TEMPERATURE AND PRESSURE HAD DECREASED CAUSING VOIDS TO FORM IN THE SYSTEM PIPING. WHEN THE "A" PUMP SUCTION ISOLATION VALVE WAS OPENED TO PLACE THE PUMP INTO SERVICE, THE SYSTEM WAS UNISOLATED AND THE PARTIALLY VOIDED PIPING RAPIDLY FILLED. A HIGH DIFFERENTIAL FLOW CONDITION (SYSTEM INFLUENT WITH NO EFFLUENT) RESULTED. ISOLATION VALVES 1G31-F001 AND F004 CLOSED ON AN RWCU SYSTEM HIGH DIFFERENTIAL FLOW ISOLATION

SIGNAL PER DESIGN. THIS TERMINATED FLOW INTO THE SYSTEM AND THE ISOLATION SIGNAL CLEARED. THE CAUSE OF THIS EVENT WAS AN INADEQUATE PROCEDURE. SPECIAL PURPOSE PROCEDURE 42SP-101791-RZ-1-1N DID NOT CONTAIN ADEQUATE INSTRUCTIONS AND CAUTIONS TO ENSURE AN ISOLATION DID NOT OCCUR WHEN THE "A" PUMP WAS PLACED INTO SERVICE. THE PROCEDURE WILL NOT BE USED AGAIN. IT WAS INTENDED ONLY AS A FUNCTIONAL TEST FOR THE "A" RWCU PUMP (THE SEAMLESS PUMP) AND THE FUNCTIONAL TEST OF THE PUMP HAS BEEN COMPLETED SATISFACTORILY.

[151] HATCH 1 DOCKET 50-321 LER 91-032
PERSONNEL ERROR RESULTS IN MISSED TECHNICAL SPECIFICATIONS REQUIRED SURVEILLANCE.
EVENT DATE: 122891 REPORT DATE: 012792 NSSS: GE TYPE: BWR

(NSIC 223857) ON 12/28/91 AT 2230 CST, UNIT 1 WAS IN THE RUN MODE AT A POWER LEVEL OF 2436 CMWT (100% RATED THERMAL POWER). AT THAT TIME, IT WAS DETERMINED FROM A ROUTINE REVIEW OF COMPLETED SURVEILLANCE PROCEDURE DATA PACKAGES THAT A DAILY CHECK OF THE UNIT 1 TORUS OXYGEN CONCENTRATION HAD NOT BEEN PERFORMED THE PREVIOUS DAY AS REQUIRED BY UNIT 1 TECHNICAL SPECIFICATIONS SECTION 4.7.A.5. THE CHECK HAD BEEN PERFORMED ON 12/26/91 AT 2245 CST, BUT WAS NOT PERFORMED AGAIN UNTIL 12/28/91 AT 1136 CST. THE TIME BETWEEN THESE TWO CONSECUTIVE TORUS OXYGEN CONCENTRATION CHECKS WAS ABOUT 37 HOURS, EXCEEDING THE TIME ALLOWED BY THE UNIT 1 TECHNICAL SPECIFICATIONS (24 HOURS PLUS A 25% GRACE PERIOD). WHEN THIS EVENT WAS DISCOVERED, THE REQUIRED SURVEILLANCE WAS CURRENT; THEREFORE, NO LIMITING CONDITIONS FOR OPERATION HAD TO BE ENTERED. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR. LICENSED OPERATIONS PERSONNEL PERFORMING SURVEILLANCE PROCEDURE 34SV-SW-019-1S, "SURVEILLANCE CHECKS," ON 12/27/91 INCORRECTLY MARKED THE TORUS OXYGEN CONCENTRATION CHECK AS "NOT REQUIRED." THIS ERROR WAS NOT CAUGHT UNTIL AFTER THE GRACE PERIOD FOR PERFORMING THE SURVEILLANCE HAD EXPIRED. A REVIEW OF THE COMPLETE DATA PACKAGE FROM PROCEDURE 34SV-SW-019-1S FOR 12/27/91 REVEALED NO OTHER MISSED SURVEILLANCES. INVOLVED PERSONNEL WERE COUNSELED.

[152] HATCH 1 DOCKET 50-321 LER 91-033
HIGH PRESSURE COOLANT INJECTION SYSTEM INOPERABLE DUE TO COMPONENT FAILURE.
EVENT DATE: 123091 REPORT DATE: 012792 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 223856) ON 12/30/91, AT 1015 CST, AND ON 12/31/91, AT 1800 CST, UNIT 1 WAS IN THE RUN MODE AT 2436 CMWT (100 PERCENT OF RATED THERMAL POWER). AT EACH OF THESE TIMES, THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM EXPERIENCED FLOW OSCILLATIONS OF APPROXIMATELY 2000 GALLONS PER MINUTE WHILE BEING TESTED IN ACCORDANCE WITH PROCEDURE 34SV-E41-002-1S, "HPCI PUMP OPERABILITY". THE PROCEDURE WAS BEING PERFORMED IN ORDER TO MEET THE SURVEILLANCE REQUIREMENT OF UNIT 1 TECHNICAL SPECIFICATIONS SECTION 4.5.D.1.B.(1). IN EACH CASE, THE SYSTEM WAS SECURED AND DECLARED INOPERABLE. APPROPRIATE LIMITING CONDITIONS FOR OPERATION (LCO) WERE INITIATED AND THE REQUIRED TECHNICAL SPECIFICATIONS ACTIONS WERE IMPLEMENTED. CORRECTIVE MAINTENANCE WAS PERFORMED ON THE SYSTEM FOLLOWING EACH INCIDENT AND FUNCTIONAL TESTING OF THE SYSTEM WAS SATISFACTORILY COMPLETED ON 12/30/91, AT 1900 CST, AND ON 1/1/92, AT 0330 CST, RESPECTIVELY. IN EACH CASE, THE SYSTEM WAS RETURNED TO STANDBY AND THE LCO TERMINATED. THE CAUSE OF EACH EVENT WAS INTERMITTENT FAILURE OF A TRANSFER RELAY(S) INTERNAL TO THE HPCI SYSTEM FLOW CONTROL UNIT, 1E41-K615. THE RELAYS FUNCTION TO TRANSFER FLOW CONTROL BETWEEN THE MANUAL AND AUTOMATIC MODES. FAILURE OF THE RELAY(S) CAUSED HPCI SYSTEM FLOW TO OSCILLATE EXCESSIVELY IN EACH OF THE EVENTS.

[153] HATCH 1 DOCKET 50-321 LER 92-001
COMPONENT FAILURE CAUSES AN UNPLANNED ENGINEERED SAFETY FEATURE ACTUATION.
EVENT DATE: 011392 REPORT DATE: 020692 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 2 (BWR)
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 223956) ON 1/13/92, AT 1548 CST, UNIT 1 AND UNIT 2 WERE BOTH IN THE RUN MODE AT 2436 CMWT (100% OF RATED THERMAL POWER). AT THAT TIME, AREA RADIATION MONITOR (ARM) 1D21-K601D TRIPPED RESULTING IN THE MAIN CONTROL ROOM ENVIRONMENTAL CONTROL (MCREC) SYSTEM AUTOMATICALLY TRANSFERRING TO THE PRESSURIZATION MODE AS

DESIGNED. LICENSED PERSONNEL INITIATED INVESTIGATION INTO THE SITUATION AND FOUND THAT THE RADIATION LEVEL IN THE AREA MONITORED BY THE ARM WAS 6 MR/HR; 9 MR/HR BELOW THE ARM DESIGN TRIP SETPOINT OF 15 MR/HR. THE ARM SETPOINT WAS CHECKED AND FOUND TO BE AT 8MR/HR. THE SETPOINT WAS ADJUSTED TO WITHIN TOLERANCE PER PROCEDURE 57CP-CAL-005-1S, "ARM SYSTEM CALIBRATION." SUBSEQUENTLY, THE ARM WAS RESET AND THE MCREC SYSTEM WAS RETURNED TO THE NORMAL MODE OF OPERATION BY 2345 CST. FURTHER INVESTIGATION OF THE ARM SYSTEM BY I&C PERSONNEL REVEALED THAT INTERMITTENT FAILURE OF ARM DC POWER SUPPLY UNIT 1D21-K603A CAUSED THE ARM TRIP. THE POWER SUPPLY UNIT WAS REPLACED. THE SETPOINTS OF THE ARMS POWERED BY THIS UNIT WERE CHECKED AND ADJUSTED. THE CAUSE OF THE EVENT WAS INTERMITTENT FAILURE OF POWER SUPPLY UNIT 1D21-K603A. THE OUTPUT OF THE 1D21-K603A 24 VDC MODULE WAS INTERMITTENTLY DRIFTING BELOW THE ALLOWABLE TOLERANCE, RESULTING IN A PREMATURE TRIP OF ARM 1D21-K601D. CORRECTIVE ACTIONS FOR THE EVENT INCLUDE REPLACING THE POWER SUPPLY, ADJUSTING ARM SETPOINTS, AND PERFORMING A FAILURE ANALYSIS ON THE POWER SUPPLY.

[154] HATCH 1 DOCKET 50-321 LER 92-002
PERSONNEL ERRORS RESULT IN MISSED TECHNICAL SPECIFICATION SURVEILLANCES.
EVENT DATE: 011592 REPORT DATE: 020692 NSSS: GE TYPE: BWR

(NSIC 223916) ON 1/15/92, AT 1200 CST, UNIT 1 WAS IN-THE RUN MODE AT 2436 CMWT (APPROXIMATELY 100 PERCENT OF RATED THERMAL POWER). LICENSED PERSONNEL WERE PREPARING A REVISION TO PROCEDURE 34SV-S W -019-1S, "SURVEILLANCE CHECKS," AND NOTED THAT IN TWO SEPARATE INSTANCES THE PROCEDURE REQUIRED INSTRUMENT CHECKS TO BE PERFORMED LESS FREQUENTLY THAN THAT REQUIRED BY THE UNIT 1 TECHNICAL SPECIFICATIONS. SPECIFICALLY, UNIT 1 TECHNICAL SPECIFICATIONS TABLE 4.2-11, ITEM 7 REQUIRES AN INSTRUMENT CHECK TO BE PERFORMED ON THE SUPPRESSION CHAMBER WATER TEMPERATURE PARAMETERS OF TEMPERATURE RECORDERS LT47-R611 AND R612 ONCE PER SHIFT. ALSO, UNIT 1 TECHNICAL SPECIFICATIONS TABLE 4.1-1, ITEM 8 REQUIRES AN INSTRUMENT CHECK TO BE PERFORMED ON AVERAGE POWER RANGE MONITOR (APRM) INDICATORS 1CSL-K605A THROUGH F ONCE PER SHIFT. HOWEVER, FOR EACH OF THESE INSTRUMENT CHECKS, PROCEDURE 34SV-SW-019-1S REQUIRED THEY BE PERFORMED DAILY RATHER THAN ONCE PER SHIFT RESULTING IN MISSED TECHNICAL SPECIFICATION SURVEILLANCES. A DEFICIENCY CARD WAS WRITTEN AND ON-SHIFT LICENSED PERSONNEL WERE NOTIFIED. THE INSTRUMENT CHECKS WERE THEN PERFORMED OR VERIFIED TO BE CURRENT. THE CAUSE OF THE EVENTS WAS COGNITIVE PERSONNEL ERROR ON THE PART OF LICENSED PERSONNEL. SPECIFICALLY, INDIVIDUALS MADE INCORRECT CHANGES TO PROCEDURE 34SV-SW-019-1S WHILE USING THE EDITORIAL CORRECTION PROCESS.

[155] HATCH 2 DOCKET 50-366 LER 92-001
ERRORS IN PLANT DRAWING AND FSAR RESULT IN MISSED TECHNICAL SPECIFICATIONS SURVEILLANCES.
EVENT DATE: 010992 REPORT DATE: 020392 NSSS: GE TYPE: BWR

(NSIC 223941) ON 1/9/92 AT 1400 CST, UNIT 2 WAS IN THE RUN MODE AT A POWER LEVEL OF 2436 CMWT (100% RATED THERMAL POWER). AT THAT TIME, NON-LICENSED PLANT ENGINEERING PERSONNEL DETERMINED THAT A LOCAL LEAK RATE TEST OF PRIMARY CONTAINMENT PENETRATION X-222A HAD NOT BEEN PERFORMED AS REQUIRED BY UNIT 2 TECHNICAL SPECIFICATIONS SECTION 4.6.1.2 D. ALSO, IT WAS DETERMINED THAT THE PENETRATION HAD NOT BEEN VERIFIED TO BE CLOSED AT LEAST ONCE EVERY 31 DAYS AS REQUIRED BY UNIT 2 TECHNICAL SPECIFICATIONS SECTION 4.6.1.1.A.1. A VISUAL INSPECTION OF THE PENETRATION'S SEALING DEVICE PERFORMED ON 1/9/92 REVEALED IT TO BE INTACT, WITH NO VISIBLE OR AUDIBLE SIGNS OF LEAKAGE OR SIGNS OF DETERIORATION. GEORGIA POWER COMPANY DETERMINED THAT THE PENETRATION DID NOT HAVE TO BE DECLARED INOPERABLE UNTIL A LEAK RATE TEST COULD BE PERFORMED. THE NRC WAS INFORMED OF THIS ON 1/9/92. THE LEAK TEST WAS DONE ON 1/10/92 AFTER A TEST VOLUME WAS INSTALLED IN ORDER TO ALLOW THE PENETRATION TO BE TESTED. THE PENETRATION'S AS-FOUND LEAKAGE RATE WAS ZERO ACTUAL CUBIC CENTIMETERS PER MINUTE (ACCM). THIS EVENT WAS CAUSED BY ERRORS IN A PLANT DRAWING AND THE UNIT 2 FINAL SAFETY ANALYSIS REPORT (FSAR) PLANT DRAWING S-28719 INCORRECTLY SHOWED THE PENETRATION'S SEALING DEVICE AS ONE NOT REQUIRING A LOCAL LEAK RATE TEST OR PERIODIC INSPECTION.

[156] HOPE CREEK 1 DOCKET 50-354 LER 91-019 REV 01
UPDATE ON "E" AND "F" FILTRATION RECIRCULATION VENTILATION SYSTEM RECIRC FAN START.
EVENT DATE: 112391 REPORT DATE: 012292 NSSS: GE TYPE: BWR
VENDOR: AMERICAN WARMING & VENTILATING INC.

(NSIC 223834) ON 11/23/91 AT 0500 HOURS, CONTROL ROOM OPERATORS (RO LICENSED) DISCOVERED THAT THE "E" FILTRATION RECIRCULATION VENTILATION (FRVS) FAN WAS RUNNING. AFTER VERIFYING NO VALID START SIGNALS WERE PRESENT THE FAN WAS SECURED. AT 0638 HOURS DURING SHIFT TURNOVER THE "E" FRVS FAN STARTED A SECOND TIME. OPERATORS NOTICED THE FAN START DURING PANEL WALK DOWN. THEY AGAIN VERIFIED NO VALID START SIGNALS WERE PRESENT AND SECURED THE FAN. OPERATORS COULD NOT DETERMINE THE CAUSE OF THE FAN START AND REQUESTED INSTRUMENT AND CONTROL TECHNICIANS TO INVESTIGATE. ON 12/1 3/91 THE EFRVS WAS INSTRUMENTED AND THE FRVS SYSTEM (A,B,C AND D FRVS RECIRC FANS) WAS PLACED IN SERVICE TO DETERMINE THE ROOT CAUSE OF THE SPURIOUS FAN STARTS IN CONJUNCTION WITH DOP/HALIDE TESTING. THE FRVS SYSTEM WAS IN SERVICE FOR 6.5 HOURS WITH NO AUTO FAN STARTS OCCURRING. THE FANS WERE SECURED AT 1222 HOURS AT THE COMPLETION OF THE DOP/HALIDE TEST. AT 1557 HOURS THE EFRVS RECIRC FAN SPURIOUSLY STARTED AND WAS SUBSEQUENTLY SECURED BY OPERATIONS PERSONNEL. SUBSEQUENT INVESTIGATION HAS REVEALED AN ACCUMULATION OF WATER IN THE LOW FLOW SWITCH SENSING LINES AS THE MOST PROBABLE CAUSE OF THE SPURIOUS FAN STARTS. THE TESTING DID RESULT IN ADDITIONAL FAN STARTS, ON 12/23/91 THE STANDBY "E" FRVS RECIRC UNIT, ON 1/2/92 THE STANDBY "F" FRVS RECIRC UNIT AUTO STARTED AND ON 1/14/92 "E" FRVS RE CIRC UNIT AUTO STARTED.

[157] HOPE CREEK 1 DOCKET 50-354 LER 91-020
HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION DUE TO IMPROPER WIRING OF TEMPERATURE DETECTOR.
EVENT DATE: 120791 REPORT DATE: 010692 NSSS: GE TYPE: BWR

(NSIC 223693) ON 12/07/91 AT 1900 HRS CONTROL ROOM PERSONNEL RECEIVED INDICATION OF A HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI) ISOLATION DURING STEAM LINE WARMUP FOLLOWING A HPCI OUTAGE. AFTER VERIFYING NO VALID SIGNALS WERE PRESENT, CONTROL ROOM PERSONNEL REQUESTED INSTRUMENT AND CONTROLS TECHNICIANS TO INVESTIGATE THE CAUSE OF THE ISOLATION. THE TECHNICIAN DETERMINED THE ISOLATION HAD OCCURRED DUE TO AN INTERMITTENT ERROR IN THE SUBTRACTION CIRCUIT OF THE "A" CHANNEL NUMAC DRAWER. THE DRAWER WAS CHECKED AND A SELF TEST WAS INITIATED. THE DRAWER TESTED SATISFACTORY AND THE EQUIPMENT WAS RETURNED TO SERVICE. DURING SUBSEQUENT WARM-UP OF THE HPCI STEAM LINE, OPERATORS CONTINUED TO MONITOR THE NUMAC DRAWER WITH NO ABNORMAL CONDITIONS NOTED. HPCI WAS THEN PLACED IN A STANDBY LINEUP AND DECLARED OPERABLE. ON 12/19/91 TECHNICIANS PERFORMING A ROOT CAUSE INVESTIGATION INTO THE ISOLATION NOTED THAT THE FIELD WIRING OF THE THERMOCOUPLES IN THE "A" CHANNEL ISOLATION SYSTEM WERE MIS-WIRED. THE SYSTEM WAS DECLARED INOPERABLE AND THE FIELD WIRING WAS RETURNED TO PROPER CONFIGURATION. PRELIMINARY RESULTS OF SUBSEQUENT INVESTIGATIONS HAVE INDICATED THE THERMOCOUPLE LEADS MAY HAVE BEEN LANDED INCORRECTLY DURING IMPLEMENTATION OF A DESIGN CHANGE PACKAGE DURING REFUEL OUTAGE THREE. THE REDUNDANT "C" CHANNEL ISOLATION MODULE HAS BEEN VERIFIED TO BE WIRED CORRECTLY AND OPERABLE.

[158] INDIAN POINT 2 DOCKET 50-247 LER 91-022
INADVERTENT ACTUATION OF THE AMMONIA TOXIC GAS MONITOR.
EVENT DATE: 112891 REPORT DATE: 123091 NSSS: WE TYPE: PWR
VENDOR: WISCONSIN BRIDGE & IRON

(NSIC 223698) ON NOVEMBER 28, 1991, AT APPROXIMATELY 0450 HOURS, WITH REACTOR POWER AT 100%, THE ANHYDROUS AMMONIA (NH3) TOXIC GAS MONITOR CHANNEL 2 ALARMED INADVERTENTLY, RESULTING IN THE TRANSFER OF THE CENTRAL CONTROL ROOM (CCR) VENTILATION SYSTEM FROM THE NORMAL MODE TO THE INCIDENT MODE. A SUBSEQUENT NH3 TOXIC GAS MONITOR CHANNEL ACTUATION OCCURRED ON NOVEMBER 29. AS DESIGNED, THE DETECTION OF THE RESPECTIVE GAS BY EITHER CHANNELS 1 OR 2 OF THE TOXIC GAS MONITORS WILL GENERATE AN ALARM IN THE CCR AND ISOLATE THE CCR VENTILATION SYSTEM. THE TOXIC GAS MONITORING SYSTEM IS CLASSIFIED AS AN ENGINEERED SAFETY FEATURE (ESF). NO TECHNICAL SPECIFICATION OR NRC LIMITS WERE EXCEEDED.

[159] INDIAN POINT 2 DOCKET 50-247 LER 92-001
INADVERTENT ACTUATIONS OF CONTROL ROOM TOXIC GAS MONITORS.
EVENT DATE: 010892 REPORT DATE: 020792 NSSS: WE TYPE: PWR
VENDOR: WISCONSIN BRIDGE & IRON

(NSIC 224026) ON JANUARY 8, 1992, AT APPROXIMATELY 0345 HOURS, WITH REACTOR POWER AT 100%, THE CHLORINE TOXIC GAS MONITOR CHANNEL 2 ALARMED INADVERTENTLY, RESULTING IN THE TRANSFER OF THE CENTRAL CONTROL ROOM (CCR) VENTILATION SYSTEM FROM THE NORMAL MODE TO THE INCIDENT MODE. SUBSEQUENT AMMONIA AND HYDROGEN CYANIDE TOXIC GAS MONITOR CHANNEL ACTUATIONS OCCURRED ON JANUARY 20 AND 24. AS DESIGNED, THE DETECTION OF THE RESPECTIVE GASES BY EITHER CHANNELS 1 OR 2 OF THE TOXIC GAS MONITORS WILL GENERATE AN ALARM IN THE CCR AND ISOLATE THE CCR VENTILATION SYSTEM. THE CCR VENTILATION SYSTEM IS CLASSIFIED AS AN ENGINEERED SAFETY FEATURE. NO TECHNICAL SPECIFICATION OR NRC LIMITS WERE EXCEEDED.

[160] INDIAN POINT 3 DOCKET 50-286 LER 92-001
EMERGENCY DIESEL GENERATOR AUTO VOLTAGE CONTROL RHEOSTAT FOUND OUT OF POSITION.
EVENT DATE: 121691 REPORT DATE: 011792 NSSS: WE TYPE: PWR
VENDOR: ALCO ENGINE DIVISION, WHITE IND.

(NSIC 223791) WITH INDIAN POINT 3 AT 100 PERCENT POWER ON DECEMBER 16, 1991, 33 EMERGENCY DIESEL GENERATOR (EDG) WAS DETERMINED TO BE INOPERABLE WHEN THE AUTO VOLTAGE CONTROL RHEOSTAT WAS FOUND OUTSIDE THE OPERATING BAND. THE AUTHORITY HAS ESTABLISHED THAT THE LAST TIME 33 EDG OPERABILITY WAS VERIFIED WAS DURING AN OPERATIONAL CHECK ON DECEMBER 9, 1991. THE EXACT CAUSE OF THE EVENT CANNOT BE DETERMINED, THEREFORE THE AUTHORITY IS ASSUMING THE RHEOSTAT WAS OUTSIDE THE OPERABLE BAND FROM DECEMBER 9, 1991 TO DECEMBER 16, 1991. BASED ON THIS ASSUMPTION, THERE WERE TWO OCCASIONS DURING THE PERIOD FROM DECEMBER 9 THROUGH DECEMBER 16, 1991 THAT THE PLANT WOULD HAVE BEEN IN A CONDITION OUTSIDE THE DESIGN BASIS. DURING THE INVESTIGATION, A NEED TO STRENGTHEN THE DOCUMENTATION PROCESS FOR OPERABILITY CHECKS WAS IDENTIFIED. A NEW SURVEILLANCE TEST WILL BE IMPLEMENTED BY FEBRUARY 3, 1992. ADDITIONALLY, REFERENCE MARKS HAVE BEEN PLACED ON THE RHEOSTAT'S ESCUTCHEON PLATES ON ALL EDG CONTROL PANELS, AND THE PANELS ARE CHECKED EVERY SHIFT.

[161] KEWAUNEE DOCKET 50-305 LER 91-012
UNMARKED STEAM EXCLUSION DOORS TO TURBINE DRIVEN AUXILIARY FEEDWATER PUMP ROOM RESULTED IN POTENTIAL INABILITY TO MEET HIGH ENERGY LINE BREAK CRITERIA.
EVENT DATE: 120391 REPORT DATE: 010392 NSSS: WE TYPE: PWR

(NSIC 223771) AT 0935 ON DECEMBER 4, 1991, WITH THE PLANT AT 100% POWER, WISCONSIN PUBLIC SERVICE CORPORATION DETERMINED THAT FOR A PERIOD OF APPROXIMATELY ONE-HALF HOUR ON DECEMBER 3, 1991, THE KEWAUNEE NUCLEAR POWER PLANT HAD BEEN IN A CONDITION OUTSIDE OF THE DESIGN BASIS FOR THE STEAM EXCLUSION SYSTEM. THE STEAM EXCLUSION SYSTEM IS DESIGNED TO PREVENT THE STEAM ENVIRONMENT THAT WILL EXIST FOLLOWING A HIGH ENERGY LINE BREAK FROM ENTERING ADJOINING AREAS WHICH CONTAIN EQUIPMENT NECESSARY TO MITIGATE THE EVENT. IT WAS DETERMINED THAT DURING THE TIME PERIOD OF 1058 - 1115 ON DECEMBER 3, CONTRACTED WELDERS RAN A WELDING CABLE THROUGH THE TWO DOORWAYS FOR THE TURBINE DRIVEN AUXILIARY FEEDWATER PUMP ROOM. THE CABLE WOULD HAVE PREVENTED CLOSING OF THE DOORS WHICH ARE PART OF THE STEAM EXCLUSION BOUNDARY FOR THE ADJOINING AREAS. THE CABLE WAS REMOVED AND THE TWO DOORS CLOSED BY 1129 ON DECEMBER 3. THE EVENT OCCURRED BECAUSE THE DOORS WERE NOT IDENTIFIED AS STEAM EXCLUSION BOUNDARIES. SIGNS WITH THE APPROPRIATE WORDING HAVE BEEN PERMANENTLY AFFIXED TO THE DOORS WHICH SHOULD PREVENT A RECURRENCE OF THIS TYPE EVENT.

[162] KEWAUNEE DOCKET 50-305 LER 91-013
PROCEDURAL DEFICIENCY RESULTS IN CLOSURE OF STEAM GENERATOR BLOWDOWN ISOLATION VALVES.
EVENT DATE: 120691 REPORT DATE: 010692 NSSS: WE TYPE: PWR

(NSIC 223714) THIS REPORT DESCRIBES AN UNPLANNED AUTOMATIC ACTUATION OF THE STEAM GENERATOR BLOWDOWN VALVES AND STEAM GENERATOR BLOWDOWN SAMPLE VALVES, ENGINEERED

SAFETY FEATURES (ESF). AT 0924 ON DECEMBER 6, 1991 WITH THE PLANT AT 100% POWER, THE VALVES CLOSED AND ISOLATED STEAM GENERATOR BLOWDOWN AND BLOWDOWN SAMPLING. THE VALVES CLOSED AS DESIGNED ON A HIGH RADIATION SIGNAL FROM THE STEAM GENERATOR BLOWDOWN MONITOR (R-19). THE HIGH RADIATION SIGNAL WAS GENERATED DURING THE MONTHLY SOURCE CHECK OF R-19 WHEN A NUCLEAR CONTROL OPERATOR IN TRAINING, DIRECTLY SUPERVISED BY A LICENSED OPERATOR, POSITIONED THE "OPERATION SELECTOR SWITCH" FROM "OPERATE" TO "CHECK SOURCE". AS EXPECTED, AN UPSCALE METER RESPONSE WAS OBSERVED. HOWEVER, THE OPERATOR IN TRAINING DID NOT RETURN THE "OPERATION SELECTOR SWITCH" TO THE "OPERATE" POSITION PRIOR TO REACHING THE R-19 NOMINAL HIGH ALARM SETPOINT (450 CPM) THUS CAUSING THE UNPLANNED ESF ACTUATION. PAST PRACTICE WAS TO HOLD THE "OPERATION SELECTOR SWITCH" IN THE "CHECK SOURCE" POSITION ONLY LONG ENOUGH TO OBSERVE A RESPONSE TO THE SOURCE AND THEN RETURN TO THE "OPERATE" POSITION. TO PREVENT RECURRENCE OF THIS TYPE EVENT, OPERATIONS WILL REVIEW AND REVISE THEIR PROCEDURES AS NECESSARY TO ENSURE THE PROCEDURES ACCURATELY INSTRUCT OPERATORS ON THE POTENTIAL OF REACHING HIGH ALARM SETPOINT WHEN PERFORMING SOURCE CHECKS.

[163] LA SALLE 1 DOCKET 50-373 LER 91-015 REV 01
UPDATE ON INADEQUATE TESTING OF DIESEL GENERATORS DUE TO INADEQUATE PROCEDURES/TECHNICAL SPECIFICATION MISINTERPRETATION.
EVENT DATE: 110791 REPORT DATE: 120691 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 2 (BWR)

(NSIC 223760) ON NOVEMBER 7, 1991 AT APPROXIMATELY 1500 HOURS, WITH UNIT 1 AND UNIT 2 IN OPERATIONAL CONDITION 1 (RUN) AT 100% POWER, DURING A NUCLEAR REGULATORY COMMISSION (NRC) ELECTRICAL DISTRIBUTION SYSTEM FUNCTIONAL INSPECTION (EDSFI) IT WAS DETERMINED, THAT CERTAIN EMERGENCY SAFETY FEATURE (ESF) BUS UNDERVOLTAGE RELAY CONTACTS WERE NOT FUNCTIONALLY TESTED AS REQUIRED BY PLANT TECHNICAL SPECIFICATION 4.8.1.1 .2.D.4. THE STATION WAS PLACED IN A 24 HOUR TIMECLOCK AND THE REQUIRED CONTACT TESTING WAS COMMENCED TO FULFILL THE TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS. ON NOVEMBER 8, 1991, AT 0900 HOURS IT WAS DETERMINED THAT THE ALLOWANCE OF THE 24 HOUR PERIOD TO PERFORM THE REQUIRED SURVEILLANCES WAS INAPPROPRIATE AND A LATE NOTIFICATION WAS MADE. TECHNICAL SPECIFICATION 3.0.5 WAS ENTERED. ALL TESTING WAS COMPLETED AT 1030 HOURS ON NOVEMBER 8, 1991. THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE TESTING PROCEDURES. THE SAFETY CONSEQUENCES OF THIS EVENT WERE MINIMAL. THE 0, 1A, 2A, 1B, AND 2B DIESEL GENERATORS WERE INOPERABLE SOLELY DUE TO THE MISSED SURVEILLANCE AND WERE FULLY FUNCTIONAL THROUGHOUT THE EVENT, EXCEPT WHILE EACH WAS UNDERGOING TESTING IN ACCORDANCE WITH STATION CORRECTIVE ACTIONS. THIS EVENT IS BEING REPORTED PURSUANT TO 10CFR50.73(A)(2)(I) DUE TO A DEVIATION FROM PLANT TECHNICAL SPECIFICATIONS.

[164] LA SALLE 1 DOCKET 50-373 LER 92-001
AVERAGE POWER RANGE MONITORS (APRM) SET NONCONSERVATIVELY DUE TO COMMUNICATION ERROR.
EVENT DATE: 010792 REPORT DATE: 020592 NSSS: GE TYPE: BWR

(NSIC 223911) ON JANUARY 7 1992, DURING LASALLE INSTRUMENT SURVEILLANCE LIS-NR-109 "UNIT 1 AVERAGE POWER RANGE MONITOR (APRM) GAIN ADJUSTMENT", A MISCOMMUNICATION EVENT OCCURRED. AT 0740, A CONTROL SYSTEM TECHNICIAN (CST) COMMENCED TO PERFORM THE GAIN ADJUSTMENT FOR THE APRM NEUTRON MONITORING SYSTEM (NR) (IG). HE CONTACTED A QUALIFIED NUCLEAR ENGINEER (QNE) FOR THE VALUE TO WHICH THE GAINS SHOULD BE SET. THE QNE ASSUMED THE CST WAS REFERRING TO LIS-NR-107 "UNIT 1 APRM/ROD BLOCK MONITOR FLOW CONVERTER TO TOTAL CORE FLOW ADJUSTMENT". THE CST PROVIDED THE QNE WITH THE DRIVE FLOWS FROM THE CORE MONITORING CODE'S CORE POWER TO FLOW LOG, AND THE QNE INSTRUCTED HIM TO SET THE GAINS TO 93 PERCENT WHILE THE REACTOR POWER WAS ACTUALLY 98 PERCENT. THE EVENT RESULTED IN ALL SIX APRMS EXCEEDING THEIR ALLOWABLE TECHNICAL SPECIFIC ATION TOLERANCE, AND THE INTENDED FUNCTION OF THE REACTOR PROTECTION SYSTEM (RPS, RP) "JC" WAS COMPROMISED. THE TOTAL TIME ELAPSED FROM THE TIME AT WHICH THE FIRST APRM WAS SET NONCONSERVATIVELY TO THE TIME WHEN ALL SIX APRMS WERE SET CORRECTLY WAS 52 MINUTES. THE INDIVIDUALS WERE COUNSELLED ON THE IMPORTANCE OF COMMUNICATION AND HAVING A QUESTIONING ATTITUDE. PROCEDURE REVISIONS WILL BE IMPLEMENTED PROVIDING

ADDITIONAL GUIDANCE TO THE CSTS WHILE SETTING THE APRM GAINS, AND REQUIRING OPERATIONS REVIEW OF DESIRED APRM SETTINGS.

[165] LA SALLE 2 DOCKET 50-374 LER 91-005 REV 02
UPDATE ON REACTOR CORE ISOLATION COOLING DECLARED INOPERABLE DUE TO STEAM LINE HIGH FLOW SWITCH FAILURE.
EVENT DATE: 062191 REPORT DATE: 112791 NSSS: GE TYPE: BWR
VENDOR: STATIC-O-RING

(NSIC 223694) ON 6/21/91, AT APPROXIMATELY 1500 HOURS, DURING THE PERFORMANCE OF LASALLE INSTRUMENT SURVEILLANCE, LIS-RI-401, "UNIT 2 STEAM LINE HIGH FLOW REACTOR CORE ISOLATION COOLING (RCIC) ISOLATION FUNCTIONAL TEST," PRESSURE DIFFERENTIAL SWITCH (PDS) 2E31-N013BA WAS FOUND FAILED APPARENTLY FROM A RUPTURED DIAPHRAGM. THIS SWITCH FUNCTIONS TO PROVIDE A DIVISION II (INBOARD) ISOLATION OF THE RCIC STEAM LINE AND TO INITIATE RCIC TURBINE TRIP UNDER A HIGH STEAM FLOW CONDITION. PDS 2E31-N013BA WAS MADE NON-FUNCTIONAL BY THE RUPTURED DIAPHRAGM; THEREFORE, THE DIVISION II ISOLATION AND THE RCIC TRIP ASSOCIATED WITH THIS SWITCH WAS UNAVAILABLE. HOWEVER PDS 2E31-N013AA WAS STILL AVAILABLE TO PROVIDE A DIVISION I (OUTBOARD) ISOLATION AND TURBINE TRIP. FURTHERMORE REDUNDANT DIVISION I AND II HIGH FLOW SWITCHES PDS 2E31-007AA AND 2E31-007BA WERE AVAILABLE TO PROVIDE ISOLATION AND TURBINE TRIP IF REQUIRED. RCIC WAS DECLARED INOPERABLE ON 6/21/91 AT 1500 HOURS IN ORDER TO PERFORM THE REQUIRED SURVEILLANCE. HIGH PRESSURE CORE SPRAY REMAINED OPERABLE THROUGHOUT THE EVENT. A REPLACEMENT SWITCH WAS INSTALLED, CALIBRATED, AND FUNCTIONALLY TESTED SATISFACTORILY. RCIC WAS DECLARE OPERABLE ON 6/22/91 AT 0515 HOURS. THIS EVENT IS BEING REPORTED TO THE NRC PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(V) DUE TO RCIC DECLARED INOPERABLE (LOSS OF A SAFETY SYSTEM FUNCTION).

[166] LA SALLE 2 DOCKET 50-374 LER 91-013 REV 01
UPDATE ON LOSS OF AUXILIARY ELECTRIC EQUIPMENT ROOM VENTILATION SUPPLY FAN DUE TO OVERHEATING OF STARTING COIL FOR THE BREAKER.
EVENT DATE: 100791 REPORT DATE: 110191 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 1 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223921) ON 10/7/91, AT 0400 HOURS WITH UNIT 1 AND 2 IN OPERATIONAL CONDITION 1 (RUN) AT 95% AND 85% POWER RESPECTIVELY, A SEVEN DAY TIMECLOCK WAS ENTERED PER TECH SPEC 3.7.2 DUE TO THE VE(VI) SUPPLY FAN BREAKER TRIPPING. AT THIS TIME THE "A" VC/VE(VI) TRAIN OUT OF SERVICE FOR SCHEDULED MAINTENANCE AND A ONE HOUR TIMECLOCK WAS ENTERED AT THIS TIME PER TECH SPEC 3.0.3, SINCE BOTH VC/VE(VI) EMERGENCY MAKE-UP TRAINS WERE INOPERABLE AT THE SAME TIME. THE "A" TRAIN WAS RETURNED TO SERVICE AT 0450 NOTIFICATION WAS MADE AT 0755 HOURS ON 10/7/91, ON THE BASIS OF A LOSS OF A SAFETY SYSTEM FUNCTION, SINCE BOTH AUXILIARY ELECTRIC ROOM VENTILATION SYSTEMS WERE UNAVAILABLE FOR SERVICE. WORK REQUEST L10621 WAS INITIATED AT THIS TIME FOR THE ELECTRICAL MAINTENANCE DEPARTMENT TO INVESTIGATE AND CORRECT THE PROBLEM. THE 52X RELAY, THE STARTING COIL, WAS FOUND TO BE THE CAUSE OF THE PROBLEM. THIS RELAY WAS REPLACED AND ON 10/7/91, AT 2030 HOURS THE VE(VI) SUPPLY FAN, OVE01CB, WAS SUCCESSFULLY RUN. ON 10/7/91, AT 2120 HOURS THE TIMECLOCK WAS EXITED. THIS EVENT IS REPORTABLE PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)92)(V)(C) DUE TO THE LOSS OF A SAFETY SYSTEM FUNCTION.

[167] LIMERICK 1 DOCKET 50-352 LER 91-008 REV 01
UPDATE ON LER REPORTS A CONDITION PROHIBITED BY TS IN THAT EIGHT PRIMARY CONTAINMENT ISOLATION VALVES WERE INOPERABLE DUE TO INADEQUATE TORQUE SWITCH SETTING.
EVENT DATE: 030891 REPORT DATE: 121191 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 223863) ON MARCH 8, 1991, LIMERICK GENERATING STATION MAINTENANCE PERSONNEL PERFORMED DIAGNOSTIC TESTING TO SUPPORT THE DESIGN BASIS ENGINEERING REVIEW FOR GENERIC LETTER 89-10, "SAFETY-RELATED MOTOR-OPERATED VALVE TESTING AND SURVEILLANCE." THIS TESTING WAS PERFORMED ON A UNIT 2 PRIMARY CONTAINMENT (PC)

MOTOR-OPERATED BUTTERFLY VALVE (MOBV) ASSOCIATED WITH THE PC PURGE AND EXHAUST SYSTEM. HOWEVER, MAINTENANCE PERSONNEL DISCOVERED THAT THIS VALVE AND SEVEN OTHER SIMILAR TYPE VALVES HAD TORQUE SWITCH SETTINGS THAT WOULD RESULT IN THE MOBV TRIPPING ON HIGH TORQUE BEFORE THE VALVE COULD ADEQUATELY CLOSE AND SEAT IN THE EVENT THESE VALVES WERE REQUIRED TO CLOSE DURING A DIFFERENTIAL PRESSURE CONDITION. THESE EIGHT MOBVS WERE DECLARED INOPERABLE AND THE APPLICABLE UNIT 1 AND UNIT 2 PC PENETRATIONS WERE ISOLATED BY DEACTIVATING AND SECURING THE REDUNDANT PC ISOLATION VALVES IN THE CLOSED POSITION. THE ACTUAL CONSEQUENCES OF THIS CONDITION WERE MINIMAL AND THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL TO THE ENVIRONMENT. THE CAUSE OF THIS EVENT IS THE RESULT OF AN INADEQUATE INDUSTRY TESTING METHOD USED FOR TORQUE SEATED MOBVS INSTALLED DURING THE UNIT 1 AND UNIT 2 CONSTRUCTION AND STARTUP TESTING PROGRAMS. THE INVESTIGATION INTO THE CAUSE OF THIS EVENT CONCLUDED THAT GENERIC IMPLICATIONS ARE LIMITED TO TORQUE SEATED MOBVS.

[168] LIMERICK 1 DOCKET 50-352 LER 91-025
MAIN CONTROL ROOM ISOLATIONS IN RESPONSE TO A HIGH TOXIC CHEMICAL CONCENTRATION ALARM OF AN INDETERMINATE NATURE.
EVENT DATE: 110391 REPORT DATE: 120391 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 223617) ON NOVEMBER 3, 1991, AND AGAIN ON NOVEMBER 18, 1991, MAIN CONTROL ROOM (MCR) PERSONNEL RECEIVED A MCR ANNUNCIATOR ALARM INDICATING HIGH TOXIC GAS CONCENTRATION IN THE MCR FRESH AIR INTAKE. IN EACH INSTANCE, MCR PERSONNEL THEN ENTERED SPECIAL EVENT PROCEDURE SE-2, "TOXIC GAS," DONNED SELF-CONTAINED BREATHING APPARATUS, AND MANUALLY INITIATED A MCR VENTILATION SYSTEM ISOLATION, AN ENGINEERED SAFETY FEATURE (ESF). IN CONJUNCTION WITH THE MCR VENTILATION SYSTEM ISOLATION, THE CONTROL ROOM EMERGENCY FRESH AIR SUPPLY (CREFAS) SYSTEM, ALSO AN ESF INITIATED AS DESIGNED AND PROVIDED TOTAL RECIRCULATION OF THE MCR AIR WITHOUT ANY FRESH AIR INTAKE FROM THE OUTSIDE ATMOSPHERE. CHEMISTRY PERSONNEL WERE NOTIFIED AND AIR SAMPLES FROM THE MCR WERE OBTAINED. NO TOXIC GAS CONCENTRATIONS WERE DETECTED. THE CONSEQUENCES OF THESE EVENTS WERE MINIMAL IN THAT NO TOXIC GAS ACTUALLY EXISTED. THE CAUSE OF THESE EVENTS WAS MOMENTARY HIGH TOXIC CHEMICAL CONCENTRATION INDICATION OF AN INDETERMINATE NATURE. SUBSEQUENT TROUBLESHOOTING FAILED TO CLARIFY THE EXACT CAUSE OF THE HIGH INDICATIONS. WE ARE INVESTIGATING SEVERAL OPTIONS TO DECREASE THE VULNERABILITY OF THE TOXIC GAS DETECTION SYSTEM TO A SINGLE FAILURE OR SPURIOUS SIGNAL TO MINIMIZE THE REQUIRED MCR ISOLATIONS.

[169] LIMERICK 1 DOCKET 50-352 LER 91-026
AN ENGINEERED SAFETY FEATURE ACTUATION DUE TO AN INADEQUATE REVIEW OF A BLOCKING PERMIT FOR AN ISOLATION VALVE ASSOCIATED WITH A PRIMARY CONTAINMENT H2/O2 COMBUSTIBLE GAS ANALYZER.
EVENT DATE: 111291 REPORT DATE: 121191 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 223664) ON NOVEMBER 12, 1991, AN ELECTRICAL LEAD FROM A PRIMARY CONTAINMENT ISOLATION VALVE (PCIV) IN ONE OF THE SAMPLE LINES TO THE H2/O2 COMBUSTIBLE GAS ANALYZER (CGA) 10S206 WAS INADVERTENTLY GROUNDED DURING MAINTENANCE ACTIVITIES, CAUSING AN ACTUATION OF THE PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM, AN ENGINEERED SAFETY FEATURE. THE INADVERTENT GROUNDING RESULTED IN A LOSS OF INDICATION ON THE MAIN CONTROL ROOM (MCR) PANEL 10C601 FOR OTHER PCIVS ASSOCIATED WITH THE SAMPLE LINES TO THE CGA 10S206. A LOSS OF VALVE POSITION INDICATION ON SEVERAL RELATED VALVES IS AN INDICATOR OF A LOSS OF ELECTRICAL POWER TO THE VALVES. SINCE THE PCIVS FAIL CLOSED WITH A LOSS OF ELECTRICAL POWER, MCR PERSONNEL REMOVED THE CGA 10S206 FROM OPERATION, AND LATER DISCOVERED THAT A FUSE HAD BLOWN IN PANEL 10C601 AS A RESULT OF THE INADVERTENT GROUNDING. WITHIN 23 MINUTES, THE CGA 10S206 WAS RETURNED TO OPERABLE STATUS. THE ACTUAL CONSEQUENCES OF THIS EVENT WERE MINIMAL. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR RESULTING FROM AN INADEQUATE UNDERSTANDING OF THE FULL SCOPE OF THE MAINTENANCE ACTIVITIES TO BE PERFORMED ON THE PCIV FOR THE CGA 10S206. THE PERSONNEL INVOLVED IN THIS EVENT WERE COUNSELED REGARDING THE PROPER DEVELOPMENT AND REVIEW OF A BLOCKING PERMIT.

[170] LIMERICK 1 DOCKET 50-352 LER 91-027
AN ENGINEERED SAFETY FEATURE ACTUATION DUE TO AN UNKNOWN CAUSE FOR AN ISOLATION VALVE ASSOCIATED WITH A PRIMARY CONTAINMENT H2/02 COMBUSTIBLE GAS ANALYZER.
EVENT DATE: 111291 REPORT DATE: 121191 NSSS: GE TYPE: BWR
VENDOR: CONTROL DIVISION OF AMERACE

(NSIC 223618) ON NOVEMBER 12, 1991, MAIN CONTROL ROOM OPERATORS OBSERVED THE INITIATION OF A SPURIOUS ACTUATION OF THE PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM (PCRVICS) AN ENGINEERED SAFETY FEATURE. THE INITIATION RESULTED IN CLOSURE OF THE PRIMARY CONTAINMENT H2/02 COMBUSTIBLE GAS ANALYZER SAMPLE LINES TO THE DRYWELL, AND AN ISOLATION SIGNAL TO THE NORMALLY CLOSED HYDROGEN RECOMBINER IN BOARD ISOLATION VALVES AND THE LOW FLOW NITROGEN MAKEUP ISOLATION VALVE. THE ACTUAL CONSEQUENCES OF THE EVENT WERE MINIMAL AND THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL TO THE ENVIRONMENT AS A RESULT OF THIS EVENT. THE PROXIMATE CAUSE OF THE ACTUATION WAS A RELAY INTERMITTENTLY CHANGING STATE; HOWEVER, THE CAUSE OF THE SPURIOUS RELAY OPERATION COULD NOT BE DETERMINED. THE DEFECTIVE RELAY WAS REPLACED. THE NEW RELAY WAS TESTED AND THE AFFECTED EQUIPMENT WAS DECLARED OPERABLE ON NOVEMBER 14, 1991. BENCH TESTING AND MONITORING OF THE ENERGIZED DEFECTIVE RELAY FOLLOWING REPLACEMENT IDENTIFIED NO PROBLEMS. SINCE THE TESTING OF THE RELAY REVEALED NO EQUIPMENT PROBLEMS AND NO SIMILAR FAILURES HAVE OCCURRED, THE FAILURE IS CONSIDERED AN ISOLATED EVENT AND NO FURTHER CORRECTIVE ACTIONS ARE PLANNED.

[171] LIMERICK 1 DOCKET 50-352 LER 91-028
INOPERABILITY OF THE HIGH PRESSURE COOLANT INJECTION SYSTEM DUE TO FAILURE OF THE INBOARD STEAM SUPPLY ISOLATION VALVE IN THE CLOSED POSITION.
EVENT DATE: 121891 REPORT DATE: 011792 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 223832) ON DECEMBER 18, 1991, FOLLOWING COMPLETION OF MAINTENANCE ON THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM STEAM SUPPLY OUTBOARD ISOLATION VALVE, HV-55-1F003, OPERATIONS PERSONNEL ATTEMPTED, UNSUCCESSFULLY, TO OPEN THE INBOARD ISOLATION VALVE, HV-55-1F002. SUBSEQUENT INVESTIGATION BY MAINTENANCE PERSONNEL LED TO THE DETERMINATION THAT THE VALVE MOTOR FAILED UPON CLOSURE OF THE VALVE ON DECEMBER 10, 1991. A UNIT 1 SHUTDOWN WAS INITIATED ON DECEMBER 18, 1991, AT 1800 HOURS, TO FACILITATE REPAIR OF THE VALVE. THE INABILITY TO OPEN HV-551F002 CONSTITUTED A CONDITION WHICH ALONE COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF THE HPCI SYSTEM. THE ACTUAL CONSEQUENCES OF THIS EVENT WERE MINIMAL IN THAT AN ACCIDENT CONDITION DID NOT OCCUR DURING THE TIME IN WHICH THE HPCI SYSTEM WAS INOPERABLE. THE CAUSE OF THIS EVENT WAS THE FAILURE OF THE HPCI STEAM SUPPLY INBOARD ISOLATION VALVE MOTOR. INVESTIGATION BY MAINTENANCE PERSONNEL LED TO THE CONCLUSION THAT THE HV-55-1F002 MOTOR FAILED DUE TO THE SPRING PACK BELLEVILLE WASHERS BINDING ON THE BEARING CARTRIDGE STEM. THE WASHERS PREVENTED THE TORQUE SWITCH FROM INTERRUPTING ELECTRICAL POWER TO THE MOTOR AND RESULTED IN ELECTRICAL POWER BEING CONTINUOUSLY SUPPLIED TO THE STALLED MOTOR UNTIL THE MOTOR WINDINGS FAILED. THE VALVE MOTOR AND SPRING PACK WERE REPLACED ON DECEMBER 20, 1991.

[172] LIMERICK 1 DOCKET 50-352 LER 91-029
PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM ACTUATIONS RESULTING FROM THE FAILURE OF A PCRVICS FUSE.
EVENT DATE: 122391 REPORT DATE: 012192 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)

(NSIC 223833) ON DECEMBER 23, 1991, VARIOUS ACTUATIONS OF THE UNIT 1 AND 2 PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEMS (PCRVICS), AND A UNIT 2 REACTOR ENCLOSURE SECONDARY CONTAINMENT ISOLATION OCCURRED DUE TO A BLOWN FUSE ON UNIT 2. THESE ARE ENGINEERED SAFETY FEATURE ACTUATIONS. THE FUSE WAS REPLACED AND ALL ISOLATIONS WERE RESET WITHIN 30 MINUTES. THE ACTUAL CONSEQUENCES OF THIS EVENT WERE MINIMAL. ALL SYSTEMS RESPONDED AS DESIGNED AND THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL TO THE ENVIRONMENT AS A RESULT OF THIS EVENT. THE CAUSE OF THE ISOLATIONS WAS THE FAILURE OF UNIT 2 PCRVICS FUSE B21H-F15A. THE ROOT CAUSE OF THE FUSE FAILURE IS UNKNOWN. THE BLOWN FUSE WILL BE SENT TO THE

MANUFACTURER FOR FAILURE ANALYSIS. A REVISION TO THIS LER WILL BE PROVIDED IF ANYTHING SIGNIFICANT IS DISCOVERED DURING THE FAILURE ANALYSIS.

[173] LIMERICK 1 DOCKET 50-352 LER 91-030
TECHNICAL SPECIFICATION LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENT FOR TEMPERATURE AND TIME EXCEEDED DURING REACTOR RECIRCULATION PUMP START.
EVENT DATE: 123091 REPORT DATE: 012992 NSSS: GE TYPE: BWR

(NSIC 223867) ON 12/30/91, AT 1023 HOURS, OPERATIONS PERSONNEL STARTED THE '1A' REACTOR RECIRCULATION PUMP IN CONJUNCTION WITH THE PERFORMANCE OF SURVEILLANCE TEST (ST) PROCEDURE ST-6-043-390-1, "REACTOR RECIRCULATION PUMP IDLE LOOP STARTUP TEMPERATURE AND FLOW CHECK." ST-6-043-390-1 SATISFIES TECHNICAL SPECIFICATIONS (TS) LIMITING CONDITION FOR OPERATION (LCO) 3.4.1.4 AND TS SURVEILLANCE REQUIREMENT (SR) 4.4.1.4. ON 12/30/91, AT APPROX. 1300 HOURS PLANT PERSONNEL DISCOVERED THAT INACCURATE TEMPERATURE AND TIME READINGS WERE RECORDED INTO THE ST, CAUSING THE 50 DEGREES FAHRENHEIT (F) TEMPERATURE DIFFERENTIAL LIMIT OF TS LCO3.4.1.4 AND THE 15 MINUTE TIME LIMIT OF TS SR 4.4.1.4 TO BE EXCEEDED BY 2 DEGREES F AND 4 MINUTES RESPECTIVELY; CONDITIONS PROHIBITED BY TS. THE '1A' RECIRCULATION PUMP WAS LEFT IN OPERATION SINCE THESE TS VIOLATIONS WERE DISCOVERED APPROX. 3 HOURS AFTER THE PUMP WAS PLACED IN SERVICE, AND IT WAS DETERMINED THAT SECURING THE PUMP AND A SUBSEQUENT RESTART WOULD ONLY RESULT IN FURTHER UNNECESSARY THERMAL TRANSIENTS. THE PRIMARY CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THE ACTUAL CONSEQUENCES OF THIS EVENT WERE MINIMAL. THE INDIVIDUAL INVOLVED IN THIS EVENT WAS COUNSELED, AND SPECIFIC PLANT PROCEDURES AND THE LICENSED OPERATOR TRAINING PROGRAM WILL BE EVALUATED TO PREVENT RECURRENCE OF A SIMILAR EVENT.

[174] LIMERICK 2 DOCKET 50-353 LER 91-017
HIGH PRESSURE COOLANT INJECTION SYSTEM DISCOVERED IN A DEGRADED CONDITION BECAUSE OF A LEAK IN AN OIL LINE TO THE TURBINE STOP VALVE.
EVENT DATE: 111591 REPORT DATE: 121691 NSSS: GE TYPE: BWR

(NSIC 223644) ON 11/15/91, THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM ENGINEER DISCOVERED AN OIL LEAK DURING A ROUTINE SYSTEM WALKDOWN WHICH WOULD HAVE PREVENTED THE HPCI SYSTEM FROM FULFILLING ITS SAFETY FUNCTION NEEDED TO MAINTAIN THE REACTOR IN A SAFE SHUTDOWN CONDITION AND MITIGATE THE CONSEQUENCES OF AN ACCIDENT. THE ACTUAL CONSEQUENCES OF THIS EVENT WERE MINIMAL BECAUSE AN ACCIDENT CONDITION DID NOT OCCUR DURING THE TIME IN WHICH THE HPCI SYSTEM WAS INOPERABLE OR COULD HAVE BEEN DEGRADED AND BECAUSE SUFFICIENT EMERGENCY CORE COOLING SYSTEMS WERE AVAILABLE. THE PERIOD OF TIME IN WHICH THE HPCI SYSTEM MAY HAVE BEEN DEGRADED DUE TO THE OIL LEAK IS LIMITED TO THE TIME BETWEEN 11/13/91, WHEN THE HPCI SYSTEM WAS VERIFIED TO BE OPERABLE BY SUCCESSFUL PERFORMANCE OF ST-6-055-230-2 , "HPCI PUMP, VALVE AND FLOW TEST," AND 11/15/91, WHEN THE OIL LEAK WAS DISCOVERED BY THE HPCI SYSTEM ENGINEER. THE CAUSE OF THE LEAK IN A LINE THAT SUPPLIES OIL TO THE TURBINE STOP VALVE IS STILL UNDER INVESTIGATION. THE LINE CONSISTING OF A BRAIDED STAINLESS STEEL HOSE WAS REMOVED AND SENT TO A LABORATORY FOR PERFORMANCE OF A FAILURE ANALYSIS. THE RESULTS OF THE INVESTIGATION WILL BE PROVIDED IN A SUPPLEMENT TO THIS LER WHICH IS EXPECTED TO BE SUBMITTED BY 4/1/92. THE LINE WAS REPLACED WITH A HARD PIPE LINE SIMILAR TO THE LINES ON THE UNIT 1 HPCI SYSTEM.

[175] LIMERICK 2 DOCKET 50-353 LER 92-001
HIGH PRESSURE COOLANT INJECTION SYSTEM, D24 EMERGENCY DIESEL GENERATOR, 'D' RESIDUAL HEAT REMOVAL SYSTEM, AND 'D' CORE SPRAY INOPERABLE DUE TO A BLOWN FUSE.
EVENT DATE: 010492 REPORT DATE: 012892 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 223868) ON JANUARY 4, 1992, A UNIT 2 PRESSURE INDICATING SWITCH SHORTED TO GROUND CAUSING A BLOWN POWER SUPPLY FUSE THAT MADE THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM, THE D24 EMERGENCY DIESEL GENERATOR, THE 'D' RESIDUAL HEAT REMOVAL SYSTEM, AND THE 'D' CORE SPRAY SYSTEM INOPERABLE. AFTER REMOVING THE FAILED PRESSURE INDICATING SWITCH, THE FUSE WAS REPLACED. WHEN THE HPCI SYSTEM

INSTRUMENTATION WAS REENERGIZED AN ANTICIPATED HPCI SYSTEM ISOLATION OCCURRED. THE HPCI SYSTEM ISOLATION WAS RESET AND THE HPCI SYSTEM WAS DECLARED OPERABLE. THE ACTUAL CONSEQUENCES OF THIS EVENT WERE MINIMAL. THE FAILED PRESSURE INDICATING SWITCH WAS REPLACED ON JANUARY 4, 1991. SIMILAR PRESSURE INDICATING SWITCHES IN THE SAME CARD FILE WERE INSPECTED AND CONFIRMED TO NOT HAVE THE SAME CONDITION. THE FAILED PRESSURE INDICATING SWITCH HAS BEEN SENT TO THE MANUFACTURER FOR PERFORMANCE OF A FAILURE ANALYSIS. THE RESULTS OF THE FAILURE ANALYSIS CONDUCTED BY THE MANUFACTURER WILL BE REVIEWED TO DETERMINE IF ADDITIONAL CORRECTIVE ACTION NEEDS TO BE TAKEN .

[176] MAINE YANKEE DOCKET 50-309 LER 91-012
PLANT TRIP DUE TO CONDENSATE PUMP MOTOR FAILURE.
EVENT DATE: 112291 REPORT DATE: 122091 NSSS: CE TYPE: PWR
VENDOR: ALLIS CHALMERS

(NSIC 223674) A REACTOR TRIP FROM 100% POWER OCCURRED ON NOVEMBER 22, 1991 AT 0957 DUE TO THE FAILURE OF A CONDENSATE PUMP MOTOR (P-27B). INTERNAL MOTOR LEAD ARCING CAUSED A DROP IN VOLTAGE ON THE BUS. THE CONDENSATE PUMP MOTOR BREAKER TRIPPED ON OVERCURRENT. THE TRANSIENT RESULTED IN A TRIP OF THE STEAM TURBINE DRIVEN FEEDWATER PUMP. THIS ACTION TRIPPED THE MAIN TURBINE. THE REACTOR PROTECTIVE SYSTEM TRIPPED THE REACTOR ON LOSS OF LOAD. ALL SAFETY SYSTEMS PERFORMED AS EXPECTED. THIS LER IS SIMILAR TO NUMBER 91-010, HOWEVER, THE CAUSE OF THE LER 91-010 MOTOR FAILURE IS NOT IDENTICAL TO THIS LER.

[177] MCGUIRE 1 DOCKET 50-369 LER 91-014
TRAIN "A" OF THE CONTAINMENT SPRAY SYSTEM WAS INOPERABLE DUE TO A MISPOSITIONED VALVE AS A RESULT OF AN UNKNOWN.
EVENT DATE: 100491 REPORT DATE: 121191 NSSS: WE TYPE: PWR

(NSIC 223621) ON SEPTEMBER 30, 1991, VALVE 1RN-951, CONTAINMENT SPRAY (NS) SYSTEM PUMP 1A AIR HANDLING UNIT (AHU) OUTLET CONTROL, WAS FOUND MISPOSITIONED IN THE CLOSED POSITION DURING MAINTENANCE ACTIVITIES. UNIT 1 WAS IN MODE 6 (REFUELING), IN DAY 10 OF A PROJECTED 71 DAY OUTAGE AT THE TIME OF DISCOVERY. THE MISPOSITION OF VALVE 1RN-951 WAS BROUGHT TO THE ATTENTION OF OPERATIONS (OPS) CONTROL ROOM PERSONNEL DURING DISCUSSION BY OPS AND MAINTENANCE (MNT) PERSONNEL OF VALVE 1RN-949, RESIDUAL HEAT REMOVAL (ND) SYSTEM PUMP 1A AHU OUTLET CONTROL, WHICH WAS DISCOVERED MISPOSITIONED ON OCTOBER 4, 1991. DURING INVESTIGATION OF THE EVENT, IT WAS DETERMINED THAT VALVE 1RN-949 WAS IN THE CORRECT POSITION DURING THE TIME THAT THE ND SYSTEM WAS REQUIRED TO BE OPERABLE. VALVE 1RN-951 HAD BEEN IN THE INCORRECT POSITION FOR AN UNKNOWN PERIOD OF TIME AFTER SEPTEMBER 4, 1991. THIS RESULTED IN TRAIN 1A OF THE NS SYSTEM BEING INOPERABLE FOR AN UNKNOWN PERIOD BETWEEN SEPTEMBER 4, 1991, AND OCTOBER 4, 1991. THE NS SYSTEM HAD BEEN REQUIRED TO BE OPERABLE SEPTEMBER 4, 1991, THROUGH SEPTEMBER 21, 1991, UNTIL UNIT 1 ENTERED MODE 5 (COLD SHUTDOWN). VALVE 1RN-949 AND 1RN-951 WERE SUBSEQUENTLY RETURNED TO THE CORRECT THROTTLED POSITION. THIS EVENT IS DESIGNED A CAUSE OF UNKNOWN BECAUSE VALVE 1RN-951 WAS FOUND IN THE CLOSED POSITION AND NO DEFINITE OR PROBABLE CAUSE COULD BE DETERMINED.

[178] MCGUIRE 1 DOCKET 50-369 LER 91-017
THE CONTROL AREA VENTILATION SYSTEM WAS INOPERABLE DUE TO A DESIGN DEFICIENCY.
EVENT DATE: 101691 REPORT DATE: 120291 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 223537) ON OCTOBER 16, 1991, MCGUIRE NUCLEAR STATION SYSTEM ENGINEERING PERSONNEL CONDUCTED A SELF TRAINING REVIEW OF THE CONTROL AREA VENTILATION SYSTEM. DURING THE REVIEW, SYSTEM ENGINEERING PERSONNEL DISCOVERED THE SMOKE PURGE EXHAUST FAN WOULD NOT RECEIVE A TRIP SIGNAL ON A DIESEL GENERATOR SEQUENCER ACTUATION. SYSTEM ENGINEERING PERSONNEL SUSPECTED THE OPERATION OF THE SMOKE PURGE EXHAUST FAN COULD ADVERSELY AFFECT THE ABILITY OF THE CONTROL AREA VENTILATION TO FUNCTION AS DESIGNED DURING AN EMERGENCY SITUATION. OPERATIONS MANAGEMENT WAS NOTIFIED BY SYSTEM ENGINEERING PERSONNEL OF THE POTENTIAL PROBLEM. SUBSEQUENT TESTING CONFIRMED THE CONTROL AREA VENTILATION SYSTEM WOULD NOT MAINTAIN A POSITIVE PRESSURE IN THE CONTROL ROOM AS DESIGNED WITH THE SMOKE PURGE

EXHAUST FAN IN OPERATION. THIS EVENT IS ASSIGNED A CAUSE OF DESIGN DEFICIENCY DUE TO AN UNANTICIPATED COMPONENT INTERACTION. OPERATIONS PERSONNEL PLACED A SAFETY TAG ON THE CONTROL SWITCH FOR THE SMOKE PURGE EXHAUST FAN TO PREVENT OPERATION OF THE FAN. UNIT 1 WAS DEFUELED AND UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER AT THE TIME OF DISCOVERY OF THE EVENT.

[179] MCGUIRE 1 DOCKET 50-369 LER 91-016
QUALIFIED FUEL ASSEMBLIES WERE STORED IMPROPERLY IN THE SPENT FUEL POOL DUE TO A DEFECTIVE PROCEDURE.
EVENT DATE: 102491 REPORT DATE: 112591 NSSS: WE TYPE: PWR

(NSIC 223536) WHILE REVIEWING TECHNICAL SPECIFICATION SECTION 3.9.12, MCGUIRE REACTOR UNIT PERSONNEL IDENTIFIED 11 FUEL ASSEMBLIES THAT HAD BEEN STORED IN THE UNIT 1 SPENT FUEL POOL IN A MANNER CONTRARY TO THE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.9.12. THIS LIMITING CONDITION FOR OPERATION REQUIRES, IN PART, THAT FUEL STORED IN REGION 2 OF THE SPENT FUEL POOL SHALL UNDERGO 16 DAYS OF DECAY, AND IF A CHECKERBOARD PATTERN IS EMPLOYED FOR UNQUALIFIED FUEL, ONE ROW BETWEEN NORMAL STORAGE LOCATIONS AND CHECKERBOARD STORAGE LOCATIONS WILL BE VACANT. THE VACANT ROW PROVISION OF THE SPECIFICATION WAS NOT SATISFIED FROM MARCH 23, 1990 THROUGH OCTOBER 23, 1991. AT THE TIME OF DISCOVERY AT 0900 ON OCTOBER 24, 1991, UNIT 1 WAS DEFUELED, AND UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER. THIS EVENT HAS BEEN ASSIGNED A CAUSE OF DEFECTIVE PROCEDURE. THE FUEL ASSEMBLIES IN QUESTION WERE IMMEDIATELY MOVED TO POSITIONS TO ESTABLISH THE REQUIRED VACANT ROW.

[180] MCGUIRE 1 DOCKET 50-369 LER 91-018
BOTH TRAINS OF THE ANNULUS VENTILATION SYSTEM WERE INOPERABLE DUE TO MANAGEMENT DEFICIENCY AND DEFICIENT COMMUNICATION.
EVENT DATE: 121891 REPORT DATE: 011792 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 223800) ON DECEMBER 18, 1991, INSTRUMENTATION AND ELECTRICAL PERSONNEL WERE PERFORMING THE SEMIANNUAL CALIBRATION ON LOOP 2IAE-9060, LOWER PERSONNEL AIR LOCK LEAK RATE MONITOR. DURING THIS CALIBRATION, DOOR AD 3321, ANNULUS VENTILATION BYPASS DOOR, WAS LATCHED OPEN TO ALLOW FOR COMMUNICATION BETWEEN PERSONNEL WORKING ON THE PNEUMATIC MODULE AND REMOTE CONTROL UNIT ASSOCIATED WITH THE LOOP. THIS DOOR IS A SELECTED LICENSEE COMMITMENT FIRE DOOR AND SERVES AS A PRESSURE BOUNDARY FOR THE ANNULUS VENTILATION (VE) SYSTEM. THE DOOR MUST BE POSTED WITH A FIRE BARRIER TAG, AND ADEQUATE COMPENSATORY MEASURES MUST BE IMPLEMENTED TO ENSURE OPERABILITY OF THE VE SYSTEM IF THE DOOR IS OPENED FOR PERIODS LONGER THAN NORMAL ACCESS DURING MODES 1 (POWER OPERATION), 2 (STARTUP), 3 (HOT STANDBY), AND 4 (HOT SHUTDOWN). DURING THE CALIBRATION NO FIRE TAGS OR VE SYSTEM COMPENSATORY MEASURES WERE IN EFFECT. THEREFORE, BOTH TRAINS OF THE VE SYSTEM WERE INOPERABLE WHILE THE DOOR WAS OPEN. UNIT 2 WAS IN MODE 1 AT 100 PERCENT POWER AT THE TIME OF THE EVENT DISCOVERY. FURTHER INVESTIGATION HAS REVEALED THAT DOOR AD 3321 ON UNIT 2 AND DOOR AD 3311, ANNULUS VENTILATION BYPASS DOOR, ON UNIT 1 HAVE BEEN LATCHED OPEN WHENEVER THE SEMI-ANNUAL CALIBRATION OF THE RESPECTIVE LOWER PERSONNEL AIR LOCK LEAK RATE MONITOR HAS BEEN PERFORMED.

[181] MILLSTONE 1 DOCKET 50-245 LER 91-002 REV 01
UPDATE ON LOW PRESSURE COOLANT INJECTION AND CORE SPRAY SYSTEM DESIGN TEMPERATURES.
EVENT DATE: 020891 REPORT DATE: 010692 NSSS: GE TYPE: BWR

(NSIC 223722) ON 2/8/91, AT 1725 HOURS, WITH THE PLANT AT 100% POWER (530F AND 1030 PSIG), AN INCONSISTENCY BETWEEN THE DESIGN TEMPERATURES ASSOCIATED WITH THE LOW PRESSURE COOLANT INJECTION (LPCI) AND CORE SPRAY PIPING, PIPING SUPPORTS, AND COMPONENTS WAS REPORTED. IT WAS DISCOVERED THAT TEMPERATURES FOR PORTIONS OF THE LPCI AND CORE SPRAY SYSTEM USED TO ANALYZE PIPING THAT WOULD BE SUBJECTED TO HIGH WATER TEMPERATURES DURING A DESIGN BASIS LOSS OF COOLANT ACCIDENT WERE NOT CONSISTENT WITH THE ACCIDENT ANALYSIS TEMPERATURE PROFILE. THE ORIGINAL DESIGN BASIS TEMPERATURE FOR THE POST ACCIDENT PEAK TORUS TEMPERATURES WAS ESTABLISHED AT 203F AND WAS BASED UPON AN INITIAL SERVICE WATER TEMPERATURE OF 75F.

ENGINEERING EVALUATIONS WERE IN PROGRESS TO QUALIFY THE VARIOUS COMPONENTS TO TEMPERATURES ABOVE THE EXISTING DESIGN BASIS VALUE OF 203F. DURING THE PERFORMANCE OF THIS ANALYSIS, ENGINEERS DISCOVERED A DISCREPANCY IN THE PIPING DESIGN TEMPERATURE. PORTIONS OF THE LPCI AND CORE SPRAY SYSTEMS WERE DESIGNED FOR 165F INSTEAD OF THE DESIGN BASIS TEMPERATURE OF 203F. AN OPERABILITY DETERMINATION WAS COMPLETED WHICH DEMONSTRATED THAT ALL COMPONENTS ASSOCIATED WITH THE LPCI AND CORE SPRAY SYSTEMS WERE OPERABLE. NO SAFETY SYSTEMS WERE REQUIRED TO FUNCTION AS A RESULT OF THIS EVENT AND NO SAFETY CONSEQUENCES RESULTED FROM THIS EVENT.

[182] MILLSTONE 1 DOCKET 50-245 LER 91-019 REV 01
UPDATE ON EMERGENCY CORE COOLING SYSTEM EQUIPMENT ROOM TEMPERATURE.
EVENT DATE: 061091 REPORT DATE: 010692 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223723) ON 6/10/91, AT 1530 HOURS, WITH THE PLANT IN COLD SHUTDOWN (82F AND 0 PSIG), AN ONGOING ENGINEERING ANALYSIS CONCLUDED THAT THE LOW PRESSURE COOLANT INJECTION (LPCI) SYSTEM AND THE CORE SPRAY SYSTEM MOTORS COULD NOT BE DEMONSTRATED TO BE OPERABLE DURING POSTULATED ACCIDENT CONDITIONS. THIS CONCLUSION WAS BASED UPON HIGHER CALCULATED POST ACCIDENT TORUS TEMPERATURES COMBINED WITH HIGHER THAN PREVIOUSLY ASSUMED AMBIENT ROOM TEMPERATURES. THE COOLING WATER SUPPLY FOR THE LPCI AND CORE SPRAY MOTOR BEARINGS IS SUPPLIED FROM THE CONTAINMENT TORUS WATER VIA THE PUMP DISCHARGE PIPING. THE POSTULATED INCREASE IN THE TORUS AND REACTOR BUILDING TEMPERATURES ADVERSELY AFFECTED THE OPERATION OF THE UPPER MOTOR BEARINGS. ON 6/19/91, AT 1510, WITH THE PLANT IN COLD SHUTDOWN, IT WAS ALSO DETERMINED AS A RESULT OF THE POSTULATED INCREASE IN REACTOR BUILDING TEMPERATURES NEAR THE TORUS AREA THAT THE WIDE RANGE TORUS LEVEL INSTRUMENTS DID NOT MEET THE ENVIRONMENTAL TEMPERATURE CONDITIONS POSTULATED DURING ACCIDENT CONDITIONS. ON 12/11/91, AT 1100 HOURS, WITH THE PLANT IN COLD SHUTDOWN, IT WAS DETERMINED THAT INSULATION REMOVED FROM THE LPCI AND CORE SPRAY SYSTEM PIPING LOCATED IN THE REACTOR BUILDING COULD ADVERSELY AFFECT THE POST ACCIDENT REACTOR BUILDING TEMPERATURE PROFILE. THE PIPING INSULATION WAS REMOVED AS A RESULT OF AN ASBESTOS REMOVAL PROGRAM.

[183] MILLSTONE 1 DOCKET 50-245 LER 91-025 REV 01
UPDATE ON CONTROL ROD FAILURE TO SCRAM.
EVENT DATE: 081691 REPORT DATE: 122091 NSSS: GE TYPE: BWR
VENDOR: ASCO VALVES

(NSIC 223724) ON 8/16/91, WHILE AT 23% POWER AND 530F, 8 CONTROL RODS EXHIBITED SIGNIFICANT DELAYS IN SCRAM INSERTION TIMES AND APPROACHED THE TECH SPEC REQUIREMENT FOR A MAXIMUM SCRAM TIME OF 7 SECONDS FOR 90% INSERTION. SUBSEQUENT REVIEW OF THE TWO CONTROL RODS PREVIOUSLY REPORTED AS FAILING THE 90% INSERTION TIME INDICATED THAT, ALTHOUGH THE FULL INSERTION TIMES EXCEED 7 SECONDS, THE 90% INSERTION TIMES WERE ACCEPTABLE. HOWEVER, A 2 X 2 ARRAY WAS IDENTIFIED WHICH WOULD HAVE FAILED THE 2 X 2 ARRAY ANALYSIS. THE AFFECTED ARRAY CONTAINED TWO OF THE SLOW CONTROL RODS AND WOULD HAVE FAILED THE TECH SPEC REQUIREMENTS FOR THE 5, 20, AND 50% TIMES FOR THE 3 FASTEST RODS OF A 2 X 2 ARRAY. THE SLOW RODS WERE IDENTIFIED DURING THE PERFORMANCE OF BEGINNING OF CYCLE CONTROL ROD SCRAM TIME TESTING. AS EACH WAS IDENTIFIED, THE CONTROL ROD WAS LEFT FULLY INSERTED AND ELECTRICALLY DISARMED. THE DELAYED SCRAM WAS DETERMINED TO BE CAUSED BY THE SLOW VENTING OF AIR FROM THE SCRAM PILOT SOLENOID VALVES (117 AND 118) LOCATED ON THE ASSOCIATED HYDRAULIC CONTROL UNITS. THE SOLENOID VALVES WERE REBUILT AND THE CONTROL RODS SCRAM TIMED. ALL AFFECTED CONTROL RODS SUCCESSFULLY PASSED WITH NOMINAL SCRAM TIMES AND WERE RETURNED TO SERVICE. THERE WERE NO SAFETY CONSEQUENCES FROM THIS EVENT.

[184] MILLSTONE 1 DOCKET 50-245 LER 91-027 REV 01
UPDATE ON 345KV SYSTEM INSTABILITY.
EVENT DATE: 110191 REPORT DATE: 013192 NSSS: GE TYPE: BWR

(NSIC 223953) ON 11/1/91, AT 1340 HOURS, WITH THE PLANT SHUTDOWN (142F AND 0 PSIG), AN ENGINEERING EVALUATION CONCLUDED A SINGLE ELECTRICAL FAULT COULD RESULT

IN A LOSS OF ALL OFFSITE POWER. THIS CONDITION IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(II). DURING A RECENT DESIGN REVIEW IN AUGUST 1991 OF THE MILLSTONE UNIT ONE 345KV TRANSMISSION SYSTEM FOR STABILITY, IT WAS RECOGNIZED A TRANSFER TRIP TIME OF 8.5 CYCLES FOR THE SWITCHYARD BREAKERS TO OPEN UNDER A THREE PHASE BOLTED FAULT INSIDE THE MAIN STEP-UP TRANSFORMER COULD RESULT IN LOSS OF OFFSITE POWER SOURCES TO MILLSTONE ONE AND LOSS OF THE MILLSTONE SWITCHYARD. THE LOSS OF THE MILLSTONE SWITCHYARD COULD THEN CAUSE INSTABILITY IN THE 345KV SYSTEM IN THE NORTHEAST. A PLANT DESIGN CHANGE HAS BEEN IMPLEMENTED TO CORRECT THE DEFICIENCY. THIS EVENT WAS THE SUBJECT OF A REPORTABILITY EVALUATION IN ACCORDANCE WITH NORTHEAST UTILITIES PROCEDURE (NEO 2.25). THE EVALUATION WAS INITIATED AUGUST 21, 1991, HOWEVER, THE ACTUAL REPORT WAS NOT MADE UNTIL 11/1/91 DUE TO A FAILURE TO RECOGNIZE THE EXISTENCE OF SUFFICIENT TECHNICAL DATA TO MAKE A REPORTABILITY DETERMINATION. IN ADDITION THE REPORTABILITY TIMELINESS LIMITS WERE NOT MET PER PROCEDURE NEO 2.25. THESE WERE AN OVERSIGHT. RECENT IMPROVEMENTS HAVE BEEN INITIATED TO CORRECT THE TIMELINESS ISSUE.

[185] MILLSTONE 1 DOCKET 50-245 LER 91-028
SEISMIC INTERACTION ON TURBINE BUILDING SECONDARY CLOSED COOLING WATER SYSTEM.
EVENT DATE: 110791 REPORT DATE: 120991 NSSS: GE TYPE: BWR

(NSIC 223599) ON 11/7/91, AT 1300 HOURS, WITH THE PLANT IN COLD SHUTDOWN (142 DEG. F AND 0 PSIG) FOR INSPECTION OF PIPING SUSCEPTIBLE TO EROSION/CORRSOION, IT WAS DETERMINED THAT A POSTULATED SEISMIC EVENT COULD RESULT IN INOPERABILITY OF THE EMERGENCY DIESEL GENERATOR AND THE FEEDWATER COOLANT INJECTION SYSTEM. THE POSTULATED SEISMIC EVENT COULD HAVE DISRUPTED INADEQUATELY ANCHORED ROOM COOLERS AND RESULTED IN A SUBSEQUENT LOSS OF TURBINE BUILDING SECONDARY CLOSED COOLING WATER (TBSCCW). TBSCCW HEAT REMOVAL CAPABILITY IS REQUIRED FOR LONG TERM OPERATION OF THE EMERGENCY DIESEL GENERATOR AND THE FEEDWATER COOLANT INJECTION PUMPS. MODIFICATIONS WERE PERFORMED TO RESTORE THE ROOM COOLERS ANCHORAGE TO ITS CORRECT CONFIGURATION. NO SAFETY SYSTEMS WERE REQUIRED TO FUNCTION AS A RESULT OF THIS EVENT AND NO SAFETY CONSEQUENCES RESULTED FROM THIS EVENT.

[186] MILLSTONE 1 DOCKET 50-245 LER 91-029
FIRE DOOR FOUND OPEN WITH NO FIRE WATCH PRESENT.
EVENT DATE: 121491 REPORT DATE: 011392 NSSS: GE TYPE: BWR

(NSIC 223845) ON 12/14/91, AT 1030 HOURS, WITH THE PLANT IN COLD SHUTDOWN (130F AND 0 PSIG), A HATCHWAY DOOR FROM THE TURBINE BUILDING 34'6" ELEVATION TO THE CONDENSER BAY MEZZANINE AREA WAS FOUND OPEN WITH NO FIRE WATCH POSTED. THE HATCHWAY DOOR WAS IMMEDIATELY CLOSED. THE TURBINE BUILDING 34'6" ELEVATION CONTAINS SWITCHGEAR THAT SUPPLIES EQUIPMENT REQUIRED TO BE OPERABLE WITH FUEL IN THE REACTOR VESSEL. THERE IS NO EQUIPMENT IMPORTANT TO SAFETY ON THE CONDENSER BAY SIDE OF THE HATCHWAY DOOR. THE AUTOMATIC FIRE DETECTION SYSTEM FOR THE SWITCHGEAR AREA AND THE AUTOMATIC FIRE DETECTION AND SUPPRESSION SYSTEMS FOR THE CONDENSER BAY WERE OPERABLE AT THE TIME OF THE EVENT. THE AVAILABLE FIRE PROTECTION FEATURES AND THE LOW COMBUSTIBLE LOADING OF THE AREAS MINIMIZED ANY POTENTIAL ADVERSE IMPACT OF MAINTAINING THE PLANT IN A SAFE SHUTDOWN CONDITION. NO SAFETY SYSTEMS WERE REQUIRED TO FUNCTION AS A RSULT OF THIS EVENT AND NO SAFETY CONSEQUENCES RESULTED FROM THIS EVENT.

[187] MILLSTONE 2 DOCKET 50-336 LER 90-020 REV 01
UPDATE ON MISSED SERVICE WATER SURVEILLANCE.
EVENT DATE: 120490 REPORT DATE: 012892 NSSS: CE TYPE: PWR

(NSIC 223922) ON 12/4/90, WITH THE PLANT IN MODE 1 AT 575F, 2260 PSI, AND 100% POWER, OPERATIONS DEPARTMENT PERSONNEL DISCOVERED THAT SURVEILLANCE PROCEDURE SP 2612D-1, FACILITY 2 SERVICE WATER SYSTEM LINEUP AND OPERABILITY TEST, HAD NOT BEEN COMPLETED IN ITS ENTIRETY DURING THE REQUIRED SURVEILLANCE INTERVAL. THIS SURVEILLANCE TESTS THE OPERABILITY OF SEVERAL REMOTELY OPERATED VALVES, AND VERIFIES THE CORRECT POSITION OF A LARGE NUMBER OF MANUALLY OPERATED SERVICE WATER VALVES. THE POSITION OF TWO VALVES HAD NOT BEEN RECORDED DUE TO ONGOING WORK IN EACH CASE. EXCEPT FOR THESE TWO VALVES, THE SURVEILLANCE HAD BEEN COMPLETED AS EXPECTED. SIMILAR EVENTS: NONE.

[188] MILLSTONE 2 DOCKET 50-336 LER 91-010 REV 01
UPDATE ON REANALYSIS OF MAIN STEAM LINE BREAK EXCEEDS CONTAINMENT DESIGN LIMITS.
EVENT DATE: 101891 REPORT DATE: 011792 NSSS: CE TYPE: PWR
VENDOR: COPES-VULCAN, INC

(NSIC 223873) ON 10/18/91, AT 1305 HOURS, WITH THE PLANT IN MODE 1 AT 100% POWER, A REPORTABILITY DETERMINATION WAS MADE CONCERNING A REANALYSIS OF THE MAIN STEAM LINE BREAK EVENT INSIDE THE CONTAINMENT. THE REANALYSIS HAS CONFIRMED THAT THE ASSUMPTIONS MADE FOR THE EXISTING (1979) MAIN STEAM LINE BREAKER ANALYSIS WERE NON-CONSERVATIVE WITH RESPECT TO POWER LEVEL, BREAK SIZE, AND SINGLE ACTIVE FAILURE. USING MORE RESTRICTIVE ASSUMPTIONS, DESIGN LIMITS FOR CONTAINMENT PRESSURE AND TEMPERATURE COULD BE EXCEEDED. NNECO DETERMINED THAT THIS CONDITION WAS REPORTABLE AS A CONDITION OUTSIDE THE DESIGN BASIS OF THE PLANT. AN IMMEDIATE REPORT WAS MADE TO THE NRC, AND THE UNIT IMMEDIATELY COMMENCED AN ORDERLY DOWNPOWER TO APPROXIMATELY 3% POWER (MODE 2). THE EXISTING MAIN STEAM LINE BREAKER ANALYSIS IS ACCEPTABLE FOR MODE 2 OPERATION. A JUSTIFICATION FOR CONTINUED OPERATION (JCO) WAS DEVELOPED TO ALLOW THE UNIT TO RETURN TO POWER OPERATION BY STATIONING A DEDICATED REACTOR OPERATOR TO CLOSE THE MAIN FEEDWATER BLOCK VALVES FOLLOWING ANY REACTOR TRIP. THIS JCO DOCUMENTS THE BASIS FOR REASONABLE ASSURANCE THAT, WITH THE ACTIONS OF A DEDICATED OPERATOR, CONTAINMENT PRESSURE WILL REMAIN BELOW THE DESIGN BASIS VALUE FOR ALL POSTULATED MAIN STEAM LINE BREAK EVENTS.

[189] MILLSTONE 2 DOCKET 50-336 LER 91-012
MANUAL REACTOR TRIP DUE TO PLANT CONDITIONS RESULTING FROM A RUPTURE IN THE REHEATER DRAIN TANK TO HIGH PRESSURE FEEDWATER HEATER PIPE.
EVENT DATE: 110691 REPORT DATE: 120691 NSSS: CE TYPE: PWR

(NSIC 223614) ON NOVEMBER 6, 1991, WITH THE PLANT OPERATING AT 100% POWER, AN EIGHT INCH DIAMETER PIPE RUPTURED AT APPROXIMATELY 0613. THE SUBJECT LINE CONTAINS PRESSURIZED, HEATED WATER AND SERVES AS A DRAIN LINE FROM THE FIRST STAGE REHEATER DRAIN TANK NO. 1B TO THE 1B HIGH PRESSURE FEEDWATER HEATER. THE WATER IN THIS LINE IS NORMALLY 470 PSIG AT 463F. THE LEAK RELEASED STEAM INTO THE TURBINE BUILDING. THE CONTROL ROOM RECEIVED A CALL THAT THERE WAS A "LOUD NOISE" IN THE TURBINE BUILDING. THIS INITIATED A REVIEW OF CONTROL ROOM INSTRUMENTATION WHICH INDICATED MOISTURE SEPARATOR REHEATER CHANGES, SLIGHT COOLDOWN, SLIGHT MWE OUTPUT DECREASE AND A 15 SECOND CALORIMETRIC INCREASE TO 2722 MWTH. A VISUAL INSPECTION OF THE TURBINE BUILDING CONFIRMED THE STEAM LEAK. AT 0649 THE SCO ORDERED A MANUAL REACTOR TRIP BASED ON ALARMS AND INDICATIONS. THE REACTOR AND TURBINE WERE TRIPPED. THE SHIFT CARRIED OUT EOP 2525, "STANDARD POST TRIP ACTIONS." APPROXIMATELY ONE MINUTE AFTER THE TRIP THE SS ORDERED THE MAIN STEAM ISOLATION VALVES SHUT AND THE OPENING OF THE CONDENSER DUMP VALVES TO DEPRESSURIZE THE MAIN STEAM HEADER. AFTER EOP 2525 WAS COMPLETED, AND IN-HAND VERIFIED, THE SHIFT ENTERED EOP 2536, "EXCESS STEAM DEMAND." IN ORDER TO ASSESS THE SOURCE OF THE LEAK, THE REMAINING CONDENSATE PUMP WAS SECURED AND CONDENSER VACUUM WAS BROKEN.

[190] MILLSTONE 3 DOCKET 50-423 LER 91-024 REV 01
UPDATE ON TRIAXIAL PEAK RECORDING ACCELEROGRAPH RANGE NOT IN COMPLIANCE WITH TECHNICAL SPECIFICATIONS DUE TO DEFICIENT PROCEDURE.
EVENT DATE: 031891 REPORT DATE: 123191 NSSS: WE TYPE: PWR
VENDOR: TERRA TECHNOLOGY CORP.

(NSIC 223758) ON 9/5/91, AT 1045 HOURS, WHILE SHUTDOWN IN MODE 5 (COLD SHUTDOWN), AT 94F AND ATMOSPHERIC PRESSURE, INSTRUMENTATION & CONTROLS PERSONNEL DISCOVERED THAT THE +/-2G RANGE OF THE "A" SAFETY INECTION ACCUMULATOR TANK TRIAXIAL PEAK ACCELEROGRAPH, 3ERS-PAS28, WAS NOT IN COMPLIANCE WITH THE +/-1G RANGE REQUIREMENT OF TECHNICAL SPECIFICATIONS (TS) TABLE 3.3-7, SEISMIC MONITORING INSTRUMENTATION. THE INCORRECT +/-2G ACCELEROGRAPH WAS IDENTIFIED DURING A CALIBRATION DOCUMENTATION REVIEW AS PART OF THE INVESTIGATION INTO RELIABILITY CONCERNS WITH RESPECT TO THE UNITS INSTALLED IN THE PLANT. THE INCORRECT RANGE ACCELEROGRAPH HAD BEEN INSTALLED SINCE 3/18/87. NO IMMEDIATE ACTIONS WERE REQUIRED BY PLANT OPERATORS IN RESPONSE TO THIS EVENT. THE ROOT CAUSE OF THE EVENT IS PROCEDURAL DEFICIENCY. THE SURVEILLANCE PROCEDURE DID NOT REQUIRE VERIFICATION OF NAMEPLATE

DATA AGAINST THE TS REQUIREMENTS AS PART OF THE INSTALLATIO EVOLUTION. A +/-1G RANGE ACCELEROGRAPHY WAS INSTALLED BY 9/6/91, TO COMPLY WITH THE TS. PROCEDURE GUIDANCE HAS BEEN STRENGTHENED TO REQUIRE VERIFICATION OF RANGE. SEPARATE STOCK CODES HAVE BEEN PROVIDED FOR THE DIFFERENT RANGES. THE RVIEW OF THE EVENT REVEALED THAT A +/-2G UNIT IS MORE SUITABLE. A +/-2G UNIT WAS INSTALLED ON 11/7/91. A TS CHANGE REQUEST HAS BEEN SUBMITTED.

[191] MILLSTONE 3 DOCKET 50-423 LER 91-019 REV 01
UPDATE ON COMPLETION OF SHUTDOWN PER TECHNICAL SPECIFICATION DUE TO SERVICE WATER MUSSEL FOULING.
EVENT DATE: 072591 REPORT DATE: 122791 NSSS: WE TYPE: PWR

(NSIC 223757) AT 2000 HOURS ON 7/25/91, A PLANT SHUTDOWN WAS COMPLETED AS REQUIRED BY PLANT TECH SPEC 3.7.4 WHEN THE PLANT ENTERED MODE 3 (HOT STANDBY), AT APPROX. 558F AND 2260 PSIA. TS 3.7.4 REQUIRES 2 OPERABLE SERVICE WATER (SWP) TRAINS WHILE IN MODES 1 (POWER OEPRATIONS) THROUGH 4 (HOT SHUTDOWN), OR AFTER 72 HOURS, AT LEAST BE IN HOT STANDBY (MODE 3) WITHIN THE NEXT 6 HOURS, AND BE IN COLD SHUTDOWN (MODE 5) WITHIN THE FOLLOWING 30 HOURS. A PLANT SHUTDOWN WAS INITIATED FROM 100% POWER, ON 7/25/91 AT 1140 HOURS. THE PLANT SHUTDOWN WAS INITIATED AFTER THE B TRAIN SWP WAS DECLARED INOPERABLE AND PLANT MANAGEMENT CONCLUDED THAT THE INOPERABLE B SWP TRAIN COULD NOT BE RESTORED WITHIN THE 72 HOUR PERIOD ALLOWED BY THE TS. THE B SWP TRAIN HAD BEEN DECLARED INOPERABLE ON 7/25/91, AT 1045 HOURS DUE TO FOULING OF THE B AND C REACTOR PLANT COMPONENT COOLING WATER (CCP) HEAT EXCHANGERS AND THE B EMERGENCY DIESEL GENERATOR HEAT EXCHANGERS DUE TO THE ACCUMULATION OF MUSSEL DEBRIS. THE ROOT CAUSE IS DESIGN DEFICIENCY. THE CHLORINATION SYSTEM DID NOT TREAT A PORTION OF THE PIPING SYSTEM. THIS ALLOWED MUSSELS GROWTH. THE PIPING WAS CLEANED AND FLUSHED PRIOR TO BEING RETURNED TO SERVICE. THE CHLORINATION INJECTION POINTS WERE MOVED TO THE SUCTION OF THE SERVICE WATER PUMPS. A MACROFOULING OPERATING STRATEGY HAS BEEN PUT IN PLACE TO DETERMINE WHEN THE SERVICE WATER SYSTEM IS AT RISK.

[192] MILLSTONE 3 DOCKET 50-423 LER 91-025 REV 01
UPDATE ON FAILURE TO VERIFY DE-ENERGIZATION OF SOLID STATE PROTECTION INPUT RELAYS FOR COLD OVERPRESSURE PROTECTION DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 091091 REPORT DATE: 012192 NSSS: WE TYPE: PWR

(NSIC 223851) AT 1300 HOURS ON 9/10/91, WILE IN MODE 5 (COLD SHUTDOWN) AT 94F AND ATMOSPHERIC PRESSURE, A PROCEDURE DEFICIENCY IN THE TEST FOR SOLID STATE PROTECTION SYSTEM INPUT RELAYS RESULTED IN AN INCOMPLETE SURVEILLANCE. INSTRUMENTATION AND CONTROLS (I&C) DEPARTMENT PERSONNEL REVIEWING THE TEST METHODOLOGY DURING A SURVEILLANCE PROCEDURE RE-WRITE DISCOVERED THAT DE-ENERGIZATION OF THE SSPS INPUT RELAYS FOR COLD OVERPRESSURE PROTECTION (COPS) WAS NOT VERIFIED DURING OVERLAP TESTING FOR CHANNEL CALIBRATION. TESTING OF THIS RELAY IS REQUIRED BY PLANT TECH SPEC. NO IMMEDIATE OPERATOR ACTION WAS REQUIRED SINCE THE PLANT WAS SHUT DOWN. THE ROOT CAUSE OF THE EVENT IS A PROCEDURAL DEFICIENCY. A NON-STANDARD CIRCUIT DESIGN IS A CONTRIBUTING FACTOR. THE PROCEDURE ASSOCIATED WITH THE SOLID STATE PROTECTION SYSTEM DOES NOT VERIFY THAT THE COPS INPUT RELAY HAS BEEN DE-ENERGIZED. THERE IS NO MAIN CONTROL BOARD ANNUNCIATOR TO VERIFY THAT THIS RELAY IS OPERATIONAL. THE SSPS OPERATION TEST PROCEDURES WERE MODIFIED TO VERIFY THE RELAY'S OPERABILITY. THE RELAY WAS SATISFACTORILY TESTED IN THE AS-FOUND CONDITION. A REVIEW OF SOLID STATE PROTECTION SYSTEM LOOPS AND THEIR ASSOCIATED SURVEILLANCE PROCEDURES HAS BEEN COMPLETED. THREE DEFICIENCIES WERE IDENTIFIED AND THE APPLICABLE PROCEDURES HAVE BEEN MODIFIED.

[193] MILLSTONE 3 DOCKET 50-423 LER 91-029
UNSEALED FIRE STOP AND SEAL PENETRATION IN THE ENGINEERED SAFETY FEATURES BUILDING.
EVENT DATE: 112191 REPORT DATE: 122391 NSSS: WE TYPE: PWR

(NSIC 223759) ON 11/21/91, AT APPROX. 1230 HOURS, WITH THE PLANT AT 0% POWER IN MODE 5 (COLD SHUTDOWN), AN UNIDENTIFIED UNSEALED FIRE PENETRATION WAS DISCOVERED BETWEEN THE "A" TRAIN CONTAINMENT RECIRCULATION SPRAY SYSTEM (RSS) AND RESIDUAL

HEAT REMOVAL (RHR) CUBICLES LOCATED IN THE ENGINEERED SAFETY FEATURES (ESF) BUILDING. AS IMMEDIATE CORRECTIVE ACTION, THE UNIDENTIFIED UNSEALED FIRE PENETRATION WAS DECLARED INOPERABLE, THE ASSOCIATED FIRE DETECTORS WERE VERIFIED OPERABLE AND A COMPENSATORY FIRE WATCH WAS IMMEDIATELY ESTABLISHED IN THE AFFECTED AREA. THE ROOT CAUSE OF THE EVENT IS INCOMPLETE WORK PRACTICES DURING CONSTRUCTION. THE SUBJECT FIRE PENETRATION WAS NOT IDENTIFIED ON CONSTRUCTION DRAWINGS. THE POST CONSTRUCTION SEAL VERIFICATION PROCESS DID NOT PROPERLY RECONCILE THE WALL CONFIGURATION AGAINST THE ASSOCIATED FIRE STOP AND SEAL DRAWING. ON 12/21/91, THE SUBJECT PENETRATION WAS SEALED IN ACCORDANCE WITH DESIGN SPECIFICATIONS. THE PENETRATION WAS DECLARED OPERABLE AND THE FIRE WATCH PATROL TERMINATED ON 12/23/91. AT 100% INSPECTION OF ALL WALL AND FLOOR BOUNDARY PENETRATIONS IN THE ESF BUILDING WILL BE PERFORMED.

[194] MILLSTONE 3 DOCKET 50-423 LER 91-030
MOTOR CONTROL CENTER AUXILIARY CONTROL RELAY FAILURE DUE TO THERMAL AGING.
EVENT DATE: 121991 REPORT DATE: 012092 NSSS: WE TYPE: PWR
VENDOR: ITE/GOULD

(NSIC 223876) ON 12/19/91, AT 1300 HOURS, WHILE SHUTDOWN IN MODE 5 (COLD SHUTDOWN), AN ENGINEERING EVALUATION CONCLUDED THE FAILURE OF THREE NORMALLY ENERGIZED AUXILIARY (CONTROL) RELAYS FOR MOTOR OPERATED VALVES (MOVS) WAS CAUSED BY INSULATION BREAKDOWN AND ELECTRICAL SHORTING OF THE RELAY COIL. THE FAILED TELEMECANIQUE MODEL J10 RELAYS CAUSED A LOSS OF CONTROL POWER AND RENDERED THE MOVS INOPERABLE. THE INVESTIGATION CONCLUDED THAT A VERY HIGH POTENTIAL EXISTED FOR ADDITIONAL RELAY FAILURES DUE TO THE DEGREE OF THERMAL AGING OBSERVED DURING INSPECTIONS. THE ROOT CAUSE OF THE RELAY FAILURE IS ACCELERATED THERMAL AGING. THE J10 RELAYS HAD BEEN ENERGIZED FOR APPROXIMATELY SEVEN YEARS. THE VENDOR EQUIPMENT QUALIFICATION REPORT QUALIFIED THE RELAYS FOR THE LIFE OF THE PLANT (I.E., 40 YEARS). THE RELAY FAILURES WERE ISOLATED TO NORMALLY ENERGIZED RELAYS MOUNTED IN A GANGED ARRANGEMENT. ALL SAFETY RELATED (SR) J10S WERE INSPECTED AND ALL NORMALLY ENERGIZED J10S IN THE SR MCCS WERE REPLACED. ALL NON-SR J10S WERE EVALUATED AND THOSE THAT COULD POSSIBLY CHALLENGE THE PLANT WERE REPLACED. A SURVEILLANCE PROGRAM TO DETECT FUTURE PROBLEMS WILL BE IMPLEMENTED PRIOR TO REFUELING OUTAGE NO. 4. THE ACCELERATED THERMAL AGING OF THE J10 RELAYS APPEARS TO HAVE GENERIC RAMIFICATIONS DUE TO THE RELAYS OPERATION, MOUNTING AND MAINTENANCE BEING CONSISTENT WITH VENDOR RECOMMENDATIONS.

[195] MILLSTONE 3 DOCKET 50-423 LER 91-031
MISSED TEMPORARY SURVEILLANCE ON VENTILATION VENT RADIATION MONITOR DUE TO ADMINISTRATIVE OVERSIGHT AND INADEQUATE PLANNING.
EVENT DATE: 122191 REPORT DATE: 012192 NSSS: WE TYPE: PWR

(NSIC 223877) AT 0745 HOURS ON 12/21/91, WHILE IN MODE 5 (COLD SHUTDOWN), AT 96F AND ATMOSPHERIC PRESSURE, A TECH SPEC SURVEILLANCE WAS MISSED. A MANUAL GRAB SAMPLE SURVEILLANCE IS REQUIRED TO BE PERFORMED AT LEAST ONCE EVERY 12 HOURS WHENEVER THE VENTILATION VENT RADIATION MONITOR IS OUT OF SERVICE. THE GRAB SAMPLE WAS NOT PERFORMED FOR APPROX. 21 HOURS. A SAMPLE WAS OBTAINED AFTER THE EVENT WAS DISCOVERED. THE SAMPLES PRIOR TO AND FOLLOWING THE EVENT WERE BELOW MINIMUM DETECTABLE ACTIVITY. THE ROOT CAUSE OF THE EVENT IS ADMINISTRATIVE OVERSIGHT. A REVIEW OF THE DAILY CHEMISTRY LOG REVEALED THAT THE TWELVE HOUR SAMPLING SURVEILLANCE HAD BEEN MISSED. THE IMMEDIATE CORRECTIVE ACTION WAS TO OBTAIN A SAMPLE. PROCEDURE CHANGES WERE IMPLEMENTED TO PROVIDE A MORE CONSERVATIVE SURVEILLANCE FREQUENCY. THE COMMUNICATION OF TECHNICAL DATA BETWEEN CHEMISTRY DEPARTMENT SHIFT PERSONNEL DURING SHIFT TURNOVER WAS ALSO IMPROVED.

[196] MILLSTONE 3 DOCKET 50-423 LER 92-001
LEAKAGE MONITORING CONNECTION CONTAINMENT ISOLATION VALVES NOT LOCKED CLOSED DUE TO PROGRAM FAILURE.
EVENT DATE: 011392 REPORT DATE: 021292 NSSS: WE TYPE: PWR

(NSIC 224045) ON 1/13/92, AT 0800 HOURS WITH THE PLANT AT 0% POWER IN MODE 5 (COLD SHUTDOWN), 93F AND APPROXIMATELY 40 PSIA (NTIROGEN FLOAT), SIX VALVES WERE IDENTIFIED WHICH WERE MISSING LOCKING DEVICES. THE VALVES WERE FOUND CLOSED BUT

NOT LOCKED AS REQUIRED. THESE VALVES ARE LEAKAGE MONITORING CONNECTION (LMC) VENT VALVES WHICH ARE USED TO TEST THE CONTAINMENT ISOLATION VALVES ASSOCIATED WITH THE RHR COLD LEG INJECTION LINES. THE ROOT CAUSE OF THE EVENT IS PROGRAM FAILURE-PROCEDURE DEFICIENCY, ADMINISTRATIVE ERROR. THE SYSTEM LINEUPS DID NOT INCLUDE THE SUBJECT VALVES AMONG THOSE REQUIRED TO BE LOCKED CLOSED. AS AN IMMEDIATE CORRECTIVE ACTION THE VALVES WERE LOCKED CLOSED AND THE PENETRATION SURVEILLANCES HAVE BEEN UPDATED TO INCLUDE THESE VALVES. A CHANGE WILL BE SUBMITTED TO UPDATE THE FSAR CONTAINMENT PENETRATION TABLE.

[197] MONTICELLO DOCKET 50-263 LER 90-001 REV 03
UPDATE ON POTENTIAL EMERGENCY FILTER TRAIN SYSTEM INOPERABILITY DUE TO INTERACTION WITH NON-SAFETY RELATED EQUIPMENT.
EVENT DATE: 031390 REPORT DATE: 121891 NSSS: GE TYPE: BWR

(NSIC 223704) 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.

[198] MONTICELLO DOCKET 50-263 LER 91-012 REV 01
UPDATE ON IMPROPER FUSE SETTING AND WIRE ERROR CAUSES UNPLANNED EMERGENCY FILTRATION TRAIN ACTUATION.
EVENT DATE: 052491 REPORT DATE: 121191 NSSS: GE TYPE: BWR

(NSIC 223600) DURING THE 1991 REFUELING OUTAGE, BOTH TRAINS OF THE CONTROL ROOM VENTILATION EMERGENCY FILTRATION TRAIN SYSTEM WERE TRIPPED INTO THE EMERGENCY MODE OF OPERATION. THE CAUSE OF THE EVENT WAS AN IMPROPER FUSE SETTING AND A MISWIRED GROUND DETECTION RELAY WHICH RESULTED IN A MOMENTARY LOSS OF POWER TO CONTROL LOGIC DURING AN AUTOMATIC TRANSFER OF STATION POWER FROM THE 2R TRANSFORMER TO THE 1R TRANSFORMER. CORRECTIVE ACTIONS TAKEN INCLUDE REVISING THE METHOD OF SETTING THE TRIP SETPOINT FOR THE CURRENT LIMITING FUSES, CORRECTING THE WIRING ERROR ON THE GROUND DETECTION RELAY AND TESTING THE CHANGES, AND DISCUSSING THE WIRING ERROR WITH SUBSTATION CONSTRUCTION. CORRECTIVE ACTIONS PLANNED INCLUDE VERIFYING THE WIRING ON THE 1ARS TRANSFORMER GROUND DETECTION RELAY AND PURSUING THE FEASIBILITY OF ELIMINATING THE AUTOMATIC ACTUATION OF THE CONTROL ROOM VENTILATION-EMERGENCY FILTRATION TRAIN SYSTEM.

[199] MONTICELLO DOCKET 50-263 LER 91-018 REV 02
UPDATE ON DUE TO NON-CONSERVATIVE ASSUMPTIONS IN THE INTERNAL FLOODING ANALYSIS A BREAK IN THE PLANT SERVICE WATER SYSTEM MAY HAVE RESULTED IN A LOSS OF REDUNDANT TRAINS OF SAFETY RELATED EQUIPMENT.
EVENT DATE: 082391 REPORT DATE: 012292 NSSS: GE TYPE: BWR

(NSIC 223850) ON 8/23/91 IT WAS DETERMINED THAT A POSTULATED LINE BREAK OF A SERVICE WATER LINE COULD AFFECT REDUNDANT TRAINS OF SAFETY RELATED ELECTRICAL EQUIPMENT. A FLOOD WATCH WAS STATIONED, PROCEDURES WERE REVISED, AND AN ANALYSIS WAS PERFORMED TO VERIFY THAT THE LINE IS EQUIVALENT TO SEISMIC CATEGORY I EQUIPMENT. THE ROOT CAUSE OF THIS EVENT WAS A PERSONNEL ERROR IN THE PREPARATION AND REVIEW OF THE INTERNAL FLOODING EVALUATION. THE FOLLOWING ACTIONS ARE BEING TAKEN: REVIEW OF THE INTERNAL FLOODING DESIGN BASES DOCUMENT IS CONTINUING, THE Q-LIST EXTENSION AND THE PIPING INSPECTION PROGRAM ARE BEING UPDATED, THE SUPPORT FOR SW-13 WILL BE MODIFIED AND THE EVENT WAS PRESENTED IN ENGINEERING TECHNICAL STAFF CONTINUING TRAINING. CONTINUING EVALUATIONS IDENTIFIED POTENTIAL PROBLEMS WITH FIRE PROTECTION, EMERGENCY SERVICE WATER, AND DEMINERALIZED WATER LINES. THE FIRE PROTECTION LINE WAS UPGRADED TO SEISMIC CLASS I, A FLOOD WATCH WAS STATIONED, AND PROCEDURES WERE REVISED TO MAINTAIN THE DEMINERALIZED WATER

STORAGE TANK LEVEL BELOW 26,000 GALLONS. THE EMERGENCY SERVICE WATER LINES WERE SUBSEQUENTLY FOUND TO BE ACCEPTABLE.

[200] MONTICELLO DOCKET 50-263 LER 92-001
SHUTDOWN REQUIRED BY TECHNICAL SPECIFICATIONS DUE TO INOPERABLE BELLOWS LEAK DETECTION SYSTEM FOR SAFETY RELIEF VALVES.
EVENT DATE: 010292 REPORT DATE: 020392 NSSS: GE TYPE: BWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)

(NSIC 223929) DURING THE PERFORMANCE OF THE BELLOWS LEAK DETECTION SURVEILLANCES THE SELF-ACTUATING FUNCTION OF TWO SAFETY RELIEF VALVES WERE DECLARED INOPERABLE. THE PLANT ENTERED A LIMITING CONDITION FOR OPERATION TECHNICAL SPECIFICATION ACTION STATEMENT REQUIRING A SHUTDOWN TO LESS THAN 345 DEGREES AND 110 PSIG WITHIN 24 HOURS. THE PLANT WAS PLACED IN A COLD SHUTDOWN CONDITION. INVESTIGATION CONCLUDED THAT THE CAUSE WAS THE FAILURE OF THE BELLOWS LEAK DETECTION SOLENOIDS VALVES TO PROPERLY FUNCTION DUE TO DEGRADATION OF THE SEATING MATERIAL FROM EXPOSURE TO TEMPERATURES NEAR THE MANUFACTURERS RATED TEMPERATURE. ALL SOLENOID VALVES FOR THE BELLOWS LEAK DETECTION SYSTEM WERE REPLACED WITH SOLENOID VALVES HAVING A HIGHER TEMPERATURE RATING FOR THE SEATING MATERIALS. THE BELLOWS LEAK DETECTION SYSTEM WAS TESTED AND THE PLANT WAS RETURNED TO POWER OPERATIONS.

[201] MONTICELLO DOCKET 50-263 LER 92-002
REACTOR PROTECTION SYSTEM ACTUATION CAUSED BY SPURIOUS INTERMEDIATE RANGE MONITOR SIGNAL.
EVENT DATE: 010392 REPORT DATE: 020392 NSSS: GE TYPE: BWR

(NSIC 223903) WITH THE PLANT SHUTDOWN AND THE MODE SWITCH IN REFUEL TO PERFORM AN INTERMEDIATE RANGE MONITOR SURVEILLANCE, A SPURIOUS HI-HI SIGNAL FROM INTERMEDIATE RANGE MONITOR 18 CAUSED A REACTOR PROTECTION SYSTEM TRIP. ALL RODS WERE FULLY INSERTED AT THE TIME OF THE EVENT AND NO ROD MOVEMENT RESULTED. THE TRIP WAS RESET AND THE SURVEILLANCE TEST COMPLETED. THE CAUSE OF THE SPURIOUS SIGNAL WAS INVESTIGATED BUT COULD NOT BE DETERMINED.

[202] MONTICELLO DOCKET 50-263 LER 92-003
INOPERABLE OFFGAS RADIATION MONITORS CAUSED BY INADEQUATE PROCEDURE.
EVENT DATE: 010792 REPORT DATE: 020592 NSSS: GE TYPE: BWR

(NSIC 223930) ON JANUARY 7, 1992, AT 0630 HOURS IT WAS DETERMINED THAT THE OFFGAS RADIATION MONITORS WERE NOT OPERABLE AS REQUIRED BY TECHNICAL SPECIFICATIONS. THE CAUSE WAS AN INADEQUATE PROCEDURE FOR STARTUP OF THE OFFGAS MONITOR SAMPLE SYSTEM. THE PROCEDURE FOR PLACING THE SAMPLE SYSTEM IN SERVICE DID NOT REQUIRE CHECKING FOR MOISTURE DURING STARTUP. CONTRIBUTING FACTORS WERE AN ERROR IN LOGGING BY A CONTROL ROOM OPERATOR, AND MISLEADING INDICATIONS AT THE LOCAL SAMPLE SYSTEM CONTROLS. CORRECTIVE ACTIONS INCLUDED DRAINING AND RETURNING THE SAMPLE SYSTEM TO SERVICE, AND REVISING THE CONTROL ROOM LOGS AND THE SYSTEM STARTUP PROCEDURE. ACTIONS TO BE COMPLETED ARE CHANGES TO CHEMISTRY COMPUTER DISPLAYS, TRAINING FOR SHIFT CHEMISTS, AND CLEANING OF THE OFFGAS FLOW MONITOR.

[203] NINE MILE POINT 1 DOCKET 50-220 LER 91-013
BREACH OF SECONDARY CONTAINMENT INTEGRITY DUE TO INADEQUATE DESIGN.
EVENT DATE: 120291 REPORT DATE: 122391 NSSS: GE TYPE: BWR

(NSIC 223762) ON DECEMBER 2, 1991, AT 1100 HOURS, WITH THE MODE SWITCH IN THE "RUN" POSITION AND THE REACTOR POWER LEVEL AT APPROXIMATELY 98% OF RATED, SECONDARY CONTAINMENT INTEGRITY WAS MOMENTARILY BREACHED. THIS BREACH WAS THE RESULT OF BOTH REACTOR BUILDING AIRLOCK DOORS, PROVIDING ACCESS FROM THE TURBINE BUILDING TO THE REACTOR BUILDING, BEING OPEN SIMULTANEOUSLY. NINE MILE POINT UNIT 1 (NMP1) TECHNICAL SPECIFICATIONS (T.S.), SECTIONS 1.12 AND 3.4.3, STATE THAT AT LEAST ONE DOOR IN EACH OF THE DOUBLE DOOR ACCESS WAYS SHALL BE CLOSED WHENEVER SECONDARY CONTAINMENT INTEGRITY IS REQUIRED. THE CAUSE OF THIS EVENT IS MAN-MACHINE INTERFACE. THE LIMITED TIME COMBINED WITH SEVEN DISTINCT ACTIONS THE

INDIVIDUAL MUST PERFORM TO OPEN THE DOOR, RESULTS IN AN INCREASED PROBABILITY THAT BOTH DOORS COULD BE OPENED AT THE SAME TIME. THE REACTOR BUILDING AIRLOCK DOORS ARE NOT PHYSICALLY INTERLOCKED TO PREVENT SIMULTANEOUS OPENING. INITIAL CORRECTIVE ACTIONS TAKEN WERE TO CLOSE AT LEAST ONE AIRLOCK DOOR TO RESTORE SECONDARY CONTAINMENT. ADDITIONAL CORRECTIVE ACTIONS INCLUDE THE EVALUATION OF THE INSTALLATION OF PHYSICAL INTERLOCKS AND THE PROCESSING OF A TECHNICAL SPECIFICATION AMENDMENT TO INCORPORATE A LIMITING CONDITION OF OPERATION (LCO) FOR A SPECIFIC TIME PERIOD TO ALLOW FOR THIS EVENT.

[204] NINE MILE POINT 1 DOCKET 50-220 LER 91-014
REACTOR SCRAM DUE TO EQUIPMENT FAILURE.
EVENT DATE: 120491 REPORT DATE: 122691 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223696) ON 12/4/91, AT 0842 HOURS, WITH THE MODE SWITCH IN "RUN" AND REACTOR POWER LEVEL AT APPROXIMATELY 96.5% OF RATED, THE NINE MILE POINT UNIT 1 (NMP1) REACTOR EXPERIENCED A REACTOR SCRAM. THE REACTOR SCRAM WAS THE RESULT OF A LEVEL TRANSIENT THAT LOWERED REACTOR WATER LEVEL BELOW THE SCRAM SETPOINT OF GREATER THAN OR EQUAL TO 53 INCH. THE ROOT CAUSES WERE MANUFACTURING DEFECTS AND DESIGN INADEQUACIES. THE IMMEDIATE CAUSE OF THE EVENT WAS EQUIPMENT FAILURE. A SOLDER CONNECTION FAILED ON THE TOTAL STEAM FLOW METER IN THE FEEDWATER LEVEL CONTROL CIRCUITRY, CAUSING AN IMBALANCE IN THE SIGNALS TO THE 3 ELEMENT CONTROLLER, WHICH CLOSED 13A AND 13B FEEDWATER FLOW CONTROL VALVES RESPONSE. INITIAL CORRECTIVE ACTIONS WERE TO RESPOND TO THE REACTOR SCRAM AND INCREASE REACTOR WATER LEVEL TO ITS NORMAL RANGE. THE CAUSE OF THE LEVEL TRANSIENT WAS INVESTIGATED, AND THE FAILURE TOTAL STEAM FLOW METER WAS IDENTIFIED AND REPLACED. THE REMAINING METERS IN THE FEEDWATER CONTROL SYSTEM WERE EVALUATED AND THOSE WITHOUT ELECTRICAL SHUNTS WERE REPLACED. EVALUATION WILL BE PERFORMED OF METERS IN OTHER CONTROL CIRCUITS TO PREVENT SIMILAR EVENTS.

[205] NINE MILE POINT 2 DOCKET 50-410 LER 91-015
STANDBY LIQUID CONTROL SYSTEM POTENTIALLY INOPERABLE DURING SURVEILLANCE TESTING.
EVENT DATE: 022091 REPORT DATE: 121391 NSSS: GE TYPE: BWR

(NSIC 223646) ON FEBRUARY 20, 1991, NINE MILE POINT UNIT 2 (NMP2) IDENTIFIED THAT INCORRECT TECHNICAL SPECIFICATION ACTION REQUIREMENTS WERE BEING ENTERED DURING TECHNICAL SPECIFICATION REQUIRED SURVEILLANCE TESTING OF THE STANDBY LIQUID CONTROL SYSTEM (SLS). THE VALVE ALIGNMENTS ESTABLISHED BY THE PROCEDURE COULD, FOR A BRIEF PERIOD OF TIME, RENDER BOTH SLS DIVISIONS (1 AND 2) POTENTIALLY INOPERABLE. PRIOR TO THIS DISCOVERY, ONLY ONE DIVISION WAS BEING DECLARED INOPERABLE. AT THE TIME THIS CONDITION WAS REPORTED, THE MODE SWITCH WAS IN THE "RUN" POSITION (CONDITION 1) WITH THE REACTOR OPERATING AT APPROXIMATELY 100 PERCENT RATED THERMAL POWER. THE ROOT CAUSE FOR THIS CONDITION WAS DETERMINED TO BE INADEQUATE PROCEDURE DEVELOPMENT AND REVIEW. CORRECTIVE ACTIONS INCLUDE: 1) REVISING THE QUARTERLY SURVEILLANCE TEST PROCEDURE TO PROVIDE TEST PERSONNEL WITH A CAUTION STATEMENT SPECIFYING REQUIRED ACTIONS IF A SYSTEM INITIATION OCCURS, AND A PLANT IMPACT STATEMENT TO SPECIFY SYSTEM OPERABILITY STATUS DURING TESTING; 2) PROVIDING INFORMATION FOR INPUT INTO THE INSTITUTE OF NUCLEAR POWER OPERATIONS (INPO) NUCLEAR NETWORK; AND 3) REVIEWING SYSTEM SURVEILLANCE TESTS FOR SIMILAR TEST CONDITIONS.

[206] NINE MILE POINT 2 DOCKET 50-410 LER 91-022
REACTOR SCRAM CAUSED BY A TURBINE CONTROL SYSTEM MALFUNCTION.
EVENT DATE: 120791 REPORT DATE: 010692 NSSS: GE TYPE: BWR

(NSIC 223737) ON DECEMBER 7, 1991, AT 0935 HOURS, WITH THE REACTOR MODE SWITCH IN THE "RUN" POSITION (MODE 1), AND THE PLANT OPERATING AT APPROXIMATELY 90 PERCENT RATED THERMAL POWER (905 MWE), NINE MILE POINT UNIT 2 EXPERIENCED AN ENGINEERED SAFETY FEATURE ACTUATION. SPECIFICALLY, AN AUTOMATIC REACTOR SCRAM OCCURRED CAUSED BY A TURBINE GENERATOR STOP VALVE CLOSURE, WHICH WAS INITIATED BY (MOST PROBABLE CAUSE) AN ELECTROHYDRAULIC CONTROL (EHC) SYSTEM MALFUNCTION. THE ROOT CAUSE INVESTIGATION IS STILL UNDERWAY AND HAS NOT YET DETERMINED THE EXACT CAUSE; HOWEVER, THE MOST PROBABLE CAUSE IS A DEFECTIVE RELAY ACTUATION. THE IMMEDIATE

CORRECTIVE ACTION WAS TO RESPOND TO THE REACTOR SCRAM AND TURBINE TRIP IN ACCORDANCE WITH PLANT PROCEDURES. A WORK REQUEST WAS ISSUED TO INVESTIGATE THE EHC MALFUNCTION, WHICH LED TO THE REPLACEMENT OF THE RELAY BOARD CONTAINING THE SUSPECTED FAULTY RELAY.

[207] NINE MILE POINT 2 DOCKET 50-410 LER 91-023
REACTOR SCRAM ON LOW REACTOR WATER LEVEL DUE TO LOSS OF FEEDWATER PUMPS.
EVENT DATE: 121291 REPORT DATE: 011392 NSSS: GE TYPE: BWR

(NSIC 223804) ON 12/12/91, AT 0322, WITH THE REACTOR MODE SWITCH IN THE "RUN" POSITION AND THE PLANT OPERATING AT APPROX. 55% RATED THERMAL POWER, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED A REACTOR SCRAM ON A REACTOR VESSEL LOW WATER LEVEL SIGNAL. SPECIFICALLY, FOLLOWING THE STARTUP OF FEEDWATER SYSTEM PUMP 2FWS-P1A IN SUPPORT OF RAISING PLANT POWER, A CONDENSATE SYSTEM (CNM) AND FEEDWATER SYSTEM (FWS) TRANSIENT OCCURRED, RESULTING IN THE LOSS OF BOTH FWS PUMPS. REACTOR VESSEL WATER LEVEL LOWERED TO 159.3 INCHES (LEVEL 3 TRIP SET POINT), INITIATING AN AUTOMATIC REACTOR SCRAM SIGNAL. ON 12/12/91, AT 1023, WHILE OPERATORS WERE ATTEMPTING TO RETURN THE REACTOR WATER CLEANUP (RWCS) SYSTEM TO AN OPERABLE STATUS FOLLOWING THE SCRAM, WCS ISOLATED ON A HIGH DIFFERENTIAL FLOW SIGNAL (ENGINEERED SAFETY FEATURE ACTUATION). THE ROOT CAUSE FOR THE REACTOR SCRAM WAS DETERMINED TO BE POOR WORK PRACTICES. THE PRELIMINARY CAUSE FOR THE WCS ISOLATION IS INADEQUATE SYSTEM DESIGN. IMMEDIATE ACTIONS INCLUDED RESTORING REACTOR VESSEL INVENTORY AND COMMENCING A CONTROLLED PLANT SHUTDOWN. ADDITIONAL CORRECTIVE ACTIONS INCLUDE: 1) ADMINISTERING DISCIPLINARY ACTION FOR INDIVIDUALS INVOLVED IN THE SCRAM; 2) EVALUATING AND REVISING OPERATING PROCEDURE N2-OP3, 3) TROUBLESHOOTING ELECTRICAL CIRCUITRY FOR CONDENSATE PUMP 2CNM-PC; AND 4) COMPLETING THE WCS ISOLATION ROOT CAUSE ANALYSIS.

[208] NINE MILE POINT 2 DOCKET 50-410 LER 92-001
MULTIPLE ENGINEERED SAFETY FEATURE ACTUATIONS DUE TO FAILURE OF A PANALARM TEMPERATURE SWITCH.
EVENT DATE: 010492 REPORT DATE: 020392 NSSS: GE TYPE: BWR
VENDOR: RILEY-BEAIRD, INC.

(NSIC 223943) AT APPROXIMATELY 0236 HOURS ON JANUARY 4, 1992, WITH THE NINE MILE POINT UNIT 2 (NMP2) REACTOR MODE SWITCH IN THE "RUN" POSITION (OPERATIONAL CONDITION 1) AND THE PLANT OPERATING AT APPROXIMATELY 89 PERCENT RATED THERMAL POWER, NMP2 EXPERIENCED TWO ENGINEERED SAFETY FEATURE (ESF) ACTUATION SIGNALS, CONSISTING OF HIGH AREA TEMPERATURE ISOLATION SIGNALS, FOR THE REACTOR CORE ISOLATION COOLING SYSTEM (ICS) AND REACTOR WATER CLEANUP SYSTEM (WCS) DUE TO A FAILED PANALARM (RILEY) TEMPERATURE MONITORING SWITCH, E31-N620A. THE ROOT CAUSE OF THE HIGH AREA TEMPERATURE SIGNAL HAS BEEN DETERMINED TO BE A FAILURE OF A PANALARM (RILEY) TEMPERATURE SWITCH. ONE OF THE PREDOMINATE FAILURE MODES IS TO DRIFT IN AND OUT OF THE ALARM STATE. CORRECTIVE ACTIONS INCLUDED VERIFYING THAT NO STEAM LEAKS OR HIGH AREA TEMPERATURES EXISTED IN THE ICS/WCS PIPE CHASE, AND REPLACING THE FAILED TEMPERATURE SWITCH. ADDITIONALLY, A PLANT MODIFICATION IS SCHEDULED FOR THE SECOND REFUELING OUTAGE TO BEGIN REPLACING PANALARM (RILEY) TEMPERATURE SWITCHES IN THE PLANT WITH A MORE RELIABLE DESIGN.

[209] NINE MILE POINT 2 DOCKET 50-410 LER 92-002
TECHNICAL SPECIFICATION REQUIREMENT MISSING DURING INITIAL SURVEILLANCE DEVELOPMENT DUE TO INADEQUATE MANAGERIAL METHODS.
EVENT DATE: 010992 REPORT DATE: 021092 NSSS: GE TYPE: BWR

(NSIC 224042) ON JANUARY 9, 1992, AT APPROXIMATELY 1200 HOURS, IT WAS RECOGNIZED THAT NINE MILE POINT UNIT 2 (NMP2) HAD NOT BEEN IN COMPLIANCE WITH TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS 4.3.1.1 AND 4.3.1.2. SPECIFICALLY, THE CHANNEL FUNCTIONAL TEST AND LOGIC SYSTEM FUNCTIONAL TEST OF THE REACTOR PROTECTION SYSTEM (RPS) DID NOT VERIFY OPERATION OF THE NON-COINCIDENT NEUTRON MONITORING SYSTEM (NMS) HIGH NEUTRON FLUX TRIP LOGIC. THE VERIFICATION IS REQUIRED PRIOR TO AND DURING THE TIME THE SHORTING LINKS ARE REMOVED. AT THE TIME OF DISCOVERY, THE REACTOR MODE SWITCH WAS IN THE "RUN" POSITION (OPERATIONAL CONDITION 1) WITH THE PLANT OPERATING AT 100 PERCENT RATED THERMAL POWER. THE

ROOT CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE A PERSONNEL ERROR DUE TO INADEQUATE MANAGERIAL METHODS. THE INITIAL CORRECTIVE ACTION WAS TO INITIATE A DEVIATION/EVENT REPORT (DZR) IDENTIFYING THE OMISSION OF TESTING THE NON-COINCIDENT LOGIC. OTHER CORRECTIVE ACTIONS WILL INCLUDE: 1) REVISING INSTRUMENT SURVEILLANCE PROCEDURES TO PERFORM THE TESTING; 2) PERFORMING THE TEST TO VERIFY OPERATION OF THE RELAYS; AND 3) CHANGING A DRAWING TO MATCH ACTUAL PLANT OPERATION.

[210] NORTH ANNA 1 DOCKET 50-338 LER 91-020 REV 01
UPDATE ON SERVICE WATER SYSTEM CONFIGURATION DURING LOGIC TESTING.
EVENT DATE: 103191 REPORT DATE: 122391 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)

(NSIC 223816) AT 1117 HOURS ON 10/31/91, WITH UNITS 1 AND 2 OPERATING AT 100% POWER (MODE 1), IT WAS DISCOVERED THAT THE SHARED SERVICE WATER (SW) SYSTEM MAY NOT HAVE BEEN ABLE TO PROVIDE DESIGN FLOW TO AN ACCIDENT UNIT'S RECIRCULATION SPRAY HEAT EXCHANGERS. THIS DISCOVERY WAS MADE DURING THE PERFORMANCE OF A PERIODIC TEST WHICH REMOVED THE AUTOMATIC START FUNCTION OF THE UNIT 2 EMERGENCY DIESEL GENERATOR WHICH PROVIDES EMERGENCY POWER TO THE UNIT 2 "B" SW PUMP DURING AN ACCIDENT. THE TEST ALSO REMOVED THE AUTOMATIC START FUNCTION OF THE UNIT 1 "B" SW PUMP. OPERATING PROCEDURES REQUIRE THAT THE SW PUMP DISCHARGE PROCEDURE BE ADJUSTED TO 250 PSIG BY MANUALLY THROTTLING COMPONENT COOLING HEAT EXCHANGER CCHX VALVES WHEN LESS THAN 4 PUMPS ARE OPERABLE; HOWEVER, THE SYSTEM WAS NOT THROTTLED AS REQUIRED. IT WAS INITIALLY DETERMINED THAT SUFFICIENT SW FLOW MAY NOT HAVE BEEN AVAILABLE TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT, AND A FOUR HOUR REPORT WAS MADE ON 10/31/91, PURSUANT TO 10 CFR 50.72(B)(2)(III)(D). ON 11/26/91, AN ENGINEERING CALCULATION WAS COMPLETED AND REVIEWED BY THE STATION NUCLEAR SAFETY AND OPERATING COMMITTEE (SNSOC) WHICH DETERMINED THAT SUFFICIENT SW FLOW WAS AVAILABLE TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED DURING THE EVENT.

[211] NORTH ANNA 1 DOCKET 50-338 LER 91-021
MISSED SURVEILLANCE OF OFFSITE AC SOURCES.
EVENT DATE: 112691 REPORT DATE: 122091 NSSS: WE TYPE: PWR

(NSIC 223731) AT 1330 HOURS, ON NOVEMBER 26, 1991, UNIT 1 WAS OPERATING AT 100% POWER (MODE 1) WHEN IT WAS DISCOVERED THAT AN EMERGENCY DIESEL GENERATOR (EDG) BATTERY HAD AN INDIVIDUAL CELL VOLTAGE (ICV) WHICH WAS LESS THAN THE TECHNICAL SPECIFICATION (TS) LOW LIMIT. THE DISCOVERY WAS MADE FOLLOWING THE ADDITION OF WATER TO CELLS WHICH HAD LOW ELECTROLYTE LEVELS. ENGINEERING PERSONNEL DETERMINED THE LOW ICV READING WAS INVALID DUE TO THE RECENT ADDITION OF WATER. AT 1630 HOURS, MANAGEMENT DETERMINED THAT THE LOW READING WAS VALID. TS 3.8.1.1, ACTION STATEMENT (B), REQUIRES VERIFICATION OF THE A.C. OFF-SITE POWER SOURCES WITHIN ONE HOUR WHEN ONE EDG IS DECLARED INOPERABLE. SINCE THE SURVEILLANCE WAS NOT PERFORMED WITHIN ONE HOUR, THIS EVENT IS REPORTABLE PURSUANT TO 10 CFR 50.73(A)(2)(I)(B). THE CAUSE OF THE EVENT WAS PERSONNEL ERROR DUE TO A MISUNDERSTANDING OF THE ICV READING RESULTING IN A DELAYED REPORT TO THE OPERATIONS SHIFT. FOLLOWING DISCOVERY OF THE MISSED SURVEILLANCE, THE OFF-SITE A.C. POWER SOURCE VERIFICATION WAS SATISFACTORILY COMPLETED AT 2015 HOURS. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS SINCE OFF-SITE POWER SOURCES REMAINED OPERABLE DURING THE PERIOD. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED AT ANY TIME DURING THIS EVENT.

[212] NORTH ANNA 1 DOCKET 50-338 LER 91-023
INADVERTENT FLOW FROM LHSI SYSTEM TO RCS DURING SHUTDOWN WITH RCS IN LOW PRESSURE.
EVENT DATE: 122791 REPORT DATE: 012292 NSSS: WE TYPE: PWR

(NSIC 223831) ON DECEMBER 27 1991, AT 1545 HOURS, UNIT 1 WAS AT COLD SHUTDOWN (MODE 5) WITH REACTOR COOLANT SYSTEM (RCS) PRESSURE AT 30 PSIG AND SLOWLY DECREASING WHEN THE PRESSURIZER LEVEL INCREASED FROM 60% TO 82% COLD CALIBRATED LEVEL. INVESTIGATION REVEALED THAT THE LOW HEAD SAFETY INJECTION (LHSI) PUMP DISCHARGE VALVES HAD BEEN LEFT OPEN AFTER TYPE "C" TESTING. AS RCS PRESSURE SLOWLY DROPPED, THE REFUELING WATER STORAGE TANK (RWST) STATIC HEAD BECAME

SUFFICIENT TO CAUSE BORATED WATER TO FLOW THROUGH THE LHSI PUMPS INTO THE RCS. THIS EVENT IS REPORTABLE PURSUANT TO 10 CFR 50.73 (A)(2)(IV), AND A FOUR HOUR REPORT WAS MADE PURSUANT TO 10 CFR 50.72 (B)(2)(II). THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE THERE WAS NO OVERPRESSURE CONCERN. IN ADDITION, THE MINIMUM SHUTDOWN MARGIN REQUIRED BY TS 3.1.1.2 WAS MAINTAINED. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED AT ANY TIME DURING THIS EVENT.

[213] NORTH ANNA 1 DOCKET 50-338 LER 92-001
STEAM GENERATOR TUBE DEFECTS.
EVENT DATE: 011092 REPORT DATE: 020792 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 224039) DURING THE 1992 MID-CYCLE STEAM GENERATOR (S/G) TUBE INSPECTION OUTAGE ON UNIT 1, ONE HUNDRED PERCENT OF THE ACCESSIBLE TUBES IN THE "B" AND "C" STEAM GENERATORS (S/GS) WERE INSPECTED USING THE STANDARD EDDY CURRENT (E/C) BOBBIN PROBE. IN "A" S/G, ALL BUT ONE TUBE (DUE TO TUBING RESTRICTIONS) HAVE BEEN INSPECTED. ACTIONS ON THE REMAINING TUBE ARE BEING EVALUATED TO COMPLETE THE INSPECTION. ADDITIONALLY, INSPECTIONS ARE BEING PERFORMED USING A ROTATING PANCAKE COIL (RPC) PROBE. AS A RESULT OF THESE INSPECTIONS, GREATER THAN 1% OF THE TUBES IN EACH S/G WERE IDENTIFIED AS HAVING PLUGGABLE INDICATIONS. THESE INSPECTION RESULTS REQUIRED THE THREE S/GS TO BE CLASSIFIED AS CATEGORY C-3. ALL DEFECTIVE TUBES ARE BEING REMOVED FROM SERVICE. THE DEFECTS IDENTIFIED IN THE S/GS ARE REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(V)(C). FOUR HOUR REPORTS WERE MADE PURSUANT TO 10CFR50.72(B)(2)(I). THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE THE PRIMARY TO SECONDARY LEAKAGE RATES WERE CLOSELY MONITORED AND WERE RELATIVELY STEADY AND LOW IN MAGNITUDE. IN ADDITION, A COMPREHENSIVE SAFETY ANALYSIS WAS PERFORMED TO JUSTIFY THE CURRENT OPERATING CYCLE AND TO ADDRESS THE POTENTIAL FOR S/G TUBE DEGRADATION. THIS EVENTUALLY INCLUDED AN ASSESSMENT OF POTENTIAL INDICATIONS LEFT INSERVICE. THEREFORE, THE HEALTH AND SAFETY OF THE GENERAL PUBLIC WAS NOT AFFECTED DUE TO THIS EVENT.

[214] NORTH ANNA 1 DOCKET 50-338 LER 92-002
PRESSURIZER SAFETY VALVE SETPOINT OUT OF TOLERANCE DUE TO SETPOINT DRIFT.
EVENT DATE: 011092 REPORT DATE: 020592 NSSS: WE TYPE: PWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV

(NSIC 223950) AT 1415 HOURS ON 1/10/92, WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN), THE "AS FOUND" SET PRESSURE FOR THE "A" PRESSURIZER SAFETY VALVE WAS FOUND TO BE OUTSIDE THE SETPOINT TOLERANCE ALLOWED BY TECH SPECS 3.4.2 AND 3.4.3. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I))(B) FOR A CONDITION PROHIBITED BY THE TECH SPECS. THE THREE PRESSURIZER SAFETY VALVES WERE SENT TO WYLE LABS FOR TESTING. THE "AS FOUND" SET PRESSURE FOR THE "A" PRESSURIZER SAFETY VALVE WAS FOUND TO BE OUTSIDE THE ALLOWABLE TECH SPEC TOLERANCE OF 2485 PSIG +/- 1%. IN ADDITION, LEAKAGE FROM THE THREE SAFETY VALVES WAS NOTED FOLLOWING "AS FOUND" TESTING. THE SAFETY VALVES WILL BE REPAIRED AND READJUSTED, AS NECESSARY TO BE WITHIN THE CORRECT SETPOINT TOLERANCE ALLOWED BY TECH SPECS. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE THE SAFETY VALVE WOULD HAVE PERFORMED ITS INTENDED SAFETY FUNCTION IN THE EVENT OF AN OVERPRESSURE CONDITION. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED AT ANY TIME DURING THIS EVENT.

[215] OCONEE 2 DOCKET 50-270 LER 91-004
BREACH OF FIRE BARRIER DUE TO UNKNOWN CAUSE RESULTS IN TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 120491 REPORT DATE: 011092 NSSS: BW TYPE: PWR

(NSIC 223786) ON DECEMBER 4, 1991 AT APPROXIMATELY 1940 HOURS, OPERATIONS PERSONNEL, ON A ROUTINE PLANT TOUR, DISCOVERED THAT A FIRE BARRIER ON UNIT 2 HAD BEEN BREACHED WITHOUT TAKING COMPENSATORY ACTIONS AS REQUIRED BY TECHNICAL SPECIFICATIONS. THE UNIT WAS OPERATING AT 100% FULL POWER WHEN THIS DISCOVERY WAS MADE. THE BREACH WAS A HOLE APPROXIMATELY THREE QUARTERS INCH IN DIAMETER IN THE CABLE SPREADING ROOM WALL AT PENETRATION NUMBER 2ME4. A FIRE BARRIER INSPECTION WAS PERFORMED IN AUGUST 1990 THAT DOCUMENTED OPERABILITY OF THIS PENETRATION. IT

IS UNKNOWN WHEN THE PENETRATION BECAME INOPERABLE AFTER THAT DATE. CABLES HAD BEEN PULLED IN THE IMMEDIATE AREA IN PREPARATION FOR UNIT 2 OUTAGE WORK. UPON THE DISCOVERY OF THIS BREACH, COMPENSATORY MEASURES WERE TAKEN UNTIL THE BREACH WAS RESEALED AT APPROXIMATELY 0940 HOURS ON DECEMBER 5, 1991. THE CAUSE OF THIS EVENT IS ASSIGNED UNKNOWN SINCE IT COULD NOT BE DETERMINED HOW OR WHEN THE BREACH ORIGINATED. CORRECTIVE ACTIONS WILL INCLUDE A REVIEW OF THIS EVENT BY PERSONNEL WHO PERFORM MAINTENANCE OR MODIFICATIONS IN THE AREAS REQUIRING FIRE BARRIERS.

[216] OCONEE 3 DOCKET 50-287 LER 91-008
EXCESSIVE REACTOR COOLANT LEAK, REACTOR TRIP, AND INADVERTENT PROTECTIVE SYSTEM ACTUATION RESULT FROM MANAGEMENT DEFICIENCIES AND EQUIPMENT FAILURE.
EVENT DATE: 112391 REPORT DATE: 122391 NSSS: BW TYPE: PWR
VENDOR: PARKER HANNIFIN CORP.

(NSIC 223769) ON NOVEMBER 23, 1991, OCONEE UNIT 3 WAS OPERATING AT 100% FULL POWER (FP) WHEN THE CONTROL ROOM OPERATORS (CROS) RECEIVED SEVERAL ALARMS AT 0141 HOURS WHICH INDICATED FAILED INSTRUMENTS INSIDE THE REACTOR BUILDING (RB). AT 0143 HOURS, THE CROS OBSERVED SYMPTOMS OF EXCESSIVE REACTOR COOLANT SYSTEM (RCS) LEAKAGE AND BEGAN ASSESSING THE LEAK RATE. AT 0203 HOURS, THEY STARTED A RAPID CONTROLLED SHUTDOWN. AT 0214 HOURS, THE SHIFT SUPERVISOR CONCLUDED THAT LEAKAGE WAS APPROXIMATELY 60 TO 70 GPM, AND DECLARED AN ALERT. AT 0327 HOURS, THE UNIT TRIPPED FROM 33% FP DUE TO A CONTROL OSCILLATION WHILE THE CROS WERE ATTEMPTING TO SECURE A FEEDWATER PUMP. AT 0641 HOURS, AN ADDITIONAL UNANTICIPATED REACTOR PROTECTIVE SYSTEM ACTUATION OCCURRED DUE TO OPERATOR ERROR. AT 1720 HOURS THE UNIT REACHED COLD SHUTDOWN AND THE ALERT WAS TERMINATED. THE LEAK WAS DETERMINED TO BE A FAILED FITTING ON AN INSTRUMENT LINE AT THE TOP OF A STEAM GENERATOR. A TOTAL OF APPROXIMATELY 87,000 GALLONS OF RCS LEAKAGE WAS CONFINED WITHIN THE RB. THE INSTRUMENT LINE WAS REPLACED, AND ADDITIONAL FITTINGS INSPECTED. THE ROOT CAUSES ARE MANAGEMENT DEFICIENCY AND EQUIPMENT FAILURE.

[217] OCONEE 3 DOCKET 50-287 LER 91-009
TECHNICAL SPECIFICATION REQUIRED CONTAINMENT INTEGRITY VALVE FOUND MISPOSITIONED DURING FORCED OUTAGE DUE TO UNKNOWN CAUSE, POSSIBLE INAPPROPRIATE ACTION.
EVENT DATE: 113091 REPORT DATE: 010192 NSSS: BW TYPE: PWR

(NSIC 223792) THE UNIT 3 REACTOR BUILDING CONTAINMENT IS SUPPLIED WITH INSTRUMENT AIR (IA) THROUGH A THREE-INCH LINE WITH NORMALLY CLOSED ISOLATION VALVES (3IA-90 AND 3IA-91) ON EITHER SIDE OF CONTAINMENT. ON DECEMBER 1, 1991 AT APPROXIMATELY 2130 HOURS AND WITH UNIT 3 AT COLD SHUTDOWN CONDITIONS, A NON-LICENSED OPERATOR WHO HAD BEEN SENT TO OPEN 3IA-91 (INSIDE THE REACTOR BUILDING) DISCOVERED THE VALVE IN THE OPEN POSITION. INVESTIGATION COULD NOT DETERMINE WHEN THE VALVE WAS LAST OPENED. THE OTHER ISOLATION VALVE, OUTSIDE THE RB, WAS FOUND CLOSED. IT WAS CONSERVATIVELY ASSUMED THAT 3IA-91 HAD BEEN OPEN SINCE MARCH 22, 1991 AT 1850 HOURS. UNIT 3 OPERATED WITH THE REACTOR COOLANT SYSTEM (RCS) ABOVE 200 DEGREES F AND 300 PSIG, THE CONDITIONS REQUIRED BY TECHNICAL SPECIFICATIONS FOR CONTAINMENT INTEGRITY, FROM MARCH 22, 1991 AT 1850 HOURS TO MARCH 24, 1991 AT 1130 HOURS AND FROM MARCH 27, 1991 AT 1900 HOURS TO NOVEMBER 23, 1991 AT 1720 HOURS. THE ROOT CAUSE OF THIS EVENT IS CONSIDERED UNKNOWN, POSSIBLE INAPPROPRIATE ACTION. CORRECTIVE ACTIONS INCLUDED CHANGES TO THE METHOD OF DOCUMENTATION OF THIS AND OTHER ROUTINELY OPERATED CONTAINMENT INTEGRITY VALVES.

[218] OYSTER CREEK DOCKET 50-219 LER 91-002 REV 01
UPDATE ON LOCAL LEAK RATE TEST RESULTS IN EXCESS OF LIMITS DUE TO VALVE DEGRADATION.
EVENT DATE: 022191 REPORT DATE: 121391 NSSS: GE TYPE: BWR
VENDOR: ATWOOD & MORRILL CO., INC.
FISHER CONTROLS CO.

(NSIC 223669) DURING THE 13R REFUELING OUTAGE, LOCAL LEAK RATE TESTING (IN ACCORDANCE WITH 10CFR50, APPENDIX J) IDENTIFIED A MAIN STEAM ISOLATION VALVE (MSIV) WITH A LEAK RATE IN EXCESS OF THE ACCEPTANCE CRITERIA OF 12.08 SCFH AT 20 PSIG AS SPECIFIED IN TECHNICAL SPECIFICATIONS 4.5.F.2. THE SAFETY SIGNIFICANCE OF THIS DISCOVERY IS CONSIDERED MINIMAL SINCE THE OTHER MSIV IN THE SAME HEADER MET

THE LOCAL LEAK RATE TEST REQUIREMENTS. SUBSEQUENT LOCAL LEAK RATE TESTING IDENTIFIED A PAIR OF ISOLATION VALVES (IN SERIES) WITH A POTENTIAL LEAK RATE IN EXCESS OF THE ACCEPTANCE CRITERIA OF 60% OF THE MAXIMUM ALLOWABLE LIMIT (LA) AT 35 PSIG AS SPECIFIED IN TECHNICAL SPECIFICATIONS 4.5.F.1. LEAKAGE WAS CAUSED BY THE AGE OF A SOFT DISC SEAT ON ONE VALVE AND BY A DEFORMED SEAT DUE TO IMPROPER VALVE INSTALLATION ON THE OTHER. THESE LEAKING VALVES WERE REPAIRED AND SUBSEQUENT LOCAL LEAK RATE TESTING VERIFIED THAT LEAKAGE RATES WERE WITHIN THE ACCEPTANCE CRITERIA. DUE TO AN IMPROPER INTERPRETATION OF REPORTING REQUIREMENTS IMPLEMENTED BY A RECENT TECHNICAL SPECIFICATION AMENDMENT, THIS CONDITION WAS NOT REPORTED WITHIN 30 DAYS AS REQUIRED BY 10CFR50.73.

[219] OYSTER CREEK DOCKET 50-219 LER 91-007
AIR EJECTOR OFF-GAS ISOLATION TIME DELAY FOUND OUT OF SPECIFICATION HIGH DURING SURVEILLANCE TESTING.
EVENT DATE: 110291 REPORT DATE: 112791 NSSS: GE TYPE: BWR

(NSIC 223596) DURING SURVEILLANCE TESTING PERFORMED ON NOVEMBER 2, 1991, IT WAS NOTED THAT THE OFF-GAS SYSTEM ISOLATION FUNCTION EXCEEDED THE 15 MINUTE DELAY LIMIT SPECIFIED BY TECHNICAL SPECIFICATIONS. THIS CONDITION IS CONSIDERED TO HAVE EXISTED SINCE AUGUST 24, 1991, WHEN A FAULTY TIME DELAY RELAY WAS REPLACED AS PART OF A MODIFICATION. A TECHNICAL EVALUATION OF THE REPLACEMENT TIME DELAY WAS NOT ADEQUATELY PERFORMED DURING THE MODIFICATION PROCESS. VENDOR DOCUMENTATION ON INSTRUMENT SETPOINT ACCURACY AND TEMPERATURE EFFECTS ON REPEATABILITY WERE NOT INCLUDED IN THE SURVEILLANCE TEST "AS LEFT" SETPOINT DETERMINATION UNTIL AFTER THE NOVEMBER 2, 1991, SURVEILLANCE. THE SURVEILLANCE PROCEDURE "AS-LEFT" SETPOINT WAS CHANGED ON NOVEMBER 4, 1991, TO PRECLUDE EXCEEDING A TECHNICAL SPECIFICATION LIMIT.

[220] OYSTER CREEK DOCKET 50-219 LER 91-008
FEEDWATER NOZZLE CALIBRATION REVEALS OPERATION OF REACTOR IN EXCESS OF LICENSED LIMIT.
EVENT DATE: 111291 REPORT DATE: 121391 NSSS: GE TYPE: BWR

(NSIC 223668) ON NOVEMBER 12, 1991 A DIFFERENCE BETWEEN TEST MEASURED AND INDICATED FEEDWATER FLOW WAS DISCOVERED AS A RESULT OF A SPECIAL LITHIUM NITRATE FLOW TEST. THIS DIFFERENCE WAS 1.19% IN THE NONCONSERVATIVE DIRECTION WITH RESPECT TO THE FEEDWATER FLOW IMPACT ON THE HEAT BALANCE. THE EXACT CAUSE OF THE 1.19% DIFFERENCE IN FEEDWATER FLOW CANNOT BE DETERMINED. EXISTING PLANT INSTRUMENTATION ACCURACY, CALIBRATION TECHNIQUES, A CHANGE IN NOZZLE DISCHARGE COEFFICIENT OR FEEDWATER PIPING ERROSION UPSTREAM OF THE FEEDWATER NOZZLES MAY HAVE CONTRIBUTED TO THE DIFFERENCE OBSERVED. OYSTER CREEK THERMAL LIMITS ARE BASED, IN PART, ON A 1.76% STANDARD DEVIATION FOR THE UNCERTAINTY IN FEEDWATER FLOW. THEREFORE, THE ERROR FOUND IN THE FEEDWATER FLOW WAS WITHIN THE ACCEPTED UNCERTAINTY AND THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL. INDICATED REACTOR POWER WAS REDUCED BY 1.2% TO ACCOUNT FOR THE FEEDWATER FLOW DIFFERENCE. THE HEAT BALANCE CALCULATIONS WERE REVISED TO ACCOUNT FOR THE OBSERVED DIFFERENCE IN FEEDWATER FLOW.

[221] PALISADES DOCKET 50-255 LER 91-020
INADEQUATE DOCUMENTATION IN ENVIRONMENTAL PROTECTION PLAN.
EVENT DATE: 111291 REPORT DATE: 121391 NSSS: CE TYPE: PWR

(NSIC 223658) ON NOVEMBER 12, 1991, THE NRC RESIDENT INSPECTOR NOTIFIED THE PALISADES LICENSING DEPARTMENT THAT DURING A ROUTINE INSPECTION HE REVIEWED AN EXAMPLE OF OUR EVALUATION OF AN UNREVIEWED ENVIRONMENTAL QUESTION (UEQ). AT THE TIME THE PLANT WAS OPERATING AT 100% POWER. HE FOUND NO DOCUMENTATION THAT AN UNREVIEWED ENVIRONMENTAL QUESTION EVALUATION HAD BEEN PERFORMED IN THE MANNER REQUIRED BY APPENDIX B, THE ENVIRONMENTAL PROTECTION PLAN (EPP), OF THE PALISADES OPERATING LICENSE. FURTHER INVESTIGATION BY PALISADES STAFF DETERMINED THAT NO UEQ EVALUATION HAD BEEN PERFORMED OR DOCUMENTED AS REQUIRED BY THE ENVIRONMENTAL PROTECTION PLAN (EPP) SINCE FEBRUARY 1987 WHEN THE ADMINISTRATIVE PROCEDURE COVERING THIS PROCESS WAS REVISED (I.E. THERE WAS NOT PROPER DOCUMENTATION TO SUPPORT THE CONCLUSIONS THAN NO UEQS EXISTED). THE PROCESSING OF ENVIRONMENTAL

IMPACT REVIEWS WAS IMMEDIATELY SUSPENDED AND THE CONDITION REPORTED TO THE NRC WITHIN 24 HOURS PER THE REQUIREMENTS OF THE OPERATING LICENSE. FAILURE TO FOLLOW THE EVALUATION AND DOCUMENTATION REQUIREMENTS OF THE EPP WAS DUE TO THE DELETION, DURING THE REVISION OF ADMINISTRATIVE PROCEDURE 4.22 IN FEBRUARY 1987, OF A FORM WHICH ASKED SPECIFIC QUESTIONS REQUIRED BY THE EPP FOR EVALUATING UEQS. ADMINISTRATIVE PROCEDURE 4.22 IS BEING REVISED TO REQUIRE AN UEQ EVALUATION TO BE PERFORMED IN THE MANNER REQUIRED BY THE EPP.

[222] PALISADES DOCKET 50-255 LER 92-004
LOSS OF CONTAINMENT INTEGRITY DUE TO THE FAILURE OF THE EMERGENCY ESCAPE AIRLOCK EQUALIZING VALVE.
EVENT DATE: 010792 REPORT DATE: 020692 NSSS: CE TYPE: PWR
VENDOR: WOOLLEY, W. J. COMPANY

(NSIC 223904) ON JANUARY 7, 1992, AT 0435 HOURS, WITH THE PLANT OPERATING AT 100% POWER, IT WA S DETERMINED THAT CONTAINMENT INTEGRITY, AS DEFINED IN TECHNICAL SPECIFICATIONS 1.4 AND 3.6.1A, HAD BEEN VIOLATED. AT THE TIME OF DISCOVERY THE PLANT STAFF WAS PERFORMING TECHNICAL SPECIFICATIONS SURVEILLANCE PROCEDURE (TSSP) SO-4B, "ESCAPE AIR LOCK PENETRATION LEAK TEST." THE ESCAPE AIR LOCK WAS LAST SATISFACTORILY TE STED ON JULY 10, 1991. THIS EVENT IS NOT CONSIDERED A SAFETY SIGNIFICANT ISSUE SINCE A SIGNIFICANT RADI OLOGICAL SOURCE TERM DID NOT EXIST WHEN CONTAINMENT INTEGRITY WAS VIOLATED ON JA NUARY 6, 1992. THEREFORE, THE IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC WAS INSIGNIFICANT. THIS IS FURTHER SUPPORTED BY THE FACT THAT THE OUTER DOOR OF THE ESCAPE AIR LOCK WAS OPERABLE AND ONLY ONE DOOR OF AN AIR LOCK IS REQUIRED FOR CO NTAINMENT INTEGRITY. THE PROXIMATE CAUSE OF THIS EVENT WAS THAT THE ESCAPE AIR LOCK INNER DOOR EQUALI ZING VALVE WAS STUCK IN THE OPEN POSITION, THEREBY DEFEATING THE ISOLATION CAPAB ILITY OF THE INNER DOOR. NO ROOT CAUSE FOR THE VALVE STICKING OPEN HAS BEEN DETE RMINED AT THIS TIME. CORRECTIVE ACTION FOR THIS EVENT INCLUDES REVISING THE MAINTENANCE PROCEDURE FOR THE AIRLOCKS TO REQUIRE THAT THE EQUALIZING VALVES BE VISUALLY CHECKED FOLLOWIN G THE COMPLETION OF THE SEAL CONTACT CHECK, PERFORMING MAINTENANCE OR REPLACING THE ESCAPE AIR LOCK INNER DOOR EQUALIZING VALVE DURING T

[223] PALO VERDE 1 DOCKET 50-528 LER 91-012
INADVERTENT CONTROL ROOM ESSENTIAL FILTRATION ENGINEERED SAFETY FEATURE ACTUATION.
EVENT DATE: 120591 REPORT DATE: 122691 NSSS: CE TYPE: PWR

(NSIC 223751) ON DECEMBER 5, 1991, AT APPROXIMATELY 0950 MST, PALO VERDE UNIT 1 WAS IN MODE 1 (POWER OPERATIONS) AT APPROXIMATELY 100 PERCENT POWER WHEN AN INADVERTENT CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNAL (CREFAS) BALANCE OF PLANT ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (BOP ESFAS) ACTUATION OCCURRED. THE ACTUATION WAS A RESULT OF AN IMPROPER RESTORATION BY A CONTROL ROOM OPERATOR OF THE CHANNEL B CREFAS MODULE THAT HAD BEEN PLACED IN BYPASS MODE FOR SCHEDULED SURVEILLANCE TESTING. THE CREFAS ACTUATION WAS VERIFIED TO NOT BE THE RESULT OF A VALID SIGNAL. THE EQUIPMENT ACTUATED BY THE CREFAS WAS VERIFIED TO OPERATE AS DESIGNED AND SUBSEQUENTLY RETURNED TO NORMAL CONFIGURATION. THE CAUSE OF THE INADVERTENT CREFAS ACTUATION WAS A COGNITIVE PERSONNEL ERROR BY THE OPERATOR WHO REMOVED THE CREFAS MODULE FROM BYPASS MODE PRIOR TO RESETTING THE TRIP INPUTS. THE ERROR WAS A RESULT OF NOT FOLLOWING AN APPROVED STATION PROCEDURE. AS CORRECTIVE ACTION, THE OPERATOR WAS TEMPORARILY REMOVED FROM SHIFT AND DISCIPLINE WAS ADMINISTERED IN ACCORDANCE WITH THE PVNGS POSITIVE DISCIPLINE PROGRAM. SIMILAR EVENTS WERE REPORTED IN LERS 528/86-022, 529/85-006, 529/86-039, 529/90-011, AND 530/91-009.

[224] PALO VERDE 1 DOCKET 50-528 LER 92-001
REACTOR SHUTDOWN REQUIRED BY TECHNICAL SPECIFICATIONS.
EVENT DATE: 010292 REPORT DATE: 020392 NSSS: CE TYPE: PWR
VENDOR: COMBUSTION ENGINEERING, INC.

(NSIC 223947) AT APPROXIMATELY 1739 MST ON JANUARY 2, 1992, PALO VERDE UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 100 PERCENT POWER WHEN A CONDITION IDENTIFIED AS A PRESSURE BOUNDARY LEAK WAS DISCOVERED. ACTION (A) OF TECHNICAL SPECIFICATION 3.4.5.2 REQUIRED THE PLANT TO BE IN MODE 3 (HOT STANDBY) WITHIN SIX

HOURS AND MODE 5 (COLD SHUTDOWN) WITHIN THE FOLLOWING 30 HOURS. THE PLANT WAS SHUTDOWN AND COOLED DOWN USING APPROVED PROCEDURES. NO SAFETY SYSTEM RESPONSES OCCURRED AND NONE WERE REQUIRED. THE PLANT WAS STABILIZED IN MODE 5 (COLD SHUTDOWN) AND REPAIRS WERE MADE TO THE PRESSURIZER STEAM SPACE NOZZLE THAT WAS LEAKING. THE CAUSE OF THIS EVENT IS BELIEVED TO BE PRIMARY WATER STRESS CORROSION CRACKING IN AN INCONEL 600 PRESSURIZER STEAM SPACE INSTRUMENT NOZZLE. A PREVIEW EVENT INVOLVING PRESSURE BOUNDARY LEAKAGE WAS DESCRIBED IN LER 528/87-018.

[225] PALO VERDE 2 DOCKET 50-529 LER 91-004 REV 01
UPDATE ON REACTOR TRIP FOLLOWING GENERATOR/TURBINE TRIP.
EVENT DATE: 081691 REPORT DATE: 012492 NSSS: CE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223873) ON 8/16/91, AT APPROXIMATELY 0839 MST, UNIT 2 WAS IN MODE 1 (POWER OPERATION), OPERATING AT APPROXIMATELY 64% POWER WHEN A REACTOR TRIP OCCURRED DUE TO HIGH PRESSURIZER PRESSURE. IMMEDIATELY PRIOR TO THE RESCTOR TRIP, THE MAIN GENERATOR TRIPPED INITIATING A MAIN TURBINE TRIP AT APPROX. 0900 MST ON 8/16/91, THE PLANT WAS STABILIZED IN MODE 3 (HOT STANDBY) AT NORMAL OPERATING TEMPERATURE AND PRESSURE. THE EVENT WAS DIAGNOSED AS AN UNCOMPLICATED REACTOR TRIP. NO OTHER SAFETY SYSTEM RESPONSES OCCURRED AND NONE WERE REQUIRED. THE REACTOR TRIP ON HIGH PRESSURIZER PRESSURE HAS BEEN DETERMINED TO BE THE NORMAL PLANT RESPONSE TO A LOAD REJECTION AT 64% POWER WITH 7 OF THE 8 STEAM BYPASS CONTROL VALVES (SBCVS) IN SERVICE. THE CAUSE OF THE REACTOR TRIP WAS A GENERATOR/TURBINE TRIP IN COMBINATION WITH A COGNITIVE PERSONNEL ERROR BY THE CONTROL ROOM PERSONNEL WHO DID NOT COMPLY WITH PROCEDURAL REQUIREMENTS TO HAVE THE 8 SBCVS IN SERVICE AT POWER LEVELS BELOW 75%. CONTROL ROOM PERSONNEL WERE DISCIPLINED IN ACCORDANCE WITH THE PVNGS POSITIVE DISCIPLINE PROGRAM. THE MAIN GENERATOR TRIP WAS CAUSED WHEN A GENERREX A.C./D.C. GATE BOARDS STATIC SWITCH FAILED AND THE GENERREX EXCITATION CONTROL SYSTEM ENTERED AN ABNORMAL STATE (I.E., EXCESSIVELY HIGH FIELD VOLTAGE, CURRENT AND VARS). THE GATE BOARD WAS REPLACED AND THE GENERREX SYSTEM WAS SUCCESSFULLY RETESTED AND RETURNED TO SERVICE.

[226] PALO VERDE 2 DOCKET 50-529 LER 91-007
CONTAINMENT INTEGRITY VIOLATION DURING CORE ALTERATIONS.
EVENT DATE: 120591 REPORT DATE: 010292 NSSS: CE TYPE: PWR

(NSIC 223752) ON DECEMBER 5, 1991, DURING A PLANNED REFUELING OUTAGE, PALO VERDE UNIT 2 WAS IN MODE 6 (REFUELING) WITH THE REACTOR VESSEL HEAD REMOVED AND THE REACTOR COOLANT SYSTEM AT APPROXIMATELY 100 DEGREES FAHRENHEIT AND ATMOSPHERIC PRESSURE WHEN MAINTENANCE PERSONNEL DISCOVERED THAT A CONTAINMENT ISOLATION VALVE (SIA-UV-673) WHICH WAS REQUIRED TO BE CLOSED DURING CORE ALTERATIONS WAS IN THE OPEN POSITION. TECHNICAL SPECIFICATION LIMITING CONDITION FOR OPERATION (TS LCO) 3.9.4.C.1 REQUIRES THAT EACH PENETRATION PROVIDING DIRECT ACCESS FROM THE CONTAINMENT ATMOSPHERE TO THE OUTSIDE ATMOSPHERE SHALL BE CLOSED BY AN ISOLATION VALVE, BLIND FLANGE, OR MANUAL VALVE DURING CORE ALTERATIONS. IF TS LCO 3.9.4.C.1 IS NOT SATISFIED, THE ASSOCIATED TS LCO ACTION REQUIRES THE IMMEDIATE SUSPENSION OF ALL OPERATIONS INVOLVING CORE ALTERATIONS. FROM NOVEMBER 30, 1991 TO DECEMBER 5, 1991, CORE ALTERATIONS WERE IN PROGRESS. THEREFORE, THE ACTION REQUIREMENTS OF TS LCO 3.9.4 WERE NOT MET. THE CAUSE OF SIA-UV-673 BEING OPEN WAS DUE TO PROCEDURE INADEQUACY IN THAT THE VALVE STEM WAS NOT REQUIRED BY PROCEDURE TO BE MATCHMARKED OR LOCKED IN THE CLOSED POSITION WHEN THE ACTUATOR WAS REMOVED. THE VALVE DISC UNEXPECTEDLY OPENED. THE ACTUATOR WAS REINSTALLED WHEN THE VALVE DISC WAS OPEN. THIS RESULTED IN THE VALVE BEING OPEN WHEN THE INDICATED POSITION SHOWED THE VALVE TO BE CLOSED.

[227] PALO VERDE 2 DOCKET 50-529 LER 92-001
REACTOR TRIP DUE TO LOW STEAM GENERATOR LEVEL.
EVENT DATE: 010992 REPORT DATE: 020792 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 1 (PWR)
PALO VERDE 3 (PWR)
VENDOR: FOXBORO CO., THE

(NSIC 224050) AT APPROXIMATELY 1653 MST ON JANUARY 9, 1992, PALO VERDE UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 17 PERCENT POWER WHEN AN AUTOMATIC REACTOR TRIP OCCURRED DUE TO LOW LEVEL IN THE NUMBER 1 STEAM GENERATOR BECAUSE A FEEDWATER SUPPLY VALVE DID NOT OPEN AUTOMATICALLY AS REQUIRED. THE REACTOR TRIP OCCURRED DURING POWER ASCENSION FOLLOWING A REFUELING OUTAGE. FOLLOWING THE REACTOR TRIP, THE PLANT WAS STABILIZED IN MODE 3 (HOT STANDBY) AT NORMAL OPERATING TEMPERATURE AND PRESSURE. THE REACTOR TRIP WAS DIAGNOSED AS AN UNCOMPLICATED REACTOR TRIP IN ACCORDANCE WITH THE EMERGENCY PLAN IMPLEMENTING PROCEDURES. PRIOR TO THE AUTOMATIC REACTOR TRIP, CONTROL ROOM PERSONNEL BEGAN REDUCING POWER, STARTED THE "B" MOTOR DRIVEN AUXILIARY FEEDWATER PUMP AND ESTABLISHED AUXILIARY FEEDWATER FLOW TO THE NUMBER 1 STEAM GENERATOR. NO OTHER SAFETY SYSTEM RESPONSES OCCURRED AND NONE WERE REQUIRED. THE CAUSE OF THE FEEDWATER SUPPLY VALVE NOT OPENING WAS FATIGUE FAILURE OF WIRES CONNECTED TO A SWITCH ASSEMBLY ON THE AUTO/MANUAL CONTROLLER FOR THE VALVE. THE FATIGUE FAILURE OF THE WIRES PREVENTED THE FEEDWATER SUPPLY VALVE FROM RECEIVING AN OPEN SIGNAL FROM THE MAIN FEEDWATER CONTROL SYSTEM. THE CAUSE OF THE SWITCH ASSEMBLY FAILURE IS BEING INVESTIGATED. THE ROOT CAUSE OF FAILURE INVESTIGATION IS EXPECTED TO BE COMPLETED BY MARCH 9, 1992.

[228] PALO VERDE 3 DOCKET 50-530 LER 91-008
REACTOR TRIP DUE TO LIGHTNING INDUCED ELECTRICAL FAULT.
EVENT DATE: 111491 REPORT DATE: 121391 NSSS: CE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 223654) AT APPROXIMATELY 1449 MST ON NOVEMBER 14, 1991, PALO VERDE UNIT 3 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 100 PERCENT POWER WHEN A LIGHTNING INDUCED ELECTRICAL FAULT ON THE A PHASE MAIN TRANSFORMER CAUSED A GENERATOR TRIP, TURBINE TRIP, AND REACTOR POWER CUTBACK (RPCB). APPROXIMATELY 35 SECONDS FOLLOWING THESE EVENTS, THE REACTOR TRIPPED ON LOW DEPARTURE FROM NUCLEATE BOILING RATIO (DNBR) SIGNALS. THE PLANT WAS STABILIZED IN MODE 3 (HOT STANDBY) AT NORMAL OPERATING TEMPERATURE AND PRESSURE. THE CONTROL ROOM SHIFT SUPERVISOR CLASSIFIED THE EVENT AS AN UNCOMPLICATED REACTOR TRIP IN ACCORDANCE WITH THE EMERGENCY PLAN IMPLEMENTING PROCEDURE. NO OTHER SAFETY SYSTEM RESPONSES OCCURRED AND NONE WERE REQUIRED. THE POST TRIP REVIEW FOUND THAT THE LOW DNBR REACTOR TRIP WAS DUE TO A CONTROL ELEMENT ASSEMBLY (CEA) SUBGROUP DEVIATION. DURING THE POST TRIP INVESTIGATION, A PS ENGINEERING DISCOVERED THAT A PROBLEM WITH THE CONTROL ELEMENT ASSEMBLY CALCULATOR (CEAC) SOFTWARE DESIGN MAY HAVE DELAYED THE REACTOR TRIP FOR UP TO 16 SECONDS WHEN A SECOND CPC TIME DELAY WAS INITIATED. AT THE TIME OF THE SECOND TIME DELAY, ONE CEA SUBGROUP WAS THOUGHT TO BE MISALIGNED GREATER THAN THE ALLOWED LIMIT IN THE CPCS. THE CAUSE OF THE SECOND TIME DELAY WAS THAT THE CEAC SOFTWARE DESIGN DID NOT ANTICIPATE THAT THERE WOULD BE CEA SLIPS LASTING LESS THAN 0.5 SECONDS.

[229] PALO VERDE 3 DOCKET 50-530 LER 91-010 REV 01
UPDATE ON ESF ACTUATIONS CAUSED BY MANUAL DEENERGIZATION OF OFFSITE POWER.
EVENT DATE: 111591 REPORT DATE: 013092 NSSS: CE TYPE: PWR

(NSIC 223948) ON NOVEMBER 15, 1991, AT APPROXIMATELY 0913 MST, PALO VERDE UNIT 3 WAS IN MODE 3 (HOT STANDBY) WHEN CONTROL ROOM PERSONNEL, RESPONDING TO A REPORT THAT A MOBILE CRANE WAS IN CONTACT WITH AN ENERGIZED 13.8 KV OVERHEAD POWER LINE, SECURED POWER TO THE TRAIN B 13.8 KV NON-CLASS 1E INTERMEDIATE SWITCHGEAR BUS (NAN-S06). THIS RESULTED IN THE EXPECTED LOSS OF POWER TO THE TRAIN B CLASS 1E 4.16 KV BUS, AND A LOSS OF POWER (LOP) ENGINEERED SAFETY FEATURE ACTUATION SYSTEM (ESFAS) ACTUATION. THE TRAIN B EMERGENCY DIESEL GENERATOR (EDG) STARTED AND LOADED PER DESIGN. AT APPROXIMATELY 0914 MST, THE CONTROL ROOM WAS NOTIFIED THAT THE LINE THE CRANE WAS IN CONTACT WITH WAS STILL ENERGIZED AND THAT THE TRAIN A 13.8 KV NON-CLASS 1E INTERMEDIATE SWITCHGEAR BUS (NAN-S05) NEEDED TO BE DEENERGIZED. CONTROL ROOM PERSONNEL PROCEEDED TO REENERGIZE THE NAN-S06 BUS PRIOR TO DEENERGIZING THE NAN-S05 BUS. AT APPROXIMATELY 0922 MST, CONTROL ROOM PERSONNEL DEENERGIZED THE NAN-S05 BUS WHICH RESULTED IN THE EXPECTED LOSS OF POWER TO THE TRAIN A CLASS 1E 4.16 KV BUS AND A TRAIN A LOP ESFAS ACTUATION. THE TRAIN A EDG STARTED AND LOADED PER DESIGN. ALL EQUIPMENT FUNCTIONED AS DESIGNED. NO OTHER SAFETY SYSTEM RESPONSES OCCURRED AND NONE WERE REQUIRED. BASED ON

INVESTIGATION RESULTS, THE CAUSE OF THE CRANE COMING IN CONTACT WITH AN ENERGIZED 1?.8 KV POWER LINE WAS DETERMINED TO BE PERSONNEL ERROR.

[230] PALO VERDE 3 DOCKET 50-530 LER 91-009
INADVERTENT CONTROL ROOM ESSENTIAL FILTRATION ESF ACTUATION.
EVENT DATE: 111891 REPORT DATE: 121391 NSSS: CE TYPE: PWR

(NSIC 223655) ON NOVEMBER 18, 1991, AT APPROXIMATELY 0859 MST, PALO VERDE UNIT 3 WAS IN MODE 3 (HOT STANDBY) AT APPROXIMATELY 565 DEGREES FAHRENHEIT AND 2252 POUNDS PER SQUARE INCH-ABSOLUTE WHEN AN INADVERTENT CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNAL (CREFAS) BALANCE OF PLANT ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (BOP ESFAS) ACTUATION OCCURRED. THE ACTUATION WAS A RESULT OF AN IMPROPER RESTORATION BY A CONTROL ROOM OPERATOR OF THE CHANNEL A CREFAS MODULE THAT HAD BEEN PLACED IN BYPASS MODE FOR SCHEDULED SURVEILLANCE TESTING. THE CREFAS ACTUATION WAS VERIFIED NOT TO BE THE RESULT OF A VALID SIGNAL. THE EQUIPMENT ACTUATED BY THE CREFAS WAS VERIFIED TO OPERATE AS DESIGNED AND SUBSEQUENTLY RETURNED TO NORMAL CONFIGURATION. THE CAUSE OF THE INADVERTENT CREFAS ACTUATION WAS A COGNITIVE PERSONNEL ERROR BY THE OPERATOR WHO REMOVED THE CREFAS MODULE FROM BYPASS MODE PRIOR TO RESETTING THE TRIP INPUTS. THE ERROR WAS A RESULT OF NOT FOLLOWING AN APPROVED STATION PROCEDURE. AS CORRECTIVE ACTION, THE OPERATOR WAS TEMPORARILY REMOVED FROM SHIFT AND DISCIPLINE WAS ADMINISTERED IN ACCORDANCE WITH THE PVNGS POSITIVE DISCIPLINE PROGRAM. SIMILAR EVENTS WERE REPORTED IN LERS 528/86-022, 529/85-006, 529/86-039, AND 529 /90-011.

[231] PEACH BOTTOM 2 DOCKET 50-277 LER 91-036
TECHNICAL SPECIFICATION VIOLATION OCCURRED WHEN THE 'A' WASTE SAMPLE TANK WAS RELEASED WITHOUT PROPER SAMPLING DUE TO FAILURE TO FOLLOW PROCEDURE.
EVENT DATE: 110391 REPORT DATE: 112991 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 223479) ON 11/03/91 AT 0530 HOURS, IT WAS DISCOVERED THAT THE "A" WASTE SAMPLE TANK (WST) TREATED LIQUID WAS RELEASED WITHOUT THE APPROPRIATE SAMPLE AND RELEASE APPROVAL. THE "B" WST WAS SAMPLED INSTEAD OF THE "A" WST. THIS CONDITION RESULTED IN A VIOLATION OF TECHNICAL SPECIFICATION 4.8.B.1B BECAUSE THE SURVEILLANCE REQUIREMENTS WERE NOT PERFORMED. THE CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE THAT THE RADWASTE PLANT OPERATOR (PO) FAILED TO PROPERLY FOLLOW THE SURVEILLANCE TEST (ST) USED TO SAMPLE AND RELEASE THE WST CONTENTS. ADDITIONALLY, MISCOMMUNICATION BETWEEN THE RADWASTE PO AND THE SHIFT CHEMIST LED TO THE WRONG WST BEING SAMPLED. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS RESULT OF THIS EVENT. THE RELEASE WAS TERMINATED AND A LIQUID SAMPLE WAS OBTAINED FROM THE "A" WST. THE PERTINENT INFORMATION FROM THIS EVENT WILL BE PROVIDED TO THE APPROPRIATE PERSONNEL. THE ST WILL BE REVISED WITH FOCUS ON HUMAN FACTORS CONSIDERATIONS. THERE WAS ONE PREVIOUS SIMILAR LER IDENTIFIED.

[232] PEACH BOTTOM 2 DOCKET 50-277 LER 91-039
TECHNICAL SPECIFICATION REQUIRED SHUTDOWN FOR A RESIDUAL HEAT REMOVAL SYSTEM VALVE REPAIR.
EVENT DATE: 120691 REPORT DATE: 010692 NSSS: GE TYPE: BWR

(NSIC 223689) UNIT 2 WAS BROUGHT TO A COLD SHUTDOWN ON 12/6/91 DUE TO A DETERMINATION MADE ON 12/3/91 BY PLANT STAFF ENGINEERING THAT THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM "B" LOOP INJECTION CHECK VALVE AND/OR ITS ASSOCIATED BYPASS VALVE HAD EXCESSIVE THROUGH VALVE LEAKAGE. THE RHR "B" LOOP OUTBOARD ISOLATION VALVE WAS THEN DE-ENERGIZED CLOSED IN ACCORDANCE WITH THE TECHNICAL SPECIFICATIONS. THE CONDITION MADE THE RHR "B" LOOP INOPERABLE AND THE APPROPRIATE SEVEN DAY LIMITING CONDITION FOR OPERATION WAS ENTERED. IT WAS THEN DECIDED TO SHUT DOWN UNIT 2 SINCE THE RHR "B" LOOP INJECTION CHECK VALVE OR ITS ASSOCIATED BYPASS VALVE REPAIRS COULD NOT BE COMPLETE WITHOUT ENTERING THE INSERTED DRYWELL. THE CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE THAT THE RHR "B" LOOP INJECTION CHECK BYPASS VALVE HAD EXCESSIVE THROUGH VALVE LEAKAGE OF APPROXIMATELY 5 GPM. THE VALVE PLUG WAS CUT AND VALVE SEAT IMPERFECTIONS WERE IDENTIFIED. IT APPEARS THAT SOME SMALL OBSTRUCTION WAS PRESENT IN THE VALVE SEATING AREA. AN INVESTIGATION IS CONTINUING. NO ACTUAL SAFETY CONSEQUENCES

OCCURRED AS A RESULT OF THIS EVENT. THE RHR "B" LOOP INJECTION CHECK BYPASS WAS DISASSEMBLED AND REPAIRED. THE APPROPRIATE POST MAINTENANCE TESTING WAS COMPLETED SATISFACTORILY AND THE PLANT WAS RETURNED TO POWER OPERATION ON 12/18/91. THERE WERE NO PREVIOUS SIMILAR EVENTS IDENTIFIED.

[233] PEACH BOTTOM 3 DOCKET 50-278 LER 91-017
AUTOMATIC DEPRESSURIZATION SYSTEM INOPERABILITY DUE TO COMPONENTS BEING UNQUALIFIED AS A RESULT OF INSULATION NOT BEING INSTALLED PROPERLY.
EVENT DATE: 092491 REPORT DATE: 120691 NSSS: GE TYPE: BWR

(NSIC 223602) ON 9/24/91 AT 1300 HOURS THE MAIN STEAM RELIEF VALVE (MSRV) SOLENOID VALVES (SV) WIRING INSULATION WAS DISCOVERED TO BE DEGRADED. AN INVESTIGATION REVEALED THAT THE MSRV THERMAL INSULATION WAS IMPROPERLY INSTALLED. THIS CAUSED AN UNUSUALLY HIGH TEMPERATURE ENVIRONMENT IN THE IMMEDIATE VICINITY OF THE SVS AND ASSOCIATED WIRING. ON 11/8/91 IT WAS DETERMINED BY ENGINEERING ANALYSIS THAT THERE WAS NO LONGER A REASONABLE ASSURANCE THAT THE AUTOMATIC DEPRESSURIZATION SYSTEM RELIEF VALVES WERE OPERABLE. THE TEMPERATURE INCREASE FROM THE INSULATION INSTALLATION ERROR CAUSED THE EXPIRATION OF THE ENVIRONMENTAL QUALIFICATION LIFE OF THE COMPONENTS. THE CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE THAT THE MSRV INSULATION WAS IMPROPERLY INSTALLED. THE MAINTENANCE PROCEDURE DID NOT PROVIDE THE NECESSARY LEVEL OF DETAIL. THE INSULATION WAS PROPERLY REINSTALLED AND THE SVS WERE REPLACED. THE MAINTENANCE PROCEDURE WILL BE REVISED. INSULATION ON UNIT 2 WAS VERIFIED TO BE PROPERLY INSTALLED. EIGHT OF THE UNIT 3 MSRV SVS AND ASSOCIATED WIRING WERE SENT TO A TEST FACILITY TO UNDERGO TESTING IN AN ACCIDENT ENVIRONMENT. A FORMAL ROOT CAUSE INVESTIGATION IS CURRENTLY UNDERWAY. THE SAFETY CONSEQUENCES IS CURRENTLY UNDER REVIEW. NO PREVIOUS SIMILAR LERS IDENTIFIED.

[234] PEACH BOTTOM 3 DOCKET 50-278 LER 91-018
ENGINEERING SAFEGUARD FEATURE ACTUATION AS A RESULT OF LESS THAN ADEQUATE PROGRAMMATIC GUIDANCE.
EVENT DATE: 121691 REPORT DATE: 011492 NSSS: GE TYPE: BWR

(NSIC 223825) ON 12/16/91 AT 2205 HOURS, AN ISOLATION OF THE MAIN STEAM LINE DRAIN AND THE RECIRCULATION SYSTEM SAMPLE VALVES OCCURRED UNEXPECTEDLY DURING THE PERFORMANCE OF AN IN-SERVICE LEAK TEST. AT THE TIME OF THE EVENT, OPERATIONS PERSONNEL WERE VALVING INTO SERVICE THE INSTRUMENT VALVES ON A FEEDWATER CONTROL SYSTEM FLOW TRANSMITTER. THE CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE A PRESSURE TRANSIENT IN THE INSTRUMENT LINE. THIS WAS A RESULT OF THE INSTRUMENT LINE NOT BEING BACKFILLED PRIOR TO THE INSTRUMENT BEING RETURNED TO SERVICE. NO DOCUMENTED STATION GUIDANCE EXISTED TO ADDRESS OPERATION OF INSTRUMENT VALVES AND RESPONSIBILITIES. A CONTRIBUTING FACTOR TO THIS EVENT WAS LESS THAN ADEQUATE COMMUNICATION. AFTER THE GROUP 1 PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) ISOLATION OCCURRED, THE PC IS LOGIC WAS RESET AT 2220 HOURS AND THE AFFECTED VALVES WERE RESTORED TO THE PROPER POSITION. STATION GUIDANCE WILL BE DEVELOPED. THE EVENT WILL BE REVIEWED TO DETERMINE ADEQUACY OF THE PRESENT METHODS USED TO PERFORM IN-SERVICE LEAK TESTING. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. THERE WERE NO PREVIOUS SIMILAR EVENTS IDENTIFIED.

[235] PEACH BOTTOM 3 DOCKET 50-278 LER 91-019
ENGINEERED SAFEGUARD FEATURE ACTUATION AS A RESULT OF A RELAY COIL FAILURE.
EVENT DATE: 121891 REPORT DATE: 011692 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223826) ON 12/18/91 AT 1822 HOURS, AN ISOLATION OF THE SHUT DOWN COOLING (SDC) MODE OF THE RESIDUAL HEAT REMOVAL SYSTEM OCCURRED WHEN POWER WAS LOST TO THE PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) CONTROL LOGIC. THE UNIT 3 REACTOR OPERATOR IMMEDIATELY MAXIMIZED REACTOR WATER CLEANUP, CONTROL ROD DRIVE, AND CORE SPRAY STAYFULL FLOWS. THE CAUSE OF THE SDC ISOLATION HAS BEEN DETERMINED TO BE A PCIS RELAY COIL FAILURE. THIS CONDITION CAUSED THE PCIS CONTROL LOGIC FUSE TO BLOW. THE BLOWN FUSE AND RELAY WERE REPLACED. SUBSEQUENTLY, THE ISOLATION WAS RESET AND SDC WAS RESTORED TO SERVICE. THERE WERE NO ACTUAL SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT. THERE WERE NO PREVIOUS SIMILAR EVENTS IDENTIFIED.

[236] PEACH BOTTOM 3 DOCKET 50-278 LER 92-001
REACTOR CORE ISOLATION COOLING SYSTEM CONTROL LOGIC FUSE FAILURE DUE TO A FAILED LIGHT SOCKET.
EVENT DATE: 010392 REPORT DATE: 020192 NSSS: GE TYPE: BWR
VENDOR: DIALCO

(NSIC 223933) ON 01/03/92 AT 0005 HOURS, DURING THE PERFORMANCE OF THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM OPERATING PROCEDURE USED TO RESET THE LOW PRESSURE ISOLATION DURING REACTOR STARTUP, THE DIVISION I CONTROL POWER FUSE FAILED WHEN THE LOGIC RESET SWITCH WAS DEPRESSED AS SPECIFIED IN THE PROCEDURE. SUBSEQUENTLY, ON 01/04/92, THE DIVISION I CONTROL POWER FUSES FAILED WHEN THE LOGIC RESET SWITCH WAS DEPRESSED AS SPECIFIED IN A SURVEILLANCE TEST. THE LOSS OF POWER TO THE CONTROL CIRCUIT CAUSED THE CONDENSATE STORAGE TANK LOW LEVEL RELAY TO DE-ENERGIZE WHICH CAUSED THE RCIC TORUS SUCTION VALVES TO AUTOMATICALLY OPEN. THE CAUSE OF THE RCIC CONTROL LOGIC FUSE FAILURE HAS BEEN DETERMINED TO BE THE RESULT OF A SHORTED STATUS LIGHT SOCKET IN THE RCIC CONTROL LOGIC. A REVIEW OF THE STATUS LIGHTS IN THIS AND OTHER SYSTEMS WILL BE PERFORMED TO DETERMINE IF THE VOLTAGE SPIKES THAT OCCUR WILL ADVERSELY AFFECT THE FUNCTION OF THE CONTROL LOGIC. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. THERE WERE THREE PREVIOUS SIMILAR EVENTS IDENTIFIED.

[237] PERRY 1 DOCKET 50-440 LER 91-020 REV 01
UPDATE ON CABLE TRAY RACEWAYS FOUND TO BE IMPAIRED AS A FIRE BARRIER, ADVERSELY AFFECTING SAFE SHUTDOWN REQUIREMENTS.
EVENT DATE: 100791 REPORT DATE: 010392 NSSS: GE TYPE: BWR

(NSIC 223849) ON 10/7/91, DISCREPANCIES IN THE INSTALLATION OF THE MECHANICAL FASTENERS ON APPENDIX R RACEWAYS WAS DETERMINED TO BE A FIRE BARRIER IMPAIRMENT WHICH COULD ADVERSELY AFFECT SAFE SHUTDOWN REQUIREMENTS. ON 12/4/91, DURING CORRECTIVE ACTIONS FOR THIS CONDITION, DEFICIENCIES IN THE LENGTH AND THICKNESS OF FIRE BARRIER MATERIAL ON THE RACEWAYS SUPPORTS/HEAT TRANSFER ITEMS WERE ALSO DISCOVERED. TECHNICAL SPECIFICATIONS REQUIRE THESE CONDITIONS BE REPORTED TO THE NRC THROUGH THE LER PROCESS. THE MECHANICAL FASTENER DISCREPANCY IS ALSO CONSIDERED REPORTABLE UNDER 10CFR21. THE CAUSE OF THE MECHANICAL FASTENER DISCREPANCY WAS INADEQUATE DESIGN. THE INFORMATION, PROVIDED BY GILBERT COMMONWEALTH, RESULTED IN VARIOUS DESIGN DOCUMENTS NOT REFLECTING THE VENDOR'S, THERMAL SCIENCE, INC. (TSI), INSTALLATION CRITERIA CONCERNING MAXIMUM SPACING OF THE MECHANICAL FASTENERS. AN INADEQUATE INSTALLATION INSTRUCTION AND INSTALLATION ERRORS RESULTED IN THE HEAT TRANSFER ITEMS THICKNESS AND LENGTH DISCREPANCIES. THE IMMEDIATE CORRECTIVE ACTION WAS TO INITIATE HOURLY FIRE WATCHES FOR THE FIRE IMPAIRMENT SITUATION, WHICH WILL CONTINUE UNTIL THE APPENDIX R RACEWAYS ARE UPGRADED TO ENSURE COMPLIANCE WITH ALL SPECIFICATIONS. THE AFFECTED DESIGN DOCUMENTS WILL BE REVISED TO REFLECT THE INSTALLATION CRITERIA SPECIFIED BY THE TSI VENDOR MANUAL.

[238] PERRY 1 DOCKET 50-440 LER 91-024
LOSS OF EMERGENCY SERVICE WATER SYSTEM LOOP DUE TO INADVERTENT ISOLATION OF KEEPFILL SYSTEM.
EVENT DATE: 120491 REPORT DATE: 010692 NSSS: GE TYPE: BWR

(NSIC 223806) ON DECEMBER 4, 1991, A PERRY PLANT OPERATOR (PPO) PERFORMING A WEEKLY CHECK OF THE EMERGENCY SERVICE WATER (ESW) KEEPFILL SYSTEM DISCOVERED THE KEEPFILL SYSTEM PRESSURE READING APPROXIMATELY 3.5 PSIG. THE KEEPFILL SYSTEM PRESSURE IS REQUIRED TO BE >/= 13 PSIG WHEN THE ESW A LOOP IS IN STANDBY. THE PPO CHECKED THE POSITION OF KEEPFILL ISOLATION VALVE 1P45-F720A AND FOUND IT CLOSED. THE ESW A LOOP AND ASSOCIATED LOADS WERE DECLARED INOPERABLE IN ACCORDANCE WITH THE APPLICABLE TECHNICAL SPECIFICATION ACTION STATEMENTS. THE MISPOSITIONING OF VALVE 1P45-F720A WAS ATTRIBUTED TO PERSONNEL ERROR. THE LAST AUTHORIZED REPOSITIONING OF 1P45-F720A OCCURRED ON NOVEMBER 21, 1991. INTERVIEWS OF PERSONNEL PERFORMING WORK IN THE VICINITY OF THE KEEPFILL ISOLATION VALVE DID NOT REVEAL THE SOURCE OF THE ERROR. IT WAS THEREFORE ASSUMED THAT THE VALVE MISPOSITIONING WAS AN UNINTENTIONAL ERROR BY AN UNIDENTIFIED PERSON. TO PREVENT RECURRENCE OF A SIMILAR INCIDENT INVOLVING THE KEEPFILL ISOLATION VALVE, THE REQUIRED VALVE POSITION WILL BE CHANGED FROM NORMALLY OPEN TO LOCKED OPEN.

ADDITIONALLY, ALL LICENSED AND NON-LICENSED PLANT OPERATORS WILL RECEIVE TRAINING ON THIS EVENT AS PART OF REQUALIFICATION TRAINING.

[239] PERRY 1 DOCKET 50-440 LER 91-025
CRACKED WELD ON VALVE APPENDAGE RESULTS IN HIGH PRESSURE CORE SPRAY SYSTEM INOPERABILITY.
EVENT DATE: 121291 REPORT DATE: 011392 NSSS: GE TYPE: BWR

(NSIC 223805) ON DECEMBER 12, 1991, AT 1100, LIQUID PENETRANT TESTING REVEALED A CRACK IN A WELD UPSTREAM OF THE HIGH POINT VENT VALVE ON THE HIGH PRESSURE CORE SPRAY (HPCS) SYSTEM TEST RETURN LINE. THE CONTROL ROOM WAS CONTACTED AND HPCS, WHICH HAD BEEN IN STANDBY READINESS, WAS DECLARED INOPERABLE ON DECEMBER 13, 1991, AT 1542. THE CRACKED WELD WAS REWORKED TO SPECIFICATIONS AND HPCS WAS DECLARED OPERABLE ON DECEMBER 15, 1991, AT 0510. THE WELD FAILURE IS ATTRIBUTED TO A COMBINATION OF FACTORS. THE INSPECTED WELDS WERE BEING CHECKED UNDER A PLANT INITIATIVE TO INSPECT WELDS AS A RESULT OF A PREVIOUS LER. THIS WELD WAS CHOSEN DUE TO BEING LOCATED IN AN AREA SUBJECT TO RELATIVELY HIGH VIBRATION. THE WELD WAS AT THE BRANCH CONNECTION BETWEEN THE VENT PIPING AND TEST RETURN PIPING. THE APPENDAGE HAD TWO 3/4 INCH VENT VALVES SUPPORTED SOLELY BY THIS BRANCH CONNECTION. ADDITIONALLY, FIELD INSPECTION OF THE APPENDAGE BY ENGINEERING PERSONNEL DETERMINED THAT THE CRACKED WELD DID NOT FULLY MEET THE SPECIFICATIONS REQUIRED BY THE DESIGN DRAWINGS, HAVING INSUFFICIENT WELD METAL AND CONTOUR. THE SCHEDULE TO INSPECT OTHER WELDS IDENTIFIED BY THE ACTIVITIES RESULTING FROM THE PREVIOUS LER HAS BEEN ACCELERATED. ENGINEERING PERSONNEL ARE IN THE PROCESS OF MONITORING THE REWORKED WELD ON THE HPCS SYSTEM FOR VIBRATION TO DETERMINE IF ADDITIONAL CORRECTIVE ACTIONS MAY BE REQUIRED.

[240] PERRY 1 DOCKET 50-440 LER 91-026
DIVISION I REACTOR CORE ISOLATION COOLING ISOLATION DURING SURVEILLANCE ACTIVITIES.
EVENT DATE: 121691 REPORT DATE: 011092 NSSS: GE TYPE: BWR

(NSIC 223742) ON DECEMBER 16, 1991, AT 0148, A DIVISION I NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM SIGNAL RESULTED IN AN UNEXPECTED REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM ISOLATION DURING THE PERFORMANCE OF A SURVEILLANCE. ON DECEMBER 16, 1991, AT 0135, INSTRUMENT AND CONTROLS (I&C) TECHNICIANS BEGAN MONTHLY SURVEILLANCE, (SVI-E31-T0100-A) "RCIC STEAM SUPPLY PRESSURE-LOW CHANNEL A FUNCTIONAL FOR 1E31-N685A." THE I&C LEAD PERFORMER OF THIS SURVEILLANCE INCORRECTLY READ THAT THE STEP TO LIFT LEADS DISABLING THE RCIC ISOLATION DID NOT HAVE TO BE PERFORMED. WHEN THE TRIP SIGNAL WAS APPLIED THE SYSTEM ISOLATED AS DESIGNED. THE CAUSE OF THIS EVENT IS PERSONNEL ERROR, INATTENTION TO DETAIL. THE TECHNICIAN PERFORMING THE SURVEILLANCE WAS COUNSELED, INVOLVED IN THIS INVESTIGATION, AND IS FULLY AWARE OF THE CONSEQUENCES OF HIS ACTIONS. THIS LER WILL BE DISCUSSED WITH ALL I&C TECHNICIANS AS PART OF THEIR CONTINUING TRAINING PROGRAM.

[241] PERRY 1 DOCKET 50-440 LER 91-027
PLANT SHUTDOWN DUE TO CIRCULATION WATER SYSTEM PIPE RUPTURE.
EVENT DATE: 122291 REPORT DATE: 012192 NSSS: GE TYPE: BWR
VENDOR: GOULD INC.
ITT HAMMEL DAHL CONOFLOW

(NSIC 223840) ON DECEMBER 22, 1991, AT 0205 HOURS, A MANUAL REACTOR SCRAM WAS INSERTED PER INTEGRATED OPERATING INSTRUCTION (IOI) - 8 DUE TO A NON-ISOLABLE BREAK IN THE CIRCULATING WATER SYSTEM (N71) PIPING. THE BREAK OCCURRED ON A SECTION OF 36 INCH FIBERGLASS REINFORCED PLASTIC PIPE WHICH SUPPLIES COOLING WATER TO THE AUXILIARY CONDENSERS. AT 0259, THE SHIFT SUPERVISOR DECLARED AN ALERT BASED ON REPORTS OF RISING WATER LEVEL RECEIVED AND INDICATIONS AVAILABLE IN THE CONTROL ROOM. ALL REQUIRED NOTIFICATIONS WERE MADE REGARDING THE ALERT DECLARATION. EQUIPMENT ANOMALIES AND MALFUNCTIONS WHICH OCCURRED AFTER THE MANUAL SCRAM WAS INSERTED ARE SUMMARIZED IN THE TEXT OF THIS LER. THE CAUSE OF THE PIPE RUPTURE WAS ATTRIBUTED TO A COMBINATION OF FACTORS WHICH INCLUDED A PRE-EXISTING FLAW IN THE FIBERGLASS, FUNCTIONAL DEGRADATION OF A PIPE SUPPORT AND

IMPROPER INSTALLATION OF AN O-RING GASKET AT THE FIBERGLASS TO STEEL TRANSITION FLANGE. THE PLANT WAS RESTARTED ON JANUARY 3, 1992 AFTER REPAIRS WERE MADE TO THE AFFECTED PIPING AND ASSOCIATED SUPPORTS. ADDITIONALLY, THE EQUIPMENT ANOMALIES AND MALFUNCTIONS WHICH OCCURRED AFTER PLANT SHUTDOWN WERE INVESTIGATED AND CORRECTIVE ACTIONS TAKEN WHERE APPROPRIATE.

[242] POINT BEACH 1 DOCKET 50-266 LER 91-015 REV 01
UPDATE ON "A" STEAM GENERATOR MAIN STEAM ISOLATION BYPASS VALVE LEFT OPEN.
EVENT DATE: 112791 REPORT DATE: 012392 NSSS: WE TYPE: PWR

(NSIC 223823) ON NOVEMBER 28, 1991, UNIT 1 WAS IN HOT SHUTDOWN; AND OPERATING PROCEDURE OP-13B , "SECONDARY SYSTEMS SHUTDOWN," WAS IN PROGRESS. THIS PROCEDURE WAS BEING PERFORMED IN ORDER TO CONDUCT A CONDENSER INSPECTION FOLLOWING A 17 GALLON PER MINUTE CONDENSER TUBE LEAK. AFTER SHUTTING BOTH MAIN STEAM ISOLATION VALVES (MSIV), PER THE PROCEDURE, THE OPERATORS NOTICED THAT THE MAIN STEAM HEADER WAS STILL PRESSURIZED TO APPROXIMATELY 200 PSIG. AN INVESTIGATION REVEALED THAT 1MS-234, THE MAIN STEAM ISOLATION BYPASS VALVE FOR THE "A" STEAM GENERATOR, WAS OPEN. THE VALVE WAS IMMEDIATELY SHUT, CORRECTING THE PROBLEM.

[243] POINT BEACH 2 DOCKET 50-301 LER 91-001 REV 01
UPDATE ON FAILURE OF MAIN STEAM ISOLATION VALVES TO CLOSE.
EVENT DATE: 092991 REPORT DATE: 012492 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: POINT BEACH 1 (PWR)
VENDOR: ATWOOD & MORRILL CO., INC.

(NSIC 223827) AT 0930 ON SEPTEMBER 29, 1991, AN ATTEMPT TO SHUT THE POINT BEACH NUCLEAR PLANT UNIT 2 MAIN STEAM ISOLATION VALVES (MSIVS) WAS MADE FROM THE CONTROL ROOM. BOTH MSIVS FAILED TO LEAVE THE FULLY OPEN POSITION. AN OPERATOR WAS DISPATCHED TO THE VALVES AND SHUT THEM BY APPLYING MECHANICAL FORCE TO THE VALVE OPERATORS. AFTER THE VALVE OPERATORS WERE FREED BY MECHANICAL FORCE, THE VALVES SHUT UNASSISTED. UNIT 2 HAD BEEN SHUT DOWN AND COOLED DOWN TO APPROXIMATELY 325F FOR THE BEGINNING OF ITS ANNUAL MAINTENANCE AND REFUELING OUTAGE WHEN THIS EVENT OCCURRED. AN EXTENSIVE INVESTIGATION INTO THE CAUSE OF THE FAILURE TO CLOSE IS BEING PERFORMED. THE CAUSE FOR THE VALVES FAILURE TO CLOSE HAS INITIALLY BEEN ATTRIBUTED TO DEGRADATION OF THE VALVE OPERATORS DUE TO CORROSION. STEPS ARE BEING TAKEN TO PREVENT RECURRENCE OF THE CORROSION AND RETURN THE OPERATORS TO SERVICE. MODIFICATIONS TO THE VALVES AND ASSOCIATED OPERATORS AND CHANGES TO THE VALVE MAINTENANCE PROGRAM ARE BEING CONSIDERED. AN INDEPENDENT NUCLEAR POWER DEPARTMENT TEAM INVESTIGATION WAS ALSO PERFORMED TO ASSESS MSIV PERFORMANCE AND PERCEPTIONS OF VALVE PERFORMANCE.

[244] POINT BEACH 2 DOCKET 50-301 LER 91-005
BOTH SAFETY INJECTION PUMP BREAKERS RACKED IN WITH REACTOR COOLANT TEMPERATURE LESS THAN 275F.
EVENT DATE: 110991 REPORT DATE: 120691 NSSS: WE TYPE: PWR

(NSIC 223605) ON NOVEMBER 11, 1991, A REACTOR COOLANT SYSTEM (RCS) HEATUP WAS IN PROGRESS FOR UNIT 2. AT APPROXIMATELY 1330, WITH THE RCS AT APPROXIMATELY 345F AND 350 PSIG, IT WAS DISCOVERED THAT BOTH HIGH HEAD SAFETY INJECTION (SI) PUMP BREAKERS WERE RACKED IN, MAKING BOTH SI PUMPS OPERABLE. IT WAS DETERMINED THAT THIS CONDITION EXISTED SINCE NOVEMBER 9, 1991, WHEN THE RCS WAS CLOSED WITH RCS COLD LEG TEMPERATURE 275F. HAVING BOTH SI PUMPS OPERABLE UNDER THIS CONDITION IS A VIOLATION OF TECHNICAL SPECIFICATION SECTION 15.3.15.B.1. TECHNICAL SPECIFICATION 15.3.15.B.1 REQUIRES THAT ONE SI PUMP BE RENDERED INOPERABLE BY REMOVING (RACKING OUT) ITS SUPPLY BREAKER OR BY SHUTTING ITS DISCHARGE VALVE AND REMOVING CONTROL POWER ANY TIME THE RCS IS NOT OPEN TO ATMOSPHERE AND TEMPERATURE IN EITHER OR BOTH RCS COLD LEGS IS <275F. RCS TEMPERATURE HAD BEEN LESS THAN 275F UNTIL APPROXIMATELY 1145 ON NOVEMBER 11, 1991 DURING THE PLANT HEATUP.

[245] POINT BEACH 2 DOCKET 50-301 LER 91-006
REACTOR TRIP DURING MODIFICATION WORK ON D11.
EVENT DATE: 121791 REPORT DATE: 011592 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: POINT BEACH 1 (PWR)
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 223794) AT 1008 ON DECEMBER 17, 1991, POINT BEACH NUCLEAR PLANT UNIT 2 EXPERIENCED A REACTOR TRIP DURING MODIFICATION WORK ON DC DISTRIBUTION PANEL D11. WHILE A CABLE WAS BEING PULLED FROM PANEL D11 AS PART OF THE MODIFICATION, A CABLE CONNECTED TO BREAKER 32 WAS DISPLACED BY THE CABLE BEING REMOVED. THIS MOVEMENT CAUSED THE TERMINATION IN BREAKER 32 TO LOOSEN AND ALLOW ITS ASSOCIATED CABLE TO BECOME DISCONNECTED. THIS CABLE SUPPLIES POWER TO DC DISTRIBUTION PANEL D22. PANEL D22 SUPPLIES POWER TO THE UNIT 2 "A" TRAIN REACTOR PROTECTION CIRCUITRY. PANEL D22 BECAME DEENERGIZED, RESULTING IN A REACTOR TRIP ON UNIT 2 AND ACTUATION OF THE CROSSOVER STEAM DUMPS ON UNITS 1 AND 2. THE UNIT 1 TURBINE WAS MANUALLY RUN BACK TO 73% POWER TO ALLOW CLOSURE OF THE UNIT 1 CROSSOVER STEAM DUMP WITHOUT EXCEEDING REACTOR POWER LIMITATIONS. THE UNIT 2 CONDENSER STEAM DUMPS WERE NOT ENABLED BECAUSE OF THE LOSS OF DC POWER TO THE ARMING CIRCUIT. PANEL D22 WAS REENERGIZED AT 1037. UNIT 1 WAS RETURNED TO FULL POWER AT 1350. AFTER THE ELECTRICAL CONNECTIONS IN PANELS D11 AND D13 WERE CHECK-TIGHTENED, UNIT 2 WAS PLACED BACK ON LINE AT 0346 ON DECEMBER 18. UNIT 2 REACHED FULL POWER AT 2245 ON DECEMBER 18, 1991. THIS EVENT IS AN ACTUATION OF THE REACTOR PROTECTION SYSTEM (RPS). THEREFORE, A FOUR-HOUR NOTIFICATION TO THE NRC WAS MADE IN ACCORDANCE WITH 10 CFR 50.72(B)(2)(II).

[246] QUAD CITIES 1 DOCKET 50-254 LER 91-016
SUPPORT EMBEDMENT PLATE OUTSIDE DESIGN BASIS DUE TO A PRESERVICE ERROR INVOLVING CONTRACTOR AND ENGINEERING PERSONNEL.
EVENT DATE: 083091 REPORT DATE: 092791 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 223814) ON 8/30/91 AT 1500 HOURS, UNIT ONE WAS IN THE RUN MODE AT 99% RATED CORE THERMAL POWER (RCTP) AND UNIT TWO WAS IN THE RUN MODE AT 97% RCTP. AT THIS TIME THE STATION WAS INFORMED BY THE NUCLEAR ENGINEERING DEPARTMENT (NED) THAT EMBEDMENT PLATES FOR SUPPORT (SPT) NM-RS-06 LOCATED ON UNIT ONE DRYWELL/TORUS (BF) VACUUM RELIEF LINE 1-1606-18-LX AND SUPPORT NM-RS-07 LOCATED ON UNIT TWO DRYWELL/TORUS VACUUM RELIEF LINE 2-1604-18-LX FAILED FSAR ALLOWABLE STRESS LIMIT REQUIREMENTS. BOTH EMBEDMENT PLATES WERE, HOWEVER, OPERABLE. THE NRC WAS NOTIFIED AT 1533 HOURS VIA ENS PER THE REQUIREMENTS OF 10CFR50.72(B)(1)(II)(B). THE CAUSE OF THIS EVENT WAS A DESIGN ERROR DURING A MODIFICATION IN 1980 BECAUSE THE AS-BUILT CONFIGURATIONS WERE NOT ACCURATELY DOCUMENTED ON DRAWINGS USED FOR THE ORIGINAL PIPING STRESS ANALYSIS. THIS ERROR WAS IDENTIFIED DURING PIPING CONFIGURATION VERIFICATION PROGRAM (PCVP) THAT COMMONWEALTH EDISION (CECO) UNDERTOOK IN 4/87. THE PCVP WAS TO VERIFY THE EXISTENCE AND LOCATION OF PIPE SUPPORTS AS WELL AS THE DETAILS UTILIZED FOR THE CONSTRUCTION OF BRANCH CONNECTIONS WITH AS-DESIGNED/ANALYZED CONFIGURATIONS FOR QUAD CITIES UNIT ONE AND TWO. THIS CONDITION WAS INITIALLY IDENTIFEDY BY S&L IN THE 88-89 TIME FRAME, HOWEVER, DUE TO A WEAKNESS IN THE INTERFACE BETWEEN SARGENT & LUNDY, NUTECH AND COMMONWEALTH EDISON THE ERROR WAS NOT REPORTED.

[247] QUAD CITIES 1 DOCKET 50-254 LER 91-023
CONTROL ROOM HVAC ISOLATION DUE TO A DRIED OUT CHLORINE ANALYZER PROBE DUE TO UNKNOWN CAUSE.
EVENT DATE: 112691 REPORT DATE: 122091 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 223699) ON NOVEMBER 26, 1991, UNIT ONE WAS IN THE RUN MODE AT 100 PERCENT OF RATED CORE THERMAL POWER. UNIT TWO WAS ALSO IN THE RUN MODE AT 74 PERCENT OF RATED CORE THERMAL POWER. AT 1330 HOURS, THE CONTROL ROOM (CR) HVAC TOXIC GAS ANALYZER ANNUNCIATED THE "CONTROL ROOM STANDBY HVAC SYSTEM MAJOR TROUBLE" ALARM IN THE CR AND CAUSED THE CR VENTS TO ISOLATE. THE INSTRUMENT MAINTENANCE DEPARTMENT WAS GETTING READY TO PERFORM THEIR FUNCTIONAL TEST WHEN THE EVENT OCCURRED. AN EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE CALL WAS COMPLETED AT 1454 HOURS PER 10CFR50.72(B)(2)(II). THE APPARENT CAUSE OF THE EVENT IS UNKNOWN.

THE POTENTIAL CAUSE OF THE EVENT IS ALLOWING THE PROBE TO DRY OUT BY NOT PERFORMING THE PREVENTATIVE MAINTENANCE. CORRECTIVE ACTIONS INCLUDE DECLARING THE TOXIC GAS ANALYZER INOPERABLE, THE INSTRUMENT MAINTENANCE DEPARTMENT (IMD) SPEAKING WITH THE VENDOR, AND ORDERING PROBES . FUTURE CORRECTIVE ACTIONS WILL INCLUDE REVISING PROCEDURES TO INCLUDE VENDOR INFORMATION ON THE CARE AND OPERATION OF THE PROBES IN THE STOREROOM. A NEW PROCEDURE WILL BE WRITTEN ON HOW TO REVIVE A PROBE ONCE IT HAS BEEN DRIED OUT. THIS REPORT IS BEING SUBMITTED TO COMPLY WITH 10CFR50.73(A)(2)(IV).

[248] QUAD CITIES 1 DOCKET 50-254 LER 91-024
FAILURE OF CO2 EXTINGUISHING SYSTEMS DUE TO INADEQUATE FLOW RATES.
EVENT DATE: 120391 REPORT DATE: 12309° NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)
VENDOR: CHEMETRON CORP.

(NSIC 223874) ON 11/21/91 AT 0225 HOURS, UNIT ONE WAS IN THE RUN MODE AT 98% OF RATED CORE THERMAL POWER. WHILE PERFORMING SPECIAL TEST 1-170, THE CO2 FLOODING SYSTEM (KQ) FOR THE UNIT ONE DIESEL GENERATOR (EK) (DG) ROOM DID NOT MEET THE SPECIFIED CONCENTRATION REQUIREMENT, AND THE SYSTEM WAS DECLARED INOPERABLE. ON 12/3/91 AT 1400 HOURS, THE UNIT 1/2 DG ROOM CO2 SYSTEM WAS DECLARED INOPERABLE. BASED ON FLOW RATE CALCULATIONS, IT WAS DETERMINED THAT THE REQUIRED DISCHARGE TIME FOR THE CO2 SYSTEMS FOR THE UNIT ONE AND UNIT 1/2 DG ROOMS EXCEEDED THE REQUIRED TIME OF ONE MINUTE. ON 12/3, AT 1429 HOURS, AN EMERGENCY NOTIFICATION SYSTEM CALL WAS MADE IN ACCORDANCE WITH 10CFR50.72(B)(1)(II)(B). THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)92)(II)(B) AND TECHNICAL SPECIFICATION 3.12/4.12.D.5. THE APPARENT CAUSE OF THE EVENT WAS IMPROPER SYSTEM INSTALLATION. THE IMMEDIATE CORRECTIVE ACTION INCLUDED ESTABLISHING BACKUP FIRE SUPPRESSION AND FIRE WATCHES FOR THE AFFECTED AREAS. THE DISCHARGE NOZZLES WILL BE REPLACED WITH LARGER NOZZLES, AND THE SYSTEMS WILL BE FURTHER TESTED.

[249] QUAD CITIES 1 DOCKET 50-254 LER 91-026
BREACH OF SECONDARY CONTAINMENT 1A AND 2A DRYWELL TO TORUS PURGE FAN DAMPERS.
EVENT DATE: 121091 REPORT DATE: 010892 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 223700) BEGINNING ON DECEMBER 9 AND CONTINUING INTO DECEMBER 10, 1991, OPERATIONS STAFF PLACED THE 1A AND 2A DRYWELL/TORUS PURGE FANS OUT-OF-SERVICE FOR MECHANICAL MAINTENANCE TO SEAL ACCESS DOOR LEAKS AND REPLACE FILTERS. DUE TO A MISCONCEPTION BY THE EQUIPMENT ATTENDANTS, AND THE LACK OF LABELS ON THE DAMPER OPERATORS OF THE PURGE SYSTEM, THE DAMPERS WERE WIRED OPEN INSTEAD OF WIRED CLOSED AS THE OUT-OF-SERVICE REQUEST REQUIRED. WHEN MECHANICAL MAINTENANCE PULLED THE ACCESS COVER FROM THE 1A PURGE FAN, A BREAK IN SECONDARY CONTAINMENT WAS CREATED VIA THE REACTOR BUILDING VENTILATION EXHAUST SYSTEM THROUGH THE WIRED OPEN DISCHARGE DAMPERS AND OUT THE OPEN ACCESS COVER. DEINERTING AND REINSERTING OF THE UNIT 2 DRYWELL WAS UNDERWAY SIMULTANEOUS TO THIS MAINTENANCE WORK. SOON AFTER MECHANICAL MAINTENANCE PULLED THE ACCESS COVER TO THE 2A PURGE FAN THE CONTROL ROOM RECEIVED A HIGH DIFFERENTIAL PRESSURE ALARM ON THE PURGE FILTERS. AN EQUIPMENT ATTENDANT SENT TO INVESTIGATE RECOGNIZED THAT THE DAMPERS WERE OOS IN THE WRONG POSITION AND INITIATED THE ACTIONS WHICH PLACED THE DAMPERS IN THE CORRECT, CLOSED, POSITION. THIS ACTION RESTORED SECONDARY CONTAINMENT. CORRECTIVE ACTIONS INCLUDE LABELLING PURGE FAN DAMPER POSITIONS, REVISING STATION PROCEDURES TO MORE CLEARLY DEFINE THE VERIFICATION PROCESS FOR DAMPERS, AND INVESTIGATE THE INSTALLATION OF DAMPER INSPECTION DOORS.

[250] QUAD CITIES 1 DOCKET 50-254 LER 91-027
U-1 SHUTDOWN FROM WATER LEAKING ONTO BUS 14-1.
EVENT DATE: 121691 REPORT DATE: 011592 NSSS: GE TYPE: BWR

(NSIC 223821) ON DECEMBER 16, 1991, UNIT ONE WAS IN THE RUN MODE AT 100 PERCENT OF RATED CORE THERMAL POWER. A LARGE LEAK WAS DISCOVERED TO BE COMING FROM AN OPEN DRAIN VALVE IN THE U1 REACTOR BUILDING VENTILATION (VA) HEATING STEAM SYSTEM. THE WATER DRAINED THROUGH A FLOOR PIPE PENETRATION ONTO EMERGENCY BUS (BU) 14-1 SWITCHGEAR (SWGR). SEVERAL ALARMS WERE RECEIVED IN THE CONTROL ROOM,

INCLUDING "4KV OVERCURRENT," "125 V BATTERY GROUND," AND "DIESEL GEN 1 RELAY TRIP." THE U1 DIESEL GENERATOR (DG) WAS THEN DECLARED INOPERABLE. THE LEAK WAS ISOLATED, BUS 14-1 WAS COVERED WITH PLASTIC, AND A LOAD DROP WAS STARTED AT 1430 HOURS FOR REMOVAL OF BUS 14-1 FROM SERVICE. BUS 14-1 WAS REMOVED FROM SERVICE AT 1450 HOURS. THE STATION ENTERED TECHNICAL SPECIFICATIONS (TS) LCO 3.0.A DUE TO ONE WHOLE DIVISION OF EMERGENCY CORE COOLING SYSTEM (ECCS) PUMPS AND THE U1 DG BEING INOPERABLE. A GENERATING STATION EMERGENCY PLAN (GSEP) UNUSUAL EVENT WAS DECLARED DUE TO TS LCO 3.0.A. UNIT ONE WAS MANUALLY SCRAMMED AT 0113 HOURS ON DECEMBER 17, 1991. UNIT ONE WAS IN COLD SHUTDOWN AT 0622 HOURS. TS LCO 3.0.A AND THE GSEP UNUSUAL EVENT WERE THEN TERMINATED.

[251] QUAD CITIES 1 DOCKET 50-254 LER 91-028
LOSS OF POWER TO 1A RPS BUS CAUSED BY EPA 1A-1 TRIPPING ON UNDERVOLTAGE DUE TO LOW M-G SET OUTPUT.
EVENT DATE: 122091 REPORT DATE: 011792 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.
OHMITE

(NSIC 223853) AT 1720 HOURS ON DECEMBER 20, 1991, UNIT ONE WAS IN THE RUN MODE AT 100% OF RATED CORE THERMAL POWER. REACTOR PROTECTION SYSTEM (RPS) BUS 1A UNEXPECTEDLY LOST POWER. THIS CAUSED A 1/2 SCRAM AND PRIMARY CONTAINMENT ISOLATION (PCI) GROUPS I, II, AND III ISOLATION SIGNALS TO BE RECEIVED IN THE CONTROL ROOM. IT ALSO CAUSED REACTOR WATER CLEAN-UP (RWCU) PUMPS TO TRIP, AND CONTROL ROOM VENTS AND REACTOR BUILDING VENTS TO ISOLATE AS DESIGNED. THE LOSS OF POWER WAS DUE TO AN UNDERVOLTAGE TRIP ON AN ELECTRICAL PROTECTION ASSEMBLY FOR THE RPS 1A MOTOR-GENERATOR (M-G) SET. AT 1735 HOURS, RPS BUS 1A WAS ENERGIZED FROM ITS RESERVE FEED. THE 1/2 SCRAM AND PCI GROUP ISOLATIONS WERE RESET, AND RWCU, CONTROL ROOM VENTS, AND REACTOR BUILDING VENTS WERE RETURNED TO SERVICE. THE APPARENT CAUSE OF THE EVENT IS LOW GENERATOR OUTPUT VOLTAGE. THE EXACT CAUSE FOR THE DROP IN VOLTAGE IS UNKNOWN. NO APPARENT ELECTRICAL LOAD FAILURE WAS DISCOVERED DURING THE INVESTIGATION. THE POSSIBLE CAUSES ARE BELIEVED TO BE NORMAL WEAR OR CORROSION OF THE VOLTAGE REGULATOR RHEOSTAT AT THE POINT OF PREVIOUS ADJUSTMENT, OR A SPURIOUS VOLTAGE REGULATOR FAILURE. AFTER ADJUSTING THE RHEOSTAT FOR A HIGHER GENERATOR OUTPUT VOLTAGE, THE M-G SET FUNCTIONED PROPERLY AND WAS RETURNED TO SERVICE. THE STATION IS INVESTIGATING REPLACEMENT OF THE VOLTAGE REGULATOR AND RHEOSTAT.

[252] QUAD CITIES 2 DOCKET 50-265 LER 92-004
INADVERTENT CLOSURE OF U2 REACTOR BUILDING ISOLATION DAMPER.
EVENT DATE: 011691 REPORT DATE: 020592 NSSS: GE TYPE: BWR

(NSIC 224028) ON JANUARY 16, 1992 AT 0040 HOURS, UNIT TWO WAS IN THE SHUTDOWN MODE FOR A REFUELING OUTAGE. TECH STAFF WAS PERFORMING QTS 105-9, "PNEUMATIC ACCUMULATOR SYSTEM PRESSURE DECAY AND FAIL SAFE TEST," UNDER TEMPORARY PROCEDURE #7371. DAMPER (DMP) 2-5742A WAS BEING TESTED. FOR THE PRESSURE DECAY PORTION OF THE TEST, THE TEST DIRECTOR AND INSTRUMENT MECHANIC (IM) AGREED TO PLACE THE JUMPER ACROSS TERMINALS 160 AND 161 IN PANEL 2252-24X WHICH WOULD BE EQUIVALENT TO PLACING A JUMPER ACROSS THE CONTACT AT THE PRESSURE SWITCH. A DIRECT GROUND RESULTED, WHICH BLEW FUSE (FU) F-3, CAUSING THE REACTOR BUILDING VENT (VA) ISOLATION DAMPERS TO GO CLOSED. THE APPARENT CAUSE OF THE EVENT IS PERSONNEL ERROR. TECH STAFF AND IM PERSONNEL INCORRECTLY READ THE WIRING DIAGRAM. THE PROPER TERMINAL POINTS SHOULD HAVE BEEN 150 AND 151. ALL DAMPERS WERE TESTED SUCCESSFULLY ON JANUARY 18, 1992 AT 1040 HOURS. FURTHER CORRECTIVE ACTIONS WILL INCLUDE REVISING QTS 105-9 AND OTHER LOCAL LEAK RATE TEST PROCEDURES AS APPLICABLE.

[253] QUAD CITIES 2 DOCKET 50-265 LER 91-014
2A RHR HEAT EXCHANGER SUPPORT BEAMS FOUND TO BE OUTSIDE BASIS DURING LIFTING DUE TO NOTCHES IN THE FLANGE AREA.
EVENT DATE: 121891 REPORT DATE: 011692 NSSS: GE TYPE: BWR

(NSIC 223846) ON 12/18/91, AT 0810 HOURS UNIT TWO WAS IN THE RUN MODE AT 66% OF RATED CORE THERMAL POWER. AT THIS TIME THE STATION WAS NOTIFIED BY ENGINEERING

OF AN ANALYSIS WHICH DETERMINED THAT TWO BEAMS SUPPORTING THE 2A RESIDUAL HEAT REMOVAL (HR) (BO) HEAT EXCHANGER (HX) EXCEEDED FINAL SAFETY ANALYSIS REPORT (FSAR) ALLOWABLE LIMITS. THIS OVER STRESSED CONDITION WAS DUE TO NOTCHES WHICH HAD BEEN CUT IN THE BEAMFLANGES. ENGINEERING EXAMINED THE DEGRADED SUPPORT BEAMS, AND DETERMINED THAT NO EXCESSIVE DEFORMATION OR DEFLECTION EXISTED. IT WAS DETERMINED THAT THE DEGRADATION DID NOT DIMINISH THE OPERABILITY OF THE HEAT EXCHANGER OR THE RHR SYSTEM. WORK REQUEST Q96769 WAS INITIATED TO REINFORCE THE DEGRADED SUPPORT BEAMS. THIS WORK WILL INSURE THAT THE SUPPORTS ARE WITHIN THE FSAR ALLOWABLE DESIGN LIMITS. THE REINFORCEMENT OF THE BEAMS WILL BE PERFORMED DURING THE UNIT TWO REFUEL OUTAGE, JANUARY 1992 (Q2R11).

[254] QUAD CITIES 2 DOCKET 50-265 LER 91-016
DESIGN DISCREPANCY BETWEEN FSAR AND AS-BUILT FOR TIP BALL VALVES.
EVENT DATE: 122791 REPORT DATE: 012392 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 1 (BWR)

(NSIC 223822) ON DECEMBER 27, 1991, AT 1600 HOURS, UNIT TWO WAS IN THE RUN MODE AT 63% OF RATED CORE THERMAL POWER. AT THIS TIME, TECHNICAL STAFF ENGINEERS PERFORMED A WALKDOWN OF THE TRAVERSING INCORE PROBE (TIP) SYSTEM AND DISCOVERED THAT THE 902-3, A- 16 AND 901-3 F-8 ANNUNCIATORS WOULD BE ACTIVATED BY LOSS OF POWER OR SHEAR VALVE ACTUATION AND NOT BY A BALL VALVE (ISV) OPEN WITH GROUP II ISOLATION SIGNAL PRESENT AS STATED IN THE UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) SECTION 7.4.5 .4. FURTHER REVIEWS DID NOT REVEAL ANY ADVERSE EFFECT ON THE OPERATION OF THE TIP BALL VALVES WITH A GROUP II ISOLATION SIGNAL PRESENT. THE DISCREPANCY RESULTED FROM FAILURE TO RECONCILE THE UFSAR WITH THE AS-BUILT CONDITION. THE UFSAR WILL BE CHANGED TO REFLECT THE AS-BUILT CONDITION. THIS REPORT IS PROVIDED TO SATISFY THE REQUIREMENTS OF 10CFR50.73(A)(2)(II)(B).

[255] QUAD CITIES 2 DOCKET 50-265 LER 92-001
GROUP I ISOLATION/REACTOR SCRAM FROM ACTUATION OF MSL LOW PRESSURE SWITCHES DUE TO PRESSURE FLUCTUATIONS IN SENSING LINES.
EVENT DATE: 010192 REPORT DATE: 012892 NSSS: GE TYPE: BWR

(NSIC 223862) ON JANUARY 1, 1992, QUAD CITIES UNIT TWO WAS IN THE RUN MODE AT 24 PERCENT OF RATED CORE THERMAL POWER. SPECIAL TEST PROCEDURE 2-104, TURBINE TRIP MAIN STEAM LINE LOW PRESSURE TEST, WAS IN PROGRESS IN AN ATTEMPT TO DISCOVER THE CAUSE FOR HISTORICAL GROUP I PRIMARY CONTAINMENT ISOLATIONS (PCI) FOLLOWING TURBINE TRIPS. AT 0014 HOURS, THE UNIT TWO MAIN TURBINE WAS MANUALLY TRIPPED FROM 200 MWE. WITHIN 0.5 SECONDS OF THE TURBINE TRIP, A GROUP I PCI OCCURRED. A REACTOR SCRAM THEN OCCURRED UPON TEN PERCENT CLOSURE OF THE MAIN STEAM ISOLATION VALVES (MSIV). ALL SAFETY FEATURE ACTUATIONS OCCURRED AS DESIGNED. AN EMERGENCY NOTIFICATION SYSTEM (ENS) NOTIFICATION WAS COMPLETED AT 0223 HOURS ON JANUARY 1, 1992. TEST DATA OBTAINED REVEALED THE CAUSE OF THE GROUP I PCI TO BE MAIN STEAM LINE (MSL) PRESSURE WAVE AND INSTRUMENT LINE RESONANCE DUE TO TURBINE STOP VALVE FAST CLOSURE. THIS CAUSED LARGE PRESSURE FLUCTUATIONS AT THE LOCATION OF THE MSL LOW PRESSURE SWITCHES. THIS FLUCTUATION WAS LARGE ENOUGH TO ACTUATE ALL FOUR SWITCHES AND CAUSE THE GROUP I PCI. GENERAL ELECTRIC HAS BEEN CONTRACTED TO EVALUATE THE TEST RESULTS AND PROVIDE RECOMMENDED CORRECTIVE ACTIONS.

[256] QUAD CITIES 2 DOCKET 50-265 LER 92-002
EXCEEDANCE OF TECHNICAL SPECIFICATION LOCAL LEAK RATE TEST LIMIT 0.6 LA WHILE TESTING THE CONTAINMENT PURGE FOUR VALVE VOLUME 2-1601-21, 22, 55 AND 56.
EVENT DATE: 010392 REPORT DATE: 013092 NSSS: GE TYPE: BWR

(NSIC 223931) ON JANUARY 1, 1992, QUAD CITIES UNIT TWO WAS SHUTDOWN FOR REFUELING AND MAINTENANCE (Q2R11). ON JANUARY 3, 1992 AT 0715 HOURS WHILE PERFORMING LOCAL LEAK RATE TESTING (LLRT) OF THE CONTAINMENT PURGE FOUR VALVE VOLUME 2-1601-21, 22, 55 AND 56, IT WAS DETERMINED THAT THE TECHNICAL SPECIFICATION 3.7.A.2.A.2 LIMIT OF 293.75 SCFH (0.6 LA) WAS EXCEEDED. AN EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE CALL WAS COMPLETED ON JANUARY 3, 1992 AT 1029 (EST) HOURS IN ACCORDANCE WITH 10CFR50.72(B)(2)(I). THE CAUSE OF THE EXCESSIVE LEAKAGES WILL NOT BE KNOWN UNTIL REPAIRS HAVE BEEN COMPLETED. A SUPPLEMENTAL REPORT WILL BE ISSUED TO ADDRESS THE CAUSES AND CORRECTIVE ACTIONS TAKEN TO BRING THE COMBINED

LEAKAGE TO WITHIN TECHNICAL SPECIFICATION LIMITS. THIS REPORT IS BEING SUBMITTED TO COMPLY WITH 10CFR50.73(A)(2)(II).

[257] QUAD CITIES 2 DOCKET 50-265 LER 92-003
REACTOR BLDG. VENTILATION ISOLATION AND SBGT START DUE TO PULLED FUSE DURING OUT OF SERVICE.
EVENT DATE: 010692 REPORT DATE: 012892 NSSS: GE TYPE: BWR

(NSIC 223863) ON JANUARY 6, 1992 AT 1154 HOURS, UNIT TWO WAS IN THE SHUTDOWN MODE FOR A REFUELING OUTAGE. THE OPERATING DEPARTMENT WAS TAKING THE UNIT 2 TORUS TO OXYGEN ANALYZER SOLENOID (SOL) (2-8801-D) OUT OF SERVICE (OOS). DURING OOS #3938 FUSE (FU) F-6 WAS PULLED FROM THE 902-40 PANEL (PL). THIS FUSE PULL CAUSED THE 595-133 RELAY (RLY) TO DEENERGIZE WHICH GAVE AN AUTOSTART SIGNAL TO STANDBY GAS TREATMENT (SBGT) (VI), ISOLATED THE REACTOR BUILDING VENTS (VA), AND A PARTIAL GROUP II ISOLATION WAS RECEIVED. AT 1221 HOURS THE SBGT TRAIN WAS SECURED AND THE REACTOR BUILDING VENTS WERE RESET AND RESTARTED. AT 1314 HOURS, OOS #3938 WAS SUCCESSFULLY COMPLETED WITHOUT PULLING THE FUSE. ON JANUARY 6, 1992, MEMO 92-1 WAS ISSUED TO OPERATING PERSONNEL WHICH PUT A STOP TO ALL FUSE PULLS IN THE 901(2)-40 AND 901(2)-41 PANELS UNTIL AN OPERATOR AIDE ADDRESSING INADVERTENT ACTUATION IS MADE UP. THIS EVENT IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV).

[258] RIVERBEND 1 DOCKET 50-458 LER 90-003 REV 04
UPDATE ON INADEQUATE THERMO-LAG FIRE BARRIER ENVELOPES SURROUNDING SAFE SHUTDOWN CIRCUITS PER TECHNICAL SPECIFICATIONS 3/4.7.7.
EVENT DATE: 020690 REPORT DATE: 122791 NSSS: GE TYPE: BWR

(NSIC 223779) DURING THE PERFORMANCE OF SURVEILLANCE TEST PROCEDURE STP-00-3602 ON 02/06/90 THROUGH 02/08/90 WITH THE UNIT IN OPERATIONAL CONDITION 1 (FULL POWER), IT WAS FOUND THAT SEVERAL MINOR DEFICIENCES EXISTED IN THE THERMO-LAG FIRE BARRIER ENVELOPES AROUND REDUNDANT SAFE SHUTDOWN CIRCUITS. THESE DEFICIENCIES CONSISTED OF SMALL HOLES, CRACKS AND UNFILLED SEAMS IN THE THERMO-LAG MATERIAL. A FIRE WATCH HAD ALREADY BEEN ESTABLISHED IN AREAS UTILIZING THERMO-LAG AS A FIRE BARRIER. GSU IS CURRENTLY WORKING WITH THE VENDOR TO RESOLVE THE IDENTIFIED DISCREPANCIES WHICH OCCURRED DURING CONSTRUCTION AND THE DEFICIENT THERMO-LAG BARRIERS. FIRE TESTS WERE CONDUCTED DURING NOVEMBER AND DECEMBER 1990 ON TYPICAL CONFIGURATIONS OF INSTALLED BARRIERS USED IN THE PLANT. REPAIR M ETHODS WERE DEVELOPED WHERE APPLICABLE. THIS SUPPLEMENT TO LER 90-003 IS SUBMITTED TO PROVIDE A STATUS OF THE THERMO-LAG ISSUE AT RIVER BEND STATION. THE COMBINATION OF THE CABLE JACKET PROPERTIES, THE CONTROL OF TRANSIENT COMBUSTIBLES, THE USE OF SUPPRESSION SYSTEMS IN THE PLANT AND THE USE OF COMPENSATORY FIRE WATCHES PROVIDES ASSURANCE THAT PLANT SAFETY AND THE HEALTH AND SAFETY OF THE PUBLIC HAS NOT BEEN JEOPARDIZED.

[259] RIVERBEND 1 DOCKET 50-458 LER 91-017 REV 01
UPDATE ON DESIGN DISCREPANCY IN THE WIRING DIAGRAM FOR TWO HYDROGEN MIXING SYSTEM VALVES.
EVENT DATE: 091891 REPORT DATE: 103091 NSSS: GE TYPE: BWR
VENDOR: LIMITORQUE CORP.

(NSIC 223780) AT 1500 HOURS ON 9/18/91, WITH REACTOR IN OPERATIONAL CONDITION 1 (POWER OPERATION), WHILE PERFORMING A REVIEW OF STATION OPERATING PROCEDURE (SOP)-0040 "HYDROGEN MIXING, PURGE, RECOMBINERS AND IGNITORS," A DISCREPANCY IN THE WIRING DIAGRAM OF THE HYDROGEN MIXING SYSTEM (*BB*) WAS DISCOVERED. THE WIRING DIAGRAM OF THE CIRCUIT SHOWED THAT THE OUTLET VALVES (*20*) 1CMP*MOV1A(B) AND 3A(B) COULD NOT BE MANUALLY BYPASSED FOLLOWING AN ISOLATION IN RESPONSE TO A LOCA SIGNAL. ALTHOUGH THE DISCREPANCY WAS DISCOVERED ON 9/18/91, IT IS BELIEVED THAT THE CONDITION EXISTED SINCE THE LAST MODIFICATION OF THE CIRCUIT ON 7/12/85. THUS, THE MANUAL BYPASS HAS BEEN INOPERABLE SINCE 07/12/85; THEREFORE, THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(I)(B) AS OPERATION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. BASED ON A LICENSE-BASIS ANALYSIS CONSIDERING REGULATORY R EQUIREMENTS, A MECHANISTIC ANALYSIS OF HYDROGEN GENERATION, A HUMAN RELIABILITY ANALYSIS, AND A PROBABILISTIC RISK ASSESSMENT OF THE HHYDROGEN

GENERATION SCENARIO, GSU HAS CONCLUDED THE SAFETY SIGNIFICANCE OF THIS EVENT IS LOW.

[260] RIVERBEND 1 DOCKET 50-458 LER 91-021
REACTOR PROTECTION SYSTEM ACTUATION DUE TO OPERATOR FAILURE TO VERIFY THE SCRAM DISCHARGE VOLUME LEVEL.
EVENT DATE: 120891 REPORT DATE: 011092 NSSS: GE TYPE: BWR

(NSIC 223807) AT APPROXIMATELY 1524 ON 12/08/91, WITH THE REACTOR IN OPERATIONAL CONDITION 3, AN UNPLANNED REACTOR PROTECTION SYSTEM (RPS) ACTUATION OCCURRED WHEN THE AT-THE-CONTROLS (ATC) OPERATOR PLACED THE CONTROL ROD DRIVE (CRD) SCRAM DISCHARGE VOLUME HIGH WATER LEVEL BYPASS SWITCHES FROM "BYPASS" TO "NORMAL" WITHOUT ASSURING THAT THE SCRAM DISCHARGE VOLUME (SDV) WATER LEVEL WAS LESS THAN 49". ALL CONTROL RODS WERE INSERTED PREVIOUSLY AND NO ADDITIONAL ROD MOTION OCCURRED DUE TO THE RPS ACTUATION. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV), AS AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION. ABNORMAL OPERATING PROCEDURE (AOP) - 0001 ("REACTOR SCRAM") IS BEING REVISED TO INCLUDE A DEDICATED SECTION TO PROVIDE SPECIFIC INSTRUCTIONS FOR RESETTING A REACTOR SCRAM OR RPS (*JE*) ACTUATION. THIS SECTION WILL REQUIRE THE ATC OPERATOR TO PLACE THE SDV HIGH WATER LEVEL BYPASS SWITCHES TO "BYPASS" AS THE FIRST STEP AND VERIFY THE SDV HIGH LEVEL ALARM IS CLEAR PRIOR TO PLACING THE SWITCHES IN THE "NORMAL" POSITION.

[261] RIVERBEND 1 DOCKET 50-458 LER 91-022
INDETERMINATE EQUIPMENT QUALIFICATION STATUS OF RESISTANCE TEMPERATURE DETECTORS IN THE FUEL BUILDING FILTER TRAINS.
EVENT DATE: 122391 REPORT DATE: 012292 NSSS: GE TYPE: BWR
VENDOR: PYCO

(NSIC 223869) ON 12/23/91, WITH THE REACTOR IN OPERATIONAL CONDITION 4, IT WAS DISCOVERED THAT FROM 6/4/91 TO 6/8/91, FUEL WAS MOVED IN THE FUEL BUILDING WHILE RESISTANCE TEMPERATURE DETECTORS (RTDS) INSTALLED IN THE FUEL BUILDING HEATER TRAINS WERE NOT ENVIRONMENTALLY AND SEISMICALLY QUALIFIED. THUS, THE FUEL BUILDING FILTER TRAINS ARE CONSIDERED TO HAVE BEEN INOPERABLE. THEREFORE, THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(I)(B) AS OPERATION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. THE SUBJECT RTDS HAVE BEEN REPLACED WITH QUALIFIED MODELS. TRAINING WILL BE PROVIDED ON THE IMPORTANCE OF COMMUNICATION WHEN PERFORMING OPERABILITY ANALYSES. GSU HAS PERFORMED AN AMBIENT HUMIDITY ANALYSIS AND A DECAY TIME ANALYSIS FOR THE FUEL IN THE SPENT FUEL POOL. THE RESULTS SHOW THAT IT WAS UNLIKELY THAT FUEL BUILDING CHARCOAL FILTER EFFICIENCY WAS DEGRADED AND THAT IF A FUEL HANDLING ACCIDENT HAD OCCURRED, OFFSITE DOSES WOULD HAVE BEEN BOUNDED BY THE DESIGN BASIS FUEL HANDLING ACCIDENT CALCULATION.

[262] RIVERBEND 1 DOCKET 50-458 LER 91-023
REACTOR WATER CLEANUP PUMPS OUTBOARD SUCTION VALVE ISOLATION DURING STANDBY LIQUID CONTROL VALVE OPERABILITY SURVEILLANCE DUE TO AN INADEQUATE PROCEDURE.
EVENT DATE: 122691 REPORT DATE: 012792 NSSS: GE TYPE: BWR

(NSIC 223870) AT 1212 ON 12/26/91 WITH THE UNIT IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN), 1G33*MOVF004 (REACTOR WATER CLEANUP PUMPS OUTBOARD SUCTION VALVE) RECEIVED AN ISOLATION SIGNAL FROM A STANDBY LIQUID CONTROL SYSTEM MANUAL INITIATION SIGNAL. THE VALVE WENT FULLY CLOSED AND THE REACTOR WATER CLEANUP (RWCU) PUMPS TRIPPED ON LOW FLOW. THE ISOLATION OF THE VALVE CONSTITUTED AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION. THEREFORE, THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(IV). THE ROOT CAUSE OF THIS EVENT WAS PROCEDURE INADEQUACY. CORRECTIVE ACTIONS INCLUDED PROCEDURE CLARIFICATION AND A REVIEW OF ALL RELATED PROCEDURES. THE STANDBY LIQUID CONTROL (SLC) SYSTEM AND THE REACTOR WATER CLEANUP (RWCU) SYSTEM RESPONDED PER DESIGN AND 1G33*MOVF004 (REACTOR WATER CLEANUP PUMPS OUTBOARD SUCTION VALVE) ISOLATED PROPERLY.

[263] ROBINSON 2 DOCKET 50-261 LER 91-011
REACTOR TRIP DUE TO FAILURE OF CONDENSATE PUMP SHAFT
EVENT DATE: 083091 REPORT DATE: 091891 NSSS: WE TYPE: PWR
VENDOR: BYRON JACKSON PUMPS, INC.

(NSIC 223783) ON AUGUST 30, 1991, AT 0700 HOURS WITH UNIT NO. 2 OPERATING AT ONE HUNDRED PERCENT POWER, AN AUTOMATIC REACTOR TRIP WAS RECEIVED FROM A "C" STEAM GENERATOR LOW LEVEL COINCIDENT WITH A STEAM FLOW/FEED FLOW MISMATCH. THE CAUSE OF THIS TRIP IS ATTRIBUTED TO FAILURE OF THE "B" CONDENSATE PUMP SHAFT, WHICH RESULTED IN INADEQUATE FEEDWATER FLOW TO THE STEAM GENERATORS. THE PLANT WAS BROUGHT TO SHUTDOWN CONDITIONS USING EMERGENCY OPERATING PROCEDURES, AND WAS STABILIZED AT 0711 HOURS. THE NRC WAS NOTIFIED OF THIS EVENT VIA THE ENS AT 0830 HOURS PURSUANT TO 10 CFR 50.72(B)(2)(II). REPAIRS TO THE "B" CONDENSATE PUMP WILL REQUIRE USE OF A PUMP THAT HAS BEEN PREVIOUSLY IN SERVICE, AND IS AWAITING REFURBISHMENT. UNTIL THIS PUMP CAN BE PLACED BACK IN SERVICE, THE PLANT WILL OPERATE AT APPROXIMATELY SIXTY FIVE PERCENT POWER, WHICH CAN BE ACHIEVED WITH ADEQUATE FEEDWATER BEING SUPPLIED WITH ONLY ONE CONDENSATE PUMP IN OPERATION. THIS LICENSEE EVENT REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(IV).

[264] ROBINSON 2 DOCKET 50-261 LER 92-001
DEGRADED CONDITION DUE TO INOPERABILITY OF CONTAINMENT ISOLATION VALVE.
EVENT DATE: 011692 REPORT DATE: 012992 NSSS: WE TYPE: PWR
VENDOR: COPES-VULCAN, INC.

(NSIC 223861) ON 1/13/92, UNIT 2 WAS OPERATING AT 100% POWER WHEN VALVE PS-956F FAILED ITS STROKE TIME TEST. THE TECH SPEC ACTION STATEMENT FOR CONTAINMENT INTEGRITY WAS ENTERED BY CLOSING THE REDUNDANT ISOLATION VALVE, PS-956E, AND REMOVING ITS POWER SOURCE. ON 1/14/92, THE LINE WAS BEING CLEARED FOR REPAIR OF PS-956F WHEN A LEAK RATE OF 890CC PER MINUTE WAS MEASURED FROM PS-956E. SINCE THIS LEAK RATE 46 PSIG, OPERABILITY OF PS-956E COULD NOT BE DETERMINED IN ACCORDANCE WITH PLANT PROCEDURES AN 15, AFTER APPROXIMATELY MINUTE, WHICH IS LOW WAS AT 2235 PSIG AND NORMAL TEST ACCEPTANCE VALUES ARE 46 PSIG, OPERABILITY OF PS-956E COULD NOT BE DETERMINED. IN ACCORDANCE WITH OPERABILITY DETERMINATION WAS REQUESTED. ON JANUARY EIGHTEEN HOURS, THE LEAKAGE WAS REDUCED TO 13 CC PER ENOUGH TO MEET OPERABILITY REQUIREMENTS. ON 1/16, THE OPERABILITY DETERMINATION RESULTS CONCLUDED THAT DURING THIS 18-HR PERIOD, THE OBSERVED LEAKAGE WAS SUCH THAT THE PENETRATION COULD NOT BE CONSIDERED OPERABLE BY PLANT TECH SPECS. THE CAUSE OF THIS CONDITION IS ATTRIBUTED TO COMPONENT FAILURE. THE NRC WAS NOTIFIED OF THIS CONDITION VIA THE ENS AT 1540 HOURS ON 1/16/92. THE VALVE IS SCHEDULED TO BE REPAIRED DURING THE UPCOMING REFUELING OUTAGE. UNTIL THAT TIME, THE CONTAINMENT PENETRATION IS OPERABLE BECAUSE THE LEAK RATE ON PS-956E IS WITHIN LEAK RATE LIMITS, AND THE POWER FOR THIS VALVE HAVE BEEN REMOVED.

[265] SALEM 1 DOCKET 50-272 LER 91-027
ENGINEERED SAFETY FEATURE ACTUATION; SAFEGUARD EQUIPMENT CONTROL CABINET "MODE OP" SIGNAL DUE TO EQUIPMENT FAILURE.
EVENT DATE: 081591 REPORT DATE: 091291 NSSS: WE TYPE: PWR
VENDOR: VITRO ENGINEERING DIVISION

(NSIC 223812) ON 8/15/91, TWO PARTIAL "MODE OP I" (LOSS OF COOLANT ACCIDENT) 1A SAFEGUARD EQUIPMENT CONTROL (SEC) CABINET SIGNAL ACTUATIONS OCCURRED. THE FIRST OCCURRED AT 1908 HOURS AND THE SECOND OCCURRED AT 1914 HOURS. THE 1A SEC WAS DECLARED INOPERABLE AND TECHNICAL SPECIFICATION 3.3.2.1 ACTION 13 WAS ENTERED UPON THE FIRST "MODE OP I" SIGNAL ACTUATION. IN ACCORDANCE WITH THE TECHNICAL SPECIFICATION 3.3.2.1 ACTION 13 WAS EXITED. UNIT LOAD HAD BEEN REDUCED TO 43%. THE CAUSE OF THESE PARTIAL "MODE OP I" EVENTS IS EQUIPMENT FAILURE. TROUBLESHOOTING OF THE 1A SEC CHASSIS CIRCUITS IDENTIFIED TWO FAILED CIRCUIT CARDS. THESE CARDS ARE BEING SENT BACK TO THE MANUFACTURER TO HELP DETERMINE CAUSE OF FAILURE. SYSTEM ENGINEERING HAS INITIATED A REVIEW OF SEC FAILURES (INCLUDING LERS 272/90-008-00 AND 272/89-026-00) TO DETERMINE IF THERE ARE ANY FACTORS COMMON TO THE FAILURES. UPON COMPLETION OF THIS INVESTIGATION, APPROPRIATE CORRECTIVE ACTION WILL BE IMPLEMENTED.

[266] SALEM 1 DOCKET 50-272 LER 91-028
VALVE 12GB4 FAILED CLOSED DUE TO EQUIPMENT FAILURE.
EVENT DATE: 082391 REPORT DATE: 091991 NSSS: WE TYPE: PWR
VENDOR: MASONEILAN INTERNATIONAL, INC.

(NSIC 223811) ON 8/23/91, AT 1046 HOURS, THE NO. 12 STEAM GENERATOR BLOWDOWN OUTLET ISOLATION VALVE (12GB4) FAILED CLOSED. THE FOUR BLOWDOWN ISOLATION VALVES (11-14GB4) HAD BEEN OPENED, AT 1004 HOURS, AFTER COMPLETION OF 1R19A RADIATION MONITORING SYSTEM (RMS) BLOWDOWN MONITOR CHANNEL CALIBRATION. THE 11-14GB4 VALVES ARE CONTAINMENT ISOLATION VALVES. CONTAINMENT ISOLATION IS AN ENGINEERED SAFETY FEATURE. THE CAUSE OF THE VALVE CLOSURE IS EQUIPMENT FAILURE. THE VALVE ACTUATOR'S DIAPHRAGM HAD FAILED. SYSTEM ENGINEERING IS CONTINUING TO INVESTIGATE TO DETERMINE WHY THE 12GB4 VALVE DIAPHRAGM FAILED. INCLUDED IS A REVIEW OF THE PREVENTIVE MAINTENANCE REQUIREMENTS FOR THE GB4 VALVES. VALVE REPAIR WAS COMPLETED ON 8/24/91 AT WHICH TIME THE VALVE WAS RETURNED TO SERVICE.

[267] SALEM 1 DOCKET 50-272 LER 91-032
ESF ACTIVATION SIGNAL FOR CONTAINMENT VENTILATION ISOLATION DUE TO INADEQUATE DESIGN.
EVENT DATE: 111491 REPORT DATE: 121291 NSSS: WE TYPE: PWR

(NSIC 223633) ON 11/11/91, A MAINTENANCE INSTRUMENT AND CONTROLS (I&C) TECHNICIAN BEGAN A PREVENTIVE MAINTENANCE (PM) TASK TO OPEN, CLEAN, INSPECT AND CLOSE THE SAMPLE BOX FOR THE RADIATION MONITORING SYSTEM (RMS) CONTAINMENT PARTICULATE RADIATION MONITOR, 1R11A. AT 0842 HOURS THE SAME DAY, DURING THE PERFORMANCE OF THE PM TASK, A CONTAINMENT PURGE/PRESSURE-VACUUM RELIEF SYSTEM (CP/P-VR) ISOLATION SIGNAL ACTUATED AS A RESULT OF A HIGH CHANNEL SPIKE ON 1R11A. THE ASSOCIATED CONTAINMENT ISOLATION VALVES WERE CLOSED AT THE TIME AND DID NOT CHANGE POSITION AS A RESULT OF THE ISOLATION SIGNAL. THE CP/P-VR SYSTEM ISOLATION IS AN ENGINEERED SAFETY FEATURE (ESF). THE ROOT CAUSE OF THE CHANNEL SPIKE IS ATTRIBUTED TO A DESIGN INADEQUACY. AS INDICATED BY PRIOR LERS, THE RMS CHANNELS ARE SUBJECT TO INTERMITTENT SPIKES DUE TO EXTERNAL INFLUENCES. INVESTIGATION COULD NOT IDENTIFY THE SPECIFIC CAUSE OF THIS CHANNEL SPIKE. THE 1R11A CHANNEL WAS RESTORED FOLLOWING THE INVESTIGATION. A CHANNEL CALIBRATION CHECK WAS COMPLETED SATISFACTORILY. SUBSEQUENTLY, THE ACTION STATEMENT WAS EXITED NOVEMBER 11, 1991, AT 1105 HOURS. THE EVENT WAS DISCUSSED WITH THE APPROPRIATE MAINTENANCE I&C RMS QUALIFIED TECHNICIANS. THE MAINTENANCE DEPARTMENT WORK STANDARD "PAPERCHANGE" HAS BEEN REVISED TO PROVIDE INSTRUCTIONS FOR BLOCKING THE CHANNEL TRIP FUNCTION.

[268] SALEM 1 DOCKET 50-272 LER 91-033
1B DIESEL GENERATOR START DUE TO PERSONNEL ERROR.
EVENT DATE: 111691 REPORT DATE: 121691 NSSS: WE TYPE: PWR

(NSIC 223601) ON NOVEMBER 16, 1991, AT 0105 HOURS, DURING THE INSTALLATION OF A TEST RECORDER TO MONITOR 1B SAFEGUARDS EQUIPMENT CONTROL (SEC) CIRCUITRY, A MAINTENANCE DEPARTMENT ELECTRICAL SUPERVISOR IMPROPERLY RECONNECTED TEST LEADS, AT THE RECORDER, CAUSING AN UNPLANNED START OF 1B DIESEL GENERATOR (D/G). THE TEST RECORDER WAS BEING INSTALLED IN SUPPORT OF 1B SEC MONITORING. THE 1B SEC HAD ITS CHASSIS REPLACED AS A RESULT OF AN AUTO TEST FAULT ALARM INDICATION. THE START OF A D/G IS AN ENGINEERED SAFETY FEATURE (ESF). THE ROOT CAUSE OF THIS EVENT IS PERSONNEL ERROR (IMPROPER USE OF TEST EQUIPMENT). A CONTRIBUTING FACTOR TO THIS EVENT WAS THE INCORRECT LABELING OF THE VOLTAGE INDICATIONS ON THE TEST RECORDER. THE CONNECTIONS AT THE RECORDER WERE REVERSED TO THEIR CORRECT CONFIGURATION AND THE FAULTY INDICATOR LABELS WERE CORRECTED. THE MAINTENANCE SUPERVISOR AND THE APPLICABLE MAINTENANCE DEPARTMENT CONTROLS TECHNICIANS WERE COUNSELED ABOUT THE IMPORTANCE OF ATTENTION TO DETAIL, IN THE USE OF TEST EQUIPMENT, RELATIVE TO THIS EVENT. THE EVENT WAS DISCUSSED WITH THE APPLICABLE MAINTENANCE DEPARTMENT PERSONNEL.

[269] SALEM 1 DOCKET 50-272 LER 91-034
TWO ENGINEERED SAFETY FEATURES SIGNAL ACTUATIONS DUE TO EQUIPMENT FAILURE.
EVENT DATE: 112691 REPORT DATE: 122391 NSSS: WE TYPE: PWR
VENDOR: LFE CORP.

(NSIC 223705) THIS LER ADDRESSES 2 ENGINEERED SAFETY FEATURE (ESF) SIGNALS INITIATED THROUGH THE RADIATION MONITORING SYSTEM (RMS). THESE SIGNALS WERE CONTAINMENT PURGE/PRESSURE-VACUUM RELIEF (CP/P-VR) SYSTEM ISOLATION ON 11/26/91, THE 1R11A CONTAINMENT PARTICULATE RMS MONITOR CHANNEL SPIKED HIGH RESULTING IN A CP/P-VR SYSTEM ISOLATION SIGNAL. THE ISOLATION VALVES WERE CLOSED AT THE TIME OF BOTH EVENTS AND DID NOT CHANGE POSITION AS A RESULT OF THE ISOLATION SIGNAL. THE 1R11A RMS CHANNEL INDICATION RETURNED TO NORMAL APPROXIMATELY 4 MINUTES AFTER SPIKE. THE 1R11A RMS CHANNEL WAS DECLARED INOPERABLE AND AN INVESTIGATION TO DETERMINE THE CAUSE OF THE EVENT WAS INITIATED. ON 11/26/91, A CHANNEL CALIBRATION CHECK WAS SATISFACTORILY PERFORMED FOR THE 1R11A CHANNEL AND THE CHANNEL WAS RETURNED TO SERVICE. ON 12/26/91, THE 1R11A RMS CHANNEL AGAIN SPIKED HIGH, MOMENTARILY, RESULTING IN A CP/P-VR SYSTEM ISOLATION SIGNAL. THE 1R11A RMS CHANNEL WAS DECLARED INOPERABLE AND AN INVESTIGATION TO DETERMINE THE CAUSE THE EVENT WAS INITIATED. THE ROOT CAUSE OF BOTH 1R11A RMS CHANNEL CP/P-VR SYSTEM ISOLATION SIGNAL ACTUATIONS HAS BEEN ATTRIBUTED TO EQUIPMENT FAILURE (DEGRADED CABLE CONNECTION). THE 1R11A DETECTOR CONNECTION WILL BE REPAIRED AND A SATISFACTORY CHANNEL CALIBRATION CHECK WILL BE PERFORMED. THE FEASIBILITY OF REPLACING THE EXISTING CONNECTOR WITH A MORE RELIABLE CONNECTOR WILL BE EXPLORED.

[270] SALEM 1 DOCKET 50-272 LER 91-035
RADIATION MONITORING SYSTEM CHANNEL INITIATED A CONTAINMENT VENTILATION ISOLATION SIGNAL.
EVENT DATE: 112791 REPORT DATE: 123191 NSSS: WE TYPE: PWR

(NSIC 223764) ON NOVEMBER 27, 1991, AT 0042 HOURS, THE RADIATION MONITORING SYSTEM (RMS)(IL) 1R12A CONTAINMENT NOBLE GAS RADIATION MONITOR CHANNEL ALARMED RESULTING IN A CONTAINMENT PURGE/PRESSURE-VACUUM RELIEF (CP/P-VR) SYSTEM (BF) ISOLATION SIGNAL. INVESTIGATION OF THE EVENT WAS INITIATED WHICH INCLUDED TAKING A PORTABLE CONTAINMENT AIR SAMPLE. THE SAMPLE DID NOT INDICATE ANY INCREASED ACTIVITY. THE 1R12A RMS CHANNEL WAS DECLARED INOPERABLE AND TECHNICAL SPECIFICATION 3.3.3.1 TABLE 3.3-6 ACTION 20 WAS ENTERED ON NOVEMBER 27, 1991, AT 0334 HOURS. INVESTIGATION OF THE ROOT CAUSE OF THIS EVENT IS CONTINUING. PRELIMINARY RESULTS INDICATE THAT THE 1R12A MONITOR MAY HAVE RESPONDED TO ACTUAL RADIOACTIVITY, THE SOURCE OF WHICH IS NOT PRESENTLY KNOWN. THE 1R12A CHANNEL DETECTOR AND DETECTOR HOUSING WERE CLEANED AND A CHANNEL CALIBRATION CHECK WAS PERFORMED SATISFACTORILY. THE CHANNEL WAS RETURNED TO SERVICE ON NOVEMBER 29, 1991, AT 1125 HOURS, AT WHICH TIME TECHNICAL SPECIFICATION TABLE 3.3-6 ACTION 20 WAS EXITED. INVESTIGATION TO DETERMINE THE SOURCE OF THE RADON DAUGHTER PRODUCTS IS CONTINUING. INVESTIGATION OF THE 1R12A CP/PV-R SYSTEM ISOLATION SIGNAL IS CONTINUING. UPON COMPLETION A SUPPLEMENTAL LER WILL BE ISSUED.

[271] SALEM 1 DOCKET 50-272 LER 91-038
CONTROL HABITABILITY CONCERN FROM POSTULATED AMMONIUM HYDROXIDE RELEASE.
EVENT DATE: 121991 REPORT DATE: 011692 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 223824) ON 12/19/91, AN INTERIM CALCULATION CONCLUDED THAT A CONTROL ROOM HABITABILITY CONCERN WAS POSSIBLE, DUE TO POSTULATED FAILURE OF A 3000 GALLON CAPACITY TURBINE BUILDING AMMONIUM HYDROXIDE STORAGE TANK. IN ACCORDANCE WITH REGULATORY GUIDE 1 .78, THE MAXIMUM CONCENTRATION RELEASE WAS POSTULATED FROM A CATASTROPHIC FAILURE OF THE LARGEST STORAGE TANK OF AMMONIUM HYDROXIDE (27.5 WT%). THE LIQUID AMMONIUM HYDROXIDE WOULD VAPORIZE TO AMMONIA UPON RELEASE, EXHAUST TO THE ENVIRONMENT (FROM THE TURBINE BUILDING VENTILATION SYSTEM), AND DISPERSE TO THE CONTROL ROOM AIR SUPPLY INTAKES. CALCULATIONS INDICATE THAT THE REGULATORY GUIDE 1.78 TOXIC LIMIT FOR AMMONIA (100 PPM) WOULD BE EXCEEDED. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO INADEQUATE DESIGN REVIEW DURING THE PREPARATION OF THE FINAL SAFETY ANALYSIS REPORT (FSAR). COMPENSATORY ACTIONS HAVE BEEN TAKEN TO ENSURE CONTROL ROOM HABITABILITY UNDER POSTULATED ACCIDENT CONDITIONS INCLUDING THE ADDITION OF A TEMPERATURE INDICATOR WITH CHART RECORDER

BY THE STORAGE VESSEL AND ADMINISTRATIVE CONTROLS FOR AMMONIUM HYDROXIDE TANKER RECEIPT. A SURVEY FOR ADDITIONAL HAZARDOUS CHEMICAL ON-SITE STORAGE CONCERNS WAS CONDUCTED. AN ENGINEERING EVALUATION, INCLUDING A 10CFR50.59 SAFETY EVALUATION, WILL BE COMPLETED TO ADDRESS THE AMMONIUM HYDROXIDE AND ANY OTHER APPLICABLE HAZARDOUS CHEMICAL ISSUES.

[272] SALEM 1 DOCKET 50-272 LER 91-009 REV 01
UPDATE ON HIGH ENERGY LINE BREAK CONCERN BETWEEN MECHANICAL AND ELECTRICAL PENETRATION AREA.
EVENT DATE: 122091 REPORT DATE: 013092 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 223932) ON 2/15/91, A PROBABILISTIC RISK ASSESSMENT (PRA) OF A UNIT 1 UNSEALED SEISMIC GAP PORTION, BETWEEN THE INBOARD MECHANICAL PENETRATION AREA AND THE ELECTRICAL PENETRATION AREA, WAS COMPLETED. RESULTS SHOW CHANGE IN THE CORE DAMAGE FREQUENCY TO BE SIGNIFICANTLY INCREASED (I.E., DUE TO CONCERN FOR A MAIN STEAMLINE BREAK IN THE INBOARD MECHANICAL PENETRATION AREA). A SIMILAR SEISMIC GAP WAS FOUND UNSEALED FOR UNIT 2 PRIOR TO DISCOVERY OF THE UNIT 1 SEAL CONCERN. THE ROOT CAUSE OF THE SEISMIC GAP SEAL(S) (BOTH UNITS) NOT BEING INSTALLED IS "DESIGN, MANUFACTURING, CONSTRUCTION/INSTALLATION" ERROR (AS PER NUREG 1022). ANOTHER UNIT 2 SEISMIC GAP SEAL WAS INSPECTED ON 2/19/91 AND FOUND IMPROPERLY SEALED; HOWEVER, ANCHORED FLASHING WAS IN PLACE WHICH WOULD ACT AS A STEAM FLOW BARRIER. THIS UNIT 2 SEISMIC GAP WAS SEALED ON 2/22/91. THE MISSING PORTION OF THE UNIT 1 SEISMIC GAP SEAL WAS INSTALLED ON 4/12/91. AN ANALYSIS WAS COMPLETED ON 9/27/91 WHICH ASSESSED THE OPERABILITY OF THE ELECTRICAL PENETRATION AREA MCCS WITH THE SEISMIC GAP NOT SEALED. THE REMAINING SEISMIC GAP AREA (BOTH UNITS) HAS BEEN INSPECTED BY THE PSRG. TO DATE, ALL SAFETY SIGNIFICANT PENETRATIONS WITH INADEQUATE SEALS HAVE BEEN REPAIRED WITH THE EXCEPTION OF 6 FOR UNIT 2 WHICH WILL BE SEALED DURING THE CURRENT REFUELING OUTAGE.

[273] SALEM 1 DOCKET 50-272 LER 92-001
ESF COMPONENT ACTUATION 12GB4 FAILED CLOSED TWICE DUE TO EQUIPMENT FAILURE.
EVENT DATE: 010792 REPORT DATE: 020692 NSSS: WE TYPE: PWR
VENDOR: MASONEILAN INTERNATIONAL, INC.

(NSIC 223918) ON 1/7/92, THE NO. 12 STEAM GENERATOR (S/G) BLOWDOWN OUTLET ISOLATION VALVE (12G B4) (A CONTAINMENT ISOLATION VALVE) FAILED CLOSED. INVESTIGATION SHOWED THAT THE VALVE ACTUATOR'S RUBBER DIAPHRAGM HAD FAILED, RESULTING IN AIR LEAKAGE AND SUBSEQUENT VALVE CLOSURE. INSPECTION OF THE 1/7/92 FAILED DIAPHRAGM REVEALED THAT A SECTION OF THE DIAPHRAGM HAD DRIED AND TORE. THE SECTION THAT DRIED IS LOCATED NEAREST TO AN UNINSULATED S/G BLOWDOWN PIPE. ON 1/8/92, THE VALVE WAS RETURNED TO SERVICE. ON 1/28/92, THE 12GB4 AGAIN FAILED CLOSED. INVESTIGATION REVEALED THAT THE VALVE ACTUATOR'S DIAPHRAGM AGAIN HAD FAILED. INSPECTION REVEALED THAT IT TORE AT THE SAME APPROXIMATE SECTION AS THE DIAPHRAGM WHICH FAILED ON 1/7/92; HOWEVER, THE DIAPHRAGM HAD NOT DRIED. THE CAUSE OF THE 12GB4 VALVE CLOSURE EVENTS IS EQUIPMENT FAILURE. A CONTRIBUTING FACTOR TO THE CAUSE OF THE 1/7/92 DIAPHRAGM FAILURE WAS PREMATURE AGING OF THE DIAPHRAGM CAUSED BY THE PRIOR REMOVAL OF PIPING INSULATION ADJACENT TO THE VALVE ACTUATOR. THE 1/28/92 VALVE ACTUATOR DIAPHRAGM FAILURE WAS NOT CAUSED BY PREMATURE AGING. TESTING OF THIS FAILED DIAPHRAGM HAS BEEN INITIATED TO DETERMINE WHY IT FAILED. ASSESSMENT OF THE CONTRIBUTING FACTORS TO THE 12GB4 VALVE ACTUATOR DIAPHRAGM FAILURES IS CONTINUING. A REVIEW OF THE MAINTENANCE HISTORY OF THE 11-14GB4 VALVES HAS BEEN INITIATED.

[274] SALEM 2 DOCKET 50-311 LER 91-020
ENGINEERED SAFETY FEATURE ACTUATION DUE TO RADIATION MONITORING SYSTEM CHANNEL RESPONSE.
EVENT DATE: 112291 REPORT DATE: 011392 NSSS: WE TYPE: PWR

(NSIC 223716) ON 11/22/91, THE RADIATION MONITORING SYSTEM (RMS) CONTAINMENT NOBLE GAS MONITOR, 2R12A, ALARMED. THIS RESULTED IN THE INITIATION OF A CONTAINMENT PURGE/PRESSURE-VACUUM RELIEF (CP/P-VR) SYSTEM ISOLATION SIGNAL. PRIOR TO THE EVENT, THE 2R12A CHANNEL ALARM SETPOINT WAS REDUCED FROM 20,000 CPM TO 240

CPM IN PREPARATION FOR ENTRY INTO REFUELING. THE ROOT CAUSE OF THIS EVENT IS PERSONNEL ERROR AS A RESULT OF INATTENTION TO DETAIL. THE SOURCE OF THE 2R12A CHANNEL C RATE INCREASE WAS THE GASES RELEASED AS A RESULT OF WORK ON A VALVE LEAK-OFF LINE, (WHICH GOES TO THE REACTOR COOLANT DRAIN TANK (RCDT)). THE LINE HAD NOT BEEN TAGGED AND ISOLATED TO SUPPORT THE VALVE WORK SUBSEQUENTLY, WHEN MAINTENANCE WORKERS CUT INTO THE LEAK-OFF LINE THE NOBLE GASES FROM THE RCDT ESCAPED TO THE CONTAINMENT ATMOSPHERE EVENTUALLY CAUSING THE 2R12A ALARM. CORRECTIVE DISCIPLINARY ACTION BEEN TAKEN WITH OPERATIONS AND MAINTENANCE PERSONNEL INVOLVED IN THE EVENT. THIS EVENT HAS BEEN REVIEWED WITH APPLICABLE PERSONNEL STRESSING THE NEED FOR ATTENTION TO DETAIL. THE WCC REVIEWED ALL EXISTING TAGOUTS. NO SIMILAR SITUATIONS WERE IDENTIFIED. ALTHOUGH CONTAINMENT EQUIPMENT HATCH WAS OPEN DURING THIS EVENT, THE AUXILIAY BUILDING VENTILATION MAINTAINED CONTAINMENT NEGATIVE PRESSURE. AN EVALUATION IS BEING CONDUCTED TO DETERMINE IF A RELEASE TO THE ENVIRONMENT (VIA THE OPEN HATCH) OCCURRED.

[275] SALEM 2 DOCKET 50-311 LER 91-018
FOUR ENGINEERED SAFETY FEATURE SIGNALS: TWO AUTO SWITCH OF CONTROL ROOM VENTILATION AND TWO CONTAINMENT VENTILATION ISOLATION.
EVENT DATE: 112991 REPORT DATE: 123091 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 1 (PWR)
VENDOR: EBERLINE INSTRUMENT CORP.
VICTOREEN INSTRUMENT DIVISION

(NSIC 223772) THIS LER ADDRESSES FOUR ENGINEERED SAFETY FEATURE (ESF) SIGNALS INITIATED THROUGH THE RADIATION MONITORING SYSTEM (RMS). TWO WERE FOR THE AUTOMATIC SWITCHING OF THE CONTROL ROOM VENTILATION FROM NORMAL TO THE EMERGENCY MODE OF OPERATION. (11/29/91, 0835 HRS - 2R1A SPIKED HIGH) & (12/05/91, 0640 HRS - 2R1A FAILED LOW). TWO WERE FOR CONTAINMENT PURGE/PRESSURE VACUUM RELIEF SYSTEM ISOLATION. (11/30/91, 404 HRS - 2R45B SPIKED HIGH, CAUSING 2R41C TO DEENERGIZE) & (12/02/91, 100 HRS - 2R12A SPIKED HIGH). THE ROOT CAUSE OF THE 2R1A RMS CHANNEL SPIKE/FAILURE, 2R45B RMS AND 2R12A CHANNEL SPIKES IS ATTRIBUTED TO DESIGN/EQUIPMENT CONCERNS, SPECIFICALLY SUSCEPTIBILITY TO VOLTAGE TRANSIENTS. THE TYPE OF DETECTOR SYSTEM USED FOR THE MAJORITY OF THE UNIT 2 RMS CHANNELS IS MANUFACTURED BY VICTOREEN. PERIODIC PROBLEMS WITH THIS SYSTEM HAVE BEEN EXPERIENCED AS INDICATED IN PRIOR, LERS (E.G., 311/91-010-00 AND 311/91-019-00). SINCE THE 12/05/91, EVENT, THE CHANNELS HAVE BEEN OPERATING SATISFACTORILY. IN EACH CASE, THE EFFECTED CHANNELS WERE CHECKED AND RETURNED TO SERVICE. ENGINEERING HAS INVESTIGATED THE CONCERNS WITH THE UNIT 2 RMS CHANNELS. SEVERAL SYSTEM DESIGN MODIFICATIONS ARE BEING IMPLEMENTED TO ELIMINATE SPURIOUS ESF ACTUATION SIGNALS.

[276] SALEM 2 DOCKET 50-311 LER 91-019
CONTAINMENT VENTILATION ISOLATION DUE TO 2R12A RMS CHANNEL RESPONSE.
EVENT DATE: 120791 REPORT DATE: 010292 NSSS: WE TYPE: PWR

(NSIC 223715) ON 12/7/91, THE 2R12A (CONTAINMENT RADIOACTIVE NOBLE GAS MONITOR) RADIATION MONITORING SYSTEM (RMS) CHANNEL ALARMED. THIS RESULTED IN AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION SIGNAL FOR CONTAINMENT PURGE/PRESSURE-VACUUM RELIEF (CP/P-VR) SYSTEM (WF) ISOLATION. THE CHANNEL WAS NOT DECLARED INOPERABLE SINCE THE CAUSE OF THE ALARM WAS DUE TO AIRBORNE ACTIVITY ABOVE NORMAL BACKGROUND LEVELS. THIS EVENT OCCURRED AS A RESULT OF ROUTINE REFUELING ACTIVITIES. REACTOR COOLANT SYSTEM (RCS) COVER GAS (CONTAINING RADIOACTIVE NOBLE GASES) WAS PULLED INTO CONTAINMENT AFTER THE S/G PRIMARY MANWAY DIAPHRAGMS WERE REMOVED. PORTABLE VENTILATION, USING HEPA FILTRATION UNITS, IS ATTACHED TO THE PRIMARY COLD LEG MANWAY TO SUPPORT S/G WORK. IN MODE 6, THE 2R12A CHANNEL SETPOINT IS SET AT </= 2 TIMES BACKGROUND, AS REQUIRED BY TECHNICAL SPECIFICATION TABLE 3.3-6 CONDITIONAL SURVEILLANCE. DUE TO THE BACKGROUND LEVELS, PRIOR TO S/G PRIMARY MANWAY DIAPHRAGM REMOVAL, THE 2R12A CHANNEL ALARM SETPOINT WAS SET 240 CPM. THE 2R12A CHANNEL RECORDED ACTIVITY LEVELS UP TO 500 CPM AFTER THE DIAPHRAGM WAS REMOVED. THE PROCEDURE(S), WHICH ADDRESS REFUELING ACTIVITIES INVOLVING S/G WORK, WILL BE APPROPRIATELY REVISED TO ADDRESS PRIMARY MANWAY DIAPHRAGM REMOVAL AND ITS AFFECT ON THE 2R12A RMS CHANNEL.

[277] SALEM 2 DOCKET 50-311 LER 91-021
TECHNICAL SPECIFICATION SURVEILLANCE OVERDUE DUE TO PERSONNEL ERROR.
EVENT DATE: 121991 REPORT DATE: 012092 NSSS: WE TYPE: PWR

(NSIC 223828) ON 12/19/91, IT WAS DISCOVERED THAT THE WASTE GAS ANALYZER CHANNEL FUNCTIONAL, TECHNICAL SPECIFICATION SURVEILLANCE, HAD NOT BEEN PERFORMED. THIS MONTHLY SURVEILLANCE WAS OVERDUE AS OF 12/16/91. THE LAST TIME THE TEST WAS SUCCESSFULLY COMPLETED WAS ON 11/9/91. THE SURVEILLANCE WAS ORIGINALLY SCHEDULED FOR 12/9/91; HOWEVER, THE START DATE WAS SHIFTED (BY COMPUTER LOGIC) TO 12/19/91. THE PREMIS COMPUTER PROGRAM FOR OUTAGE SCHEDULING IS STRUCTURED TO CONTROL THE START DATE FOR ALL WORK DONE DURING THE OUTAGE, INCLUDING SURVEILLANCES. WHEN A JOB IS DELAYED, THE COMPUTER AUTOMATICALLY RESCHEDULES THE START DATE FOR JOBS DEPENDENT ON THE DELAYED JOB, INDEPENDENT OF TECHNICAL SPECIFICATION OR OTHER COMMITMENT REQUIREMENTS. THE ROOT CAUSE OF THIS EVENT IS PERSONNEL ERROR. NEITHER OUTAGE PLANNING DEPARTMENT PERSONNEL NOR THE TECHNICAL SPECIFICATION SURVEILLANCE ADMINISTRATOR OBSERVED THE SHIFT IN THE SURVEILLANCE START DATE UPON REVIEW OF THE "TECHNICAL SPECIFICATION OVERDUE SURVEILLANCE REPORT". THE SURVEILLANCE WAS SUCCESSFULLY COMPLETED ON 12/19/91. APPROPRIATE DISCIPLINARY ACTION HAS BEEN TAKEN WITH THOSE INDIVIDUALS INVOLVED IN THIS EVENT. THIS EVENT HAS BEEN REVIEWED WITH APPLICABLE OUTAGE PLANNING DEPARTMENT PERSONNEL. A TECHNICAL SPECIFICATION SURVEILLANCE REVIEW WILL BE CONDUCTED TO IDENTIFY THOSE SURVEILLANCES REQUIRED WHEN THE CORE IS DEFUELED.

[278] SALEM 2 DOCKET 50-311 LER 92-002
LACK OF SECONDARY OVERCURRENT PROTECTION DEVICE ON CONTAINMENT ELECTRICAL PENETRATION.
EVENT DATE: 010392 REPORT DATE: 013092 NSSS: WE TYPE: PWR

(NSIC 223936) ON 1/3/92, IT WAS DETERMINED THAT A UNIT 2 ELECTRICAL PENETRATION CIRCUIT WAS NOT PROTECTED BY A SECONDARY DEVICE AS REQUIRED BY REGULATORY GUIDE 1.63. WITHOUT A SECONDARY PROTECTIVE DEVICE, THIS CONDUCTOR COULD POTENTIALLY DAMAGE THE ELECTRICAL PENETRATION IN THE EVENT OF AN OVERCURRENT FAULT AND THE PRIMARY PROTECTIVE DEVICE (CIRCUIT BREAKER) WERE TO FAIL. THE UNIT 2 DESIGN BASE DOCUMENT, CALCULATION 5-2-CAN-DC-266-0, INCORRECTLY SHOWED 125V DC CONTROL CABLE 2PRZ55-AT AS BEING SUPPLIED BY BREAKER #12 WITH A SECONDARY PROTECTIVE DEVICE (FUSE). IN ACTUALITY, THIS CABLE IS SUPPLIED BY BREAKER 8, WITHOUT SECONDARY PROTECTION. THE ROOT CAUSE OF HAVING A CONTAINMENT ELECTRICAL PENETRATION CONDUCTOR WHOSE CIRCUITRY LACKS REDUNDANT OVERCURRENT PROTECTION IS CLASSIFIED AS "DESIGN, MANUFACTURING, CONSTRUCTION/INSTALLATION" ERROR (CAUSE CODE B, NUREG-1022). A LINE CHECK OF DC ELECTRICAL PENETRATION WIRING DIAGRAMS WAS PERFORMED AND SIMILAR CIRCUITS WITHIN DC DISTRIBUTION PANELS WERE CHECKED. NO ADDITIONAL OVERCURRENT PROTECTION CONCERNS WERE IDENTIFIED. FUSES (1 PER LEG) WILL BE INSTALLED TO CABLE 2PRZ55-AT, CONDUCTORS 3 & 4 (DCP 2EC-3107). DRAWINGS WILL BE CORRECTED TO REFLECT THE PROPER BREAKER DESIGNATOR AND THE INSTALLATION OF THE FUSE PROTECTION TO THE CIRCUIT.

[279] SALEM 2 DOCKET 50-311 LER 92-001
VITAL BUS UNDERVOLTAGE DUE TO EQUIPMENT FAILURE.
EVENT DATE: 010492 REPORT DATE: 013092 NSSS: WE TYPE: PWR

(NSIC 223935) ON JANUARY 4, 1992, AT 2143 HOURS, 2A VITAL BUS TRIPPED DURING A TRANSFER OF THE BUS FROM 22 STATION POWER TRANSFORMER (SPT) TO 21 SPT. THE 21 SPT INFEED BREAKER (21ASD) DID NOT CLOSE RESULTING IN AN AUTOMATIC START AND BLACKOUT LOADING OF 2A DIESEL GENERATOR (D/G). THE AUTOMATIC START AND LOADING OF THE D/G IS AN ENGINEERED SAFETY FEATURE (ESF). THE CAUSE OF THE 2A VITAL BUS TRIP IS EQUIPMENT FAILURE. A CARTRIDGE FUSE IN THE 21 SPT POTENTIAL TRANSFORMER SECONDARY THAT SUPPLIES POWER TO THE METERING CIRCUIT HAD FAILED. A REVIEW WAS CONDUCTED OF OTHER FUSE CONCERNS (IN SIMILAR SYSTEMS). ADDITIONAL SIMILAR FAILURES OF THIS TYPE FUSE WERE NOT IDENTIFIED. THIS EVENT IS CONSIDERED TO BE AN ISOLATED OCCURRENCE. THE 2A 4KV VITAL BUS WAS RECONNECTED TO NO. 22 SPT AT 2225 HOURS ON JANUARY 4, 1992. THE FAILED FUSE WAS REPLACED.

[280] SAN ONOFRE 1 DOCKET 50-206 LER 91-021
VOLUNTARY ENTRY INTO TECHNICAL SPECIFICATION DUE TO INADEQUATE TECHNICAL SPECIFICATIONS IN ORDER TO PERFORM A SURVEILLANCE OF A CONTAINMENT SPRAY SYSTEM PUMP.
EVENT DATE: 110791 REPORT DATE: 120691 NSSS: WE TYPE: PWR

(NSIC 223591) ON 11/7/91, BETWEEN 1724 AND 1752, WHILE UNIT 1 WAS IN MODE 1 AT APPROX. 90% POWER AND WITH THE NORTH REFUELING WATER PUMP (RWP) INOPERABLE WITH MAINTENANCE REQUIRED ON ITS CIRCUIT BREAKER (52), TECH SPEC (TS) 3.0.3 WAS ENTERED IN ORDER TO DEMONSTRATE THE OPERABILITY OF THE SOUTH RWP (G27S). IN ORDER TO INITIATE MAINTENANCE ON G27N, TS 3.3.1.C. REQUIRES THAT G27S BE TESTED TO DEMONSTRATE ITS AVAILABILITY. IN ORDER TO PERFORM THE TEST WITHOUT INITIATING CONTAINMENT SPRAY, THE PUMP'S MANUAL DISCHARGE ISOLATION VALVE MUST BE CLOSED RENDERING THE SOUTH TRAIN INOPERABLE. SINCE THERE ARE NO ACTION STATEMENTS WHICH ADDRESS THE INOPERABILITY OF BOTH CONTAINMENT SPRAY SYSTEM TRAINS, PERFORMANCE OF THE G27S SURVEILLANCE TEST WOULD CONSTITUTE AN ENTRY INTO TS 3.0.3. THE G27S DISCHARGE ISOLATION VALVE WAS CLOSED BETWEEN 1724 AND 1752 WHILE THE PUMP SURVEILLANCE WAS BEING PERFORMED. THE SURVEILLANCE TEST VERIFIED THAT PUMP G27S CONTINUED TO BE OPERABLE. THE CAUSE OF THE TS 3.0.3 ENTRY IS THE ABSENCE OF APPROPRIATE ACTION STATEMENTS. A TS AMENDMENT APPLICATION HAS BEEN SUBMITTED WHICH WILL REVISE TS 3.3.1 TO PRECLUDE TS 3.0.3 ENTRIES FOR CONDITIONS SIMILAR TO THIS EVENT. THIS TS AMENDMENT APPLICATION IS BEING REVIEWED BY THE NRC AND SCE. THE ENTRY INTO TECH SPEC HAD NO SAFETY SIGNIFICANCE.

[281] SAN ONOFRE 2 DOCKET 50-361 LER 91-017
DELINQUENT REACTOR COOLANT SYSTEM SURVEILLANCE SAMPLE.
EVENT DATE: 111291 REPORT DATE: 121291 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 223619) ON 11/13/91, IT WAS DETERMINED THAT THE UNIT 2 AND UNIT 3 REACTOR COOLANT SYSTEM SAMPLES OBTAINED ON 11/11/91, HAD NOT BEEN ANALYZED FOR FLUORIDE WITHIN THE TIME REQUIRED BY TECHNICAL SPECIFICATION (TS) 3/4.4.6, "REACTOR COOLANT SYSTEM CHEMISTRY." THE ANALYSES WERE REQUIRED TO BE COMPLETED ON 11/12/91 AT 0223 AND 0250 FOR UNIT 2 AND UNIT 3, RESPECTIVELY. THE FLUORIDE ANALYSES WERE SUBSEQUENTLY COMPLETED AT APPROXIMATELY 1100 ON 11/12/91. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE A COGNITIVE PERSONNEL ERROR. A NUCLEAR CHEMISTRY TECHNICIAN FOREMAN (UTILITY, NON-LICENSED) HAD INCORRECTLY DEFERRED COMPLETION OF THE ANALYSIS WITHOUT RECOGNIZING THAT THE SURVEILLANCE INTERVAL (INCLUDING THE EXTENSION ALLOWED BY TS 4.0.2) WOULD BE EXCEEDED. IN ADDITION, DIFFICULTIES WERE ENCOUNTERED IN THE USE OF THE FLUORIDE ANALYSIS EQUIPMENT WHICH CONTRIBUTED TO THE DELAY. CORRECTIVE ACTIONS INCLUDE 1) REVIEWING THIS EVENT WITH APPROPRIATE CHEMISTRY PERSONNEL, AND 2) PROVIDING REFRESHER TRAINING WITH RESPECT TO TS SAMPLING AND ANALYSIS REQUIREMENTS TO APPROPRIATE CHEMISTRY PERSONNEL. THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE SUBSEQUENT SAMPLE RESULTS WERE WITHIN THE TS LIMITS.

[282] SAN ONOFRE 2 DOCKET 50-361 LER 91-019
INOPERABLE CONTAINMENT PRESSURE INSTRUMENTATION DUE TO MISLABELED WIRING FROM INSTRUMENT LINE ISOLATION VALVE SOLENOID ACTUATOR.
EVENT DATE: 111691 REPORT DATE: 122491 NSSS: CE TYPE: PWR
VENDOR: VALCOR ENGINEERING CORP.

(NSIC 223735) AT 2030 ON 11/25/91, WITH UNIT 2 AT 100% POWER, IT WAS DETERMINED THAT CONTAINMENT PRESSURE INSTRUMENT LINE ISOLATION VALVE 2HV0352C HAD BEEN CLOSED (ALTHOUGH INDICATING OPEN) SINCE THE VALVE OPERATING SOLENOIDS AND POSITION INDICATION ASSEMBLY HAD BEEN REPLACED ON 11/9/91, DURING THE PREVIOUS REFUELING OUTAGE. AS A RESULT, THE CHANNEL "C" CONTAINMENT PRESSURE TRANSMITTERS HAD BEEN ISOLATED FROM CONTAINMENT, AND INOPERABLE, SINCE THAT TIME. TECHNICAL SPECIFICATION (TS) 3.3.2, "ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION," REQUIRES THAT IN MODES 1-3, AN INOPERABLE CONTAINMENT PRESSURE CHANNEL MUST BE BYPASSED WITHIN ONE HOUR. MODE 3 WAS ENTERED AT 2329 ON 11/15/91 THEREFORE, OPERATION IN MODES 1-3 AFTER 0029 ON 11/16/91, WITH PPS CHANNEL "C" NOT BYPASSED, REPRESENTED AN OPERATION PROHIBITED BY TSS. THERE IS MINIMAL SAFETY SIGNIFICANCE TO THIS EVENT SINCE THE REMAINING CHANNELS OF CONTAINMENT PRESSURE

INSTRUMENTATION REMAINED CAPABLE OF INITIATING ASSOCIATED PLANT PROTECTION SYSTEM ACTUATIONS. THE WIRING FROM THE OPERATING SOLENOIDS FOR 2HV0352C WAS FOUND TO BE MISLABELED BY THE MANUFACTURER AS A RESULT, THE VALVE ACTUATOR WAS INSTALLED SUCH THAT BOTH OPERATION AND CONTROL ROOM INDICATION OF THE VALVE WERE REVERSED.

[283] SAN ONOFRE 3 DOCKET 50-362 LER 91-008
MISSED FIREWATCH DUE TO COGNITIVE PERSONNEL ERROR.
EVENT DATE: 112991 REPORT DATE: 122391 NSSS: CE TYPE: PWR

(NSIC 223817) AT 1603 ON 11/29/91, WITH UNIT 3 IN MODE 1 AT 100% POWER, A FIREWATCH SUPERVISOR (CONTRACTOR, NON-LICENSED) CONDUCTING AN INSPECTION DISCOVERED THAT THE REQUIRED HOURLY FIREWATCH PATROL FOR THE TWO UNIT 3 EMERGENCY DIESEL GENERATOR (DG) (DG, EK) (30002 AND 30003) ROOMS HAD BEEN MISSED BETWEEN 1500 AND 1600. THE HOURLY FIREWATCH PATROL HAD BEEN ESTABLISHED AS REQUIRED BY TECH SPECS 3.7.8.2, "SPRAY AND/OR SPRINKLER SYSTEMS," AND 3.3.3.7, "FIRE DETECTION INSTRUMENTATION," FOR INOPERABLE FIRE PROTECTION EQUIPMENT (KP). THE CAUSE OF THIS EVENT WAS COGNITIVE PERSONNEL ERROR BY THE FIREWATCH (CONTRACTOR, NON-LICENSED) WHEN HE MISSED THE DG ROOMS DURING HIS ASSIGNED PATROL. THE FIREWATCH INVOLVED IN THIS EVENT HAS RECEIVED APPROPRIATE DISCIPLINARY ACTION. THIS EVENT HAS BEEN REVIEWED BY APPROPRIATE FIREWATCH PERSONNEL. AS ADDITIONAL CORRECTIVE ACTION, FIREWATCH SUPERVISORS HAVE INCREASED THEIR MONITORING OF ROVING FIREWATCH PERSONNEL. THERE WAS NO SAFETY SIGNIFICANCE ASSOCIATED WITH THIS EVENT SINCE: 1) THREE EARLY WARNING SMOKE DETECTORS LOCATED IN EACH DG ROOM WERE OPERABLE AND WOULD HAVE ALERTED APPROPRIATE PERSONNEL OF A FIRE, 2) MANUAL FIRE FIGHTING EQUIPMENT WAS AVAILABLE IN THE DG ROOMS TO EXTINGUISH A FIRE, AND 3) A CONTINUOUSLY MANNED FIRE STATION IS MAINTAINED ON-SITE TO IMMEDIATELY RESPOND TO A FIRE.

[284] SAN ONOFRE 3 DOCKET 50-362 LER 91-009
DELINQUENT TECHNICAL SPECIFICATION QUARTERLY SURVEILLANCE ON CLASS 1E 125 VOLT DC BATTERIES DUE TO PERSONNEL ERROR.
EVENT DATE: 122191 REPORT DATE: 020392 NSSS: CE TYPE: PWR

(NSIC 223940) ON 1/2/92, WITH UNIT 3 AT 100% POWER, DURING A REVIEW OF MAINTENANCE ORDERS THAT REQUIRED SCHEDULING, IT WAS IDENTIFIED THAT THE QUARTERLY SURVEILLANCE ON THE CLASS 1E 125 VOLT DC (VDC) "C" AND "D" BATTERIES, AS REQUIRED BY TECHNICAL SPECIFICATION (TS) 3/4.8.2, "DC SOURCE" WAS DELINQUENT. THE SURVEILLANCE HAD LAST BEEN COMPLETED ON 8/28/91 AND WAS OVERDUE ON 12/21/91. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE THE DELINQUENT BATTERY SURVEILLANCE WAS COMPLETED WITH SATISFACTORY RESULTS ON 1/2/91, INDICATING THAT THE BATTERIES HAD REMAINED CAPABLE OF PERFORMING THEIR SAFETY FUNCTION. PERSONNEL INVOLVED WITH ENSURING THAT TS SURVEILLANCES ARE COMPLETED WITHIN SPECIFIED TIME INTERVALS FAILED TO ADEQUATELY PERFORM THEIR ASSIGNED TASKS AS INTENDED BY ESTABLISHED MAINTENANCE PRACTICES DUE TO PERSONNEL ERROR, INCLUDING: 1) THE SURVEILLANCE PLANNER AND SCHEDULER FAILED TO ADEQUATELY REVIEW THE WEEKLY SURVEILLANCE STATUS REPORT, WHICH CLEARLY IDENTIFIED THE BATTERY SURVEILLANCE TO BE DELINQUENT; AND 2) THE PLANNER'S SUPERVISION FAILED TO ADEQUATELY PERFORM THEIR OVERSIGHT RESPONSIBILITIES. APPROPRIATE DISCIPLINARY ACTION HAS BEEN ADMINISTERED TO THE PERSONNEL INVOLVED WITH THE DELINQUENT SURVEILLANCE. THIS EVENT WAS REVIEWED WITH THE SURVEILLANCE PLANNER, SCHEDULER, THEIR SUPERVISION, AND OTHER APPROPRIATE PERSONNEL AND SUPERVISORS.

[285] SEQUOYAH 1 DOCKET 50-327 LER 91-026
INSUFFICIENT VENDOR INFORMATION RESULTS IN A LACK OF SELECTIVE BREAKER COORDINATION.
EVENT DATE: 110791 REPORT DATE: 120991 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 223608) ON 11/7/91, AT 0800 EASTERN STANDARD TIME (EST) WITH UNIT 1 IN A REFUELING OUTAGE AND UNIT 2 OPERATING AT APPROXIMATELY 100% POWER, IT WAS DETERMINED THAT VARIOUS 480-VOLT ELECTRICAL BOARDS DID NOT HAVE SELECTIVE COORDINATION BETWEEN THE FEEDER BREAKERS AND THE LOAD BREAKERS ON THE BOARD AS A

RESULT OF A BREAKER DESIGN FEATURE. THE DESIGN FEATURE HAD NOT BEEN RECOGNIZED AS A RESULT OF INSUFFICIENT VENDOR INFORMATION. TO CORRECT THE COORDINATION PROBLEM ON SAFETY-RELATED BOARDS, A MODIFICATION WAS INITIATED TO DISABLE THE DESIGN FEATURE ON SELECTED BREAKERS. ON 11/8/91, TWO ELECTRICIANS IMPLEMENTING THE MODIFICATION INTRODUCED A GLOWSTICK INTO A 1B2-B SHUTDOWN BOARD COMPARTMENT PRIOR TO CHECKING FOR FAULTS. THE ELECTRODE ASSEMBLY OF THE GLOWSTICK CONTACTED THE B AND C PHASE BUSES AND A FAULT OCCURRED CAUSING DAMAGE AND LOSS OF THE SHUTDOWN BOARD. THE SHUTDOWN BOARD WAS REPAIRED AND THE MODIFICATION COMPLETED FOR THE SELECTED BREAKERS.

[286] SEQUOYAH 1 DOCKET 50-327 LER 91-027
MANUAL CLOSURE OF THE MAIN STEAM ISOLATION VALVES AS A RESULT OF A MALFUNCTION OF A STEAM DUMP VALVE CONTROLLER RESULTING IN A COOLDOWN OF THE REACTOR COOLANT SYSTEM.
EVENT DATE: 121391 REPORT DATE: 011392 NSSS: WE TYPE: PWR
VENDOR: COPES-VULCAN, INC.
FOXBORO CO., THE

(NSIC 223798) ON DECEMBER 13, 1991, AT 1459 EASTERN STANDARD TIME (EST) UNIT 1 REACTOR COOLANT SYSTEM (RCS) EXPERIENCED AN INADVERTENT COOLDOWN EVENT. UPON OBSERVING PRESSURIZER LEVEL DECREASING, OPERATIONS INCREASED CHARGING FLOW AND MANUALLY ISOLATED LETDOWN IN AN ATTEMPT TO STABILIZE THE PRESSURIZER LEVEL. REVIEW OF PLANT PARAMETERS DETERMINED THAT THE EVENT WAS DUE TO A PROBLEM ON THE SECONDARY SIDE OF THE PLANT. OPERATIONS PROCEEDED TO MANUALLY CLOSE THE MAIN STEAM LINE ISOLATION VALVES AND MANUALLY CLOSED THE LEVEL CONTROL VALVES TO LIMIT THE COOLDOWN EFFECT OF AUXILIARY FEEDWATER. THE ORDER TO COMPLETE STEAM GENERATOR (S/G) ISOLATION, THE OPERATORS ISOLATED S/G BLOWDOWN. AFTER ISOLATION OF ALL FOUR S/GS, PLANT PARAMETERS STABILIZED. THE ROOT CAUSE OF THIS EVENT WAS THE INTERMITTENT FAILURE OF A STEAM DUMP VALVE CONTROLLER RESULTING IN THE OPENING OF TWO STEAM DUMP VALVES. THE SEVERITY OF THE COOLDOWN WAS CAUSED BY A STEAM DUMP VALVE FAILING TO CLOSE AFTER THE CONTROLLER MALFUNCTIONED. A WORK ORDER WAS PERFORMED THAT REPLACED THE PRESSURE CONTROLLER AND CONTROLLING POTENTIOMETER.

[287] SEQUOYAH 2 DOCKET 50-328 LER 91-006
REACTOR TRIP ON LOW-LOW STEAM GENERATOR LEVEL RESULTING FROM AN INADVERTENT MAIN STEAM ISOLATION VALVE CLOSURE CAUSED BY A LIMIT SWITCH FAILURE.
EVENT DATE: 110791 REPORT DATE: 120991 NSSS: WE TYPE: PWR
VENDOR: NAMCO CONTROLS

(NSIC 223609) ON NOVEMBER 7, 1991, AT 1403 EASTERN STANDARD TIME, THE UNIT 2 REACTOR TRIPPED ON LOW-LOW STEAM GENERATOR LEVEL IN LOOP 4 DURING A ROUTINE PERFORMANCE OF A MAIN STEAM ISOLATION VALVE (MSIV) PARTIAL STROKE TEST IN ACCORDANCE WITH SURVEILLANCE REQUIREMENT 4.7.1.5. DURING THE PERFORMANCE OF THE TEST, THE MSIV FULLY CLOSED BECAUSE OF THE FAILURE OF A LIMIT SWITCH TO ACTUATE, WHICH REOPENS THE VALVE UP ON REACHING THE 90 PERCENT OPEN POSITION. THE CONTROL ROOM STAFF RESPONDED AS PRESCRIBED BY EMERGENCY PROCEDURES. THEY PROMPTLY DIAGNOSED THE PLANT CONDITIONS AND TOOK ACTIONS NECESSARY TO STABILIZE THE UNIT IN A SAFE CONDITION. SAFETY SYSTEMS PERFORMED AS EXPECTED WITH MINOR ANOMALIES THAT DID NOT AFFECT THE OPERATORS' CAPABILITY TO RESPOND TO THE EVENT. THE 90 PERCENT LIMIT SWITCH WAS REPLACED, AND THE MSIV WAS RETESTED SUCCESSFULLY.

[288] SEQUOYAH 2 DOCKET 50-328 LER 91-008
FAILURE OF THE REFUELING WATER STORAGE TANK WIDE RANGE LEVEL TRANSMITTERS BECAUSE OF INADEQUATE ADMINISTRATIVE CONTROLS.
EVENT DATE: 120291 REPORT DATE: 010292 NSSS: WE TYPE: PWR
VENDOR: ITT-BARTON

(NSIC 223721) ON DECEMBER 2, 1991, AT 0050 EASTERN STANDARD TIME, UNIT 2 ENTERED THE ACTION STATE OF LIMITING CONDITIONS FOR OPERATION 3.0.3 AND 3.3.3.7 BECAUSE OF THE INOPERABILITY OF THE FOUR WIDE RANGE REFUELING WATER STORAGE TANK (RWST) LEVEL TRANSMITTERS (LTS). TRANSMITTER FAILURES OCCURRED WHEN HEAVY RAINS RESULTED IN THE FLOODING OF THE BASIS SURROUNDING THE RWST. SUBSEQUENTLY, THE LTS FAILED BECAUSE OF THE PARTIAL SUBMERGENCE OF THE CABINETS HOUSING THE TRANSMITTERS

RESULTING IN A SHORT BETWEEN THE ELECTRICAL LEADS. THE ROOT CAUSE OF THIS EVENT IS CONSIDERED TO BE INEFFECTIVE ADMINISTRATIVE CONTROLS TO COMPENSATE FOR A DESIGN DEFICIENCY. THE RWST BASIN WAS SLUMPED OUT, AND LTS AND TERMINAL STRIPS WERE DRIED. THE PROPER OPERATION OF THE TRANSMITTERS WAS VERIFIED, AND THEY WERE RETURNED TO SERVICE. CORRECTIVE ACTION IS TO STRENGTHEN THE ADMINISTRATIVE CONTROLS.

[289] SHEARON HARRIS 1 DOCKET 50-400 LER 91-020
FAILURE TO ADEQUATELY TEST SEALED SOURCES RESULTING IN TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 120491 REPORT DATE: 122091 NSSS: WE TYPE: PWR

(NSIC 223682) ON 12/4/91, DURING A ROUTINE REVIEW OF COMPLETED TEST DOCUMENTATION, RADIATION CONTROL SUPERVISORY PERSONNEL IDENTIFIED A DEFICIENCY IN RADIATION SURVEILLANCE TEST RST-010. THIS PROCEDURE IS USED FOR LEAK TESTING SEALED SOURCES TO SATISFY TECHNICAL SPECIFICATIONS (TS) 4.7.9.1 AND 4.7.9.2. THIS PROCESS INCLUDES SMEARING THESE SOURCES AND ANALYZING THE SMEARS WITH AN INSTRUMENT CAPABLE OF DETECTING .005 MICROCURIES OF CONTAMINATION PER TEST SAMPLE. IT WAS DETERMINED THAT THE INSTRUMENT USED FOR THIS ANALYSIS DID NOT HAVE THE REQUIRED SENSITIVITY LEVEL TO DETECT THE LOW ENERGY EMISSIONS PRODUCED BY SOURCES THAT CONTAIN NICKEL-63 AND IRON-55 ISOTOPES. THESE SOURCES ARE USED IN THE PLANT ACCESS EXPLOSIVE DETECTORS AND AN ALLOY ANALYZER USED FOR NON-DESTRUCTIVE EXAMINATION (RESPECTIVELY). EACH OF THE TWELVE SOURCES THAT CONTAIN THESE ELEMENTS WERE SMEARED AGAIN AND WERE ANALYZED WITH A LIQUID SCINTILLATION COUNTER THAT HAS THE REQUIRED SENSITIVITY LEVEL. NO CONTAMINATION LEVELS IN EXCESS OF TS LIMITS WERE FOUND. THE CAUSE OF THIS EVENT WAS A PROCEDURAL DEFICIENCY IN RST-010. ADEQUATE GUIDANCE WAS NOT PROVIDED IN THIS PROCEDURE TO ENSURE THAT THE PROPER TYPE OF INSTRUMENT WAS USED FOR ANALYSIS. CORRECTIVE ACTIONS WILL INCLUDE A REVISION TO RST-010 AND TRAINING FOR APPLICABLE RADIATION CONTROL PERSONNEL ON THIS REVISION.

[290] SHEARON HARRIS 1 DOCKET 50-400 LER 92-001
VENTILATION DEFICIENCES FOR SPARE CHARGING/SAFETY INJECTION PUMP ROOM.
EVENT DATE: 010292 REPORT DATE: 020392 NSSS: WE TYPE: PWR

(NSIC 223942) REACTOR AUXILIARY BUILDING VENTILATION, BOTH COOLING AND EMERGENCY EXHAUST, FOR THE STANDBY CHARGING/SAFETY INJECTION PUMP (CSIP) ROOM WAS FOUND TO BE INOPERABLE. THIS PUMP IS A SWING PUMP THAT CAN BE PLACED INTO SERVICE TO REPLACE EITHER THE NORMAL "A" OR "B" TRAIN PUMP IF THEY ARE INOPERABLE OR OUT-OF-SERVICE FOR MAINTENANCE. THE STANDBY CSIP ROOM VENTILATION WAS INOPERABLE BECAUSE ONE MANUAL DAMPER WAS MISALIGNED AND TWO BLANK FLANGES WERE NOT INSTALLED AS CALLED FOR BY PLANT DRAWINGS. AT THE TIME THESE DEFICIENCIES WERE IDENTIFIED, BOTH THE "A" AND "B" CSIPS WERE OPERABLE AND THE STANDBY CSIP WAS NOT IN-SERVICE. AS CORRECTIVE ACTIONS, THE MANUAL DAMPER WAS PLACED INTO ITS REQUIRED POSITION AND BLANK FLANGES HAVE BEEN FABRICATED AND INSTALLED AS ORIGINALLY DESIGNED. THIS EVENT IS REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(I)(B) AS A TECHNICAL SPECIFICATION VIOLATION.

[291] SOUTH TEXAS 1 DOCKET 50-498 LER 91-014 REV 02
UPDATE ON ERRATIC CONTAINMENT EXTENDED RANGE PRESSURE CHANNEL OUTPUT.
EVENT DATE: 042091 REPORT DATE: 011592 NSSS: WE TYPE: PWR
VENDOR: ITT-BARTON

(NSIC 223809) ON APRIL 20, 1991, UNIT 1 WAS IN MODE 1 AT 100% POWER. AT 0406 HOURS, WHILE CONDUCTING A CONTAINMENT SUPPLEMENTAL PURGE TO LOWER THE CONTAINMENT PRESSURE IN RESPONSE TO A CONTAINMENT HIGH PRESSURE ALARM, CONTAINMENT EXTENDED RANGE PRESSURE CHANNEL 9759 WAS FOUND TO READ 5 PSIG WHILE CHANNEL 9760 READ 0 PSIG. CHANNEL 9759 WAS DECLARED INOPERABLE AT 0407 HOURS. REVIEW OF HISTORICAL COMPUTER RECORDS INDICATED THAT THE CHANNEL HAD BEEN INOPERABLE IN EXCESS OF THE SEVEN-DAY ALLOWED OUTAGE TIME. AFTER INITIAL RECALIBRATION, SUBSEQUENT CHANNEL CHECK SURVEILLANCE REVEALED AN ADDITIONAL ERRATIC OUTPUT SIGNAL BY THE TRANSMITTER. THE TRANSMITTER CONTROL CARD WAS REPLACED AND THE TRANSMITTER WAS CALIBRATED. CHANNEL CHECKS WERE PERFORMED WEEKLY FOR ONE MONTH TO CONFIRM THE

CHANNEL WAS REPAIRED. ALTHOUGH NO GENERIC FAILURE MECHANISM HAS BEEN ESTABLISHED, THE FAILURE RATES ARE CONSISTENT WITH INDUSTRY EXPERIENCE. THESE TRANSMITTERS ARE BEING MONITORED UNDER THE FACILITY TREADING PROGRAM.

[292] SOUTH TEXAS 2 DOCKET 50-499 LER 91-007 REV 01
UPDATE ON REACTOR TRIP CAUSED BY INADVERTENT ACTUATION OF GENERATOR BREAKER EMERGENCY TRIP.
EVENT DATE: 052291 REPORT DATE: 012992 NSSS: WE TYPE: PWR
VENDOR: CONTROL COMPONENTS
PAUL-MUNROE HYDAULICS INC.

(NSIC 223871) ON 5/22/91, UNIT 2 WAS IN MODE 1 AT 100% POWER. AT APPROXIMATELY 2220 HRS, WHILE WAITING IN THE AREA OF THE MAIN GENERATOR BREAKER TO UNLOCK A LOCAL CABINET FOR AN ELECTRICAL MAINTENANCE INDIVIDUAL, A NON-LICENSED OPERATOR INADVERTENTLY ACTUATED THE LOCAL GENERATOR BREAKER EMERGENCY TRIP PUSHBUTTON. THE SUDDEN LOSS OF SECONDARY LOAD CAUSED AN AUTOMATIC OVER TEMPERATURE DELTA TEMPERATURE (OTDT) REACTOR TRIP. PRESSURIZER SPRAY WAS UNABLE TO REDUCE THE PRESSURE BEFORE THE PRESSURIZER PORVS OPENED AT APPROXIMATELY 2335 PSIG. STEAM GENERATOR 2C POWER-OPERATED RELIEF VALVE (PORV) FAILED TO OPEN EVEN THOUGH THE PRESSURE EXCEEDED THE LIFT SETPOINT. THE NON-LICENSED OPERATOR RESPONSIBLE FOR THE TRIP WAS COUNSELLED WITH REGARDS TO PAYING STRICT ATTENTION TO PERFORMANCE OF OPERATIONS ACTIVITIES. THE STEAM GENERATOR 2C PORV HAS BEEN REPAIRED. OTHER SWITCH DESIGNS HAVE BEEN REVIEWED TO IDENTIFY CHANGES THAT CAN PREVENT SIMILAR INADVERTENT ACTUATIONS.

[293] SOUTH TEXAS 2 DOCKET 50-499 LER 91-010
AUTOMATIC REACTOR TRIP AND SAFETY INJECTION ACTUATION DUE TO LOW PRESSURIZER PRESSURE.
EVENT DATE: 122491 REPORT DATE: 013092 NSSS: WE TYPE: PWR
VENDOR: BAILEY METER COMPANY
FISHER FLOW CONTROL DIV (ROCKWELL INT)

(NSIC 223872) ON DECEMBER 24, 1991, AT 1644 HOURS, UNIT 2 WAS OPERATING AT 30% RATED THERMAL POWER (RTP) WHEN PRESSURIZER SPRAY VALVE PCV-655C FAILED OPEN. THIS ULTIMATELY CAUSED AN AUTOMATIC REACTOR TRIP AND SAFETY INJECTION (SI) ACTUATION ON LOW PRESSURE AT 1648 HOURS FROM 16% RTP. THREE REACTOR COOLANT PUMPS (RCPS) WERE SECURED TO TERMINATE THE TRANSIENT. ALL AVAILABLE SAFETY EQUIPMENT PERFORMED AS DESIGNED AND NO ACTUAL INJECTION TO THE REACTOR OCCURRED. THE CAUSE WAS DISENGAGEMENT OF THE FEEDBACK ARM LINKAGE TO THE VALVE STEM CONNECTING PLATE ON THE PRESSURIZER SPRAY VALVE CONTROLLER. LOCKING NUTS WERE ADDED TO THE SPRAY VALVE FEEDBACK ARM LINKAGE CONNECTING SCREWS. ONGOING CORRECTIVE ACTIONS INCLUDE IMPROVING MAINTENANCE WORK INSTRUCTIONS, CONDUCTING PLANT MANAGEMENT REVIEWS WITH PERSONNEL TO DISCUSS THE E[illegible] AND PROVIDING TRAINING ON LESSONS LEARNED FROM THE EVENT.

[294] ST. LUCIE 1 DOCKET 50-335 LER 91-008
CORE ALTERATIONS PERFOR[illegible] CONTAINMENT INTEGRITY NOT IN COMPLIANCE WITH TECHNICAL SPECIFICATIONS D[illegible]O PERSONNEL ERROR.
EVENT DATE: 111091 REPORT DATE: 120991 NSSS: CE TYPE: PWR

(NSIC 223661) BETWEEN 1430 ON 11/10/91, AND 0210 ON NOVEMBER 11, CONTAINMENT INTEGRITY WAS NOT IN COMPLIANCE WITH TECHNICAL SPECIFICATIONS. ST. LUCIE UNIT 1 WAS IN MODE 6, REFUELING, AND FUEL ASSEMBLIES WERE BEING MOVED IN THE REACTOR CORE. CONTAINMENT INTEGRITY IS REQUIRED FOR CORE ALTERATIONS. THE CAUSE OF THE INCOMPLETE CONTAINMENT INTEGRITY WAS [illegible]ERSONNEL ERROR BY MECHANICAL MAINTENANCE DEPARTMENT PERSONNEL FOR FAILURE TO [illegible]LOW AN APPROVED PROCEDURE. PLANT PROCEDURE REQUIRES THAT NUCLEAR PLANT WORK [illegible] BE APPROVED BY THE NUCLEAR PLANT SUPERVISOR PRIOR TO STARTING WORK[illegible]UT OBTAINING THIS PRIOR APPROVAL, MAINTENANCE WORKERS REMOVED A REL[illegible]LVE ON ONE SIDE OF A CONTAINMENT PENETRATION WHILE A DRAIN VALVE [illegible]EN ON THE OTHER SIDE. THIS CREATED A DIRECT FLOW PATH FROM CONTAINMENT TO THE [illegible]ACTOR AUXILIARY BUILDING. PERMISSION TO START WORK WAS REQUESTED AFTER THE RELIEF VALVE HAD BEEN REMOVED BY THE PREVIOUS SHIFT. CORRECTIVE ACTIONS WERE: CONTAINMENT INTEGRITY WAS RE-ESTABLISHED, THE

RESPONSIBLE MAINTENANCE SUPERVISOR WAS COUNSELED ON THE NEED FOR STRICT ADHERENCE TO PROCEDURES, AND THE EVENT WILL BE INCORPORATED INTO MAINTENANCE JOURNEYMAN TRAINING. THIS EVENT WAS ALSO DISCUSSED WITH ALL MAINTENANCE CREWS AND THE IMPORTANCE OF RECEIVING PROPER AUTHORIZATION PRIOR TO BEGINNING WORK ON PLANT SYSTEMS WAS STRESSED.

[295] ST. LUCIE 1 DOCKET 50-335 LER 91-009
REMOVAL OF THE PLANT VENT STACK MONITORS FROM SERVICE RESULTED IN A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS DUE TO PERSONNEL ERROR.
EVENT DATE: 111491 REPORT DATE: 120991 NSSS: CE TYPE: PWR

(NSIC 223613) ON 11/14/91 WITH ST. LUCIE UNIT 1 IN MODE 6 COMPLETING REFUELING REQUIREMENTS, A LICENSED UTILITY OPERATOR NOTED THAT THE UNIT WAS IN A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS. THE PLANT VENT STACK PARTICULATE SAMPLER, IODINE SAMPLER, AND NOBLE GAS ACTIVITY MONITOR HAD BEEN PLACED OUT OF SERVICE AND THE CHEMISTRY DEPARTMENT WAS NOT NOTIFIED TO SAMPLE FOR ACTIVITY. TECHNICAL SPECIFICATION 3.3.3.10 REQUIRES CONTINUOUS SAMPLE COLLECTION WITH AUXILIARY SAMPLING EQUIPMENT TO BE PERFORMED IF THE PLANT VENT IS TO BE USED AS THE EFFLUENT PATHWAY. THIS CONDITION EXISTED FOR APPROXIMATELY TWELVE HOURS. DURING THIS TIME PERIOD THERE WERE NO ACTIVITIES WHICH COULD HAVE RESULTED IN AN ABNORMAL GASEOUS RELEASE. THE ROOT CAUSE OF THIS EVENT WAS A COGNITIVE PERSONNEL ERROR BY UTILITY-LICENSED OPERATORS FOR REFERENCING AN INAPPROPRIATE TECHNICAL SPECIFICATION PERTAINING TO RADIATION MONITORING INSTRUMENTATION. CORRECTIVE ACTIONS TAKEN WERE: 1) CHEMISTRY OBTAINED A GAS SAMPLE AND ANALYZED IT FOR ABNORMAL ACTIVITY; 2) OPERATIONS REVIEWED CONTAINMENT WORK FOR THE PERIOD WHEN THE PLANT STACK WAS UNMONITORED; 3) OPERATIONS SUPERVISION COUNSELLED ALL LICENSED PERSONNEL; 4) TRAINING DEPARTMENT INCORPORATED THIS EVENT INTO THE REQUALIFICATION PROGRAM.

[296] SUMMER 1 DOCKET 50-395 LER 92-001
MISSED SURVEILLANCE FOR AXIAL FLUX DIFFERENCE.
EVENT DATE: 011192 REPORT DATE: 021092 NSSS: WE TYPE: PWR

(NSIC 224040) ON JANUARY 11, 1992, AT APPROXIMATELY 1830 HOURS, OPERATIONS PERSONNEL OBSERVED THAT THE INTEGRATED PLANT COMPUTER SYSTEM (IPCS) WAS INDICATING AN AXIAL FLUX DIFFERENCE (AFD) OF 0 AS COMPARED TO MAIN CONTROL BOARD (MCB) INDICATIONS OF APPROXIMATELY +1.5%. THE COMPUTER WAS DECLARED INOPERABLE AT 1845 HOURS AND MANUAL TRACKING OF THE AFD WAS INITIATED IN ACCORDANCE WITH TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT 4.2.1.1(B) FOR MONITORING AND LOGGING OF INDICATIONS WHEN THE AFD MONITOR ALARM IS INOPERABLE. THE COMPUTER WAS SHUTDOWN, RESTARTED, AND THE FUNCTION OF SOFTWARE PROGRAMS VERIFIED PRIOR TO RETURNING THE SYSTEM TO OPERATION AT 2200 HOURS. A REVIEW OF THE SEQUENCE OF EVENTS SUMMARY DETERMINED THAT THE FAILURE TO PERFORM CERTAIN CRITICAL CALCULATIONS WAS DUE TO THE DELETION OF COMPUTER POINT U1169 (A CALCULATED VALUE THAT REPRESENTS THE STATISTICAL AVERAGE OF THE FOUR NUCLEAR POWER RANGE CHANNELS). CONTROL ROOM PERSONNEL WERE UNAWARE OF ERRORS WHICH COULD BE GENERATED WITHIN THE IPCS DUE TO THE DELETION OF CERTAIN COMPUTER POINTS FROM PROCESSING (UTILIZATION OF INPUT DATA FOR SOFTWARE CALCULATIONS). SCE&G CONSIDERS THIS EVENT TO BE ISOLATED AND NOT INDICATIVE OF DEFICIENCIES IN EITHER PLANT PROGRAMMATIC CONTROLS OR COMPUTER SOFTWARE.

[297] SURRY 1 DOCKET 50-280 LER 91-022
CONTAINMENT SPRAY PUMP SURVEILLANCE TESTING EXCEEDED TECHNICAL SPECIFICATION REQUIREMENTS DUE TO PERSONNEL ERROR.
EVENT DATE: 120491 REPORT DATE: 123091 NSSS: WE TYPE: PWR

(NSIC 223766) ON DECEMBER 4, 1991, WITH UNIT 1 AT 100% POWER, IT WAS DISCOVERED THAT THE CONTAINMENT SPRAY PUMPS (1-CS-P-1A AND 1-CS-P-1B) HAD NOT BEEN DEMONSTRATED OPERABLE BETWEEN JANUARY 1, 1991 AND MAY 14, 1991. THIS EXCEEDED THE ALLOWED 92 DAY INTERVAL BETWEEN SURVEILLANCE TESTS AND THUS WAS A VIOLATION OF TECHNICAL SPECIFICATION 4.0.3. SINCE THE CONTAINMENT SPRAY PUMPS REMAINED OPERABLE THROUGHOUT THE PERIOD, THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED. THE CAUSE OF THESE EVENTS WAS PERSONNEL ERROR IN THE CONDUCT OF THE

SURVEILLANCE TEST PROGRAM WHICH OCCURRED DURING THE LATE JANUARY/EARLY FEBRUARY 1991 TIME FRAME. A TASK TEAM HAD BEEN FORMED EARLIER THIS YEAR TO REVIEW THESE AND SIMILAR EVENTS, AND APPROPRIATE CORRECTIVE ACTION WAS TAKEN. THESE EVENTS ARE BEING REPORTED PURSUANT TO 10CFR50.73(A)(2)(I)(B).

[298] SURRY 1 DOCKET 50-280 LER 92-001
DROPPED ROD DUE TO PERSONNEL ERROR FOLLOWED BY A REQUIRED MANUAL REACTOR TRIP.
EVENT DATE: 010292 REPORT DATE: 020392 NSSS: WE TYPE: PWR
VENDOR: FISHER CONTROLS CO.
LAMBDA ELECTRONICS
POWER DESIGNS INC.
WESTINGHOUSE ELEC CORP.-NUCLEAR ENERGY SYS

(NSIC 223952) ON 1/2/92 AT 1649 HOURS, WITH UNIT 1 AT 56% REACTOR POWER, TROUBLESHOOTING WAS IN PROGRESS FOR "B" SHUTDOWN BANK CONTROL ROD E-5 TO DETERMINE WHY E-5 HAD DROPPED INTO THE CORE AT 0754 THAT MORNING DURING THE PERFORMANCE OF BIWEEKLY CONTROL ROD FREEDOM OF MOVEMENT TESTING. AS "D" CONTROL BANK WAS MANUALLY STEPPED OUT BY THE OPERATOR TO CONTROL DELTA FLUX, A SECOND ROD, "D" CONTROL BANK CONTROL ROD H-2 DROPPED. CONTROL ROD H-2 WAS VERIFIED TO BE IN THE CORE AND THE REACTOR WAS MANUALLY TRIPPED IN ACCORDANCE WITH STATION ABNORMAL PROCEDURES. THIS EVENT OCCURRED AS THE RESULT OF PERSONNEL ERROR IN THAT THE TROUBLESHOOTING GUIDE PREPARED BY ELECTRICAL MAINTENANCE FOR ROD E-5 DID NOT IDENTIFY SHARED CIRCUITRY BETWEEN ROD E-5 AND H-2 WHICH WOULD RESULT IN H-2 BEING DROPPED IF CONTROL RODS WERE STEPPED DURING TROUBLESHOOTING. FOLLOWING THE REACTOR TRIP, AN "A" MAIN FEED PUMP TRIPPED WHEN ITS RECIRCULATION VALVE FAILED TO OPEN DUE TO A FAILED SOLENOID, INTERMEDIATE RANGE NUCLEAR INSTRUMENTATION INDICATION WAS ERRATIC DUE TO HIGH VOLTAGE POWER SUPPLY PROBLEMS, STEAM GENERATOR ATMOSPHERE POWER OPERATED RELIEF VALVE FOR "A" STEAM GENERATOR RESPONDED POORLY WHILE THE VALVES FOR "B" AND "C" STEAM GENERATORS DID NOT RESPOND AS EXPECTED, AND THE TURBINE GENERATOR ELECTRO-HYDRAULIC CONTROL SYSTEM INDICATIONS WERE ERRATIC.

[299] SURRY 1 DOCKET 50-280 LER 92-002
4160 VOLT TRANSFER BUS "D", "E", AND "F" UNDERVOLTAGE RELAY TRIP SETPOINTS SET BELOW TECHNICAL SPECIFICATION LIMIT DUE TO PROCEDURE ERROR.
EVENT DATE: 010692 REPORT DATE: 013092 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 223934) ON JANUARY 6, 1992, AT 1559 HOURS, WITH UNITS 1 AND 2 AT COLD SHUTDOWN AND 100% POWER, RESPECTIVELY, IT WAS DETERMINED, BASED ON PREVIOUS CALIBRATION "AS-LEFT" VALUES, THE 4160 VOLT TRANSFER BUSES "D" AND "F" (PHASES "AB" AND "BC") AND TRANSFER BUS "E" ("BC" PHASE) UNDERVOLTAGE (UV) RELAY TRIP SETPOINTS WERE NOT WITHIN THE REQUIRED TECHNICAL SPECIFICATION (TS) LIMIT. BOTH CHANNELS FOR STATION BLACKOUT AUXILIARY FEEDWATER INITIATION WERE DECLARED INOPERABLE AND A UNIT 2 LIMITING CONDITION FOR OPERATION (LCO) REQUIRING HOT SHUTDOWN WITHIN SIX HOURS WAS ENTERED. ACTIONS WERE PROMPTLY INITIATED TO CHECK AND CALIBRATE THE SUBJECT RELAYS. AT 1755 HOURS, THE SIX HOUR LCO WAS EXITED AND A LCO REQUIRING HOT SHUTDOWN WITHIN 48 HOURS WAS ENTERED IN ACCORDANCE WITH TS TABLE 3.7-2. THE 48 HOUR LCO WAS EXITED AT 2130 HOURS WHEN THE REMAINING UV RELAYS WERE CALIBRATED AND RETURNED TO SERVICE. THE EVENT WAS CAUSED BY AN ERROR IN THE CALIBRATION PROCEDURES IN THAT AN INCORRECT UV RELAY TRIP SETPOINT WAS SPECIFIED. THE CALIBRATION PROCEDURES ARE BEING REVISED TO REFLECT THE CORRECT SETPOINT VALUES. THE EVENT IS BEING REPORTED, PURSUANT TO 10CFR 50.73(A)(2)(I)(B), SINCE THIS CONDITION IS NOT PERMITTED BY TS 3.7.D.

[300] SURRY 2 DOCKET 50-281 LER 91-011
HIGH STEAM GENERATOR LEVEL DUE TO MAIN FEEDWATER REGULATING VALVE OSCILLATIONS RESULTS IN ESF ACTUATION AND REACTOR TRIP.
EVENT DATE: 121791 REPORT DATE: 011492 NSSS: WE TYPE: PWR
VENDOR: COPES-VULCAN, INC.

(NSIC 223789) ON DECEMBER 17, 1991, AT 2254 HOURS, WITH UNITS 1 AND 2 AT 100% AND 23% POWER, RESPECTIVELY, A UNIT 2 AUTOMATIC REACTOR TRIP OCCURRED AS A RESULT OF

A TURBINE TRIP DUE TO HIGH STEAM GENERA. . (SG) LEVEL. THE REQUIRED SAFETY SYSTEMS PERFORMED AS DESIGNED, APPROPRIATE OPERATOR ACTIONS WERE TAKEN TO ENSURE THE PERFORMANCE OF SYSTEM AUTOMATIC ACTIONS, AND THE UNIT WAS QUICKLY BROUGHT TO A STABLE NO-LOAD CONDITION. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO THE FAILURE OF THE "B" MAIN FEEDWATER REGULATING VALVE (MFRV) TO MAINTAIN A DEMAND POSITION. THIS CONDITION RESULTED IN A HIGH SG LEVEL, THE MAIN FEEDWATER PUMPS TRIPPING, AUTOMATIC START OF AUXILIARY FEEDWATER (AFW) PUMPS, TURBINE TRIP, AND AN AUTOMATIC REACTOR TRIP. THE "B" MFRV WAS INSPECTED ON DECEMBER 18, 1991. THE POSITIONER WAS FOUND TO BE WORN AND WAS REPLACED AND TESTED. DURING THE SUBSEQUENT UNIT STARTUP ON DECEMBER 18, 1991, A SIMILAR LESS SEVERE OSCILLATION OF THE SUBJECT VALVE WAS NOTED. THEREFORE, A ROOT CAUSE EVALUATION IS BEING PERFORMED TO DETERMINE THE CAUSE OF THIS CONDITION. A NON-EMERGENCY FOUR-HOUR REPORT, PURSUANT TO 10CFR 5).72(B)(2)(II), WAS MADE TO THE NUCLEAR REGULATORY COMMISSION AT 0235 HOURS ON DECEMBER 18, 1991. THIS EVENT IS BEING REPORTED, PURSUANT TO 10CFR 50.73(A)(2)(IV), AS AN UNPLANNED ENGINEERED SAFETY FEATURE ACTUATION AS A RESULT OF A VALID SIGNAL.

[301] SUSQUEHANNA 1 DOCKET 50-387 LER 91-013
POTENTIAL FOR PRESSURE-LOCKING OF RESIDUAL HEAT REMOVAL LOW PRESSURE COOLANT INJECTION SYSTEM AND CORE SPRAY SYSTEM INJECTION VALVES.
EVENT DATE: 101891 REPORT DATE: 111891 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)

(NSIC 223544) ON OCTOBER 18, 1991, WITH BOTH UNIT 1 AND UNIT 2 IN CONDITION 1 AT 100 POWER, PP&L ENGINEERING DETERMINED THAT A SUSCEPTIBILITY TO A MOTOR-OPERATED VALVE PRESSURE-LOCKING PHENOMENON EXISTS FOR THE RESIDUAL HEAT REMOVAL - LOW PRESSURE COOLANT INJE TION SYSTEM (RHR-LPCI) AND CORE SPRAY SYSTEM INJECTION VALVES. BASED ON PP&L'S OPERABILITY EVALUATION, THIS DEFICIENCY HAS NO IMPACT ON THE CONTINUED SAFE OPERATION OF THE SUSQUEHANNA UNITS SINCE THE INJECTION VALVE RESPONSE TIMES REMAIN WELL WITHIN THOSE REQUIRED BY THE ACCIDENT ANALYSES CONTAINED IN THE FSAR. ALTHOUGH THE OVERALL SYSTEM RESPONSE TIME LIMITS ARE NOT AFFECTED, THE MARGIN OF SAFETY IS REDUCED. PER A PRIOR AGREEMENT BETWEEN PP&L AND NRC NRR FOR CONDITIONS OF THIS NATURE, THIS EVENT IS BEING REPORTED PURSUANT TO 10CFR50.73(A)(2)(II)(B). PP&L HAS IMPLEMENTED PRECAUTIONS TO PROVIDE ADDITIONAL ASSURANCE AGAINST THE POTENTIAL PRESSURE LOCKING SCENARIOS AND IS EVALUATING MODIFICATIONS TO RESOLVE THE PRESSURE LO ING CONCERN AS WELL AS A SCHEDULE FOR IMPLEMENTATION.

[302] SUSQUEHANNA 1 DOCKET 50-387 LER 91-014
RPS ALTERNATE POWER SUPPLY EPA BREAKER OVERVOLTAGE SETPOINTS DECLARED INOPERABLE.
EVENT DATE: 110791 REPORT DATE: 120991 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)

(NSIC 223623) AT 1900 HOURS ON NOVEMBER 7, 1991 ALL STATION ALTERNATE REACTOR PROTECTION SYSTEM (RPS) POWER SUPPLY ELECTRICAL PROTECTION ASSEMBLIES (EPA) WERE DECLARED INOPERABLE. THE CAUSE OF THE EVENT WAS DUE TO A "SHIFT" OF THE EPA OVERVOLTAGE TRIP SETPOINTS ABOVE TECHNICAL SPECIFICATION LIMITS. THE CONDITION WAS DISCOVERED WHILE INVESTIGATING AN EPA BREAKER TRIP. DURING THE INVESTIGATION, A TEST METHOD NOT PREVIOUSLY EMPLOYEED WAS USED TO CHECK THE EPA SETPOINTS. A SETPOINT "SHIFT" WAS OBSERVED WHEN THIS METHOD WAS EMPLOYEED. THE "SHIFT" WAS THE RESULT OF USING A TEST SOURCE WITH A NEAR PERFECT OUTPUT AC VOLTAGE SINE WAVE TO CALIBRATE THE SET POINTS THEN POWERING THE EPAS WITH SUPPLY TRANSFORMERS WHICH PRODUCED AN OUTPUT AC VOLTAGE SINE WAVE WITH A SLIGHT DISTORTION. THE UNDERVOLTAGE AND UNDERFREQUENCY SETPOINTS OF THE ALTERNATE EPAS REMAINED WITHIN TECHNICAL SPECIFICATION LIMITS. THE NORMAL RPS POWER SUPPLY EPA BREAKERS WERE NOT AFFECTED BY THIS CONDITION. ALL RPS DISTRIBUTION PANELS REMAINED IN SERVICE AND POWERED BY THE NORMAL SUPPLIES WHILE THE ALTERNATE EPA OVERVOLTAGE SETPOINTS WERE ADJUSTED TO WITHIN TECHNICAL SPECIFICATION LIMITS USING TRANSFORMER OUTPUT AS THE CALIBRATION SOURCE. THE EPA CALIBRATION PROCEDURE WILL BE REVISED TO ACCOUNT FOR THE "SHIFT" OF THE ALTERNATE EPA SETPOINTS.

[303] SUSQUEHANNA 1 DOCKET 50-387 LER 91-015
HIGH PRESSURE COOLANT INJECTION SYSTEM INOPERABLE DUE TO BROKEN STEAM CONTROL VALVE PILOT.
EVENT DATE: 110791 REPORT DATE: 120991 NSSS: GE TYPE: BWR
VENDOR: TERRY STEAM TURBINE COMPANY

(NSIC 223624) ON NOVEMBER 7, 1991, WITH UNIT 1 AT 100% POWER, IT WAS DETERMINED THAT THE HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI) WAS INOPERABLE IN THAT THE PRESSURE AND FLOW REQUIREMENTS OF THE QUARTERLY FLOW SURVEILLANCE COULD NOT BE MET. INSPECTION OF THE TURBINE STEAM CHEST REVEALED THAT THE HEAD OF THE #1 POPPET (PILOT VALVE) HAD BROKEN OFF. AN ENGINEERING FAILURE ANALYSIS WILL BE PERFORMED TO DETERMINE THE FAILURE MODE. THIS EVENT WAS DETERMINED TO BE REPORTABLE PER 10CFR50.73(A)(2)(V)(D) AS A CONDITION THAT ALONE COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF THE SYSTEM. HOWEVER, SUFFICIENT SAFETY MARGIN EXISTS IN THE DESIGN SUCH THAT HPCI COULD HAVE PERFORMED ITS INTENDED FUNCTION. THE BROKEN POPPET WAS REPLACED AND THE OTHER POPPETS WERE INSPECTED. THE SURVEILLANCE TEST WAS THEN SATISFACTORILY COMPLETED. ANY ADDITIONAL CORRECTIVE ACTIONS WILL BE DETERMINED DEPENDING ON THE RESULTS OF THE FAILURE ANALYSIS.

[304] SUSQUEHANNA 1 DOCKET 50-387 LER 91-016
POSTULATED APPENDIX R FIRE COULD PLACE THE PLANT OUTSIDE ITS ANALYZED DESIGN BASIS.
EVENT DATE: 112091 REPORT DATE: 122091 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)

(NSIC 223736) ON NOVEMBER 20, 1991 WITH BOTH UNITS OPERATING AT 100% POWER, IT WAS DETERMINED THAT A POSTULATED APPENDIX R FIRE IN THE CONTROL ROOM COULD PLACE THE PLANT IN A CONDITION OUTSIDE OF ITS ANALYZED DESIGN BASIS. THIS CONDITION HAS BEEN DETERMINED TO BE REPORTABLE PER 10CFR50.73(A)(2)(II)(B). THE FIRE COULD RESULT IN A HOT SHORT IN THE CONTROL CIRCUIT OF ONE OF A NUMBER OF COMPONENTS REQUIRED TO SHUT DOWN THE UNIT FROM THE REMOTE SHUTDOWN PANEL. DAMAGE TO THE COMPONENT COULD OCCUR. THE SCENARIO IS APPLICABLE TO BOTH UNITS. APPROXIMATELY NINETY VALVES ARE IMPACTED ON EACH UNIT FROM THE RHR, ESW, RHRSW AND RCIC SYSTEMS. THE POSTULATED CONDITION RELATES TO AN OVERSIGHT IN THE APPENDIX R ANALYSIS IN WHICH THE POSSIBILITY OF A MALFUNCTION OF PATH 2 SAFE SHUTDOWN COMPONENTS DURING A POSTULATED CONTROL ROOM APPENDIX R FIRE WAS NOT ANALYZED. THE SAFETY SIGNIFICANCE IS CONSIDERED MINIMAL BECAUSE OF THE LOW PROBABILITY OF OCCURRENCE AS WELL AS THE FACT THAT BACKUP SYSTEMS EXIST THAT ARE ABLE TO BRING THE PLANT TO A SUCCESSFUL SAFE SHUTDOWN, FIRE DETECTION AND SUPPRESSION SYSTEMS ARE OPERABLE IN THE CONTROL ROOM, OPERATIONS CONTROL ROOM PERSONNEL HAVE BEEN BRIEFED ON THE SCENARIO AND ARE QUALIFIED FIRE WATCH/FIRE BRIGADE PERSONNEL HAVING ACCESS TO PORTABLE FIRE FIGHTING EQUIPMENT IN THE CONTROL ROOM.

[305] SUSQUEHANNA 1 DOCKET 50-387 LER 91-017
UNPLANNED ESF ACTUATION WHEN RADIATION MONITOR POWER INTERRUPTED.
EVENT DATE: 112591 REPORT DATE: 121991 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)

(NSIC 223680) ON NOVEMBER 25, 1991 WITH UNIT 1 AT 100% POWER AND UNIT 2 AT 92% POWER AN UNPLANNED ESF ACTUATION OCCURRED WHEN A FUSE BLEW IN THE POWER SUPPLY TO RADIATION MONITORS SERVING THE ZONE III VENTILATION, STANDBY GAS TREATMENT, AND CONTROL ROOM EMERGENCY OUTSIDE AIR SUPPLY SYSTEMS. THE LOSS OF POWER CAUSED A ZONE III ISOLATION AND AUTO-START OF THE SGTS & CREOASS SYSTEMS. THE BLOWN FUSE WAS REPLACED AND ALL SYSTEMS WERE RETURNED TO NORMAL OPERATION. THE CAUSE OF THE BLOWN FUSE WAS ATTRIBUTED TO RANDOM FAILURE. THIS EVENT WAS DETERMINED TO BE REPORTABLE PER 10CFR50.73(A)(2)(IV), AS AN UNPLANNED ESF ACTUATION. THERE WERE NO SAFETY CONSEQUENCES OR COMPROMISES AS A RESULT OF THIS EVENT. ALL APPROPRIATE ELECTRICAL PARAMETERS WERE CHECKED FOLLOWING FUSE REPLACEMENT.

[306] SUSQUEHANNA 2 DOCKET 50-388 LER 91-014
MAIN STEAM ISOLATION VALVE CLOSURE CHANNEL RESPONSE TIME EXCEEDED TECHNICAL SPECIFICATION REQUIREMENTS.
EVENT DATE: 121291 REPORT DATE: 011092 NSSS: GE TYPE: BWR

(NSIC 223777) AT 1400 HOURS ON DECEMBER 12, 1991, AFTER REVIEWING AN INDUSTRY EVENT REPORT, AN INSTRUMENTATION AND CONTROL ENGINEER DISCOVERED THAT A MAIN STEAM ISOLATION VALVE (MSIV) CLOSURE CHANNEL RESPONSE TIME HAD EXCEEDED THE TECHNICAL SPECIFICATION REQUIRED VALUE OF 60 MILLI-SECONDS (MSEC) BY 2 MSEC. THE UNIT 2 "BL" CHANNEL WAS 2 MSEC OVER TECHNICAL SPECIFICATION LIMITS FROM THE FALL OF 1989 UNTIL THE SPRING OF 1991. THE CAUSE OF THE EVENT WAS THAT RESPONSE TIME TESTING PROCEDURES DID NOT ACCOUNT FOR THE TRIP CHANNEL SENSOR (LIMIT SWITCH CONTACT) RESPONSE TIME OF 10 MSEC AS IDENTIFIED IN GENERAL ELECTRIC DESIGN DATA SHEETS. THEREFORE THE MEASURED RESPONSE TIME OF 52 MSEC WAS ACTUALLY 62 MSEC. THE REMAINING CLOSURE RESPONSE CHANNELS WERE WITHIN TECHNICAL SPECIFICATION LIMITS AND THE SYSTEM WOULD HAVE PERFORMED ITS INTENDED FUNCTION. CHANGES WERE INITIATED FOR I&C RESPONSE TIME TESTING PROCEDURES TO ENSURE THAT 10 MSEC FOR CHANNEL SENSOR RESPONSE IS INCLUDED IN THE OVERALL RPS RESPONSE TIME FOR MSIV CLOSURE AND TURBINE STOP VALVE CLOSURE. THE REASON WHY THE SENSOR RESPONSE TIME ALLOTMENT WAS NOT ACCOUNTED FOR COULD NOT BE DETERMINED. A REVIEW OF RESPONSE TIME PROCEDURES OF BOTH UNITS WAS PERFORMED TO ENSURE NO OTHER RESPONSE TIME TESTS WERE AFFECTED BY THE GENERAL ELECTRIC DESIGN DATA SHEETS.

[307] SUSQUEHANNA 2 DOCKET 50-388 LER 91-015
HIGH PRESSURE COOLANT INJECTION SYSTEM INOPERABLE WHEN STEAM SUPPLY ISOLATED DUE TO LEAK.
EVENT DATE: 121691 REPORT DATE: 011492 NSSS: GE TYPE: BWR

(NSIC 223801) ON DECEMBER 16, 1991, WITH UNIT 2 IN CONDITION 1 AT 100% POWER, A SMALL LEAK WAS DISCOVERED IN A DRAIN LINE FOR THE HPCI SYSTEM STEAM SUPPLY LINE. THE LEAK WAS ISOLATED BY CLOSING TWO DRAIN VALVES UPSTREAM OF A STEAM TRAP WHICH IS UPSTREAM OF THE LEAKAGE LOCATION. IN ADDITION, THE STEAM SUPPLY ISOLATION VALVE WAS CLOSED DUE TO THE POSSIBILITY OF CONDENSATION BUILD-UP. CLOSING THE STEAM SUPPLY VALVE CAUSED THE HPCI TO BE INOPERABLE. THE CAUSE OF THE EVENT WAS DETERMINED TO BE EROSION/CORROSION OF THE CARBON STEEL HPCI DRAIN LINE PIPING. THIS EVENT WAS DETERMINED TO REPORTABLE PER 10CFR50.73(A)(2)(V)(D), AS A CONDITION THAT ALONE COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF A SYSTEM NEEDED TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT. THE DRAIN PIPING WAS REPAIRED AT THE LEAKAGE LOCATION. THE PIPING IS INCLUDED IN THE EROSION/CORROSION CONTROL PROGRAM AND HAD BEEN IDENTIFIED AS EXPERIENCING SOME DEGRADATION. DOCUMENTS WERE ALREADY IN PLACE TO REPLACE THE SUBJECT PIPING WITH UPGRADED MATERIAL DURING THE NEXT REFUELING OUTAGE IN EACH UNIT.

[308] THREE MILE ISLAND 1 DOCKET 50-289 LER 91-007
INADVERTENT EFW ACTUATION DURING PLANT HEATUP DUE TO PERSONNEL ERROR.
EVENT DATE: 111391 REPORT DATE: 121391 NSSS: BW TYPE: PWR

(NSIC 223635) TMI-1 WAS HEATING UP ON NOVEMBER 13, 1991 FOLLOWING THE 9R REFUELING OUTAGE. A SURVEILLANCE PROCEDURE "HSPS-EFW AUTO INITIATION" HAD BEEN CONDUCTED TO DEMONSTRATE OPERABILITY OF THE EFW AUTO-INITIATION CHANNELS. THE SURVEILLANCE REQUIRES AT LEAST ONE MAIN FEEDWATER PUMP NOT BE IN THE TRIPPED CONDITION. HEAT SINK PROTECTION SYSTEM (HSPS) DEFEAT/ENABLE SWITCHES WERE LEFT IN "ENABLE" BY PROCEDURE AT THE COMPLETION OF THE TEST. LATER, WHEN PLACING THE MAIN FEEDWATER PUMP IN OPERATION FROM THE RESET CONDITION, IT WAS INTENTIONALLY TRIPPED PER PROCEDURE TO VERIFY STOP VALVE CLOSURE. THE FEEDWATER PUMP TRIP CAUSED THE HSPS TO INITIATE AN AUTOMATIC START OF THE EFW SYSTEM. THE HSPS PERFORMED AS DESIGNED. BOTH MOTOR-DRIVEN EFW PUMPS STARTED. THE STEAM-DRIVEN EFW PUMP DID NOT START BECAUSE THE STEAM SUPPLY VALVES WERE TAGGED OUT FOR MAINTENANCE. EFW WAS NOT SUPPLIED TO THE OTSGS BECAUSE THE LEVELS WERE ADEQUATE. THE SURVEILLANCE WAS PERFORMED UNDER PLANT CONDITIONS DIFFERENT FROM THOSE FOR WHICH THE PROCEDURE WAS SPECIFICALLY WRITTEN. THE ROOT CAUSE OF THE EVENT WAS PERSONNEL ERROR. AN EVALUATION SHOULD HAVE BEEN MADE TO DETERMINE IF THE FINAL STATUS OF EQUIPMENT UNDER TEST WAS APPROPRIATE FOR EXISTING PLANT CONDITIONS. THE

EVENT WILL BE REVIEWED BY ALL SHIFT FOREMEN, SHIFT SUPERVISORS AND OPERATIONS ENGINEERS.

[309] TROJAN DOCKET 50-344 LER 91-030 REV 01
UPDATE ON INADEQUATE ADMINISTRATIVE CONTROLS FOR THE HIGH ENERGY LINE BREAK EVALUATION PROGRAM RESULTS IN POTENTIAL FOR BOTH TRAINS OF SAFETY RELATED SYSTEM TO BE INOPERABLE.
EVENT DATE: 082691 REPORT DATE: 121691 NSSS: WE TYPE: PWR

(NSIC 223642) ON 8/26/91, TROJAN WAS IN MODE 5 WHEN A DETERMINATION WAS MADE THAT IF A HIGH ENERGY LINE BREAK (HELB) WERE TO OCCUR IN THE MAIN STEAM SUPPORT STRUCTURE (MSSS), IT WOULD HAVE THE POTENTIAL TO RENDER THE DIESEL AUXILIARY FEEDWATER PUMP (DAFP) INOPERABLE. A SUBSEQUENT REVIEW IDENTIFIED THE TURBINE AUXILIARY FEEDWATER PUMP (TAFP) TO HAVE A SIMILAR PROBLEM FOR A DIFFERENT HELB. THE ROOT CAUSES OF THE EVENT WERE CONCLUDED TO BE: (1) A MISMATCH EXISTED BETWEEN THE EXPERIENCE LEVEL OF THE PERSONNEL AND THE COMPLEXITY OF THE ANALYSIS, AND (2) THERE WAS A LACK OF DEDICATED ORGANIZATION AND SUPPORTING PROCEDURES TO ASSURE THAT HELB DOCUMENTATION INTERFACES WERE UNDERSTOOD, REVIEWED AND MAINTAINED. CORRECTIVE ACTIONS INCLUDE(D) MODIFICATIONS TO ELIMINATE OPERABILITY CONCERNS FOR THE DAFP AND THE TAFP, ORGANIZATIONAL CHANGES TO ADDRESS THE SHORTCOMINGS, A REVISION TO THE REVIEW METHODOLOGY FOR HELB INDUCED ENVIRONMENTAL EFFECTS AND AN ASSESSMENT OF THE ADEQUACY OF PREVIOUS HELB ANALYSES. EACH AUXILIARY FEEDWATER PUMP PROVIDES A REDUNDANT CAPABILITY TO THE OTHER AND NEITHER WOULD BE RENDERED INOPERABLE AS A RESULT OF THE SAME HELB. THIS EVENT IS CONSIDERED TO HAVE MINIMAL SAFETY SIGNIFICANCE.

[310] TROJAN DOCKET 50-344 LER 91-033
INADEQUATE TRAINING AND ADMINISTRATIVE CONTROLS RESULT IN INCORRECT MAIN STEAM SAFETY VALVE CONTROL RING SETTINGS.
EVENT DATE: 102291 REPORT DATE: 112191 NSSS: WE TYPE: PWR

(NSIC 223662) ON 10/2/91, THE TROJAN NUCLEAR PLANT WAS IN MODE 5 DURING THE 1991 REFUELING OUTAGE. AN INSPECTION OF A MAIN STEAM SAFETY VALVE (MSSV) WAS BEING DONE TO RESOLVE DISCREPANCIES BETWEEN PROCUREMENT AND DESIGN DOCUMENTS REGARDING THE VALVE'S ORIFICE SIZE. THE ORIFICE SIZE WAS IN ACCORDANCE WITH PLANT DESIGN REQUIREMENTS, HOWEVER, THE VALVE CONTROL RING SETTINGS WERE INCORRECT. THE REMAINING MSSV CONTROL RING SETTINGS WERE CHECKED. ON OCTOBER 22, 1991 IT WAS CONCLUDED THAT SIX OF THE MSSVS WERE INOPERABLE DURING THE 1990 OPERATING CYCLE. THIS CONDITION WAS DETERMINED TO BE THE CUMULATIVE RESULT OF INADEQUATE TRAINING, PROCEDURES, SUPERVISORY METHODS AND EVALUATION OF PREVIOUS INDUSTRY AND PLANT SPECIFIC EVENTS. SPECIFIC TRAINING WAS DEVELOPED AND THE MSSV CONTROL RINGS WERE SET TO THE SPECIFIED VALUES, BY QUALIFIED PERSONNEL. THE TRAINING PROGRAM FOR RELIEF AND SAFETY VALVES WILL BE UPGRADED. CORRECTIVE MAINTENANCE PROCEDURES WILL BE DEVELOPED FOR THE MSSVS AND OTHER SAFETY/RELIEF VALVES IN USE AT TROJAN. MAINTENANCE AND SURVEILLANCE ACTIVITIES WILL BE EVALUATED TO IDENTIFY AREAS WHERE DATA TRENDING OR TECHNICAL REVIEWS SHOULD BE DONE TO ENSURE TIMELY DETECTION OF PROBLEMS. A STRUCTURED ROOT CAUSE EVALUATION TO DETERMINE ACTIONS NECESSARY TO IMPROVE THE TECHNICAL ADEQUACY OF REVIEWS PERFORMED UNDER THE OPERATIONAL ASSESSMENT REVIEW PROGRAM WILL BE DONE.

[311] TROJAN DOCKET 50-344 LER 91-036
RADIATION MONITOR CONTROL SWITCH MALFUNCTION RESULTS IN CONTAINMENT VENTILATION ISOLATION DURING SETPOINT ADJUSTMENT.
EVENT DATE: 111091 REPORT DATE: 121091 NSSS: WE TYPE: PWR

(NSIC 223616) ON NOVEMBER 10, 1991, THE TROJAN NUCLEAR PLANT WAS IN MODE 5 (COLD SHUTDOWN) DURING THE 1991 REFUELING OUTAGE. CONTAINMENT PURGING WAS IN PROGRESS. AT 2242, THE OPERATORS WERE ADJUSTING THE ALARM AND TRIP SETPOINTS OF THE GAS CHANNEL OF THE CONTAINMENT PROCESS AND EFFLUENT RADIATION MONITOR (PERM 1C). AFTER THE ADJUSTMENT, THE SETPOINTS WERE CHECKED. WHEN THE HIGH ALARM SETPOINT WAS CHECKED, A HIGH RADIATION ALARM WAS RECEIVED, AND A CONTAINMENT VENTILATION ISOLATION SIGNAL WAS GENERATED, RESULTING IN CLOSURE OF THE CONTAINMENT PURGE VALVES. THIS EVENT WAS CAUSED BY A FAILURE OF THE "HIGH" ALARM INDICATOR/SWITCH

TO OPERATE AS DESIGNED, PROBABLY DUE TO AGE RELATED WEAR. THIS SWITCH IS USED WHEN SETTING THE HIGH ALARM SETPOINT. WHEN THE SWITCH IS DEPRESSED, IT IS SUPPOSED TO DISABLE THE OUTPUT RELAYS AND PREVENT THE ALARM AND ISOLATION FROM OCCURRING. THE SWITCH WAS CLEANED AND REINSTALLED. DURING POSTMAINTENANCE TESTING IT WAS FOUND THAT THE SWITCH WAS STILL NOT FUNCTIONING CORRECTLY. THE SWITCH WAS THEN REPLACED AND TESTED SATISFACTORILY. THE "HIGH" ALARM PUSHBUTTON SWITCHES ON OTHER PLANT PERM'S WHICH SUPPLY ENGINEERED SAFETY FEATURE ACTUATION SIGNALS WILL BE REPLACED DURING THE NEXT REGULARLY SCHEDULED CALIBRATION ON THOSE MONITORS. THIS EVENT DID NOT AFFECT PLANT SAFETY. THIS EVENT RESULTED IN THE GENERATION OF A SPURIOUS HIGH RADIATION ALARM.

[312] TROJAN DOCKET 50-344 LER 91-035
TECHNICAL SPECIFICATION SURVEILLANCE FOR FIRE DETECTORS WAS MISSED AND FOUND NOT TO COMPLETELY IMPLEMENT TESTING REQUIREMENTS DUE TO INADEQUATE PROCEDURES AND A PERSONNEL ERROR.
EVENT DATE: 111591 REPORT DATE: 121691 NSSS: WE TYPE: PWR

(NSIC 223643) ON NOVEMBER 14, 1991 THE TROJAN NUCLEAR PLANT WAS IN MODE 5 (COLD SHUTDOWN) WHEN IT WAS DISCOVERED, DURING A QUALITY ASSURANCE AUDIT OF FIRE PROTECTION ACTIVITIES, THAT A CONTROL BUILDING SWITCHGEAR ROOM SMOKE EXHAUST AND SMOKE DETECTION SYSTEM SURVEILLANCE HAD NOT BEEN PERFORMED WITHIN ITS REQUIRED SURVEILLANCE INTERVAL. IN ADDITION, THE SURVEILLANCE TEST DID NOT FULLY IMPLEMENT TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS FOR FIRE DETECTOR INSTRUMENTATION. THE ROOT CAUSES WERE DETERMINED TO BE AN INADEQUATE ADMINISTRATIVE PROCEDURE, AN INADEQUATE TEST PROCEDURE, AND PERSONNEL ERROR. PROCEDURE CHANGES WILL BE IMPLEMENTED AS THE CORRECTIVE ACTIONS FOR THESE EVENTS. IN ADDITION, THE SURVEILLANCE TEST WAS REVISED TO INCORPORATE FULLY THE MISSING SURVEILLANCE REQUIREMENTS. SUBSEQUENT COMPLETION OF THE SURVEILLANCE TEST INDICATED THAT THE CONTROL BUILDING SWITCHGEAR ROOM'S SMOKE DETECTORS WOULD HAVE BEEN ABLE TO PERFORM THEIR DESIGN FUNCTION. IN ADDITION, AN HOURLY FIRE PATROL WAS IN EFFECT FOR THE AREA DURING THE PERIOD OF THE EXCEEDED SURVEILLANCE INTERVAL. FOR THESE REASONS THESE EVENTS WERE CONCLUDED TO NOT BE SAFETY SIGNIFICANT.

[313] TROJAN DOCKET 50-344 LER 91-038
DROPPED FILTER RESULTS IN HIGH AIRBORNE PARTICULATE ACTIVITY INSIDE CONTAINMENT AND GENERATION OF A CONTAINMENT VENTILATION ISOLATION SIGNAL.
EVENT DATE: 112791 REPORT DATE: 122491 NSSS: WE TYPE: PWR

(NSIC 223678) ON NOVEMBER 27, 1991, THE TROJAN NUCLEAR PLANT WAS IN MODE 5 (COLD SHUTDOWN) DURING THE 1991 REFUELING OUTAGE. AT APPROXIMATELY 2243, CONTROL ROOM PERSONNEL WERE NOTIFIED THAT A CONTAMINATED AIR FILTER HAD BEEN DROPPED, INSIDE CONTAINMENT, WHILE IT WAS BEING MOVED IN PREPARATION FOR DISPOSAL. THE FILTER HAD BEEN USED TO COLLECT PARTICLES GENERATED DURING THE HONING OF STEAM GENERATOR TUBES. AT 230 2, THE "HIGH" ALARM SETPOINT OF THE CONTAINMENT RADIATION MONITORING SYSTEM, PARTICULATE CHANNEL (PERM-1A), WAS REACHED AND A CONTAINMENT VENTILATION ISOLATION SIGNAL WAS GENERATED. THE CONTAINMENT PURGE AND HYDROGEN VENT VALVES WERE ALREADY CLOSED SO NO COMPONENT MOVEMENT OCCURRED. THE FILTER WAS DROPPED WHEN IT WAS LIFTED OVER A RAILING USING A LONG HANDLED TOOL. A SIMILAR EVENT OCCURRED ON SEPTEMBER 19, 1991. CORRECTIVE ACTIONS FOLLOWING THAT EVENT DID NOT INSTALL SUFFICIENT BARRIERS TO PREVENT RECURRENCE. AN IN DEPTH FAILURE ANALYSIS WAS PERFORMED FROM GENERATION OF A CONTAMINATED FILTER THROUGH PLACING IT IN A HIGH INTEGRITY CONTAINER FOR DISPOSAL. BASED ON THE ANALYSIS, A NUMBER OF CORRECTIVE ACTIONS WERE GENERATED, INCLUDING: 1) SPECIFIC WORK INSTRUCTIONS WERE WRITTEN, 2) THE FILTER LIFTING TOOL AND ITS USE WERE MODIFIED, AND 3) PERSONNEL WERE TRAINED ON THE NEW REQUIREMENTS.

[314] TROJAN DOCKET 50-344 LER 91-039
SPURIOUS CHLORINE DETECTOR ALARM CAUSES AUTOMATIC CONTROL BUILDING VENTILATION ISOLATION.
EVENT DATE: 122691 REPORT DATE: 012792 NSSS: WE TYPE: PWR

(NSIC 223858) ON DECEMBER 26, 1991, THE TROJAN NUCLEAR PLANT WAS IN MODE 5 (COLD

SHUTDOWN) DURING THE 1991 REFUELING OUTAGE, AT 1734, THE CONTROL ROOM RECEIVED A "TOXIC GAS DAMPER ISOLATION" ALARM, INDICATING ACTUATION OF A TOXIC GAS MONITOR. UPON RECEIPT OF THE ALARM, THE OPERATORS ENTERED OFF-NORMAL INSTRUCTION 54 "HIGH TOXIC GAS CONCENTRATION" AND VERIFIED THAT THE CONTROL BUILDING VENTILATION SYSTEMS HAD AUTOMATICALLY SHUT DOWN, AS REQUIRED. THE OPERATORS ALSO DETERMINED THAT THE "B" CHLORINE MONITOR WAS THE SOURCE OF THE ALARM. AT 1736, THE TURBINE BUILDING WATCH REPORTED THAT CHLORINE WAS NOT PRESENT. THE OPERATORS RESET THE CHLORINE MONITOR AND LEFT IT BYPASSED UNTIL THE CAUSE OF THE ACTUATION WAS INVESTIGATED. THE CAUSE OF THE CHLORINE MONITOR ACTUATION COULD NOT BE DETERMINED. ON DECEMBER 31, 1991, THE SENSOR ON THE "B" TRAIN CHLORINE MONITOR WAS REPLACED AND THE MONITOR WAS RETURNED TO SERVICE. ON JANUARY 18, 1992, THE SAME MONITOR BEGAN TO OPERATE ERRATICALLY. IT HAS BEEN DECLARED INOPERABLE AND PLACED IN BYPASS. INVESTIGATION INTO THE CAUSE IS CONTINUING.

[315] TROJAN DOCKET 50-344 LER 91-037 REV 02
UPDATE ON FAILURE TO COMPREHENSIVELY TEST MANUAL ENGINEERED SAFETY FEATURES ACTUATION FUNCTIONS, PROCESS RADIATION MONITOR AND PERMISSIVE AND BLOCK FUNCTIONS DUE TO DEFICIENCIES IN SURVEILLANCE TEST PROGRAM.
EVENT DATE: 123191 REPORT DATE: 013092 NSSS: WE TYPE: PWR

(NSIC 223865) ON 12/11/91, A REVIEW OF ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE PROCEDURES WAS BEING DONE IN RESPONSE TO PREVIOUSLY IDENTIFIED SURVEILLANCE DEFICIENCIES. THE REVIEW IDENTIFIED THAT SURVEILLANCE PROCEDURES DID NOT ADEQUATELY TEST THE ESFAS FUNCTIONS ASSOCIATED WITH MANUAL INITIATIONS. IT WAS ALSO FOUND THAT CONTAINMENT PROCESS AND EFFLUENT RADIOACTIVITY MONITOR SAMPLING INSTRUMENTS WERE NOT BEING PROPERLY SURVEILLED. IT WAS FURTHER FOUND THAT THE ESFAS AND REACTOR TRIP SYSTEM INTERLOCK FUNCTIONS WERE NOT BEING TESTED PROPERLY. THESE CONDITIONS WERE THE RESULT OF INADEQUATE TRANSLATION OF TECH SPEC REQUIREMENTS INTO SURVEILLANCE TEST PROCESSES. THE ROOT CAUSE WAS JUDGED TO BE A LACK OF DETAILED UNDERSTANDING OF THE EXTENT AND DEPTH OF THE TECH SPEC SURVEILLANCE REQUIREMENTS AND PAST MANAGEMENT EXPECTATIONS FOR COMPLIANCE. PROCEDURES ARE BEING DEVELOPED AND WILL BE PERFORMED TO ADDRESS THE SPECIFIC SURVEILLANCE DEFICIENCIES. SPECIAL TRAINING ON THE TECH SPECS HAS BEEN PROVIDED TO SELECTED PERSONNEL AND INCORPORATED INTO THE TECHNICAL STAFF TRAINING PROGRAM. IMPLEMENTATION OF A CONSOLIDATED PROGRAM PLAN FOR TROJAN SURVEILLANCE PROGRAM IMPROVEMENT BEGAN IN NOVEMBER OF 1991.

[316] TURKEY POINT 3 DOCKET 50-250 LER 91-013
UNCONTAMINATED LEAKAGE FROM WASTE CONTAINER DUE TO INADEQUATE CONTAINER LOWER DOOR SEAL.
EVENT DATE: 120691 REPORT DATE: 121391 NSSS: WE TYPE: PWR

(NSIC 223631) BETWEEN DECEMBER 2 AND DECEMBER 4, 1991, TWO 20-FOOT SEA-LAND CONTAINERS WERE LOADED WITH CONTAMINATED SCAFFOLDING. SEVERAL INSPECTIONS OF THE CONTAINERS WERE PERFORMED PRIOR TO LOADING TO IDENTIFY DEFECTS WHICH COULD COMPROMISE THE CONTAINERS INTEGRITY. NO LEAKAGE WAS OBSERVED DURING ANY INSPECTION. ON DECEMBER 5, 1991, THIS SHIPMENT WAS RELEASED AND CONSIGNED TO THE QUADREX RECYCLE CENTER IN OAK RIDGE, TENNESSEE. ON DECEMBER 6, 1991, AT 0908, HEALTH PHYSICS (HP) WAS NOTIFIED BY THE CONSIGNEE THAT A SMALL AMOUNT OF LIQUID HAD LEAKED ONTO THE GROUND WHERE THE VEHICLE WAS PARKED, IN JASPER, FLORIDA. HP REQUESTED THE QUADREX RECYCLE CENTER TO IMMEDIATELY DISPATCH THEIR RADIOLOGICAL EMERGENCY RESPONSE TEAM TO TAKE RADIOLOGICAL MEASUREMENTS AND REPORT THEIR FINDINGS TO FPL, AND TO CONTROL THE LEAK AND CLEAN UP ANY CONTAMINATION FOUND. THE LEAKING FLUID WAS DETERMINED TO BE UNCONTAMINATED WATER. CONDENSATION WAS OBSERVED IN THE CONTAINER, AND THERE WAS EVIDENCE THAT SOME WATER HAD ESCAPED THROUGH THE LOWER DOOR SEAL. THIS EVENT OCCURRED BECAUSE THE LOWER DOOR SEAL OF THE CONTAINER DID NOT PREVENT LEAKAGE OF CONDENSATION. THE CONDENSATION DEVELOPED AS THE TEMPERATURE OF THE AIR AROUND THE CONTAINER DROPPED FROM OVER 80F AT TURKEY POINT, TO LESS THAN 40F IN JASPER. DOOR SEALS WILL BE CAULKED WITH A FLEXIBLE SILICON SEALANT.

[317] VERMONT YANKEE DOCKET 50-271 LER 91-006 REV 02
UPDATE ON LOSS OF "B" LOOP SHUTDOWN COOLING DUE TO PRESSURE SWITCH ACTIVATION.
EVENT DATE: 031491 REPORT DATE: 010992 NSSS: GE TYPE: BWR

(NSIC 223763) ON 03/14/91 AT 0450 HOURS, WITH REACTOR VESSEL COOLDOWN IN PROGRESS FOLLOWING A REACTOR SCRAM ON 03/13/91 (SUBJECT OF LICENSEE EVENT REPORT 91-05), AND WITH THE "B" LOOP RESIDUAL HEAT REMOVAL (RHR) SYSTEM FLUSHED AND LINED UP FOR SHUTDOWN COOLING, A GROUP 4 PRIMARY CONTAINMENT ISOLATION SIGNAL (PCIS) WAS RECEIVED DURING TWO ATTEMPTED STARTS OF THE "B" RHR PUMP. THE GROUP 4 PRIMARY CONTAINMENT ISOLATION SIGNAL RESULTED IN A TRIP OF THE "B" RHR PUMP AND CLOSURE OF SHUTDOWN COOLING SUCTION ISOLATION VALVES. THE SECOND FAILED PUMP START ATTEMPT WAS INITIATED AT 0455 HOURS. AT 0504 HOURS, AFTER ISOLATIONS WERE RESET A SECOND TIME, THE "D" RHR PUMP WAS SUCCESSFULLY STARTED ON THE "B" RHR LOOP. SHUTDOWN COOLING REMAINED IN OPERATION ON THE "B" RHR LOOP FOR THE DURATION OF THE SHUTDOWN. THE REACTOR WAS RETURNED TO CRITICAL ON 03/18/91 AT 0055 HOURS. ROOT CAUSE INVESTIGATIONS AND TESTING COMPLETED TO DATE HAVE IDENTIFIED A SECTION OF RHR PIPING BETWEEN THE INBOARD DISCHARGE CHECK VALVE AND OUTBOARD INJECTION VALVE WHICH REMAINS IN A DEPRESSURIZED STATE AT THE COMPLETION OF LOOP FLUSHING AND IS RESPONSIBLE FOR THE PRESSURE SURGES OCCURRING IN THE "B" RHR LOOP. PROCEDURE CHANGES ARE BEING IMPLEMENTED WHICH WILL REPRESSURIZE THE SECTION OF PIPING PRIOR TO SYSTEM STARTUP AND ADDITIONAL TESTING PLANNED FOR THE UPCOMING 1992 REFUELING OUTAGE TO VERIFY ADEQUACY OF THE CORRECTIVE ACTIONS.

[318] VERMONT YANKEE DOCKET 50-271 LER 91-008 REV 01
UPDATE ON POTENTIAL ENVIRONMENTAL CONDITIONS NOT PREVIOUSLY EVALUATED AS A RESULT OF OMISSION FROM ORIGINAL LINE BREAK ANALYSIS.
EVENT DATE: 032591 REPORT DATE: 121391 NSSS: GE TYPE: BWR

(NSIC 223632) ON 3/25/91, AT APPROXIMATELY 1805 HOURS WITH THE PLANT AT 100% POWER, IT WAS DETERMINED THAT FAILURE OF HOUSE HEATING STEAM LINES HAD NOT BEEN ADDRESSED IN PREVIOUS HIGH ENERGY LINE BREAK (HELB) ANALYSES. THIS COULD CREATE ENVIRONMENTAL CONDITIONS OUTSIDE THOSE PREVIOUSLY EVALUATED. THIS WAS IDENTIFIED AS A RESULT OF INVESTIGATION RELATIVE TO USNRC INFO. NOTICE 90-053 THAT NOTIFIED LICENSEES OF SIMILAR CONDITIONS AT OTHER FACILITIES. TWO JUSTIFICATIONS FOR CONTINUED OPERATION (JCO) WERE PREPARED AND APPROVED, THE FIRST ON 4/1/91, SUPPORTING CONTINUED OPERATION OF THE HEATING SYSTEM FOR THE REMAINDER OF THE 1990/91 HEATING SEASON, AND THE SECOND ON 10/9/91 SUPPORTING OPERATION OF THE HEATING SYSTEM FOR THE 1991/92 HEATING SEASON. THE HOUSE HEATING SYSTEM WAS ERRONEOUSLY OMITTED FROM HELB ANALYSES PERFORMED IN 1973 AND 1974. A DESIGN MODIFICATION, PROCEDURE REVISIONS AND REVIEWS OF OTHER SYSTEMS HAVE BEEN COMPLETED AND REVISION OF DESIGN BASIS DOCUMENTATION TO ADDRESS THE HOUSE HEATING STEAM LINES AND RELATED ANALYSES IS ONGOING.

[319] VERMONT YANKEE DOCKET 50-271 LER 92-001
SPURIOUS HPCI SUCTION TRANSFER DUE TO UNKNOWN CAUSE.
EVENT DATE: 121991 REPORT DATE: 022092 NSSS: GE TYPE: BWR

(NSIC 224029) ON 12/19/91 AT 1010 HOURS, DURING NORMAL OPERATION WITH REACTOR POWER AT 100%, A SPURIOUS HIGH PRESSURE COOLANT INJECTION (HPCI) (*BJ) CONDENSATE STORAGE TANK (CST) LOW LEVEL ALARM WAS RECEIVED AND SUCTION TO HPCI TRANSFERRED FROM CST TO THE TORUS. APPROXIMATELY 15 MINUTES LATER AT 1025 HOURS, AFTER REVIEWING PLANT PARAMETERS (PROPER CST LEVEL), OPERATORS RETURNED HP SUCTION BACK TO THE CST. THE ROOT CAUSE OF THE EVENT IS UNKNOWN. A FUNCTIONAL TEST WAS PERFORMED ON THE APPLICABLE HPCI SUCTION TRANSFER LOGIC. THE TESTS DID NOT REVEAL ANY FAILED COMPONENTS, CALIBRATION COMPONENTS, OR ANY OTHER ABNORMALITIES WHICH WOULD ALLOW FOR ROOT CAUSE ANALYSIS. THE EVENT WAS DETERMINED TO BE NOT-REPORTABLE BY VERMONT YANKEE. HOWEVER, AS A RESULT SUBSEQUENT DISCUSSIONS WITH THE ON-SITE RESIDENT INSPECTOR THIS EVENT IS BEING VOLUNTARILY CONSIDERED AS AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION AND REPORTED UNDER 50.73(A)(2)(IV).

[320] VERMONT YANKEE DOCKET 50-271 LER 91-016
FAILURE TO PERFORM CORRECT DAILY INSTRUMENT CHECKS DUE TO TECH SPEC HUMAN FACTORS WEAKNESS.
EVENT DATE: 122091 REPORT DATE: 011792 NSSS: GE TYPE: BWR

(NSIC 223787) ON DECEMBER 20, 1991 IT WAS DETERMINED THAT DAILY INSTRUMENT CHECKS FOR 3 OF THE 15 POST ACCIDENT PARAMETERS (EIIS=IP) WERE PERFORMED ON INSTRUMENTS DIFFERENT THAN LISTED IN THE APPLICABLE TECHNICAL SPECIFICATION (TS) TABLE. FOR EACH PARAMETER THE INSTRUMENT CHECK WAS BEING PERFORMED ON ANOTHER CONTROL ROOM INDICATION OF THE PARAMETER WHICH ORIGINATED FROM THE SAME SENSOR INSTRUMENTATION. THEREFORE, A VALID INDICATION OF PARAMETER AVAILABILITY AND EQUIPMENT OPERABILITY, WAS DETERMINED DAILY, HOWEVER THE SPECIFIC INSTRUMENT IN TS WAS NOT BEING CHECKED AND LOGGED ON THE CONTROL ROOM OPERATORS ROUNDS SHEET. THE CAUSE OF THE EVENT WAS DETERMINED TO BE A HUMAN FACTORS WEAKNESS IN THE CONTENT AND ARRANGEMENT OF TWO TS TABLES. IMMEDIATE CORRECTIVE ACTIONS CONSISTED OF REVISION OF THE CONTROL ROOM OPERATORS ROUNDS HEAT AND REVIEWING ALL DAILY INSTRUMENT CHECKS TO VERIFY THAT NO SIMILAR CONDITIONS EXISTED. REVISION TO THE PROCEDURE GOVERNING OPERATOR ROUNDS WILL BE COMPLETED TO REFERENCE APPLICABLES TS REQUIREMENTS.

[321] VERMONT YANKEE DOCKET 50-271 LER 92-003
ADVANCED OFF-GAS RUPTURE DISK TEMPORARY REPAIR NOT WITHIN SYSTEM DESIGN BASES.
EVENT DATE: 011392 REPORT DATE: 021392 NSSS: GE TYPE: BWR

(NSIC 224031) FROM 01/13/92 AT 1335 UNTIL 01/16/92 AT 2006, WITH THE REACTOR OPERATING AT 100% POWER, THE STEAM JET AIR EJECTORS (SJAE) (EIIS=SH) WERE OPERATED WITH THE ADVANCED OFF-GAS (AOG) (EIIS=WF) SYSTEM INLET RUPTURE DISK RUPTURED. THE RUPTURE DISC RUPTURED ON 01/13/92 AT 1330 AFTER BOTH AOG RECOMBINER INLET VALVES WERE ISOLATED AS A RESULT OF A DEFICIENT PREVENTIVE MAINTENANCE PROCEDURE STEP. ON 1/16/92 AT 0515 THE TURBINE WAS TAKEN OFF-LINE AND THE RUPTURE DISC WAS REPLACED WHEN IT WAS IDENTIFIED THAT NEITHER A TEMPORARY OR ON-LINE REPAIR COULD BE MADE TO THE RUPTURE DISK. A RELEASE OF NOBLE GASSES AND ASSOCIATED PARTICULATES OCCURRED AS A RESULT OF THE AOG RECOMBINERS ISOLATING AND THE RUPTURE DISC BURSTING. THE RELEASE WAS EVALUATED AND FOUND TO NOT EXCEED ANY LIMITS. NO RELEASE OCCURRED WHILE THE SYSTEM WAS OPERATED WITH THE RUPTURE DISC RUPTURED. THE AOG INLET PIPE WAS OPERATED AT A VACUUM WITH INLEAKAGE BEING CONTROLLED WITH A METAL COVER. THE ROOT CAUSE OF THE RECOMBINER INLET VALVE CLOSURE WAS A PROCEDURE DEFICIENCY. THE PREVENTIVE MAINTENANCE PROCEDURE IS BEING REVISED TO PREVENT RECURRENCE. IN ADDITION, A TASK TEAM IS EVALUATING THE DISC RUPTURE.

[322] VERMONT YANKEE DOCKET 50-271 LER 92-002
MISSED STANDBY LIQUID CONTROL TANK BORON CONCENTRATION SURVEILLANCE DUE TO A PERSONNEL ERROR WHEN TRANSFERRING DUE DATES TO THE 1992 SCHEDULE.
EVENT DATE: 011592 REPORT DATE: 021292 NSSS: GE TYPE: BWR

(NSIC 224030) ON 1/15/92, WITH THE REACTOR OPERATING AT 100% POWER THE SURVEILLANCE TEST COORDINATOR COVERED THAT THE BORON CONCENTRATION CHECK OF THE STANDBY LIQUID CONTROL (SLC) (EIIS=BR) TANK HAD NOT BEEN PERFORMED WITHIN THE TECHNICAL SPECIFICATIONS TIME LIMITS. SUBSEQUENT TESTS SHOWED THAT THE CONCENTRATION WAS NORMAL. THE IMMEDIATE CAUSE FOR THIS EVENT WAS AN ERROR IN THE PUBLISHED SURVEILLANCE SCHEDULE. THE CAUSE OF THIS EVENT WAS A PERSONNEL ERROR DUE TO INATTENTION TO DETAIL. WHEN THE DATA TRANSFERRED FROM THE 1991 SCHEDULE TO THE 1992 SCHEDULE THE TEST IN QUESTION, WHICH HAD MOVED BACK ONE WEEK IN 1991, WAS INADVERTENTLY LEFT IN THE ORIGINALLY SCHEDULED WEEK. DISTRIBUTING CAUSE WAS AN INADEQUATE PROCEDURE IN THAT THE PROCEDURE DOES NOT INCLUDE SPECIFIC INSTRUCTIONS REGARDING THE EXTENT OF THE INDEPENDENT REVIEW REQUIRED. THE TEST SCHEDULE HAS BEEN REVIEWED, AND NO SIMILAR PROBLEMS WERE IDENTIFIED. ADDITIONALLY, PROCEDURE AP 4000, SURVEILLANCE TESTING CONTROL WILL BE REVISED TO SPECIFICALLY STATE THE DEPTH OF THE REVIEW REQUIRED FOR THE SCHEDULE.

[323] VOGTLE 1 DOCKET 50-424 LER 91-015
VALVE MANUFACTURING DEFECT LEADS TO CONTAINMENT ISOLATION VALVE FAILING OPEN.
EVENT DATE: 090591 REPORT DATE: 122391 NSSS: WE TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 223741) ON AUGUST 29, 1991, A LOCAL LEAK RATE TEST (LLRT) OF A CONTAINMENT ISOLATION VALVE (CIV), SERVICE AIR SYSTEM CHECK VALVE 12401U4034, WAS CONDUCTED. LEAKAGE LIMITS WERE EXCEEDED, THE TECHNICAL SPECIFICATION (TS) LIMITING CONDITION FOR OPERATION (LCO) WAS ENTERED, AND A DEFICIENCY CARD (DC) WAS INITIATED ON SEPTEMBER 5, 1991, AN INVESTIGATION INTO THE CAUSE OF THE LEAKAGE FOUND THE DISK OF THIS CHECK VALVE STUCK FULLY OPEN. A CASTING MARK ON THE VALVE HINGE ENGAGED THE HINGE SUPPORT WHEN THE DISK WAS IN THE FULL OPEN POSITION AND WOULD NOT ALLOW THE DISK TO RETURN TO THE CLOSED POSITION. ON DECEMBER 6, 1991, A FINAL REVIEW OF THE DC FOR ADEQUACY OF INVESTIGATION AND CORRECTIVE ACTIONS IDENTIFIED THAT 12401U4034 MAY HAVE BEEN STUCK IN THE FULL OPEN POSITION FOR A PERIOD OF TIME LONGER THAN THAT ALLOWED BY THE TS LCO ACTION STATEMENT REQUIREMENTS. HOWEVER, SINCE FIRM EVIDENCE FOR THE TIME THAT THE VALVE BECAME STUCK OPEN DOES NOT EXIST, THIS REPORT IS BEING SUBMITTED AS A VOLUNTARY LER. THE CAUSE OF THIS EVENT WAS THE MANUFACTURER'S CASTING MARK ON THE VALVE HINGE, WHICH CAUSED THE DISK TO BIND AND REMAIN IN THE OPEN POSITION. THE CASTING MARK WAS REMOVED, LLRT TESTING WAS COMPLETED SATISFACTORILY, AND 12401U4034 WAS RETURNED TO SERVICE.

[324] VOGTLE 1 DOCKET 50-424 LER 91-011
AUXILIARY FEEDWATER ACTUATION WHILE PREPARING FOR TESTING.
EVENT DATE: 111891 REPORT DATE: 121691 NSSS: WE TYPE: PWR

(NSIC 223648) ON NOVEMBER 18, 1991, PERSONNEL WERE PREPARING TO TEST THE TURBINE DRIVEN AUXILIARY FEEDWATER (AFW) PUMP FLOW RATE. ALTHOUGH STEAM GENERATOR (SG) WATER LEVELS ARE NORMALLY MAINTAINED AT 60 TO 70 PERCENT OF THE NARROW RANGE BAND, THIS PROCEDURE REQUIRES LEVELS TO BE MAINTAINED IN THE LOWER END OF THE NARROW RANGE BAND. AT APPROXIMATELY 0158 EST, THE STEAM GENERATOR WATER LEVEL CONTROL OPERATOR STOPPED FEEDWATER FLOW FOR AFW VALVE STROKE TESTING. WHEN THE OPERATOR NEXT CHECKED SG WATER LEVEL, HE FOUND THAT THE SG 1 AND SG 4 WATER LEVELS WERE NEAR 40 PERCENT OF THE NARROW RANGE BAND. HE IMMEDIATELY REINITIATED FEEDWATER FLOW TO SG1. MOMENTS LATER AT 0201 EST, AN AUTOMATIC AFW ACTUATION OCCURRED AS SG 1 REACHED THE 37.8-PERCENT, LOW-LOW LEVEL SETPOINT FOR INITIATING AN AFW ACTUATION. THE STANDBY MOTOR DRIVEN AFW PUMP STARTED, AND VALVES OPENED TO PROVIDE FULL AFW FLOW AS DESIGNED. STEAM GENERATOR WATER LEVELS WERE RESTORED, AND NORMAL OPERATIONS WERE RESUMED. THE DIRECT CAUSE OF THIS EVENT WAS A COGNITIVE PERSONNEL ERROR BY THE OPERATOR IN NOT MAINTAINING THE SG WATER LEVELS IN ACCORDANCE WITH PROCEDURE. THE OPERATOR HAS BEEN COUNSELED REGARDING THE IMPORTANCE OF ATTENTION TO DUTY AND THE NEED TO MAINTAIN CONTROL OVER WORK ACTIVITIES IN HIS AREA.

[325] VOGTLE 1 DOCKET 50-424 LER 91-012
MANUAL ACTUATION OF REACTOR PROTECTION SYSTEM DURING ROD DROP TIME TESTING.
EVENT DATE: 111891 REPORT DATE: 121691 NSSS: WE TYPE: PWR

(NSIC 223649) ON NOVEMBER 18, 1991, UNIT 1 WAS IN MODE 3 (HOT STANDBY). THE REACTOR TRIP BREAKERS WERE CLOSED, AND WITHDRAWAL OF SHUTDOWN BANK (SDB) "A" WAS IN PROGRESS PER PROCEDURE 88006-C, "ROD DROP TIME MEASUREMENT WITH ROD DROP TEST CART." AFTER WITHDRAWING SDB "A" TO 53 STEPS, IT WAS NOTICED THAT THE DIGITAL ROD POSITION INDICATION (DRPI) FOR SDB "A" ROD D-2 HAD STOPPED MOVING. SHUTDOWN BANK "A" WAS FURTHER WITHDRAWN, AND AT GROUP STEP COUNTER READINGS OF 59 AND 58, DRPI URGENT ALARMS WERE RECEIVED AND THE ROD AT BOTTOM LED ILLUMINATED FOR ROD D-2. THE OPERATORS THOUGHT THEY HAD ALSO OBSERVED INDICATION OF ROD D-2 DROPPING INTO THE CORE. THE ABNORMAL OPERATING PROCEDURE (AOP) FOR ROD CONTROL SYSTEM MALFUNCTIONS WAS ENTERED, AND AT 1010 EST, A REACTOR TRIP WAS CONSERVATIVELY INITIATED AS DISCUSSED IN EVOLUTION PREBRIEFINGS AND AS DIRECTED BY THE AOP. IT WAS VERIFIED THAT THE REACTOR TRIP BREAKERS OPENED PROPERLY, AND THE SDB "A" RODS FULLY INSERTED INTO THE CORE. SUBSEQUENT TROUBLESHOOTING INDICATED ROD D-2 DID NOT ACTUALLY DROP INTO THE CORE PRIOR TO INITIATION OF THE REACTOR TRIP. RATHER, A DRPI DATA A PROBLEM WAS INDICATED TO EXIST. WHILE THE EXACT CAUSE OF THIS PROBLEM WAS NOT DETERMINED, THE OPERABILITY OF THE DRPI SYSTEM PER TECHNICAL

SPECIFICATION REQUIREMENTS WAS DEMONSTRATED AND A REACTOR STARTUP WAS SUCCESSFULLY COMPLETED ON NOVEMBER 19, 1991.

[326] VOGTLE 1 DOCKET 50-424 LER 91-014
FUEL HANDLING BUILDING ISOLATION FROM RADIATION MONITOR SIGNAL.
EVENT DATE: 112691 REPORT DATE: 122091 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: VOGTLE 2 (PWR)

(NSIC 223687) ON NOVEMBER 26, 1991 AT 1016 EST, A FUEL HANDLING BUILDING (FHB) ISOLATION OCCURRED FROM RADIATION MONITORS ARE-2532A AND ARE-2532B. CONTROL ROOM PERSONNEL VERIFIED THAT EQUIPMENT ACTUATED PROPERLY. CHEMISTRY PERSONNEL REPORTED THAT THE ARE-2532A AND ARE-2532B DATA PROCESSING MODULE (DPM) HAD UNDERGONE A POWER RESET AND THAT NO ABNORMAL RADIATION CONDITION EXISTED. THE ARE-2532A AND ARE-2532B ENGINEERED SAFETY FEATURE (ESF) ACTUATION SIGNAL WAS BLOCKED AND NORMAL FHB VENTILATION WAS RESTORED AT 1035 EST. THE DIRECT CAUSE OF THIS EVENT WAS A LOSS OF POWER TO THE RADIATION MONITORS WHICH CAUSED THEM TO FAIL TO THE "SAFE" CONDITION, RESULTING IN A FHB ISOLATION. THE ROOT CAUSE OF THE LOSS OF POWER IS UNKNOWN. FOLLOWING TESTING AND TROUBLESHOOTING, THE RADIATION MONITORS WERE RETURNED TO SERVICE.

[327] VOGTLE 1 DOCKET 50-424 LER 91-016
FAILURE TO COMPLETE TECHNICAL SPECIFICATION REQUIRED ACTION FOR BATTERY CELL LOW VOLTAGE.
EVENT DATE: 122391 REPORT DATE: 012192 NSSS: WE TYPE: PWR
VENDOR: C & D BATTERIES, DIV OF ELTRA CORP.

(NSIC 223839) ON 12-26-91, A MAINTENANCE SUPERVISOR AND PLANT ENGINEERING PERSONNEL WERE REVIEWING RECORDED DATA FOR A CLASS 1E BATTERY CELL WHICH WAS BEING SINGLE-CELL CHARGED WHEN IT WAS NOTICED THAT THE CELL HAD DISPLAYED ABNORMAL BEHAVIOR. PER THE RECORDED DATA, AFTER THE CELL WAS PLACED ON SINGLE-CELL CHARGE ON 12-23-91, THE CELL FLOAT VOLTAGE HAD DROPPED BELOW THE TECHNICAL SPECIFICATION (TS) ALLOWABLE VALUE FOR APPROXIMATELY 22 HOURS. PER TS REQUIREMENTS, IF FLOAT VOLTAGE FOR A CONNECTED CELL IS FOUND TO BE BELOW THE ALLOWABLE VALUE, THEN BATTERY MUST BE DECLARED INOPERABLE. SUBSEQUENTLY, ENGINEERING REVIEW DETERMINED THAT A TEMPORARY INTERNAL SHORT HAD PROBABLY EXISTED IN THE CELL AND CONSEQUENTLY, AN INOPERABLE CONDITION SHOULD HAVE BEEN CONSIDERED TO EXIST. THE ROOT CAUSE OF THE EVENT WAS PROCEDURE INADEQUACY. THE PROCEDURE, WHICH PROVIDES INSTRUCTIONS FOR SINGLE-CELL CHARGING, DID NOT PROVIDE A PRECAUTION THAT TS REQUIREMENTS WOULD APPLY FOR LOW CELL VOLTAGE READINGS OBTAINED WHILE SINGLE-CELL CHARGING. IN ADDITION TO REVISING THE PROCEDURE, CORRECTIVE ACTION HAS BEEN TAKEN TO JUMPER OUT THE CELL DUE TO FURTHER UNACCEPTABLE VOLTAGE READINGS. A NEW CELL IS EXPECTED TO BE INSTALLED BY 2-15-92.

[328] WATERFORD 3 DOCKET 50-382 LER 91-008 REV 02
UPDATE ON REACTOR COOLANT SYSTEM LEAKAGE IN EXCESS OF TECHNICAL SPECIFICATIONS DUE TO CHECK VALVE LEAKAGE.
EVENT DATE: 051791 REPORT DATE: 012092 NSSS: CE TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 223837) AT 1335 HOURS ON MAY 17, 1991, WITH WATERFORD STEAM ELECTRIC STATION UNIT 3 IN HOT SHUTDOWN (MODE 4), AN UNUSUAL EVENT WAS DECLARED DUE TO REACTOR COOLANT SYSTEM (RCS) LEAKAGE FROM PRESSURIZER SPRAY CHECK VALVE RC-303, IN EXCESS OF TECHNICAL SPECIFICATION (TS) REQUIREMENTS. TS 3.4.5.2(D) REQUIRES RCS LEAKAGE TO BE LESS THAN 10 GALLON PER MINUTE (GPM). THE LEAKAGE FROM RC-303 WAS CALCULATED TO BE APPROXIMATELY 20 (GPM), REQUIRING PLANT SHUTDOWN TO COLD SHUTDOWN (MODES) PER TS'S. DURING THIS EVENT, TS 3.0.3 WAS ENTERED DUE TO CLOSING THE SAFETY INJECTION TANK (SIT) OUTLET ISOLATION VALVES. THE ROOT CAUSE OF THIS EVENT IS UNDERDEVELOPED TRAINING ON PRESSURE SEAL VALVE INSTALLATION. CORRECTIVE ACTIONS ARE TO TRAIN MECHANICAL MAINTENANCE PERSONNEL ON PRESSURE SEAL VALVE INSTALLATION AND TO REVISE THE PRESSURE SEAL VALVE TECHNICAL MANUAL TO INCLUDE MORE DETAILED INFORMATION ON PRESSURE SEAL GASKETS, AFTER RC-303 BEGAN TO LEAK, RCS PRESSURE AND TEMPERATURE WAS REDUCED, CONTAINMENT WAS EVACUATED, AND RC-303 WAS REPAIRED; THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT JEOPARDIZED.

[329] WATERFORD 3 DOCKET 50-382 LER 91-022
INADVERTENT ENGINEERED SAFETY FEATURE ACTUATIONS DUE TO PLANT PROTECTION SYSTEM TEST CIRCUIT MALFUNCTION.
EVENT DATE: 111791 REPORT DATE: 121791 NSSS: CE TYPE: PWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)
FISHER CONTROLS CO.

(NSIC 223645) AT 0324 HOURS ON NOVEMBER 17, 1991, WHILE PERFORMING OP-903-107, PLANT PROTECTION SYSTEM FUNCTIONAL TEST, THE PLANT RECEIVED A SAFETY INJECTION ACTUATION SIGNAL (SIAS), CONTAINMENT ISOLATION ACTUATION SIGNAL (CIAS), AND MAIN STEAM ISOLATION SIGNAL (MSIS). SUBSEQUENTLY, A REACTOR TRIP OCCURRED ON LOSS OF POWER TO THE CONTROL ELEMENT DRIVE MECHANISM MOTOR GENERATOR SETS DUE TO THE SIAS. OPERATIONS CONTROL ROOM PERSONNEL ENTERED THE EMERGENCY OPERATING PROCEDURES AND CARRIED OUT THE IMMEDIATE OPERATOR ACTIONS. RECOVERY WAS PERFORMED IN ACCORDANCE WITH OP-902-006, LOSS OF MAIN FEEDWATER PROCEDURE. INSPECTION OF THE TEST CABINET REVEALED A LOOSE WIRE CONNECTION IN THE CD MATRIX TEST CIRCUITRY. THE LOOSE WIRE CONNECTION WAS TIGHTENED AND A SYSTEM RETEST WAS PERFORMED SATISFACTORILY. THE CAUSE OF THIS EVENT WAS A TEST CIRCUIT MALFUNCTION WHICH WAS CAUSED BY A LOOSE CONNECTION ON TB701 TERMINAL 2. THE CAUSE OF THE LOOSE CONNECTION COULD NOT BE POSITIVELY DETERMINED AND MAY HAVE EXISTED FOR SOME TIME. WATERFORD 3 IS IMPLEMENTING A DESIGN CHANGE TO INSTALL TEST CIRCUIT ENHANCEMENTS.

[330] WATERFORD 3 DOCKET 50-382 LER 91-023
VALVE OUT OF POSITION DUE TO INADEQUATE POSITION INDICATION LABELING.
EVENT DATE: 122091 REPORT DATE: 012092 NSSS: CE TYPE: PWR

(NSIC 223836) AT 2203 HOURS ON DECEMBER 20, 1991, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% POWER WHEN A NUCLEAR AUXILIARY OPERATOR (NAO) DISCOVERED COMPONENT COOLING VALVE CC-304 A, CROSS CONNECT INLET TO AB CHILLER, OPEN. THIS VALVE MISPOSITION WAS IDENTIFIED WHILE VERIFYING POSITIONS OF ACCESSIBLE LOCKED VALVES PER OP-100-009, CONTROL OF VALVES AND BREAKERS. INVESTIGATION DETERMINED THAT THE VALVE WAS OPERATED ON DECEMBER 1, 1991, AT 1528 HOURS WHILE REPLACING THE B CHILLER WITH THE AB CHILLER, REQUIRING CC-304 B OPEN AND CC-304 A CLOSED. WITH CC-304 A AND B OPEN, COMPONENT COOLING WATER (CCW) LOOPS A AND B WERE CROSS CONNECTED AT THE INLET OF THE CHILLERS. THIS CROSS CONNECTION DEFEATS THE TWO REDUNDANT, SEPARATE TRAIN DESIGN AND RESULTS IN OPERATION OF THE SYSTEM OUTSIDE DESIGN BASIS. THE ROOT CAUSE OF THE EVENT WAS INADEQUATE POSITION INDICATION LABELING OF CC-304 A. THE INADEQUATE LABELING CAUSED A MISINTERPRETATION OF THE VALVE ACTUAL POSITION, ALLOWING THE VALVE TO BE MISPOSITIONED. CONTRIBUTING TO THE EVENT WAS THE INAPPROPRIATE ACTION TAKEN BY THREE OPERATORS WHO INTENDED ON TWO SEPARATE OCCASIONS TO CORRECTLY POSITION OR CHECK THE POSITION OF CC-304 A.

[331] WOLF CREEK 1 DOCKET 50-482 LER 89-021
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT NOT SATISFIED PRIOR TO EQUIPMENT BEING RETURNED TO SERVICE BECAUSE OF A PROCEDURAL INADEQUACY.
EVENT DATE: 111589 REPORT DATE: 122791 NSSS: WE TYPE: PWR

(NSIC 223747) ON NOVEMBER 27, 1991, FOLLOWING A REVIEW OF THE POSITION INDICATION TEST SURVEILLANCE SCHEDULING PROGRAM, IT WAS DISCOVERED THAT SURVEILLANCE TEST PROCEDURE STS BM-202, "STEAM GENERATOR B/D INSERVICE VALVE TEST", HAD NOT BEEN PERFORMED SATISFACTORILY IN NOVEMBER OF 1989 ON STEAM GENERATOR "A" SAMPLE ISOLATION VALVES BM HV019 AND BM HV035 WITHIN THE REQUIRED TIME INTERVAL, THEREBY VIOLATING TECHNICAL SPECIFICATION 4.0.2. ON NOVEMBER 27, 1991 AT 1646 CST SURVEILLANCE PROCEDURE STS BM-202 WAS PERFORMED SATISFACTORILY. THIS EVENT RESULTED FROM AN INADEQUACY IN THE PROCEDURES GOVERNING THE EOL AND THE WORK REQUEST PROCESS. WHEN VALVES BM HV019 AND BM HV035 WERE ENTERED ON THE EQUIPMENT OUT-OF-SERVICE LOG, THE REQUIREMENT TO REPERFORM STS BM-202 WAS NOT INCLUDED. ALSO, THE REQUIREMENT TO REPERFORM STS BM-202 WAS NOT INCLUDED ON THE WORK REQUEST WHICH WAS INITIATED TO INVESTIGATE THE INDICATED LEAKAGE THROUGH THE VALVES. CORRECTIVE ACTIONS HAVE ALREADY BEEN COMPLETED WHICH WILL PRECLUDE FUTURE OCCURRENCES.

[332] WOLF CREEK 1 DOCKET 50-482 LER 91-017 REV 01
UPDATE ON RESOLUTION OF POTENTIAL INOPERABILITY OF FUEL BUILDING AND AUXILIARY BUILDING EMERGENCY EXHAUST SYSTEMS.
EVENT DATE: 092091 REPORT DATE: 011592 NSSS: WE TYPE: PWR

(NSIC 223844) ON SEPTEMBER 21, 1991, AT 0748 CDT, SECURITY PERSONNEL NOTIFIED CONTROL ROOM OPERATORS THAT A FUEL BUILDING/AUXILIARY BUILDING PRESSURE BOUNDARY DOOR WAS FOUND PROPPED OPEN WITHOUT A CONTINUOUS WATCH HAVING BEEN ESTABLISHED. SUBSEQUENT EVALUATION CONCLUDED THAT WITH THIS DOOR OPEN, THE FUEL BUILDING AND AUXILIARY BUILDING EMERGENCY EXHAUST SYSTEMS (FFS) MAY NOT HAVE BEEN ABLE TO MAINTAIN A NEGATIVE PRESSURE GREATER THAN OR EQUAL TO .25 INCH WATER GAUGE INDEPENDENTLY AS REQUIRED BY TECHNICAL SPECIFICATION 3.7.7 AND 3.9.13 FOR FUEL BUILDING AND AUXILIARY BUILDING OPERABILITY. SUBSEQUENT TESTING HAS SHOWN THAT WITH THIS DOOR OPEN, THE FUEL BUILDING AND AUXILIARY BUILDING EES WERE ABLE TO MAINTAIN A NEGATIVE PRESSURE GREATER THAN OR EQUAL TO .25 INCH WATER GAUGE INDEPENDENTLY AS REQUIRED BY TECHNICAL SPECIFICATIONS. THE ROOT CAUSE OF THIS EVENT IS FAILURE TO FOLLOW PROCEDURES BY THE CRAFTSMEN WHO PROPPED THE DOOR OPEN TO MOVE SCAFFOLDING THROUGH THE DOOR. TO PREVENT RECURRENCE OF THIS EVENT, THE MANAGER MODIFICATIONS HAS DISCUSSED THIS EVENT AND ITS RAMIFICATIONS WITH HIS CRAFTSMEN'S SUPERVISION.

[333] WOLF CREEK 1 DOCKET 50-482 LER 91-022
FAILURE TO VERIFY THAT THE EMERGENCY DIESEL GENERATORS ARE CAPABLE OF REJECTING A LOAD OF 1352 KW.
EVENT DATE: 111291 REPORT DATE: 121291 NSSS: WE TYPE: PWR

(NSIC 223629) ON 11/12/91, AT 2003 CST WITH THE UNIT IN MODE 6, REFUELING, THE CONTROL ROOM WAS INFORMED THAT THE REQUIREMENTS OF TECH SPEC 4.8.1.1.2.G.2 COULD NOT BE SATISIFED FOR EMERGENCY DIESEL GENERATOR (EDG) "B". THIS TECH SPEC REQUIRES THAT THE DIESEL GENERATOR BE CAPABLE OF REJECTING A LOAD GREATER THAN OR EQUAL TO 1352 KILOWATTS (KW) (ESSENTIAL SERVICE WATER (ESW) PUMP) WHILE MAINTAINING VOLTAGE AND FREQUENCY WITHIN SPECIFIC LIMITS. IT WAS SUBSEQUENTLY DETERMINED THAT THIS TECH SPEC REQUIREMENT HAD ALSO NOT BEEN SATISFIED FOR EDG "A" SINCE THE SURVEILLANCE DID NOT REQUIRE VERIFICATION THAT THE ESW PUMP LOAD WAS EQUAL TO OR GREATER THAN 1352 KW PRIOR TO DOING THE LOAD REJECTION OF THE ESW PUMP. THE ROOT CAUSE OF THIS EVENT WAS THAT ONLY THE LOAD DESCRIPTION AND NOT THE NUMERICAL VALUE OF THE LOAD WAS INCORPORATED INTO SURVEILLANCE PROCEDURES STS KJ-001A AND STS JK-001B. TO ENSURE THAT A SIMILAR CONDITION DOES NOT EXIST FOR OTHER TECH SPEC SURVEILLANCE REQUIREMENTS, A REVIEW WILL BE PERFORMED TO IDENTIFY THOSE SURVEILLANCE REQUIREMENTS THAT SPECIFY NUMERICAL VALUES AND ENSURE THAT THE APPLICABLE PROCEDURES ACCOMPLISH THEM. THIS REVIEW WILL BE COMPLETED BY 3/30/92.

[334] WOLF CREEK 1 DOCKET 50-482 LER 91-023
ACCIDENTAL BUMPING OF 120 VOLT SUPPLY PANEL RESULTS IN ENGINEERED SAFETY FEATURES EQUIPMENT ACTUATIONS.
EVENT DATE: 111991 REPORT DATE: 121991 NSSS: WE TYPE: PWR

(NSIC 223748) ON NOVEMBER 19, 1991, AT 2106 CST, 480 VOLT SUPPLY BREAKER 52NG02BAF4 TRIPPED, RESULTING IN A LOSS OF POWER TO VARIOUS RADIATION MONITORS . THIS LOSS OF POWER INITIATED A CONTROL ROOM VENTILATION ISOLATION SIGNAL, A FUEL BUILDING ISOLATION SIGNAL, AND A CONTAINMENT PURGE ISOLATION SIGNAL. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRACTED TO INADVERTENT BUMPING OF THE OPERATING SWITCH TO BREAKER 52NG02BAF4 BY NON-LICENSED PERSONNEL. EXERCISING CAUTION WHILE WORKING NEAR PLANT EQUIPMENT HAS BEEN RE-EMPHASIZED WITH THE RESPONSIBLE WORK GROUP AND THE REMAINING GROUPS IN THE DEPARTMENT. ADDITIONALLY, THE OPERATING SWITCH HAS BEEN INSPECTED AND IT WAS DETERMINED THAT A HARDWARE PROBLEM DID NOT CONTRIBUTE TO THE BREAKER TRIPPING.

[335] WOLF CREEK 1 DOCKET 50-482 LER 91-025
POSITIVE REACTIVITY CHANGES WITH NO OPERABLE CENTRIFUGAL CHARGING PUMPS AND BORON INJECTION FLOW PATH RESULTS IN TECHNICAL SPECIFICATION VIOLATIONS.
EVENT DATE: 111991 REPORT DATE: 121991 NSSS: WE TYPE: PWR

(NSIC 223749) ON NOVEMBER 19, 1991, FROM 1541 CST UNTIL 1914 CST AND ON NOVEMBER 20, 1991, APPROXIMATELY 1930 CST UNTIL NOVEMBER 21, 1991, AT 0655 CST, POSITIVE REACTIVITY CHANGES OCCURRED WHEN BORATED WATER WAS CHARGED AT A LOWER CONCENTRATION THAN THE REACTOR COOLANT SYSTEM BORON CONCENTRATION WHILE BOTH CENTRIFUGAL CHARGING PUMPS (CCPS) WERE INOPERABLE. WITH NO CCP OPERABLE, THE ACTION STATEMENTS FOR TECHNICAL SPECIFICATIONS 3.1.2.1 AND 3.1.2.3 REQUIRE THE SUSPENSION OF ALL OPERATIONS INVOLVING POSITIVE REACTIVITY CHANGES THUS REQUIRING THE CHARGING BORON CONCENTRATION TO BE HIGHER THAN THE REACTOR COOLANT SYSTEM BORON CONCENTRATION EVEN IF THE CHARGING CONCENTRATION EXCEEDS SHUTDOWN REQUIREMENTS. SEVERAL CAUSES CONTRIBUTED TO THESE EVENTS' OCCURRENCES, INCLUDING PROCEDURAL INADEQUACY AND INCOMPLETE TRAINING ON THE REACTOR MAKEUP CONTROL SYSTEM. ONE REACTOR OPERATOR DID NOT HAVE A COMPLETED UNDERSTANDING OF THE REACTOR MAKEUP CONTROL SYSTEM AND HAS BEEN COUNSELLED AN THE PROPER OPERATION OF THE SYSTEM. TO PREVENT RECURRENCE, THE APPROPRIATE PROCEDURES WILL BE REVISED. ADDITIONALLY, INFORMATION CONCERNING THE CONTROL SYSTEM DEFAULT TO THE 120 GALLON PER MINUTE SETPOINT WHEN SELECTED TO AUTOMATIC MODE HAS BEEN ADDED TO THE LICENSED OPERATOR ESSENTIAL READING.

[336] WOLF CREEK 1 DOCKET 50-482 LER 91-020 REV 01
UPDATE ON CONTAINMENT ISOLATION VALVES FAILED LOCAL LEAK RATE TEST CAUSING TOTAL PATH LEAKAGE AND OVERALL INTEGRATED LEAKAGE TO BE ABOVE 0.6 LA AND 0.75 LA.
EVENT DATE: 120591 REPORT DATE: 010692 NSSS: WE TYPE: PWR
VENDOR: FISHER CONTROLS CO.

(NSIC 223781) ON 10/22/91, DURING REFUELING OUTAGE V, WITH ALL FUEL REMOVED FROM THE CORE, THE CONTROL ROOM WAS INFORMED THAT THE TOTAL PATH CONTAINMENT LOCAL LEAKAGE RATES FOR TYPE B AND C TESTS WAS ABOVE THE TECHNICAL SPECIFICATION LIMIT OF 0.6 LA. THIS WAS DETERMINED FOLLOWING THE PERFORMANCE OF A LOCA LEAK RATE TEST ON CONTAINMENT ISOLATION VALVES EF HV032 AND EF HV034. THESE VALVES, ASSOCIATED WITH PENETRATION 28, ISOLATE CONTAINMENT AIR COOLERS "B" AND "D" FROM ESSENTIAL SERVICE WATER TRAIN "B". ALSO, THESE VALVES ARE NORMALLY OPEN VALVES AND RECEIVE AN OPEN SIGNAL ON A SAFETY INJECTION SIGNAL. ON 12/5/91, AT 1600 CST, FOLLOWING CALCULATIONS TO DETERMINE THE "AS-FOUND" CONDITION FOR THE OVERALL CONTAINMENT INTEGRATED LEAKAGE RATE (ILR), IT WAS DETERMINED THAT THE OVERALL ILR WAS GREATER THAN 1.0 LA AFTER FACTORING IN THE TOTAL PATH LEAKAGE FOR ALL APPLICABLE PENETRATIONS INCLUDING PENETRATION 28. THE EXCESSIVE LEAKAGE THROUGH VALVES EF HV032 AND EF HV034 RESULTED FROM EROSION/CORROSION (E/C) OF THE VALVE DISCS. THE VALVE DISCS WERE REPLACED. AN EVALUATION OF E/C DAMAGE TO THE NEWLY INSTALLED DISCS WILL BE CONDUCTED DURING THE NEXT REFUELING OUTAGE.

[337] WPPSS 2 DOCKET 50-397 LER 91-033
250 VOLT DC BUS INOPERABLE DUE TO LACK OF ADEQUATE FUSE COORDINATION.
EVENT DATE: 112091 REPORT DATE: 121991 NSSS: GE TYPE: BWR

(NSIC 223681) AS PART OF THE SUPPLY SYSTEM EFFORT TO UPGRADE THE DESIGN CALCULATIONS FOR NP-2, A FUSE COORDINATION CALCULATION IS BEING PREPARED FOR THE CLASS 1E DIRECT CURRENT (DC) POWER SYSTEMS, INCLUDING THE 250 VOLT (250VDC) SYSTEM. DEVELOPMENT OF THIS CALCULATION REVEALED THAT THE TIME-CURRENT MINIMUM MELTING CURVE FOR THE CLASS 1E 250VDC BUS BATTERY MAIN FUSE, AN 800 AMP FUSE, CROSSES OVER THE TIME-CURRENT CLEARING CURVE FOR THE DOWNSTREAM BRANCH CIRCUIT FUSE THAT SUPPLIES NON-CLASS 1E LOADS, A 400 AMP FUSE. THIS MEANS THAT IT WAS POSSIBLE, UNDER THE POSTULATED FAULT CONDITIONS, THAT THE CLASS 1E BATTERY MAIN FUSE COULD HAVE MELTED PRIOR TO THE DOWNSTREAM BRANCH CIRCUIT FUSE CLEARING, ISOLATING THE 250VDC BATTERY FROM THE BUS. HAD THIS EVENT OCCURRED THE 250VDC BUS WOULD HAVE BEEN SUPPLIED BY THE BATTERY CHARGER, BUT THIS IS NOT CONSIDERED A STABLE LONG-TERM CONDITION AND IS NOT CREDITED IN THE ACCIDENT ANALYSES. BECAUSE OF THE LACK OF SELECTIVE COORDINATION, THE 250VDC SYSTEM HAS BEEN TECHNICALLY INOPERABLE SINCE THE TIME OF INITIAL PLANT STARTUP IN 1983. OPERATION WITH THE 250VDC SYSTEM INOPERABLE WAS A CONDITION PROHIBITED BY THE PLANTS TECHNICAL SPECIFICATIONS AND IS REPORTABLE PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(I)(B). THIS CONDITION, LACK OF SELECTIVE COORDINATION ON THE 250VDCBUS, WAS OUTSIDE THE DESIGN BASIS OF THE PLANT AND IS REPORTABLE IN ACCORDANCE WITH 10CFR50.73(A)(2)(II).

[338] WPPSS 2 DOCKET 50-397 LER 91-035
MANUAL SCRAM DUE TO REACHING TECHNICAL SPECIFICATION LIMIT FOR REACTOR COOLANT CONDUCTIVITY DUE TO MAIN CONDENSER TUBE LEAK.
EVENT DATE: 122091 REPORT DATE: 011592 NSSS: GE TYPE: BWR

(NSIC 223802) ON DECEMBER 20, 1991 AT 1250 HOURS THE PLANT WAS MANUALLY SCRAMMED TO COMPLETE CONTROLLED SHUTDOWN DUE TO HIGH REACTOR COOLANT CONDUCTIVITY. DURING THE VENT PERIOD, PLANT PERSONNEL WERE IN THE PROCESS OF ATTEMPTING TO IDENTIFY THE SOURCE OF A SUSPECTED MAIN CONDENSER TUBE LEAK. THIS TROUBLESHOOTING EFFORT WAS BEING PERFORMED AS A RESULT OF TWO CONDUCTIVITY EXCURSIONS THAT HAD RECENTLY OCCURRED, BUT WHERE TECHNICAL SPECIFICATION LIMITS WERE NOT REACHED. DURING TROUBLESHOOTING EFFORTS, A LEAK OF APPROXIMATELY 65 GPM DEVELOPED IN SECTION A OF THE MAIN CONDENSER, AND CONDUCTIVITY LEVELS INCREASED. EFFORTS BY PLANT PERSONNEL TO CONTROL THE CONDUCTIVITY EXCURSION BY REDUCING REACTOR POWER AND SECURING ONE OF THREE CIRCULATING WATER (CW) SYSTEM PUMPS WERE UNSUCCESSFUL. ACCORDINGLY, PLANT CONTROL ROOM OPERATORS TOOK APPROPRIATE AND TIMELY ACTION TO SHUTDOWN THE PLANT AND MANEUVER TO THE COLD SHUTDOWN CONDITION. THE CAUSE OF THIS EVENT WAS A FAILED TUBE IN THE MAIN CONDENSER. AN AXIAL RACK, APPROXIMATELY 8 TO 12 INCHES IN LENGTH, WAS LATER DISCOVERED ON AN INTERIOR TUBE. CORRECTIVE ACTIONS CONSIST OF 1) PLUGGING THE FAILED TUBE AND PERFORMING SONIC LEAK TESTING OF ALL CONDENSER TUBES, 2) PERFORMING A MANAGEMENT EVALUATION TO RESOLVE PROBLEMS ASSOCIATED WITH CONDUCTIVITY EXCURSIONS, AND 3) PERFORMING NONDESTRUCTIVE EXAMINATION OF TUBING.

[339] WPPSS 2 DOCKET 50-397 LER 92-001
HIGH PRESSURE CORE SPRAY INOPERABLE DUE TO BATTERY INOPERABILITY.
EVENT DATE: 010292 REPORT DATE: 013092 NSSS: GE TYPE: BWR

(NSIC 223910) ON JANUARY 1, 1992 PLANT ELECTRICAL MAINTENANCE TECHNICIANS PERFORMED TECHNICAL SPECIFICATION REQUIRED BATTERY CHECKS ON THE DIVISION 3 HIGH PRESSURE CORE SPRAY (HPCS) BATTERY. SEVERAL CELLS WERE FOUND TO HAVE ELECTROLYTE LEVELS BELOW THE TECHNICAL SPECIFICATION DEFINED LIMIT VALUE. THE BATTERY WAS STILL OPERABLE SINCE THE LEVELS WERE ABOVE THE ALLOWABLE VALUE LEVEL. WATER WAS ADDED TO THE CELLS PRIOR TO THE REQUIRED CHECK OF OTHER CELL PARAMETERS. WHEN THE OTHER PARAMETERS WERE CHECKED ON JANUARY 2, THE SPECIFIC GRAVITY READINGS OF THREE OF THE 58 CELLS WERE BELOW THE ALLOWABLE VALUE. THE HPCS BATTERY, HPCS DIESEL GENERATOR, AND HPCS SYSTEM WERE DECLARED INOPERABLE AND THE NRC WAS VERBALLY NOTIFIED PER THE REQUIREMENTS OF 10CFR50.72(B)(2)(III)(D). THE BATTERY WAS PLACED ON AN EQUALIZE CHARGE AND BY 1215 HOURS ON JANUARY 3, 1992 THE BATTERY PARAMETERS WERE BACK WITHIN THE LIMIT AND ALLOWABLE VALUE RANGES AND THE BATTERY, HPCS, AND HPCS DIESEL GENERATOR WERE DECLARED OPERABLE. THE ROOT CAUSE OF THIS EVENT WAS INSTRUCTIONAL PRESENTATION DEFICIENCIES IN THAT THE TECHNICAL SPECIFICATION REQUIREMENTS ARE NOT CLEARLY WORDED AND THE PROCEDURE DID NOT PROVIDE CLEAR GUIDANCE ON THE TECHNICAL SPECIFICATION REQUIREMENTS. FURTHER CORRECTIVE ACTIONS INCLUDE: 1) NOTES WERE ADDED; 2) WILL BE REVIEWED; AND 3) WRITTEN GUIDANCE WILL BE PROVIDED TO CLARIFY THE TECHNICAL SPECIFICATION REQUIREMENTS.

[340] YANKEE ROWE DOCKET 50-029 LER 91-004
REACTOR SCRAM DUE TO LOSS OF GENERATOR FIELD.
EVENT DATE: 080391 REPORT DATE: 090291 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223810) ON 8/3/91, AT 1647 HOURS, WITH THE PLANT IN MODE 1 AT 78% POWER, THE TURBINE GENERATOR STATIC EXCITER INPUT BREAKER OPENED, CAUSING A LOSS OF GENERATOR FIELD WITH RESULTANT TURBINE TRIP AND REACTOR SCRAM. THERE WERE NO INOPERABLE STRUCTURES, COMPONENTS OR SYSTEMS THAT CONTRIBUTED TO THE EVENT. THE ROOT CAUSE OF THE EVENT IS STILL UNDER INVESTIGATION. THE PLANT IS OPERATING ON THE ROTATING EXCITER. THIS EVENT IS REPORTABLE PER 10CFR50.73(A)(2)(IV) WHICH IDENTIFIES ACTUATION OF A REACTOR PROTECTION SYSTEM AS A REPORTABLE EVENT. AT NO TIME DURING THE EVENT WAS THE PLANT IN AN UNALAYZED CONDITION OR IN A CONDITION OUTSIDE THE PLANT DESIGN BASIS. ALL ENGINEERED SAFETY FEATURE SYSTEMS AND EQUIPMENT OPERATED AS DESIGNED. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT ADVERSELY AFFECTED BY THIS EVENT.

[341] ZION 1 DOCKET 50-295 LER 91-012 REV 03
UPDATE ON SNUBBER MODIFICATION MISINTERPRETATION.
EVENT DATE: 071191 REPORT DATE: 121391 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 2 (PWR)
VENDOR: BINDICATOR CORP.

(NSIC 223603) IN PREPARATION FOR A HYDRAULIC SNUBBER REPLACEMENT MODIFICATION, IT WAS NOTICED THAT THE FIELD LOCATION OF SNUBBERS 1SIRS-1018 AND 1SIRS-1019 DID NOT AGREE WITH THE DRAWING. INVESTIGATION SHOWED THAT SNUBBERS 1SIRS-1017 AND 1SIRS-1020 AS IDENTIFIED ON THE DRAWING WERE SUPPOSED TO BE REMOVED BY MODIFICATION M22-1-89-14, BUT DUE TO INCORRECT LABELS, 1SIRS-1018 AND 1SIRS-1019 WERE REMOVED INSTEAD. ON 09/30/91 AND 11/13/91 PER THE CORRECTIVE ACTIONS OF THIS ORIGINAL INVESTIGATION, WALKDOWNS OF THE INACCESSIBLE UNIT 1 AND UNIT 2 HYDRAULIC SNUBBERS WERE PERFORMED. THREE DISCREPANCIES WERE FOUND DURING THESE WALKDOWNS. SNUBBER 2RCRS-2082 WAS REMOVED INSTEAD OF 2RCRS-2083A, SNUBBER 2SIRS-2132 WAS REMOVED INSTEAD OF 2SIRS-2133, AND SNUBBER 1DIRS-1064 WAS REMOVED INSTEAD OF 1DRTS-1063. THESE EVENTS WERE CAUSED BY PERSONNEL ERRORS DURING THE INSTALLATION OF MODIFICATIONS M22-1-89-14, M22-2-88-69, AND M22-1-88-033. THE NUCLEAR ENGINEERING DEPARTMENT PERFORMED CALCULATIONS TO DETERMINE IF THE REVISED SNUBBER CONFIGURATION COULD ADEQUATELY SUPPORT THE DESIGN BASIS LOADS. THE CALCULATIONS CONCLUDED THAT ALL AFFECTED SUPPORTS WOULD HAVE REMAINED FUNCTIONAL THROUGHOUT A SEISMIC EVENT AND PIPING STRESSES WOULD NOT HAVE EXCEEDED DESIGN BASIS LIMITS. THE CORRECT SNUBBERS WERE REINSTALLED AND THE INCORRECT SNUBBERS WERE REMOVED. ALL SNUBBERS WERE VERIFIED TO BE CORRECTLY INSTALLED.

[342] ZION 1 DOCKET 50-295 LER 91-018
INADEQUATE SHIFT CONTROL ROOM ENGINEER COVERAGE PER TECHNICAL SPECIFICATION.
EVENT DATE: 120691 REPORT DATE: 011092 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 2 (PWR)

(NSIC 223818) ON 12/11/91, DURING A REVIEW OF THE SHIFT ENGINEER'S LOG BOOK, IT WAS DISCOVERED THAT ON 12/6/91 NONE OF THE LICENSED SHIFT SUPERVISOR'S (LSS) ON DUTY WERE SHIFT CONTROL ROOM ENGINEER (SCRE) QUALIFIED. TECH SPEC FIGURE 6.1-1, MINIMUM SHIFT CREW COMPOSITION, REQUIRES THAT EACH SHIFT CREW HAVE ONE QUALIFIED SCRE. THIS EVENT WAS CAUSED BY A MANAGEMENT DEFICIENCY. AN INADEQUATE PROGRAM EXISTED TO ENSURE THAT THE TECH SPEC REQUIREMENTS WERE SATISFIED FOR SCRE QUALIFICATION. THE SAFETY SIGNIFICANCE OF THE EVENT WAS MINIMAL. CORRECTIVE ACTIONS INCLUDED IDENTIFYING SCRE AND FIRE CHIEF QUALIFICATIONS ON THE POSTED LSS SCHEDULE, AND ESTABLISHING A SCRE TURNOVER LOG.

[343] ZION 2 DOCKET 50-304 LER 92-001
FAILURE TO DECLARE A CONTAINMENT PENETRATION INOPERABLE AFTER A CONTAINMENT ISOLATION VALVE EXCEEDED ITS MAXIMUM ALLOWABLE STROKE TIME.
EVENT DATE: 010792 REPORT DATE: 020692 NSSS: WE TYPE: PWR

(NSIC 223954) ON 12/14/91 AT 1900 HOURS, DURING THE PERFORMANCE OF PERIODIC TEST (PT)-300, CONTAINMENT ISOLATION VALVES STROKE TIME VERIFICATION, CONTAINMENT ISOLATION VALVE 2AOV-SS9354A (BD) STROKED IN 27.3 SECONDS WHICH EXCEEDED ITS MAXIMUM ALLOWABLE STROKE TIME OF 25 SECONDS AS DESCRIBED IN THE ACCEPTANCE CRITERIA FOR PT-300. THIS STROKE TIME WAS BASED ON THE REQUIREMENTS OF THE INSERVICE TESTING PROGRAM. THE UNIT SUPERVISOR DID NOT DECLARE THE VALVE INOPERABLE BECAUSE THE VALVE WAS STROKING WITHIN THE TECH SPEC REQUIRED STROKE TIME OF 60 SECONDS. THIS EVENT WAS CAUSED BY A PERSONNEL ERROR ON THE PART OF THE SHIFT PERSONNEL WHO DID NOT RECOGNIZE THAT 2AOV-SS9354A WAS INOPERABLE ON 12/14/91, AND TAKE THE NECESSARY ACTIONS TO ISOLATE THE CONTAINMENT PENETRATION PER TECH SPEC 3.9.3.A. THE EFFECT OF THIS EVENT ON THE SAFE OPERATION OF THE PLANT WAS MINIMAL, AND THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED. CORRECTIVE ACTIONS INCLUDE COUNSELLING THE INDIVIDUALS INVOLVED, REVIEWING THE EVENT WITH ALL LICENSED SHIFT PERSONNEL, VERIFYING ALL CONTAINMENT ISOLATION VALVE STROKE TIMES SINCE JULY 1991 WERE ACCEPTABLE, AND REVISING PT-14 TO SPECIFY THAT FAILING IST STROKE TIMES RENDERS A VALVE INOPERABLE.

COMPONENT INDEX

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

COMPONENT INDEX

COMPONENT INDEX

SYSTEM INDEX

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

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KEYWORD INDEX

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

KEYWORD INDEX

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(See instructions on the reverse)

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10. SUPPLEMENTARY NOTES

11. 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 Operations Analysis Center (NOAC) 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.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

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