NUREG/CR-2000
ORNL/NSIC-200
Vol. 7, No. 3

Licensee Event Report (LER) Compilation

For month of March 1988

Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

Available from

Superintendent of Documents
U.S. Government Printing Office
Post Office Box 37082
Washington, D.C. 20013-7082

A year's subscription consists of 12 issues for
this publication.

Single copies of this publication
are available from National Technical
Information Service, Springfield, VA 22161

NUREG/CR-2000
ORNL/NSIC-200
Vol. 7, No. 3

Licensee Event Report (LER) Compilation

For month of March 1988

Manuscript Completed: March 1988
Date Published: April 1988

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

Prepared for
Office for Analysis and Evaluation of Operational Data
U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FIN A9135

Abstract

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

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

Gary T. Mays, Director
Nuclear Safety Information Center
Oak Ridge National Laboratory
P. O. Box Y, Oak Ridge, TN 37831
Telephone: 615/574-0391, FTS Number 624-0391

Questions regarding LER searches should be directed to

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

Page

[1] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 87-008
INADEQUATE DESIGN MODIFICATION CREATED A PATHWAY FOR UNFILTERED AIR INLEAKAGE IN EXCESS OF THE DESIGN BASIS FOR CONTROL ROOM HABITABILITY FOLLOWING A LOSS OF COOLANT ACCIDENT.
EVENT DATE: 111987 REPORT DATE: 121887 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 2 (PWR)

(NSIC 207526) DURING A REVIEW OF THE DESIGN OF THE ARKANSAS NUCLEAR ONE, UNITS 1 AND 2 COMMON CONTROL ROOMS EMERGENCY VENTILATION SYSTEMS, A LEAKAGE PATHWAY FOR UNFILTERED AIR WAS IDENTIFIED ON 11/19/87. THE PATHWAY CONSISTED OF A 1/4 INCH CIRCUMFERENTIAL HOLE AROUND A MOTOR-TO-FAN DRIVE SHAFT PENETRATING A CONTROL ROOM EMERGENCY VENTILATION SYSTEM FAN HOUSING. TESTING CONDUCTED ON 11/20/87 SHOWED THAT APPROXIMATELY 16 CUBIC FEET PER MINUTE (CFM) OF UNFILTERED AIR INLEAKAGE INTO THE CONTROL ROOM EXISTED. THE CONTROL ROOM HABITABILITY DESIGN BASIS FOR THIS SYSTEM ASSUMES ONLY 3 CFM OF UNFILTERED AIR INLEAKAGE FOR CALCULATING CONTROL ROOM OPERATOR RADIATION DOSE FOLLOWING A LOSS OF COOLANT ACCIDENT. THEREFORE, THIS WAS A CONDITION OUTSIDE THE PLANT DESIGN BASIS. A MODIFICATION TO SEAL THE OPENING WAS COMPLETED ON 11/25/87. THE CONDITION WAS THE RESULT OF AN INADEQUATE/INCOMPLETE DESIGN AND REVIEW OF A MODIFICATION MADE TO THE SYSTEM IN MARCH 1977. AT THAT TIME, AN ADDITIONAL FILTER BED WAS ADDED TO THE SYSTEM WHICH REQUIRED RELOCATING THE VENTILATION FAN FROM UPSTREAM OF THE FILTER TRAIN TO DOWNSTREAM OF THE FILTER TRAIN. IT WAS APPARENTLY NOT RECOGNIZED THAT THE NEW FAN HOUSING PENETRATION NEEDED TO BE SEALED TO PREVENT UNFILTERED AIR INLEAKAGE.

[2] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 87-009
SUBCRITICAL REACTOR TRIP WHILE PERFORMING A REACTOR SHUTDOWN DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 111487 REPORT DATE: 121187 NSSS: CE TYPE: PWR

(NSIC 207531) ON 11/14/87 AT 2107 HOURS, WITH THE REACTOR SUBCRITICAL, AN AUTOMATIC REACTOR TRIP OCCURRED FROM 5E-4% FULL POWER DURING A REACTOR SHUTDOWN. THE SHUTDOWN WAS BEING PERFORMED TO ALLOW A RECALCULATION OF THE ESTIMATED CRITICAL POSITION (ECP) OF THE CONTROL ELEMENT ASSEMBLIES (CEAS), IE., CONTROL RODS. REACTOR CRITICALITY HAD BEEN ACHEIVED JUST PRIOR TO BEGINNING THE SHUTDOWN. UPON REACHING CRITICALITY, DUE TO DELAYS IN THE REACTOR STARTUP AND XENON DECAY, CEA POSITIONS WERE CLOSE TO MAXIMUM INSERTION LIMITS ALLOWED BY TECH SPECS (ALTHOUGH WITHIN ACCEPTABLE ECP ACCEPTANCE CRITERIA). AT THE TIME OF THE TRIP, A LICENSED CONTROL ROOM OPERATOR WAS MANUALLY INSERTING CEA REGULATING GROUP 4. ALL WITHDRAWN CEAS THEN INSERTED INTO THE CORE. THE CAUSE OF THE AUXILIARY CPC TRIP WAS THAT THE GROUP 4 CEA POSITION WAS BELOW THAT ALLOWED IN THE CPC ANALYZED OPERATING SPACE. THE ROOT CAUSE OF THE TRIP WAS DETERMINED TO BE PROCEDURE DEFICIENCY. THE PROCEDURE BEING USED TO PERFORM THE EVOLUTION DID NOT PROVIDE INSTRUCTIONS TO PERSONNEL TO INSURE THE CPCS WERE BYPASSED PRIOR TO INSERTING CEAS TO A LEVEL THAT WOULD INITIATE CPC AUXILIARY TRIPS. ADDITIONALLY, THE PROCEDURE LACKED ADEQUATE GUIDANCE ON ACTIONS TO BE TAKEN IF REACTOR CRITICALITY IS DELAYED FOR UNANTICIPATED REASONS DURING A STARTUP.

[3] ARNOLD DOCKET 50-331 LER 87-030
ACTUATION OF STANDBY FILTER UNIT DUE TO DOWNSCALE TRIP OF RADIATION MONITOR.
EVENT DATE: 121587 REPORT DATE: 122287 NSSS: GE TYPE: BWR
VENDOR: NUCLEAR MEASUREMENTS CORP.

(NSIC 207546) ON DECEMBER 15, 1987, AT 0013 HOURS THE PLANT WAS OPERATING AT 100% POWER WHEN THE CONTROL BUILDING VENTILATION ISOLATED AND THE 'A' CONTROL BUILDING STANDBY FILTER UNIT (SFU) INITIATED UNEXPECTEDLY. INVESTIGATION OF THE TRIP REVEALED THAT IT HAD OCCURRED DUE TO A DOWNSCALE SIGNAL BEING RECEIVED BY THE 'A' CONTROL BUILDING AIR INTAKE RADIATION MONITOR. THE CAUSE OF THE DOWNSCALE SIGNAL IS SUSPECTED TO BE A TEMPORARY MALFUNCTION IN THE ARTIFICIAL BACKGROUND SOURCE. THIS SUSPICION COULD NOT BE VERIFIED, THOUGH, DUE TO THE MONITOR BECOMING

OPERABLE BEFORE MAINTENANCE COULD BE PERFORMED. IMMEDIATE CORRECTIVE ACTIONS WERE TO DETERMINE THE SOURCE OF THE INITIATION SIGNAL AND VERIFY AUTOMATIC FUNCTIONS. LATER IN THE DAY THE MONITOR'S CALIBRATION PROCEDURE WAS PERFORMED AND THERE WAS NO INDICATION OF THE UNIT BEING OUT OF CALIBRATION IN A WAY WHICH COULD HAVE CAUSED THE DOWNSCALE TRIP. DUE TO THE CONSERVATIVE NATURE OF THIS TRIP THERE WAS NO AFFECT ON THE SAFE OPERATION OF THE PLANT.

[4] BEAVER VALLEY 1 DOCKET 50-334 LER 87-019
RADIATION MONITOR FAILURE CAUSED MAIN FILTER BANK VENTILATION ALIGNMENT.
EVENT DATE: 120987 REPORT DATE: 010888 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 207638) ON 12/8/87 AT 1646 HOURS, INSTRUMENT AND CONTROL (I&C) PERSONNEL FINISHED THE CALIBRATION OF THE A-TRAIN FUEL BUILDING (FB) VENTILATION EXHAUST (RM-VS-103A) RADIATION MONITOR. RADIATION CONTROL (RADCON) PERSONNEL THEN ATTEMPTED TO PERFORM A SOURCE CHECK OF THE MONITOR, BUT WERE UNSUCCESSFUL. I&C PERSONNEL THEN REMOVED THE MONITOR FOR TROUBLESHOOTING. THE MONITOR WAS RESTORED ON 12/9/87 AT APPROXIMATELY 0800 HOURS. AT 0930 HOURS, CONTROL ROOM PERSONNEL WERE NOTIFIED OF THE SOUNDING OF THE FUEL BUILDING EVACUATION ALARM. THE OPERATORS ALSO NOTICED THAT THE FB VENTILATION HAD SWAPPED TO THE MAIN FILTER BANK. RM-VS-103A WAS THEN DISCOVERED TO HAVE FAILED HIGH CAUSING THESE ACTIONS. THE MONITOR WAS TURNED OFF, THE VENTILATION WAS REALIGNED AND THE EVACUATION ALARM WAS SILENCED. THE CAUSE FOR THIS EVENT WAS DUE TO FAILED ELECTRONIC COMPONENTS IN THE RADIATION MONITOR (POWER SUPPLY CAPACITORS) AND A WIRING PROBLEM. THESE COMPONENTS WERE REPLACED AND THE WIRING PROBLEM WAS CORRECTED. LONG-TERM CORRECTIVE ACTIONS INVOLVE ADDITIONAL PROCEDURAL GUIDANCE FOR RADIATION MONITORS AND TO INITIATE VENTILATION SYSTEM CHANGES PRIOR TO DETECTOR RE-ENERGIZATION. THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC BECAUSE A REDUNDANT RADIATION MONITOR MONITORING THE FB VENTILATION SHOWED NO ACTUAL INCREASE IN RADIATION LEVELS TO WARRANT A VENTILATION REALIGNMENT.

[5] BEAVER VALLEY 1 DOCKET 50-334 LER 87-020
ENERGIZATION OF CONTAINMENT PURGE EXHAUST MONITOR RESULTS IN ESF ACTUATION.
EVENT DATE: 121387 REPORT DATE: 011288 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 207896) ON 12/13/87 AT 0130 HOURS, IN ACCORDANCE WITH STATION SHUTDOWN PROCEDURES, THE CONTAINMENT PURGE AND EXHAUST RADIATION MONITORS (RM-VS-104A AND B) WERE ENERGIZED. BOTH MONITORS SPIKED ABOVE THEIR HIGH-HIGH SETPOINT INITIATING FLOW THROUGH THE MAIN FILTER BANKS. APPROXIMATELY TEN MINUTES LATER BOTH MONITORS FELL BELOW THEIR ALARM SETPOINTS AND THE VENTILATION DAMPERS WERE REALIGNED. AT APPROXIMATELY 0830 HOURS, IN PREPARATION FOR A BATCH RELEASE OF CONTAINMENT ATMOSPHERE NEW ALARM SETPOINTS WERE CALCULATED. A RADIATION TECHNICIAN PROCEEDED TO ADJUST THE ALARM SETPOINTS ON (RM-VS-104A) AND PLACED THE CONTROL KNOB TO CALIBRATE. AFTER ADJUSTING THE ALARM SETPOINTS, HE TURNED THE CONTROL KNOB BACK TO OPERATE. THIS INITIATED TRAIN "A" FLOW THROUGH THE MAIN FILTER BANKS. THE OPERATOR RESET THE RADIATION MONITOR AND REALIGNED THE VENTILATION DAMPERS. THERE WERE NO SAFETY IMPLICATIONS RESULTING FROM THIS EVENT SINCE THE SYSTEMS FUNCTIONED AS DESIGNED ON RECEIPT OF A HIGH-HIGH RADIATION SIGNAL.

[6] BEAVER VALLEY 1 DOCKET 50-334 LER 87-021
INOPERABLE CHARCOAL FILTER BANK SPRINKLER NOZZLES.
EVENT DATE: 122187 REPORT DATE: 012088 NSSS: WE TYPE: PWR
VENDOR: SPRAYING SYSTEMS COMPANY

(NSIC 207991) UNIT 1 WAS IN THE SIXTH REFUELING OUTAGE. ON 12/21/87, BEAVER VALLEY TEST (BVT) 1.1-1.33.1 "MAIN FILTER BANK SPRINKLER AIR FLOW TEST" WAS

CONDUCTED TO VERIFY THE OPERABILITY OF THE TRAIN "B" CHARCOAL FILTER BANK FIRE SUPPRESSION SPRINKLER SYSTEM. AT 1050 HOURS, THE TEST RESULTS INDICATED SEVEN NOZZLES TO BE BLOCKED. THIS CONDITION VIOLATED TECH SPEC 4.7.14.2.D, WHICH REQUIRES VERIFICATION OF FLOW THROUGH ALL NOZZLES. SINCE THE SYSTEM WAS POTENTIALLY INOPERABLE LONGER THAN THE 14 DAY ACTION STATEMENT OF TECH SPEC 3.7.14.2, THIS EVENT IS BEING REPORTED UNDER 10CFR50.73.A.2.I.B, AS WELL AS TECH SPEC 6.9.2 (FIRE SUPPRESSION SPECIAL REPORT). THE NOZZLES WERE REMOVED, CLEANED AND REPLACED. UNOBSTRUCTED FLOW WAS VERIFIED BY 1135 HOURS 12/21/87. ON 1445 HOURS, 1/8/88, BVT 1.1-1.33.1 WAS COMPLETED FOR THE TRAIN "A" SPRINKLERS, WITH 34 NOZZLES FOUND TO BE BLOCKED. AFTER CLEANING, THEY WERE RETURNED TO SERVICE BY 1535 HOURS, 1/8/88. THE CAUSE OF THE CLOGGED NOZZLES IS DEBRIS RESULTING FROM MOISTURE IN THE SYSTEM. PREVIOUSLY, THOROUGH CLEANING WAS JUDGED TO BE THE SOLUTION TO THE PROBLEM (LER 86-005-01). SINCE THAT SOLUTION DID NOT PROVE SUCCESSFUL, UNIT 1 IS CONSIDERING A DESIGN CHANGE TO REPLACE THE EXISTING SPRAY SYSTEMS 9510 NOZZLES WITH A LARGER ORIFICE MODEL.

[7] BEAVER VALLEY 2 DOCKET 50-412 LER 87-034
REACTOR TRIP DUE TO A LO-LO LEVEL CONDITION IN THE 21B STEAM GENERATOR.
EVENT DATE: 102987 REPORT DATE: 113087 NSSS: WE TYPE: PWR
VENDOR: MASONEILAN INTERNATIONAL, INC.

(NSIC 207278) ON 10/29/87 AT 0135 HOURS, WITH THE PLANT AT 98% REACTOR POWER, ERRATIC LEVEL SWINGS IN THE "B" 5TH POINT HEATER WERE OBSERVED. OPERATIONS AND INSTRUMENT & CONTROL (I&C) PERSONNEL WERE DISPATCHED TO INVESTIGATE. AT 0142 HOURS, EXTRACTION STEAM TO THE "B" 5TH POINT HEATER ISOLATED ON EXTREME HIGH LEVEL. HEATER DRAIN PUMP AND MAIN FEEDWATER PUMP (MFP) PERTURBATIONS WERE ALSO NOTED. AT 0145 HOURS, CONDENSATE POLISHING WAS BYPASSED TO IMPROVE MFP FLOW. A START ATTEMPT ON THE 21C CONDENSATE PUMP WAS UNSUCCESSFUL. THE "B" 5TH POINT HEATER ISOLATION CAUSED LEVEL INCREASES AND HEATER ISOLATIONS IN THE ENTIRE "B" HEATER TRAIN. AT 0145 HOURS, THE 21A MFP TRIPPED ON LOW SUCTION PRESSURE. AN EMERGENCY SHUTDOWN WAS COMMENCED, HOWEVER AT 0147 HOURS, A REACTOR TRIP ON LO-LO LEVEL IN THE 21B STEAM GENERATOR OCCURRED. THE OPERATORS STABILIZED THE PLANT IN HOT STANDBY USING THE EMERGENCY OPERATING PROCEDURES. THE CAUSE FOR THIS EVENT WAS IMPROPER RESPONSE OF THE "B" 5TH POINT HEATER LEVEL CONTROL VALVES (LCV). I&C PERSONNEL ADJUSTED THE RESPONSE OF THE "B" 5TH POINT HEATER LCVS. THE OTHER LCVS IN THE HEATER TRAIN WERE ALSO CHECKED FOR PROPER OPERATION. ADMINISTRATIVE GUIDANCE HAS BEEN PROVIDED TO OPERATIONS PERSONNEL AS A RESULT OF THIS EVENT. THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC AS A RESULT OF THIS EVENT.

[8] BIG ROCK POINT DOCKET 50-155 LER 87-013
TECHNICAL SPECIFICATION VIOLATION CAUSED BY LIMIT FOR CRD WITHDRAW TIME.
EVENT DATE: 112287 REPORT DATE: 122287 NSSS: GE TYPE: BWR

(NSIC 207499) DURING A REACTOR STARTUP ON NOVEMBER 22, 1987 (1047 HOURS) THREE (3) CONTROL ROD DRIVES WERE OBSERVED TO HAVE "DOUBLE" NOTCHED DURING WITHDRAWAL. STARTUP WAS IMMEDIATELY TERMINATED AND ALL CONTROL RODS INSERTED TO ALLOW INVESTIGATION OF THE PROBLEM. ON NOVEMBER 22, 1987 (1743 HOURS) DURING PERFORMANCE OF CONTROL ROD DRIVE TIMING TESTS, TWO (2) CONTROL ROD DRIVES WERE FOUND TO HAVE WITHDRAW TIMES OF 21.7 AND 22.6 SECONDS. THESE VALUES EXCEEDED THE BIG ROCK POINT TECH SPEC 5.2.2(A)(IV) MINIMUM WITHDRAW TIME OF 23 SECONDS. FURTHER INVESTIGATION CONCLUDED THAT THE CRD TIMING IS AFFECTED BY RECIRCULATING PUMP VALVE POSITION AND FLOWRATE. ALL THIRTY-TWO (32) CONTROL ROD DRIVES WERE TIMED AND RESET TO ACCEPTABLE VALUES WITH THE RECIRCULATING PUMPS AT FULL FLOW. AT 0105 HOURS ON NOVEMBER 23, 1987, REACTOR STARTUP COMMENCED.

[9] BRAIDWOOD 1 DOCKET 50-456 LER 87-060
MANUAL REACTOR TRIP DUE TO PLUGGED CONDENSATE PUMP SUCTION STRAINERS.
EVENT DATE: 120687 REPORT DATE: 122487 NSSS: WE TYPE: PWR

(NSIC 207561) AT 1953 ON DECEMBER 6, 1987, UNIT 1 WAS MANUALLY TRIPPED IN RESPONSE TO RAPIDLY DECREASING STEAM GENERATOR LEVELS. CONDENSATE PUMP SUCTION STRAINER DIFFERENTIAL PRESSURE HIGH AND FEEDWATER PUMP NET POSITIVE SUCTION HEAD LOW ALARMS WERE RECEIVED IMMEDIATELY PRIOR TO THE TRIP. ADDITIONALLY, STATION PERSONNEL REPORTED CONDENSATE SYSTEM PIPING SHAKING. THE ROOT CAUSE OF THE EVENT IS ATTRIBUTED TO BLOCKAGE OF THE CONDENSATE/CONDENSATE BOOSTER PUMP SUCTION STRAINERS. CORRECTIVE ACTION INCLUDED A WALKDOWN OF THE CONDENSATE SYSTEM TO LOOK FOR PIPING/HANGER DAMAGE. PRIOR TO REACTOR STARTUP, ALL FOUR CONDENSATE PUMP SUCTION STRAINERS WERE MANUALLY CLEANED, AND ALL FOUR CONDENSATE BOOSTER PUMP STRAINERS WERE BACKWASHED. THE CONDENSER HOT WELL WILL BE CLEANED AND INSPECTED DURING THE NEXT OUTAGE. THERE HAVE BEEN NO PREVIOUS OCCURRENCES.

[10] BRAIDWOOD 1 DOCKET 50-456 LER 87-062
CONTROL ROOM AND AUXILIARY BUILDING VENTILATION SHIFT TO EMERGENCY MAKEUP MODE DUE TO LOOSE HANDLE ON UNIT 2 SWITCH.
EVENT DATE: 121187 REPORT DATE: 122387 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BRAIDWOOD 2 (PWR)

(NSIC 207725) ON DECEMBER 11, 1987, OPERATIONS WAS PREPARING TO PERFORM THE UNIT 2 TRAIN A SOLID STATE PROTECTION SYSTEM (SSPS) BI-MONTHLY SURVEILLANCE. THE INPUT ERROR INHIBIT SWITCH WAS PLACED IN THE INHIBIT POSITION AND THE OUTPUT MODE TEST SWITCH WAS PLACED IN THE OPERATE POSITION. IMMEDIATELY UPON PLACING THE OUTPUT MODE TEST SWITCH IN THE OPERATE POSITION, A SAFETY INJECTION (SI) SIGNAL WAS GENERATED. THIS CAUSED THE 2A DIESEL GENERATOR AND THE 2A ESSENTIAL SERVICE WATER PUMP TO AUTO START, AND ACTUATED AUXILIARY BUILDING CHARCOAL BOOSTER FANS OVA03CA, OVA03CF, AND OVA04CA WITH THE ASSOCIATED DAMPERS. IN ADDITION, THE CONTROL ROOM EMERGENCY MAKEUP UNIT FILTER FAN OVC03CA ALSO STARTED. CONTROL ROOM PERSONNEL RESPONDED TO THE SI SIGNAL AND RETURNED EQUIPMENT TO NORMAL. CAUSE OF EVENT WAS INSUFFICIENT TORQUING OF THE SET SCREW ON THE HANDLE OF THE INPUT ERROR INHIBIT SWITCH, THUS ALLOWING THE HANDLE TO ROTATE WITHOUT MOVING THE SWITCH SHAFT. TO PREVENT RECURRENCE, ALL SIMILAR SWITCHES IN BOTH TRAINS OF SSPS ON UNIT TWO WERE CHECKED. SIMILAR ACTIONS WILL BE TAKEN ON UNIT ONE SSPS DURING THE FIRST OUTAGE OF OPPORTUNITY.

[11] BRAIDWOOD 1 DOCKET 50-456 LER 87-061
INCOMPLETE POWER OPERATED RELIEF VALVE SURVEILLANCE DUE TO AN OVERLY RESTRICTIVE PROCEDURAL REQUIREMENT.
EVENT DATE: 122487 REPORT DATE: 011188 NSSS: WE TYPE: PWR

(NSIC 207947) AT 1250 ON DECEMBER 24, 1987, IT WAS DISCOVERED THAT THE POWER OPERATED RELIEF VALVE (PORV) AND PORV BLOCK VALVE STROKE TEST SURVEILLANCE WAS OVERDUE. PORV 1RY456 WAS INOPERABLE DUE TO EXCESSIVE SEAT LEAKAGE AND ITS ASSOCIATED BLOCK VALVE CLOSED. AT 1305 BOTH PORV BLOCK VALVES AND THE REMAINING OPERABLE PORV, 1RY455A, WERE DECLARED INOPERABLE DUE TO THE INCOMPLETE SURVEILLANCE WHICH WAS INITIATED ON DECEMBER 11, 1987. THE SURVEILLANCE WAS SATISFACTORILY PERFORMED ON BOTH PORV BLOCK VALVES AND 1RY455A AND THE VALVES WERE DECLARED OPERABLE AT 1316. THE CAUSE WAS OVERLY RESTRICTIVE PROCEDURAL REQUIREMENTS WHICH MANDATED BOTH PORV'S TO BE OPERABLE TO PERFORM THE SURVEILLANCE. THE SURVEILLANCE PROCEDURE WILL BE REVISED TO PRECLUDE THIS TYPE OF OCCURRENCE. THERE HAVE BEEN NO PREVIOUS LICENSEE EVENT REPORTS DUE TO RESTRICTIVE SURVEILLANCE PROCEDURES.

[12] BROWNS FERRY 1 DOCKET 50-259 LER 87-001 REV 01
UPDATE ON FIRE-WATCH VIOLATIONS CAUSED BY PERSONNEL ERRORS.
EVENT DATE: 012887 REPORT DATE: 120487 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 207504) ON JANUARY 27, 1987, AT 2200 PORTIONS OF THE FIRE PROTECTION SYSTEM WERE ISOLATED IN ORDER TO REPLACE A HOSE STATION VALVE IN THE UNIT 1 REACTOR BUILDING. APPROXIMATELY FIVE HOURS LATER THE SHIFT ENGINEER DISCOVERED THAT A FIRE WATCH WAS NOT POSTED, AS REQUIRED BY PLANT TECH SPECS, IN AN AREA OF THE UNIT 1 AND 2 DIESEL GENERATOR BUILDING. IMMEDIATE STEPS WERE TAKEN TO POST THE WATCH. THE INTENT OF THE WORDING ON THE FIRE EQUIPMENT REMOVAL FROM SERVICE PERMIT HAD NOT BEEN CLEARLY COMMUNICATED, AND THE EXISTING PATROLLING FIRE WATCH IN THE REACTOR BUILDING WAS DEEMED ADEQUATE. ONE PREACTION SPRINKLER SYSTEM AND FOUR FIRE HOSE STATIONS IN THE DIESEL GENERATOR BUILDING WERE INCAPACITATED AND UNMONITORED FOR THE DURATION OF THIS EVENT. FIRE DETECTION AND CO(2) FIRE SUPPRESSION FOR THE INDIVIDUAL DIESEL GENERATORS REMAINED UNAFFECTED. ANOTHER FIRE WATCH INFRACTION OCCURRED ON FEBRUARY 10, 1987, WHEN AN INDIVIDUAL REPORTED TO THE WRONG CABLE SPREADING ROOM FOLLOWING THE FAILURE OF A MASTER CONTROL VALVE FOR THE CO(2) FIRE PROTECTION SYSTEM. THE PLANT FIRE PROTECTION UNIT HAS DEVELOPED A CONCISE LISTING OF AREAS REQUIRING FIRE WATCHES, AND THE REASONS FOR THEM, FOR THE SHIFT ENGINEER'S OFFICE.

[13] BROWNS FERRY 1 DOCKET 50-259 LER 87-014 REV 01
UPDATE ON CONTROL ROOM EMERGENCY VENTILATION SYSTEM FLOW OUT OF SPECIFICATION DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 070887 REPORT DATE: 010888 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 207970) ON JULY 8, 1987 AT 1430, ENGINEERS TESTING THE CONTROL ROOM EMERGENCY VENTILATION (CREV) SYSTEM MEASURED FLOW RATES IN EXCESS OF THE DESIGN BASIS FLOW RATE. THE TEST, PERFORMED FOR THE FIRST TIME UNDER SIMULATED ACCIDENT CONDITIONS, REVEALED THAT THE THROTTLE DAMPERS ON THE TWO CREV UNITS HAD NOT BEEN SET TO DELIVER THE FLOW, REQUIRED BY TECH SPECS FOR OPTIMUM FILTRATION. THIS CONDITION HAS LIKELY EXISTED SINCE THE SYSTEM WAS INSTALLED DURING PLANT CONSTRUCTION. UPON DISCOVERY OF THE DEFICIENCY, THE DAMPER POSITIONS WERE ADJUSTED TO PROVIDE FLOWS WITHIN PROPER LIMITS. THE SPECIAL PROCEDURE WAS PROMPTED FOLLOWING RECOGNIZED DIFFERENCES BETWEEN THE PROCEDURAL METHODOLOGY FOR TESTING CREV AND CREV AUTO INITIATION FROM ACCIDENT LOGIC. SUPPLY FANS WERE ROUTINELY TRIPPED FOR PERFORMANCE OF THE FLOW TEST. THE TEST PROCEDURE ALSO SPECIFIED A DIFFERENT DAMPER LINEUP. THE CREV FLOW TEST PROCEDURE WILL BE REVISED. ADMINISTRATIVE CONTROLS NEEDED TO MAINTAIN PROPER SYSTEM ALIGNMENT WILL BE PRESCRIBED, AS APPROPRIATE, WHEN TESTING IS COMPLETE. IN ADDITION, THE RESTART TEST PROGRAM WILL VERIFY THE ABILITY OF THE CREV SYSTEM TO PERFORM ITS FUNCTION.

[14] BROWNS FERRY 1 DOCKET 50-259 LER 87-030
FAULTY RELAY COIL CAUSES AUTO START OF CONTROL ROOM EMERGENCY VENTILATION SYSTEM.
EVENT DATE: 112587 REPORT DATE: 122987 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 207658) ON NOVEMBER 24, 1987, AT 1930 HOURS, CONTROL ROOM PERSONNEL OBSERVED THAT TRAIN B OF THE CONTROL ROOM EMERGENCY VENTILATION (CREV) SYSTEM WAS RUNNING. ALL THREE UNITS WERE DEFUELED AND NO REFUELING OPERATIONS WERE IN PROGRESS. THE EXACT TIME OF THE CREV AUTO START IS UNKNOWN. AFTER VERIFYING CREV TRAIN A OPERABLE AND NO VALID INITIATION SIGNAL PRESENT, THE LICENSED OPERATOR

SECURED THE SYSTEM. FOLLOWING INVESTIGATION BY ELECTRICAL MAINTENANCE PERSONNEL IT WAS DETERMINED THAT A BURNED ELECTRICAL COIL IN THE UNIT 3 16A-K72 RELAY HAD CAUSED THE ACTUATION. THE FAULTED COIL WAS ASSESSED AS AN END-OF-LIFE FAILURE AND WAS REPLACED.

[15] BROWNS FERRY 1 DOCKET 50-259 LER 87-031
PROCEDURAL INADEQUACY RESULTS IN START OF CONTROL ROOM EMERGENCY VENTILATION SYSTEM.
EVENT DATE: 120487 REPORT DATE: 123187 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 207671) UNITS 1, 2, AND 3 WERE IN REFUELING OUTAGES AND COMPLETELY DEFUELED WHEN THIS EVENT OCCURRED. ON DECEMBER 4, 1987, AT 1455 HOURS, THE CONTROL ROOM EMERGENCY VENTILATION (CREV) SYSTEM AUTO STARTED DURING SWITCHING OPERATIONS ON UNITS 1 AND 2 4-KV SHUTDOWN BOARD A. A RESTART TEST PROCEDURE BEING UTILIZED DID NOT SPECIFY A POSITION FOR THE TRANSFER SWITCH ON THE 4-KV BOARD. WITH THE ALTERNATE POWER SUPPLY FEEDING THE BOARD, THE ASSISTANT SHIFT ENGINEER PLACED THE TRANSFER SWITCH IN AUTOMATIC. THIS CAUSED THE AUTOMATIC TRANSFER OF THE BOARD TO ITS NORMAL POWER SUPPLY. THIS RESULTED IN LOAD SHEDDING AND CREV AUTO START. SYSTEMS WERE RESTORED TO NORMAL AT 1501 HOURS. THE RESTART TEST PROCEDURE WILL BE REVISED TO SPECIFY TRANSFER SWITCH POSITION.

[16] BROWNS FERRY 3 DOCKET 50-296 LER 87-006
UNPLANNED ENGINEERING SAFETY FEATURE ACTUATIONS DUE TO RELAY FAILURE AND PERSONNEL ERROR DURING CORRECTIVE MAINTENANCE.
EVENT DATE: 112487 REPORT DATE: 122387 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 207544) AT 1945 HOURS, ON NOVEMBER 22, 1987, WITH ALL THREE UNITS DEFUELED, A RELAY COIL IN THE PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) LOGIC DEVELOPED AN ELECTRICAL SHORT AND CLEARED THE FUSE PROTECTING THAT PORTION OF THE PCIS LOGIC. AS A RESULT, THE INBOARD PCIS VALVES FOR THE DRYWELL FLOOR AND THE EQUIPMENT DRAIN SUMP DISCHARGES ISOLATED. THE FUSE AND RELAY WERE REPLACED AND ALL SYSTEMS WERE RETURNED TO THEIR ORIGINAL STATUS BY 1940 HOURS, ON NOVEMBER 24, 1987. THE CAUSE OF THE RELAY COIL FAILURE IS CONSIDERED TO BE AN END-OF-LIFE FAILURE. DURING THE REPLACEMENT OF THIS RELAY, ON NOVEMBER 24, 1987, AT 1730 HOURS, THE JUMPER WHICH WAS INSTALLED AS PART OF THIS CORRECTIVE MAINTENANCE WAS INADVERTENTLY DISLODGED BY MAINTENANCE CRAFT. THIS DEENERGIZED OTHER PCIS RELAYS. THE DEENERGIZATION CLOSED THE INBOARD ISOLATION VALVE ON THE SUCTION OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM AND TRIPPED THE RUNNING RWCU PUMP. THE JUMPER WAS REPLACED AND THE ISOLATION WAS RESET. RWCU WAS RETURNED TO SERVICE 15 MINUTES AFTER IT ISOLATED. CORRECTIVE MAINTENANCE WAS COMPLETED BY 1940 HOURS, ON NOVEMBER 24, 1987, AND ALL SYSTEMS WERE RETURNED TO THEIR ORIGINAL STATUS. THESE INDIVIDUAL VALVE CLOSURES ARE CONSIDERED UNPLANNED ACTUATIONS OF ENGINEERED SAFETY FEATURES.

[17] BRUNSWICK 1 DOCKET 50-325 LER 87-022
AUTOMATIC ISOLATIONS OF UNITS 1 AND 2 COMMON CONTROL BUILDING HEATING, VENTILATING, AIR CONDITIONING, AND EMERGENCY AIR FILTRATION SYSTEMS DUE TO CHLORINE LEAKS.
EVENT DATE: 121287 REPORT DATE: 011188 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)
VENDOR: WALLACE & TIERMAN, INC.

(NSIC 207892) AT 1959 HOURS ON 12/12/87, 0515 HOURS ON 12/26/87, AND 2330 HOURS ON 12/31/87, AUTOMATIC ISOLATIONS OF THE UNITS 1 AND 2 COMMON CONTROL BUILDING HEATING, VENTILATING, AIR CONDITIONING (CB HVAC) SYSTEM AND COMMON CONTROL

BUILDING EMERGENCY AIR FILTRATION (CBEAF) SYSTEM OCCURRED DUE TO CHLORINE LEAKAGE/ALARM CONDITIONS AT THE UNITS' COMMON CHLORINATION SYSTEM (CS) CHLORINE LOADING AREA. IN EACH CASE, THE ISOLATIONS WERE REVEALED THROUGH APPROPRIATE CONTROL ROOM ALARM ANNUNCIATION AND INDICATION. UNIT 1 WAS AT 100% AND UNIT 2 WAS RESPECTIVELY AT 73%, 56%, AND 69%, WHILE IN A FUEL DEPLETION COAST DOWN APPROACHING A UNIT REFUEL/MAINTENANCE OUTAGE IN 1988. IN EACH CASE, FOLLOWING THE CLEARING OF THE HIGH CHLORINE ALARM CONDITION, THE CB HVAC AND CBEAF SYSTEMS WERE RETURNED TO NORMAL SERVICE. THESE EVENTS DID NOT RESULT IN A SIGNIFICANT PERSONNEL HAZARD. THE 12/12/87 EVENT RESULTED FROM CHLORINE LEAKAGE PAST THE FLEX SEALS OF THE CS LIQUID EVAPORATORS' NUMBERS 1 AND 3 PRESSURE REGULATING VALVES (PRVS) 2-CW.PRV.212 AND 232. THE SUBJECT VALVES' FLEX SEALS, BOTH WALLACE & TIERNAN, PART NO. P-16903, WERE REPLACED AND THE VALVES WERE RETURNED TO SERVICE. THE EVENTS ON 12/26/87 AND 12/31/87, ARE UNDER INVESTIGATION AND WILL BE APPROPRIATELY ADDRESSED.

[18] BRUNSWICK 1 DOCKET 50-325 LER 87-023
INOPERABILITY OF HIGH PRESSURE COOLANT INJECTION HPCI SYSTEM DUE TO FAILURE OF HPCI TURBINE STEAM INLET ISOLATION VALVE E41-F001.
EVENT DATE: 123187 REPORT DATE: 012988 NSSS: GE TYPE: BWR
VENDOR: LIMITORQUE CORP.
PORTER PEERLESS MOTORS

(NSIC 208025) AT 0100 HOURS ON 12/31/87, WHILE PERFORMING THE OPERABILITY TEST OF THE UNIT 1 HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM (E41), PERIODIC TEST (PT).09.2, THE HPCI TURBINE STEAM SUPPLY ISOLATION VALVE, E41-F001, WOULD NOT OPEN. UNIT 1 WAS OPERATING AT 100% THE HPCI SYSTEM WAS DECLARED INOPERABLE AND A LIMITING CONDITION FOR OPERATION WAS ESTABLISHED. THE REACTOR CORE ISOLATION COOLING, RESIDUAL HEAT REMOVAL/LOW PRESSURE COOLANT INJECTION, AND AUTOMATIC DEPRESSURIZATION SYSTEMS, AND THE A AND B CORE SPRAY SUBSYSTEMS WERE OPERABLE. E41.F001 WOULD NOT OPEN DUE TO FAILURE OF THE VALVE MOTOR. AN INVESTIGATION COULD NOT DETERMINE THE CAUSE OF THE MOTOR FAILURE. THE VALVE MOTOR WAS REPLACED, THE VALVE WAS RETURNED TO SERVICE, AND AT 2045 HOURS ON 1/2/88, THE HPCI WAS RETURNED TO STANDBY READINESS. BY 7/1/88 A FOLLOW-UP TO THIS REPORT WILL BE SUBMITTED OUTLINING THE RESULTS OF A PLANT ENGINEERING INVESTIGATION/EVALUATION OF THIS EVENT ALONG WITH ANY FURTHER CORRECTIVE ACTIONS TAKEN OR PLANNED.

[19] BRUNSWICK 2 DOCKET 50-324 LER 87-012
UNPLANNED DIVISION I PRIMARY CONTAINMENT GROUP 5 ISOLATION OF REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM WHILE INSTALLING ELECTRICAL JUMPER DURING RCIC SURVEILLANCE TESTING.
EVENT DATE: 120787 REPORT DATE: 010688 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 207681) AT 1425 HOURS ON 12/7/87, THE UNIT 2 REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM (E51) TURBINE STEAM SUPPLY INBOARD PRIMARY CONTAINMENT VALVE (PCIV) AUTOCLOSED ON AN A LOGIC, DIVISION 1, PRIMARY CONTAINMENT GROUP 5 ISOLATION SIGNAL. UNIT 2 WAS AT 75% POWER WHILE IN A FUEL DEPLETION COASTDOWN APPROACHING A SCHEDULED UNIT REFUEL/MAINTENANCE OUTAGE IN JANUARY 1988. THE UNIT HIGH PRESSURE COOLANT INJECTION, AUTOMATIC DEPRESSURIZATION, REACTOR CORE SPRAY, AND RESIDUAL HEAT REMOVAL/LOW PRESSURE COOLANT INJECTION SYSTEMS WERE OPERABLE. PERFORMANCE OF THE CHANNEL CALIBRATION AND FUNCTIONAL TEST OF THE RCIC ISOLATION INSTRUMENTATION WAS IN PROGRESS IN ACCORDANCE WITH MAINTENANCE SURVEILLANCE TEST (MST) 2MST-RCIC22M. THIS ISOLATION RESULTED FROM PERSONNEL ERROR. AN ALLIGATOR CLIP OF AN ELECTRICAL JUMPER, UTILIZED TO MEET TECHNICAL SPECIFICATIONS REQUIREMENTS AND ALLOW THE MST PERFORMANCE, WAS INADEQUATELY ATTACHED; THEREFORE, SLIPPING OFF THE INSTRUMENT TERMINAL OF RCIC ISOLATION INSTRUMENT E51A-K49A AND MAKING MOMENTARY CONTACT WITH THE NEARBY TERMINAL OF RCIC ISOLATION INSTRUMENT E51A-K49C. THE JUMPER WAS REINSTALLED AND THE RCIC SYSTEM WAS RETURNED TO

STANDBY READINESS AT 1433 HOURS ON 12/7/87. REAL-TIME TRAINING WILL BE CONDUCTED CONCERNING THIS EVENT BY 3/31/88.

[20] BYRON 1 DOCKET 50-454 LER 87-019 REV 01
UPDATE ON SAFETY INJECTION AND REACTOR TRIP FROM LOW STEAM LINE PRESSURE DUE TO FAILED MAIN TURBINE THROTTLE VALVE DURING THE THROTTLE VALVE TO GOVERNOR VALVE TRANSFER.
EVENT DATE: 081287 REPORT DATE: 011588 NSSS: WE TYPE: PWR
VENDOR: LIMITORQUE CORP.
MOOG INC.

(NSIC 207994) ON 8/12/87 BYRON UNIT 1 WAS AT 6.5 PERCENT POWER RETURNING TO POWER OPERATIONS. DURING THE THROTTLE TO GOVERNOR VALVE TRANSFER ON THE MAIN TURBINE, A THROTTLE VALVE FAILURE INITIATED EVENTS LEADING TO A TURBINE OVERPSEED. MANUAL CONTROL OF THE TURBINE WAS TAKEN IN ORDER TO REDUCE TURBINE SPEED. ONCE THE SPEED WAS REDUCED THE OPERATOR RETURNED THE TURBINE CONTROLLER BACK IN THE AUTO MODE OF OPERATION. WHEN AUTO WAS SELECTED THE GOVERNOR VALVES WENT OPEN RESULTING IN A REACTOR TRIP AND SAFETY INJECTION. THE INTERMEDIATE CAUSE FOR THE THROTTLE VALVE FAILING CLOSED WAS DETERMINED TO BE FAILED MOOG SERVO VALVE. THE ROOT CAUSE FOR THE VALVE'S FAILURE MODE IS STILL UNDER INVESTIGATION AND WILL BE REPORTED IN A SUPPLEMENTAL REPORT. THE DEFECTIVE MOOG SERVO VALVE WAS REPLACED AND A THROTTLE VALVE TO GOVERNOR VALVE TRANSFER WAS SIMULATED REPEATEDLY TO ENSURE THAT THE CORRECTIVE ACTIONS WERE SUCCESSFUL. PLANT STARTUP PROCEDURES WILL BE REVISED TO INSTRUCT THE OPERATOR TO TRIP THE TURBINE IF MANUAL CONTROL HAS TO BE TAKEN PRIOR TO SYNCHRONIZATION IN ORDER TO REINITIALIZE THE TURBINE CONTROLLER. THERE HAS BEEN NO PREVIOUS REPORTABLE EVENTS.

[21] BYRON 1 DOCKET 50-454 LER 87-023
CONTROL ROOM VENTILATION RADIATION MONITOR INOPERABLE DUE TO IMPROPER CALIBRATION RESULTING FROM A PERSONNEL ERROR.
EVENT DATE: 122887 REPORT DATE: 012988 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 2 (PWR)

(NSIC 208044) ON DECEMBER 28, 1987, AND ON JANUARY 5, 1988, THREE SEPARATE RADIATION PROCESS MONITORS WERE CALIBRATED WITH THE WRONG RADIOACTIVE SOURCE. OF THESE THREE MONITORS, ONE OF THEM, THE MAIN CONTROL ROOM OUTSIDE AIR INTAKE GASEOUS RADIATION DETECTOR, ORE PR034B, EXCEEDED THE TECH SPEC REQUIREMENTS. THE FACT THAT THIS INSTRUMENT HAD BEEN INCORRECTLY CALIBRATED WAS DISCOVERED ON JANUARY 6, 1988 WHEN THE SOURCE CONTAINER WAS FOUND TO CONTAIN THE WRONG SOURCE. THE INVESTIGATION THAT FOLLOWED DETERMINED THAT THE MOST PROBABLE CAUSE WAS A COGNITIVE PERSONNEL ERROR ON DECEMBER 16 OR 17, 1987, THAT RESULTED IN TWO SOURCES BEING PLACED IN THE WRONG CONTAINERS. WHEN THE ERROR WAS DISCOVERED, THE SOURCES WERE RETURNED TO THE CORRECT CONTAINERS, APPLICABLE TECH SPEC ACTION REQUIREMENTS WERE INITIATED FOR THE INSTRUMENTS IMPROPERLY CALIBRATED, AND THE CORRECT SOURCE WAS USED TO RECALIBRATE THOSE INSTRUMENTS. ADDITIONALLY, SOURCES ARE NOW BEING STORED IN CLEAR UNMARKED CONTAINERS AND PROCEDURES HAVE BEEN REVISED TO BETTER ADMINISTRATE THE ISSUANCE OF SOURCES. THERE WERE NO SAFETY IMPLICATIONS FROM THIS EVENT, AS REDUNDANT RADIATION MONITORS WERE ALWAYS OPERABLE DURING THIS EVENT. THIS WAS THE FIRST TIME AT BYRON STATION THAT INSTRUMENTS HAVE BEEN CALIBRATED USING A WRONG SOURCE.

[22] BYRON 2 DOCKET 50-455 LER 87-020
UNANTICIPATED WATER DISCHARGE INTO THE PRIMARY SYSTEM FROM THE SAFETY INJECTION ACCUMULATORS DUE TO A PROCEDURAL INADEQUACY.
EVENT DATE: 120287 REPORT DATE: 123087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 1 (PWR)

(NSIC 207651) ON DECEMBER 2, 1987, BYRON UNIT 2 WAS IN THE 7TH DAY OF A PLANNED

OUTAGE WITH THE REACTOR IN MODE 5, COLD SHUTDOWN. THE REACTOR COOLANT SYSTEM WAS DEPRESSURIZED WITH PRESSURIZER LEVEL AT 54%. BOTH TRAINS OF THE SOLID STATE PROTECTION SYSTEM (SSPS) WERE IN THE TEST MODE TO PREVENT ANY INADVERTENT ENGINEERED SAFETY FEATURES ACTUATIONS. IN PREPARATION TO PERFORM THE SAFETY INJECTION ACCUMULATOR BACK-UP CHECK VALVE STROKE TEST OF VALVES 2SI8808A,B,C, AND D, A TEMPORARY LIFT PACKAGE WAS PREPARED TO RESTORE POWER TO THESE VALVES. THESE VALVES ARE ADMINISTRATIVELY TAKEN OUT OF SERVICE CLOSED WHENEVER THE UNIT IS IN COLD SHUTDOWN. IN ADDITION, ACCUMULATOR PRESSURE WAS DECREASED TO APPROXIMATELY 170 PSIG IN PREPARATION FOR THE SURVEILLANCE. AS POWER WAS BEING RESTORED TO THE VALVES, TWO VALVES STARTED TO STROKE OPEN. THE UNIT NUCLEAR STATION OPERATOR (NSO LICENSED) SAW THE VALVES START TO STROKE OPEN AND IMMEDIATELY HELD THE HAND SWITCHES IN THE CLOSED POSITION. THIS ACTION CLOSED THE TWO VALVES THAT HAD OPENED. PRESSURIZER LEVEL INCREASED TO APPROXIMATELY 90%. POWER WAS SUBSEQUENTLY REMOVED FROM THE VALVE ACTUATORS. APPROXIMATELY 4600 GALLONS OF BORATED WATER WAS DISCHARGED INTO THE REACTOR COOLANT SYSTEM. THE CAUSE OF THE EVENT WAS A DEFECTIVE PROCEDURE.

[23] CALLAWAY 1 DOCKET 50-483 LER 87-018 REV 01
UPDATE ON INOPERABLE ESSENTIAL SERVICE WATER SYSTEM AND TECHNICAL SPECIFICATIONS 3.0.3 UNKNOWINGLY ENTERED DUE TO PERSONNEL ERRORS.
EVENT DATE: 081587 REPORT DATE: 122287 NSSS: WE TYPE: PWR
VENDOR: JAMES BURY CORP.

(NSIC 207537) ON 8/15/87 AT 0510 CDT, DURING A CONTAINMENT COOLING FAN TEST, UTILITY OPERATORS DISCOVERED ESSENTIAL SERVICE WATER (ESW) TRAIN 'B' TO THE ULTIMATE HEAT SINK ISOLATION VALVE, EF-V-0117, PARTIALLY SHUT. TRAIN 'B' WAS DECLARED INOPERABLE, THE VALVE WAS OPENED, AND TRAIN 'B' WAS DECLARED OPERABLE AT 1431. AN EVALUATION CONCLUDED THAT TOTAL TRAIN 'B' FLOW WITH THIS RESTRICTION WAS LESS THAN SPECIFIED BY DESIGN. THIS CONDITION HAD EXISTED SINCE 5/11/84 AND ITS POSITION INDICATORS HAD CONFLICTED SINCE NOTED ON A WORK REQUEST (WR) ON 5/14/84. SINCE TRAIN 'A' HAS BEEN REMOVED FROM SERVICE FOR TESTING, BOTH ESW TRAINS HAVE BEEN SIMULTANEOUSLY INOPERABLE AND TECHNICAL SPECIFICATION 3.0.3 UNKNOWINGLY ENTERED. THE PLANT WAS IN MODE 1 POWER OPERATION AT 100% POWER. THIS EVENT WAS DUE TO FAILURE OF UTILITY PERSONNEL TO RECOGNIZE THE EFFECT THE INDICATION PROBLEM HAD ON ESW OPERABILITY RESULTING IN LOW WORK PRIORITY PLACED ON REPAIRING THE PROBLEM. THE CAUSE OF THE DELAY IN DISCOVERING THE FLOW PROBLEM WAS FAILURE OF UTILITY PERSONNEL TO COMPARE TOTAL FLOW TO PRE-OP FLOWS WHEN BASELINING THE PUMPS IN 1984 AND FEBRUARY 1987. THE VALVE ACTUATOR WAS REPAIRED DURING REFUEL II. VOIDED/OPEN WR'S ON SELECTED SYSTEMS WERE REVIEWED AND THE WR VOIDING PROCESS WAS REVISED. PERSONNEL INVOLVED WERE COUNSELED.

[24] CALLAWAY 1 DOCKET 50-483 LER 88-001
REACTOR TRIP ON LOW STEAM GENERATOR LEVEL OSCILLATIONS DURING TROUBLESHOOTING OF A FAULTY AMBER LIGHT CONDITION FOR MAIN FEEDWATER ISOLATION VALVE.
EVENT DATE: 010488 REPORT DATE: 020388 NSSS: WE TYPE: PWR

(NSIC 208000) ON 1/4/88 AT 0236 CST, A REACTOR TRIP OCCURRED AS A RESULT OF LOW LEVEL IN STEAM GENERATOR (S/G) 'A'. THE PLANT WAS IN MODE 1 - POWER OPERATIONS, AT 100 PERCENT REACTOR POWER. THE REACTOR COOLANT SYSTEM (RCS) TEMPERATURE WAS 589F (AVERAGE). RCS PRESSURE WAS 2234 PSIG. THE LOW S/G LEVEL OCCURRED DURING THE TROUBLESHOOTING OF THE S/G 'A' MAIN FEEDWATER ISOLATION VALVE (FWIV), AE-FV-0039, AMBER LIGHT CONDITION ON THE ENGINEERING SAFETY FEATURES (ESF) STATUS PANEL SA066X. WHILE TROUBLE-SHOOTING THE INDICATION CIRCUIT FOR THE ACCUMULATOR PRESSURE SWITCH, AE-PSL-0039A, THE CONTROL VOLTAGE FOR AE-FV-0039 WAS SHORTED BY INSTRUMENTATION AND CONTROL (I&C) TECHNICIANS CAUSING A BLOWN FUSE. THE BLOWN FUSE RESULTED IN VALVE AE-FV-0039 FAST CLOSING. THIS WAS FOLLOWED BY FLOW MISMATCH ON ALL FOUR STEAM GENERATORS, WITH AN ACCOMPANYING REACTOR TRIP ON S/G LO-LO LEVEL IN S/G 'A' AT 0236. THE OPERATORS RECOVERED FROM THE TRIP VIA PLANT PROCEDURES. ALL OF THE AVAILABLE WIRING DETAILS OF THE CIRCUIT WERE NOT USED.

TECHNICIANS REPLACED THE BLOWN FUSE AND SATISFACTORILY COMPLETED THE TROUBLESHOOTING. TO PREVENT RECURRENCE, I&C TECHNICIANS HAVE BEEN INSTRUCTED TO REVIEW DESIGN DRAWINGS BEFORE TROUBLESHOOTING. ADDITIONAL TRAINING WILL BE GIVEN TO I&C TECHNICIANS ON THE SA066 CABINET POWER DISTRIBUTION.

[25] CALVERT CLIFFS 1 DOCKET 50-317 LER 87-009 REV 02
UPDATE ON USE OF FASTENERS IN ASME CLASS 1, 2, & 3 SYSTEMS WITHOUT PROPER CERTIFICATION, SPECIAL NDE, OR SPECIAL MARKING.
EVENT DATE: 042387 REPORT DATE: 020388 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 2 (PWR)

(NSIC 208024) ON APRIL 23, 1987, WITH THE UNIT IN COLD SHUTDOWN, WE DETERMINED THERE WERE INSTANCES WHERE COMMERCIAL-QUALITY FASTENERS WITHOUT THE REQUISITE MATERIAL TRACEABILITY AND CERTIFICATION HAD BEEN INSTALLED IN ASME SECTION XI CLASS 1, 2, AND 3 SYSTEMS. THERE ARE NO SIMILAR EVENTS PREVIOUSLY REPORTED IN AN LER. WE REVIEWED ALL MAINTENANCE WORK PERFORMED ON ASME SECTION XI SYSTEMS SINCE INITIAL PLANT OPERATION (APPROXIMATELY 40,000 MAINTENANCE REQUESTS, OR MRS, WERE INVOLVED) AND FOUND THAT COMMERCIAL QUALITY FASTENERS HAD BEEN INAPPROPRIATELY USED IN 61 CASES. THESE FASTENERS, TOTALLING OVER 1600 STUDS, BOLTS, AND NUTS WERE REMOVED AND REPLACED WITH PROPERLY CERTIFIED FASTENERS. THE REMOVED FASTENERS WERE TESTED FOR STRENGTH AND CHEMISTRY. THE TESTING REVEALED ONLY 16 FASTENERS WERE JUDGED TO FAIL ASTM SPECIFICATIONS. HOWEVER, THE RESULTS SHOWED THAT 115 OF THE FASTENERS WERE MADE OF A MATERIAL DIFFERENT FROM THE MATERIAL GRADE THAT WAS SPECIFIED FOR THE LOCATIONS THEY WERE INSTALLED. ENGINEERING ANALYSES SHOWED THAT IN ALL CASES THE FASTENERS WOULD HAVE PERFORMED THEIR INTENDED FUNCTIONS UNDER ACCIDENT CONDITIONS. THE CAUSES OF THIS EVENT HAVE BEEN IDENTIFIED AND CORRECTIVE ACTION HAS BEEN TAKEN TO PREVENT RECURRENCE.

[26] CALVERT CLIFFS 1 DOCKET 50-317 LER 87-012 REV 01
UPDATE ON FAULTY 500KV CIRCUIT BREAKER OPERATION LEADS TO LOSS OF NONEMERGENCY AC POWER.
EVENT DATE: 072387 REPORT DATE: 120387 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 2 (PWR)
VENDOR: GENERAL ELECTRIC CO.
TERRY STEAM TURBINE COMPANY

(NSIC 207680) AT 1525 ON 7/23/87 A FAULT DEVELOPED ON ONE (TRANSMISSION LINE 5052) OF THE TWO 500KV TRANSMISSION LINES CONNECTING CALVERT CLIFFS NUCLEAR POWER PLANT TO THE COMPANY'S BULK POWER DISTRIBUTION GRID AT WAUGH CHAPEL STATION. THE CIRCUIT BREAKERS FOR LINE 5052 AT WAUGH CHAPEL AND CALVERT CLIFFS TRIPPED TO ISOLATE THE FAULT. IN ADDITION, CIRCUIT BREAKERS AT CALVERT CLIFFS FOR THE OTHER TRANSMISSION LINE (LINE 5051) INCORRECTLY TRIPPED OPEN. THIS RESULTED IN ISOLATING THE GENERATING PLANT FROM THE POWER GRID RESULTING IN BOTH REACTORS TRIPPING ON LOSS OF LOAD OF ALL OFF SITE NONEMERGENCY AC POWER. ALL THREE EMERGENCY DIESEL GENERATORS STARTED AUTOMATICALLY TO POWER THE VITAL 4KV BUSES. EMERGENCY OPERATING PROCEDURES 0 AND 2 WERE INITIATED TO PLACE BOTH REACTORS IN A STABLE CONDITION. AT 1530 AN EMERGENCY RESPONSE "ALERT" CONDITION WAS DECLARED. THE EVENT WAS DOWNGRADED TO AN "UNUSUAL EVENT" AT 1700 UPON COMPLETION OF A CHECKOUT OF THE 500KV SWITCHYARD. AT 1723 OFF SITE ELECTRICAL POWER WAS PROVIDED TO A VITAL 4KV BUS FROM THE ALTERNATE OFF SITE POWER LINE. NORMAL OFF SITE POWER WAS RESTORED AT 1910 VIA TRANSMISSION LINE 5051. THE FAULT ON TRANSMISSION LINE 5052 WAS CAUSED BY A TREE THAT CAME IN CONTACT WITH THE TRANSMISSION LINE. ALL TREES IN THE AREA HAVE BEEN CUT DOWN TO PREVENT RECURRENCE.

[27] CALVERT CLIFFS 2 DOCKET 50-318 LER 87-009
LOSS OF MAIN GENERATOR PERMANENT MAGNET GENERATOR.
EVENT DATE: 122187 REPORT DATE: 012088 NSSS: CE TYPE: PWR
VENDOR: COPES-VULCAN, INC.

WESTINGHOUSE ELECTRIC CORP.

(NSIC 207962) UNIT 2 TRIPPED ON DECEMBER 21, 1987 FROM 100% POWER DUE TO A LOSS OF LOAD SIGNAL. THE LOSS OF LOAD WAS CAUSED BY A FAILURE OF THE UNIT 2 MAIN GENERATOR PERMANENT MAGNET GENERATOR DUE TO THE STATOR FRAME BEING MISALIGNED. DURING THE POST-TRIP COOLDOWN, AN ATMOSPHERIC DUMP VALVE MALFUNCTIONED (FAILED TO FULLY SHUT) CAUSING A MORE RAPID COOLDOWN THAN NORMAL. THE OPERATORS SECURED THE STEAM LOADS TO MINIMIZE COOLDOWN AND SUPPLIED THE STEAM GENERATORS FROM THE AUXILIARY FEEDWATER SYSTEM. THE COOLDOWN WAS TERMINATED WHEN THE ATMOSPHERIC DUMP VALVE WAS MANUALLY ISOLATED. THE FOLLOWING CORRECTIVE ACTIONS ARE BEING TAKEN: 1. APPROPRIATE MAINTENANCE PERSONNEL WILL REVIEW THIS LER THROUGH THE MAINTENANCE AWARENESS PROGRAM. TURBINE-GENERATOR MAINTENANCE PROCEDURES HAVE RECENTLY BEEN SIGNIFICANTLY UPGRADED. 2. THE ATMOSPHERIC DUMP VALVE HAS BEEN REPAIRED AND RETURNED TO SERVICE. FAILED COMPONENTS HAVE BEEN SENT TO THE MATERIALS LABORATORY FOR TESTING.

[28] CATAWBA 1 DOCKET 50-413 LER 86-041 REV 03
UPDATE ON MISSED SURVEILLANCE OF SEVERAL FIRE DETECTION ZONES INSIDE CONTAINMENT DUE TO A MANAGEMENT DEFICIENCY.
EVENT DATE: 080186 REPORT DATE: 020188 NSSS: WE TYPE: PWR

(NSIC 208011) ON AUGUST 1, 1986, AT 1615 HOURS, DURING A SURVEILLANCE REVIEW, IT WAS DISCOVERED THAT SEVERAL REACTOR BUILDING FIRE DETECTION ZONES HAD NOT HAD A TECHNICAL SPECIFICATION REQUIRED SURVEILLANCE TEST PERFORMED AFTER THE UNIT HAD BEEN IN MODE 5, COLD SHUTDOWN FOR MORE THAN 24 HOURS FROM JUNE 15, 1986 TO JUNE 19, 1986. THE TECHNICAL SPECIFICATION WAS VIOLATED ON JUNE 19, 1986. UPON THIS DISCOVERY, THE REACTOR BUILDING ZONES WERE DECLARED INOPERABLE AND CONTAINMENT TEMPERATURE MONITORING WAS STARTED ON AN HOURLY BASIS. THIS EVENT WAS DISCOVERED WITH UNIT 1 AT 70% POWER. THIS INCIDENT IS ATTRIBUTED TO A MANAGEMENT DEFICIENCY. RESPONSIBILITIES WITHIN THE MAINTENANCE GROUP FOR SURVEILLANCE ACTIVITIES, INCLUDING OVERDUE SURVEILLANCES, WERE NOT DEFINED. THEREFORE, THE FIRE DETECTION SURVEILLANCE WAS NOT RECOGNIZED AS INCOMPLETE WHEN THE UNIT ENTERED MODE 4, HOT SHUTDOWN, AFTER BEING IN MODE 5 FOR MORE THAN 24 HOURS. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS EVENT.

[29] CATAWBA 1 DOCKET 50-413 LER 86-059 REV 02
UPDATE ON CONTAINMENT PRESSURE CHANNEL UNKNOWINGLY INOPERABLE DUE TO UNKNOWN CAUSE.
EVENT DATE: 120686 REPORT DATE: 020288 NSSS: WE TYPE: PWR

(NSIC 208012) ON 11/19/86, WITH THE UNIT IN MODE 2, STARTUP, IT WAS NOTICED THAT ONE OF THE 4 NARROW RANGE CONTAINMENT PRESSURE GAUGES WAS NOT MOVING WITH CHANGES IN OTHER CONTAINMENT PRESSURE INDICATIONS. AT 0630 HOURS, A WORK REQUEST WAS INITIATED TO INVESTIGATE AND REPAIR THE CAUSE OF THIS PROBLEM. AT 1000 HOURS, THE SHIFT TECHNICAL ADVISOR LOGGED THE PRESSURE GAUGE INOPERABLE IN THE TECH SPEC ACTION ITEM LOG UNDER THE ACCIDENT MONITORING INSTRUMENTATION TECH SPEC. SUBSEQUENT INVESTIGATION ON 12/6/86, WITH THE UNIT AT 100% POWER, REVEALED THAT THE ISOLATION VALVE FOR THE PRESSURE TRANSMITTER IN THE LOOP WAS CLOSED. THE VALVE HAD BEEN VERIFIED OPEN ON 10/28/86, SO IT WAS APPARENTLY CLOSED BETWEEN THEN AND 11/19/86. THIS VALVE BEING CLOSED RESULTED IN A VIOLATION OF TECH SPECS WITH A CONTAINMENT PRESSURE CHANNEL BEING UNKNOWINGLY INOPERABLE AND NOT PLACED IN THE TRIPPED CONDITION, WITHIN THE REQUIRED TIME PERIOD. THE EXACT TIME THE ISOLATION VALVE WAS LEFT CLOSED COULD NOT BE IDENTIFIED DUE TO LACK OF EVIDENCE. THE VALVE HANDLE WAS SUBSEQUENTLY VERIFIED TO BE CORRECTLY INSTALLED ON THE VALVE STEM AS A RESULT OF A LATER INCIDENT INVOLVING A SIMILAR VALVE (SEE LER 413/87-18). THEREFORE, THE ROOT CAUSE OF THIS INCIDENT REMAINS UNKNOWN. A PERSONNEL ERROR CONTRIBUTED TO THIS INCIDENT.

[30] CATAWBA 1 DOCKET 50-413 LER 87-041
SEMI-ANNUAL NON-PRELUBED START PERIODIC TECHNICAL SPECIFICATION REQUIRED TESTS FOR DIESEL GENERATORS 1B AND 2B MISSED DUE TO PERSONNEL ERRORS.
EVENT DATE: 061687 REPORT DATE: 121787 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 207647) ON JULY 16, 1987, WHILE PERFORMING DIESEL GENERATOR (D/G) 1B OPERABILITY TEST, AND ON JULY 22, 1987, WHILE PERFORMING D/G 2B OPERABILITY TEST, NUCLEAR EQUIPMENT OPERATORS (NEOS) PERFORMED PRELUBED D/G STARTS INSTEAD OF THE REQUIRED NON-PRELUBED STARTS AND CONSEQUENTLY MISSED THE REQUIRED SEMI-ANNUAL SURVEILLANCES. DUKE POWER PERSONNEL DETERMINED THIS EVENT TO BE REPORTABLE ON NOVEMBER 17, 1987. THE SURVEILLANCES WERE SATISFIED ON OCTOBER 7, 1987, FOR D/G 1B AND ON NOVEMBER 17, 1987, FOR D/G 2B. BOTH UNITS OPERATED IN MODE 1, POWER OPERATION, MODE 2, STARTUP, MODE 3, HOT STANDBY, MODE 4, HOT SHUTDOWN, AND MODE 5, COLD SHUTDOWN, DURING THE PERIODS IN WHICH THE SURVEILLANCES WERE NOT PERFORMED. THESE INCIDENTS ARE ATTRIBUTED TO PERSONNEL ERRORS. THE INVOLVED NEOS DID NOT PERFORM THE PROPER STEP IN EACH PROCEDURE AND THEIR SUPERVISORS DID NOT DISCOVER THE ERROR DURING THEIR REVIEWS. THE SURVEILLANCE WAS PERFORMED FOR D/G 2B FOLLOWING DISCOVERY. THE SURVEILLANCE FOR D/G 1B WAS FOUND TO HAVE BEEN SATISFIED BY A NON-PRELUBED START PERFORMED ON OCTOBER 7, 1987. THESE INCIDENTS WERE REVIEWED WITH INVOLVED PERSONNEL WITH EMPHASIS ON ATTENTION TO DETAIL. FUTURE NON-PRELUBED START REQUIREMENTS WILL BE BETTER IDENTIFIED ON THE OPERATIONS WORKLIST.

[31] CATAWBA 1 DOCKET 50-413 LER 87-035 REV 01
UPDATE ON POTENTIAL CONTROL ROOM AREA VENTILATION AND CHILLED WATER SYSTEM AND NUCLEAR SERVICE WATER PUMP INOPERABILITY DURING D/G LOAD SEQUENCER TESTING DUE TO A PROCEDURAL DEFICIENCY.
EVENT DATE: 081087 REPORT DATE: 020288 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 208055) ON AUGUST 10, 1987, AT 1000 HOURS WHILE PREPARING A TEST PROCEDURE FOR RETYPE, IT WAS DISCOVERED THAT PLACING A DIESEL GENERATOR (D/G) LOAD SEQUENCER IN TEST ON ONE UNIT WOULD PREVENT ACTUATION OF THE CONTROL ROOM AREA VENTILATION (VC) CHILLED WATER (YC) SYSTEM ON THAT TRAIN IF A LOSS OF COOLANT ACCIDENT (LOCA) SIGNAL WAS RECEIVED ON THE OTHER UNIT. ON OCTOBER 21, 1987, THE PERFORMANCE STAFF ENGINEER WAS PERFORMING ADDITIONAL INVESTIGATION FOR SIMILAR PROBLEMS AND ALSO DISCOVERED THAT PLACING A D/G LOAD SEQUENCER IN TEST ON ONE UNIT WOULD PREVENT THE START OF THAT TRAIN'S NUCLEAR SERVICE WATER (RN) SYSTEM PUMP IF A LOCA OR BLACKOUT WAS RECEIVED ON THE OTHER UNIT, OR DURING A LOW RN PIT LEVEL SWAPOVER TO THE STANDBY NUCLEAR SERVICE WATER POND (SNSWP). BOTH UNITS HAD OPERATED IN ALL MODES PRIOR TO DISCOVERY OF THE EVENT. ALTHOUGH THE EVENT IS NOT REPORTABLE, ON AUGUST 26, 1987, DUKE POWER DECIDED TO SUBMIT THIS REPORT AS A VOLUNTARY LER FOR INFORMATION PURPOSES. THIS INCIDENT IS ATTRIBUTED TO A DEFECTIVE PROCEDURE. VARIOUS STATION PROCEDURES THAT PLACED THE D/G LOAD SEQUENCERS IN TEST DID NOT ENSURE THAT THE CORRESPONDING VC/YC ON RN PUMP WAS DECLARED INOPERABLE. APPROPRIATE PROCEDURES WILL BE REVISED TO ENSURE THAT THE APPLICABLE TRAIN OF VC/YC AND RN PUMP ARE DECLARED INOPERABLE WHEN REQUIRED.

[32] CATAWBA 1 DOCKET 50-413 LER 87-037 REV 01
UPDATE ON FAILURE OF ITT GRINNELL MINI-STIFF PIPE CLAMPS DUE TO MANUFACTURING DEFICIENCY.
EVENT DATE: 090987 REPORT DATE: 011588 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)
VENDOR: GRINNELL INDUSTRIAL PIPING, INC.

(NSIC 207977) ON 9/10/86 DURING CATAWBA NUCLEAR STATION'S UNIT 1 END-OF-CYCLE 1 REFUELING OUTAGE, VISUAL INSPECTION OF SNUBBERS PER TECH SPEC 4.7.8.B REVEALED TWO BROKEN PIPE SUPPORTS WHICH INCORPORATED A FIGURE 214N "MINISTIFF CLAMP"

ESPECIALLY MANUFACTURED FOR CATAWBA NUCLEAR STATION. THERE ARE 1500 FIGURE 214N MINISTIFF CLAMPS INSTALLED AT CATAWBA RANGING FROM 1/2" TO 4" IN SIZE. THE FRACTURED PART OF THE "MINISTIFF CLAMP" WAS THE METAL STRAP WHICH WRAPS AROUND THE PIPE. THIS STRAP IS FABRICATED FROM A PRECIPITATION HARDENED STAINLESS STEEL, HEAT TREATED AT 1975F. FAILURE ANALYSES INDICATED AN IMPROPER HEAT TREATMENT OF THE STRAP. THIS RESULTED IN A HIGHER STRENGTH MATERIAL, BUT LOWERED ITS FRACTURE TOUGHNESS AND STRESS CORROSION CRACKING RESISTANCE. ON SEPTEMBER 9, 1987, DUKE POWER DISCOVERED THAT OVER SEVERAL YEARS THE ABILITY OF AFFECTED HANGERS TO PERFORM THEIR INTENDED FUNCTION COULD BE POTENTIALLY DEGRADED. AN OPERABILITY ANALYSIS WAS PERFORMED AND IT WAS DETERMINED THAT PLANT OPERATION WAS UNAFFECTED BY THE FAILED PIPE SUPPORTS. BOTH UNITS HAVE BEEN IN ALL MODES OF OPERATION SINCE THE PIPE CLAMPS WERE INSTALLED. THIS INCIDENT IS ATTRIBUTED TO A MANUFACTURING DEFICIENCY BECAUSE THE MATERIAL WAS IMPROPERLY HEAT TREATED.

[33] CATAWBA 1 DOCKET 50-413 LER 87-045
TECHNICAL SPECIFICATION VIOLATION DUE TO THE CONTAINMENT PURGE SYSTEM BEING INOPERABLE BECAUSE OF A PROCEDURAL DEFICIENCY AND A PERSONNEL ERROR.
EVENT DATE: 101887 REPORT DATE: 010788 NSSS: WE TYPE: PWR

(NSIC 207694) ON 10/18/87 WHILE DUKE POWER PERSONNEL WERE CONDUCTING THE CONTAINMENT PURGE SYSTEM (VP) AUTOMATIC VALVE ACTUATION VERIFICATION PROCEDURE, TRAIN B HIGH HUMIDITY TRIP FAILED TO ACTUATE AS REQUIRED. DUKE POWER PERSONNEL FOUND A FUSED JUMPER INSTALLED IN AN ELECTRICAL PANEL WHICH PREVENTED THE TRIP FROM ACTUATING. UPON REMOVAL OF THE JUMPER, THE PROCEDURE WAS SUCCESSFULLY COMPLETED. A REVIEW OF VP WORK PERFORMED FROM THE LAST SUCCESSFUL PERFORMANCE OF THE PROCEDURE ON 9/26/86, UNTIL THE JUMPER WAS DISCOVERED, REVEALED ONLY ONE WORK REQUEST, ISSUED ON 9/27/86, THAT REQUIRED FUSED JUMPERS TO BE PLACED IN THIS PANEL. DUKE POWER PERSONNEL DETERMINED ON 12/8/87 THIS EVENT WAS A VIOLATION OF THE TS WHICH REQUIRE THAT UPPER AND LOWER VP SUPPLY AND EXHAUST VALVES CLOSE ON A HIGH HUMIDITY SIGNAL DURING FUEL MOVEMENT OR CORE ALTERATIONS UNLESS THE CONTAINMENT ISOLATION VALVES ARE CLOSED. THE UNIT WAS IN MODE 6, REFUELING, AND CORE ALTERATIONS WERE IN PROGRESS AT THE TIME OF DISCOVERY. TRAIN A WAS AVAILABLE AT ALL TIMES DURING CORE ALTERATIONS. THIS INCIDENT IS ATTRIBUTED TO A DEFECTIVE PROCEDURE, DUE TO THE GENERAL TROUBLESHOOTING PROCEDURE NOT ADEQUATELY ADDRESSING OPERABILITY EVALUATION WHILE TROUBLESHOOTING WITH FUSED JUMPERS. THIS INCIDENT IS ALSO ATTRIBUTED TO A PERSONNEL ERROR BECAUSE THE JUMPER WAS LEFT IN PLACE FOLLOWING TROUBLE SHOOTING OR REPAIR.

[34] CATAWBA 1 DOCKET 50-413 LER 87-043
TECHNICAL SPECIFICATION VIOLATION DUE TO INOPERABLE REACTOR COOLANT SYSTEM INSTRUMENTATION BECAUSE OF A DESIGN DEFICIENCY.
EVENT DATE: 110587 REPORT DATE: 123087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 207692) ON 11/87, WHILE PERFORMING AN INSPECTION OF SAFETY RELATED INSTRUMENTATION CABLE TERMINATION SPLICES, A DUKE POWER ELECTRICAL DESIGN ENGINEER DISCOVERED THAT THE UNIT 1 REACTOR COOLANT (NC) SYSTEM WIDE RANGE (W/R) RESISTANCE TEMPERATURE DETECTOR'S (RTD'S) CABLE TERMINATION JUNCTION BOXES WERE NOT SEALED. FOLLOWING A REVIEW OF THE PROBLEM, STATION MANAGEMENT WAS NOTIFIED OF THE POSSIBLE INOPERABILITY OF THE UNIT 1 AND UNIT 2 W/R HOT AND COLD LEG RTDS ON 11/13/87. UNIT 1 AND UNIT 2 HAVE OPERATED IN ALL MODES OF OPERATION WITH THE AFFECTED W/R RTDS BEING TECHNICALLY INOPERABLE BEFORE COMPENSATORY MEASURES WERE IMPLEMENTED. THE UNIT 2 RTDS WERE CONSERVATIVELY DECLARED INOPERABLE AT 1530 HOURS. AT APPROXIMATELY 1800 HOURS, A STATEMENT OF OPERABILITY WAS ISSUED WHICH JUSTIFIED OPERABILITY OF THE UNIT 2 W/R HOT LEG RTDS THROUGH THE USE OF COMPENSATORY MEASURES IN THE EVENT OF A HIGH ENERGY LINE BREAK INSIDE CONTAINMENT. THE LOOP 2A AND 2B COLD LEG RTDS WERE DETERMINED NOT TO HAVE AN OPERABILITY CONCERN DUE TO THEIR CABLES BEING ENVIRONMENTALLY QUALIFIED. CONTROL ROOM PERSONNEL WERE TRAINED ON THE REQUIRED COMPENSATORY MEASURES AND THE UNIT 2

RTDS WERE DECLARED OPERABLE ON 11/14/87, AT 0800 HOURS. DUKE POWER STATION MANAGEMENT PERSONNEL DETERMINED THIS EVENT TO BE REPORTABLE ON 11/30/87.

[35] CATAWBA 1 DOCKET 50-413 LER 87-042
DIESEL GENERATOR AUTO START AND SUBSEQUENT FAILURE OF AN EMERGENCY LOAD GROUP TO ENERGIZE DUE TO EQUIPMENT MALFUNCTIONS.
EVENT DATE: 111787 REPORT DATE: 121787 NSSS: WE TYPE: PWR
VENDOR: CUTLER-HAMMER
WESTINGHOUSE ELECTRIC CORP.

(NSIC 207662) ON NOVEMBER 17, 1987, DUKE POWER PERSONNEL WERE TESTING A 50F COOLING FAN CONTROL RELAY IN 6.9KV BREAKER LTD-5 WHEN THE BREAKER TRIPPED AND LOCKED OUT AT 1102 HOURS DUE TO AN OUT OF ADJUSTMENT WIPER THAT SHORTS TWO TERMINALS IN THE RELAY BEING STRAINED DUE TO THE TESTING. THE MALFUNCTIONING RELAY CAUSED A PHASE TO PHASE CURRENT INVALANCE AND ACTIVATED A GROUND FAULT RELAY WHICH TRIPPED LTD-5. AS A RESULT POWER TO 4.16 KV ESSENTIAL BUS 1ETB WAS LOST AND CAUSED A DIESEL GENERATOR 1B AUTO START. ESSENTIAL 600V LOAD CENTERS 1ELVB AND 1ELXD DID NOT ENERGIZE AS REQUIRED BECAUSE AN 8.5 SECOND UNDERVOLTAGE TIMER HAD DRIFTED TO 9.7 SECONDS AT A PREVIOUS TIME. THEREFORE SINCE THE 1 SECOND LOAD SHED TRIP SIGNAL WHICH FOLLOWS THE TIMING OUT OF THE 8.5 SECONDS TEST TIMER WAS STILL AVAILABLE ON THE INCOMING BREAKERS OF LOAD GROUP 1 WHEN THE D/G SEQUENCER TRIED TO CLOSE FOLLOWING THE EXPIRATION OF ITS 10 SECOND TIMER (WHICH RUNS SIMULTANEOUSLY TO THE 8.5 SECOND UNDERVOLTAGE TIMER) 1ELXB AND 1ELXD FAILED TO ENERGIZE. THE OPERATOR AT THE CONTROLS MANUALLY ENERGIZED 1ELXB AND 1ELXD APPROXIMATELY 20 MINUTES AFTER INITIATION OF THE EVENT. THE WIPER ON THE 50F RELAY WAS READJUSTED AND SATISFACTORILY TESTED. APPROPRIATE DUKE POWER PERSONNEL HAVE BEEN TRAINED TO DISABLE GROUND FAULT RELAYS PRIOR TO TESTING PHASE RELAYS.

[36] CATAWBA 1 DOCKET 50-413 LER 87-044
TECHNICAL SPECIFICATION VIOLATION DUE TO MANAGEMENT DEFICIENCY AND PERSONNEL ERROR.
EVENT DATE: 120187 REPORT DATE: 123187 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 207693) ON 12/1/87, AT 0200 HOURS, DIESEL GENERATOR (D/G) 1B WAS DECLARED INOPERABLE AFTER IT TRIPPED DURING AN ENGINEERED SAFETY FEATURE PERIODIC TEST. AT THE TIME, D/G 1B WAS THE DESIGNATED EMERGENCY POWER SUPPLY FOR CONTROL ROOM AREA VENTILATION (VC)/CHILLED WATER (YC) SYSTEM TRAIN B. AT 0400 HOURS, VC/YC TRAIN A WAS DECLARED INOPERABLE IN ORDER TO ALLOW PREVENTIVE MAINTENANCE WORK TO BE PERFORMED. THIS SITUATION RESULTED IN BOTH TRAINS OF VC/YC BEING INCAPABLE OF PERFORMING THEIR SAFETY FUNCTION HAD A LOSS OF OFFSITE POWER OCCURRED. UNIT 1 WAS IN MODE 5, COLD SHUTDOWN, AND UNIT 2 WAS IN MODE 1, POWER OPERATION, AT 75% POWER DURING THIS INCIDENT. THIS INCIDENT HAS BEEN ATTRIBUTED TO A MANAGEMENT DEFICIENCY BECAUSE THE SHIFT SUPERVISOR ALLOWED VC/YC TRAIN A TO BE REMOVED FROM SERVICE WHILE TRAIN B HAD NO EMERGENCY POWER. ADDITIONALLY, EMERGENCY POWER REQUIREMENTS FOR VC/YC ARE NOT ADEQUATELY ADDRESSED IN TECH SPECS WHEN THE UNIT ASSOCIATED WITH AN INOPERABLE D/G IS BELOW MODE 4. A PERSONNEL ERROR WAS ALSO INVOLVED BECAUSE A NUCLEAR CONTROL OPERATOR INITIALED A STEP IN THE NUCLEAR SERVICE WATER (RN) SYSTEM PROCEDURE CERTIFYING THAT BOTH D/G 1B AND 2A HAD BEEN REMOVED FROM SERVICE PER THE DIESEL GENERATOR OPERATION PROCEDURE. HOWEVER, THE PROCEDURE WAS NOT COMPLETED UNTIL THREE HOURS LATER.

[37] CATAWBA 1 DOCKET 50-413 LER 87-046
MAIN FEEDWATER ISOLATION DURING REMOVAL OF A TEMPORARY STATION MODIFICATION DUE TO A MANAGEMENT DEFICIENCY.
EVENT DATE: 121987 REPORT DATE: 011888 NSSS: WE TYPE: PWR

(NSIC 207914) ON DECEMBER 19, 1987, AT 1322:24 HOURS, A MAIN FEEDWATER (CF)

ISOLATION OCCURRED DURING THE REMOVAL OF A TEMPORARY STATION MODIFICATION (TSM) FROM THE SOLID STATE PROTECTION SYSTEM (SSPS). THE TSM HAD BEEN INSTALLED TO PREVENT A CF ISOLATION WHEN THE STEAM GENERATORS WERE PLACED IN WET LAY-UP FOR CHEMISTRY CONTROL DURING END OF CYCLE 2 REFUELING OUTAGE. THE UNIT WAS IN MODE 5, COLD SHUTDOWN, WITH THE DECAY HEAT REMOVAL SYSTEM IN OPERATION DURING THIS INCIDENT. THIS INCIDENT IS ATTRIBUTED TO A MANAGEMENT DEFICIENCY. THE INVOLVED TECHNICIANS WERE GIVEN INCORRECT INSTRUCTIONS BY THEIR SUPERVISOR TO REMOVE THE TSM. THEY ALSO HAD NO PREVIOUS TRAINING ON THE SSPS AND WERE ASSIGNED A JOB ON EQUIPMENT THEY WERE UNFAMILIAR WITH. THE TSM WAS REMOVED AND THE CF SYSTEM WAS RETURNED TO ITS PREVIOUS ALIGNMENT. THIS INCIDENT IS BEING REVIEWED WITH ALL APPROPRIATE SUPERVISORS. ADDITIONALLY, TECHNICIANS PERFORMING MAINTENANCE ON ENGINEERED SAFEGUARDS FEATURE (ESF) OR REACTOR PROTECTION SYSTEM (RPS) EQUIPMENT ARE TO BE TRAINED ON THE EQUIPMENT OR BE UNDER THE DIRECT SUPERVISION OF TRAINED PERSONNEL AS IT MAY BE APPROPRIATE. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS EVENT.

[38] CATAWBA 1 DOCKET 50-413 LER 87-047
ENGINEERED SAFETY FEATURE ACTUATION ON TWO OCCASIONS WHILE OPENING MAIN STEAM ISOLATION VALVE DUE TO UNKNOWN CAUSE AND A PERSONNEL ERROR.
EVENT DATE: 122187 REPORT DATE: 012088 NSSS: WE TYPE: PWR

(NSIC 208056) ON DECEMBER 21, 1987, AT 1516:58 HOURS WHILE UNIT 1 WAS IN MODE 5 (COLD SHUTDOWN), A REACTOR TRIP SIGNAL OCCURRED DUE TO STEAM GENERATOR (S/G) LOW LOW LEVEL. THE TRIP SIGNAL WAS FOLLOWED BY A FEEDWATER ISOLATION SIGNAL DUE TO REACTOR TRIP WITH TEMPERATURE IN THE REACTOR COOLANT (NC) SYSTEM BELOW 564 DEGREES F. THE CONTROL ROOM OPERATORS (CROS) WERE PERFORMING THE CONTROLLING PROCEDURE FOR UNIT STARTUP AND WERE OPENING THE S/G 1D MAIN STEAM ISOLATION VALVE (MSIV) WHEN THIS INCIDENT OCCURRED. ON DECEMBER 24, 1987 WITH THE UNIT IN MODE 4 (HOT SHUTDOWN), AT 1321:21 HOURS, THE CROS WERE AGAIN OPENING THE MSIVS WHICH HAD BEEN CLOSED TO SUPPORT TURBINE TRIP TESTING. WHEN THE S/G LD MSIV WAS OPENED, SUFFICIENT DIFFERENTIAL PRESSURE EXISTED TO CAUSE A HIGH HIGH LEVEL CONDITION, WHICH ACTUATED A FEEDWATER ISOLATION/TURBINE TRIP SIGNAL. THE ROOT CAUSE OF THE FIRST PART OF THE EVENT REMAINS UNKNOWN AT THIS TIME. THE SECOND INCIDENT HAS BEEN ATTRIBUTED TO A PERSONNEL ERROR. THE ASSISTANT NUCLEAR CONTROL OPERATOR DID NOT FOLLOW AN EXISTING PROCEDURE TO EQUALIZE PRESSURE ACROSS THE MSIVS AFTER THE TURBINE TRIP TEST. IN BOTH INCIDENTS, THE CROS VERIFIED THAT THE AUTOMATIC ACTUATIONS HAD OCCURRED CORRECTLY AND THAT S/G LEVELS HAD STABILIZED.

[39] CATAWBA 2 DOCKET 50-414 LER 86-027 REV 01
UPDATE ON CONTAINMENT AIR RELEASE TERMINATION DUE TO INSTALLATION DEFICIENCY.
EVENT DATE: 062786 REPORT DATE: 020288 NSSS: WE TYPE: PWR

(NSIC 208013) ON 6/27/86, AT 0655 HOURS, THE CONTAINMENT GAS MONITOR (2EMF39L) ALARMED, RESULTING IN AN ENGINEERED SAFEGUARDS FEATURES (ESF) ACTUATION THAT TERMINATED A CONTAINMENT AIR RELEASE. ISOLATION VALVES ON THE CONTAINMENT AIR RELEASE AND ADDITION (VQ) SYSTEM WERE AUTOMATICALLY CLOSED DUE TO THE ESF SIGNAL. A GRAB SAMPLE HAD BEEN TAKEN PRIOR TO THE START OF THE RELEASE. THIS SAMPLE INDICATED THAT THE EMF SETPOINTS WERE APPROPRIATE AND THAT THE RELEASE WOULD BE WITHIN TECH SPEC LIMITS. THE UNIT WAS IN MODE 1 AT 24% POWER AT THE TIME OF THIS INCIDENT THIS INCIDENT IS ATTRIBUTED TO AN INSTALLATION DEFICIENCY. THE 2EMF39L CABINET WAS DISCOVERED TO BE IMPROPERLY GROUNDED TO THE STATION GROUND SYSTEM INSTEAD OF TO THE INSTRUMENT GROUND SYSTEM AS SPECIFIED BY DESIGN DRAWINGS. FOLLOWING THE RELEASE TERMINATION, HEALTH PHYSICS PERSONNEL SAMPLED CONTAINMENT ATMOSPHERE AND CALCULATED NEW EMF SETPOINTS. DUKE POWER PERSONNEL HAVE CORRECTED THE GROUNDING PROBLEMS. A NUCLEAR STATION MODIFICATION HAS BEEN INITIATED TO INSTALL A DIGITAL MONITOR WHICH WILL PROVIDE MORE ACCURATE SETPOINT DETERMINATION. THIS INCIDENT WAS ORIGINALLY REPORTED PURSUANT TO 10 CFR 50.73, SECTION (A)(2)(IV) AND 10 CFR 50.72, SECTION (B)(2)(II). HOWEVER, SINCE THE TIME THIS INCIDENT OCCURRED, VQ RELEASE TERMINATIONS DUE TO HIGH RADIATION SIGNALS

FROM THE CONTAINMENT RADIATION MONITOR HAVE BEEN DETERMINED TO BE A NORMAL FUNCTION OF THE VQ SYSTEM.

[40] CATAWBA 2 DOCKET 50-414 LER 87-023 REV 02
UPDATE ON UNUSUAL EVENT DECLARED BECAUSE OF UNISOLABLE CONTAINMENT VALVE DUE TO A MANAGEMENT DEFICIENCY.
EVENT DATE: 080787 REPORT DATE: 020188 NSSS: WE TYPE: PWR
VENDOR: BORG-WARNER CORP.

(NSIC 207978) ON 8/7/87, AT 0340 HOURS, WITH UNIT 2 AT 85% POWER, A UNIT SHUTDOWN WAS COMMENCED PER TECH SPECS WHEN ONE OF THE STEAM GENERATOR (S/G) MAIN FEEDWATER BYPASS TO AUXILIARY FEEDWATER (CA) NOZZLE VALVES WAS DISCOVERED PARTIALLY OPEN AND UNISOLABLE WHILE PERFORMING THE RETEST FOR THE TURBINE DRIVEN CA PUMP DISCHARGE CHECK VALVE TO S/G 2B. DUKE POWER STATION PERSONNEL DECLARED THE VALVE INOPERABLE, COMMENCED UNIT SHUTDOWN, AND DECLARED AN UNUSUAL EVENT AS REQUIRED BY TECH SPEC. THE UNIT ENTERED MODE 3, HOT STANDBY, AT 0735 HOURS. THE VALVE WAS REPAIRED AND SUCCESSFULLY RETESTED AT 1854 HOURS. THE STATION WAS SECURED FROM THE UNUSUAL EVENT AT 1915 HOURS. DUKE POWER PERSONNEL HAD ORIGINATED A WORK REQUEST ON 9/9/86, AT 1500 HOURS TO INVESTIGATE/REPAIR 2CA150 PASSING APPROXIMATELY 300 GPM WHILE INDICATING CLOSE. THE UNIT WAS IN MODE 5, COLD SHUTDOWN, AT THE TIME AND THE VALVE WAS NOT REQUIRED TO BE OPERABLE. THIS WORK REQUEST WAS SCHEDULED FOR COMPLETION DURING THE UNIT'S FIRST REFUELING OUTAGE. THIS INCIDENT IS ATTRIBUTED TO A MANAGEMENT DEFICIENCY. THE FAILURE OF THE VALVE WAS DUE TO THE IMPROPER SETTING OF THE AIR ACTUATOR FOLLOWING MAINTENANCE ACTIVITY. DUKE POWER PERSONNEL HAD NOT RECEIVED ADEQUATE TRAINING TO ACCOMPLISH VALVE MAINTENANCE AND HAD NOT BEEN QUALIFIED TO PERFORM THE PROCEDURE.

[41] CATAWBA 2 DOCKET 50-414 LER 87-031
DIESEL GENERATORS RENDERED INOPERABLE BECAUSE RETESTS WERE MISSED DUE TO PERSONNEL ERROR.
EVENT DATE: 122187 REPORT DATE: 012088 NSSS: WE TYPE: PWR
VENDOR: CALCON

(NSIC 207995) ON 1/21/87, AT 1600 HOURS WHILE PERFORMING MAINTENANCE ON DIESEL GENERATOR (D/G) 2A, DUKE POWER PERSONNEL DISCOVERED THAT D/G 2B STARTING AIR (VG) INLET VALVES HAD BEEN REPAIRED ON 11/19/87 AND THAT NO PERFORMANCE (PRF) RETEST HAD BEEN CONDUCTED CONTRARY TO TECH SPECS. SUBSEQUENT REVIEW OF WORK HISTORY FOR ALL D/G VG INLET VALVES REVEALED A TOTAL OF SIX INSTANCES OF MISSED RETESTS FOLLOWING MAINTENANCE (ALL ON THE UNIT 2 D/GS) BETWEEN 7/11/86 AND 12/21/87. UNIT 2 WAS OPERATING AT 65% POWER AT THE TIME OF DISCOVERY AND IT HAD BEEN IN ALL MODES EXCEPT MODE 6, REFUELING, WITH D/G INLET VALVES BEING UNKNOWINGLY TECHNICALLY INOPERABLE. AS A RESULT, D/G 2A WAS TECHNICALLY INOPERABLE BETWEEN 8/24/87 AND 12/21/87 AND D/G 2B WAS TECHNICALLY INOPERABLE BETWEEN 11/19/87 AND 12/21/87. THE OPERABILITY OF D/G 2A WAS VERIFIED BY 38 STARTS AND THE OPERABILITY OF D/G 2B WAS VERIFIED BY 13 STARTS DURING THE RESPECTIVE INTERVALS. THE ACTION STATEMENTS OF TECH SPEC 3.8.1.1 WERE NOT ENTERED. THE D/G 2B PRF RETEST WAS SATISFACTORILY COMPLETED ON 12/21/87 AT 1050 HOURS, AND THE D/G 2A PRF RETEST WAS SATISFACTORILY COMPLETED BY 1812 HOURS. THIS INCIDENT IS ATTRIBUTED TO PERSONNEL ERRORS IN FAILING TO ADEQUATELY REVIEW THE PUMP AND VALVE INSERVICE INSPECTION PROGRAM MANUAL.

[42] CATAWBA 2 DOCKET 50-414 LER 87-032
BOTH CHANNELS OF SOURCE RANGE DETECTORS INOPERABLE DUE TO A MANAGEMENT DEFICIENCY.
EVENT DATE: 122987 REPORT DATE: 012988 NSSS: WE TYPE: PWR

(NSIC 208041) ON 12/29/87, AT APPROXIMATELY 0925 HOURS, DUKE POWER INSTRUMENTATION AND ELECTRICAL (IAE) TECHNICIANS REMOVE THE FUSES IN THE CONTROL ROOM CABINET FOR SOURCE/INTERMEDIATE RANGE DETECTORS N31/35 IN PREPARATION FOR

DETECTOR ASSEMBLY REPLACEMENT. THE RESPONSIBLE IAE SUPERVISOR AND TECHNICIANS WENT TO THE DETECTOR LOCATION AND BEGAN TO DISCONNECT THE DETECTORS. AT 1032 HOURS, THE CONTROL ROOM LOST INDICATION ON N32/36, AND CONTROL ROOM OPERATORS (CROS) IMPLEMENTED THE MALFUNCTION OF NUCLEAR INSTRUMENTATION SYSTEM ABNORMAL PROCEDURE. AT 1040 HOURS, THE IAE PERSONNEL DISCOVERED THEY HAD DISCONNECTED THE WRONG DETECTORS AND INFORMED THE CROS. N31/35 WERE RETURNED TO SERVICE AT 1041 HOURS, AND WORK CONTINUED ON N32/36. THE UNIT WAS IN MODE 5, COLD SHUTDOWN, AT THE TIME OF THIS INCIDENT. THIS INCIDENT IS ATTRIBUTED TO A MANAGEMENT DEFICIENCY. UNDER THE DIRECTION OF THE RESPONSIBLE IAE SUPERVISOR, THE IAE TECHNICIANS DISCONNECTED THE WRONG SOURCE AND INTERMEDIATE RANGE DETECTORS, WHICH RESULTED IN ALL SOURCE RANGE DETECTORS BEING INOPERABLE WHEN REQUIRED. CROS EVACUATED CONTAINMENT AND VERIFIED ADEQUATE SHUTDOWN MARGIN AS REQUIRED BY THE MALFUNCTION OF NUCLEAR INSTRUMENTATION SYSTEM ABNORMAL PROCEDURE. N31/25 WERE RETURNED TO SERVICE BY IAE PERSONNEL.

[43]   CLINTON 1   DOCKET 50-461   LER 87-063
INOPERABLE DIVISION 1 REACTOR VESSEL PRESSURE TRANSMITTER DUE TO FAILURE OF CONTROL AND INSTRUMENTATION TECHNICIANS TO FOLLOW PROCEDURE DURING VALVE OPERATION.
EVENT DATE: 112287   REPORT DATE: 121787   NSSS: GE   TYPE: BWR

(NSIC 207532) ON NOVEMBER 22, 1987, WITH THE PLANT IN MODE 2 (STARTUP), THE INSTRUMENT ROOT VALVE FOR THE DIVISION 1 REACTOR VESSEL PRESSURE TRANSMITTER WAS DISCOVERED CLOSED, THUS ISOLATING THE TRANSMITTER. THIS WAS DISCOVERED DURING THE COURSE OF PLANT HEATUP WHEN CONTROL ROOM INDICATIONS ASSOCIATED WITH THE PRESSURE TRANSMITTER WERE NOT RESPONDING TO THE CHANGING PLANT CONDITIONS. IMMEDIATE ACTIONS WERE TAKEN BY OPERATORS TO IDENTIFY THE LIMITING CONDITIONS FOR OPERATION (LCO) NOT BEING MET, AND APPROPRIATE ACTION STATEMENTS WERE SUBSEQUENTLY ENTERED. A VALVE LINEUP WAS PERFORMED, AND CHANNEL CHECKS OF THE INSTRUMENT WERE SATISFACTORILY COMPLETED. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO THE FAILURE OF CONTROL AND INSTRUMENTATION (C&I) TECHNICIANS TO FOLLOW PROCEDURE. THE C&I TECHNICIANS INVOLVED IN THIS EVENT COMPLETED A COMPREHENSIVE VALVE LINEUP AND OPERATION RETRAINING PROGRAM. VALVE LINEUPS WERE VERIFIED FOR INSTRUMENTS THAT WERE WORKED DURING THE RECENTLY COMPLETED FALL 1987 OUTAGE. ALL C&I PERSONNEL WILL BE TRAINED ON VALVE LINEUP AND OPERATION REQUIREMENTS AND THE NECESSITY FOR PROCEDURAL COMPLIANCE. ADDITIONALLY, THE REQUIREMENT TO SEPARATELY RECORD AND VERIFY INSTRUMENT VALVE OPERATIONS WILL BE ESTABLISHED. THE EVENT WAS ASSESSED AS NOT SAFETY SIGNIFICANT SINCE THE REDUNDANT INSTRUMENT CHANNELS WERE AVAILABLE.

[44]   CLINTON 1   DOCKET 50-461   LER 87-067
OVERLY RESTRICTIVE DESIGN SETPOINT TRIP TOLERANCE FOR TEMPERATURE SWITCH RESULTS IN BROKEN LEAD WIRE AND REACTOR CORE ISOLATION COOLING ISOLATION.
EVENT DATE: 112487   REPORT DATE: 122287   NSSS: GE   TYPE: BWR
VENDOR: RAYCHEM CORP.
RILEY COMPANY, THE

(NSIC 207533) ON 11/24/87 WITH THE PLANT IN MODE 1 (POWER OPERATION), AT 40% REACTOR POWER, A DIVISION 1 REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) ISOLATION OCCURRED. THE ISOLATION OCCURRED DURING PERFORMANCE OF A MONTHLY SURVEILLANCE WHEN A CONTROL AND INSTRUMENTATION TECHNICIAN MOVED CABLES ASIDE SLIGHTLY TO GAIN ACCESS FOR LIFTING A THERMOCOUPLE LEAD WIRE. DURING THE PROCESS OF LIFTING THE LEAD WIRE, ANOTHER THERMOCOUPLE LEAD WIRE TERMINATING AT THE RCIC HIGH TEMPERATURE TERMINAL WAS BROKEN RESULTING IN THE RCIC ISOLATION. THE LIFTED LEAD WIRE WAS RE-LANDED. THE BROKEN THERMOCOUPLE LEAD WIRE WAS REWORKED. REPEATED LIFTING OF THE THERMOCOUPLE LEAD WIRE RESULTED IN A FATIGUE FAILURE. THE ROOT CAUSE IS ATTRIBUTED TO AN OVERLY RESTRICTIVE DESIGN SETPOINT TRIP TOLERANCE FOR RILEY #86 TEMPERATURE SWITCHES WHICH RESULTED IN MORE FREQUENT CALIBRATION OF THE TEMPERATURE SWITCHES THAN REQUIRED BY TECHNICAL SPECIFICATIONS

AND THUS MORE FREQUENT LIFTING OF THE LEAD WIRES. ILLINOIS POWER IS EVALUATING THIS MATTER FOR ENHANCEMENTS THAT MAY ELIMINATE OR REDUCE THE NEED TO MOVE THESE THERMOCOUPLE LEAD WIRES. THIS EVENT WAS DETERMINED NOT TO BE SAFETY SIGNIFICANT SINCE THE HIGH PRESSURE CORE SPRAY SYSTEM PROVIDES ADEQUATE BACKUP REACTOR COOLING ON FAILURE OF THE RCIC. THIS EVENT IS REPORTABLE UNDER THE PROVISIONS OF 10CFR50.73(A)(2)(IV).

[45] CLINTON 1 DOCKET 50-461 LER 87-069
LICENSED OPERATOR OVERSIGHT DURING REVIEW OF SURVEILLANCE IMPACT MATRIX RESULTS IN INOPERABLE CONTAINMENT ISOLATION FUNCTION OF VALVE.
EVENT DATE: 120387 REPORT DATE: 123187 NSSS: GE TYPE: BWR

(NSIC 207534) ON DECEMBER 3, 1987, WITH THE PLANT IN MODE 1 (POWER OPERATION) THE DIVISION 3 HIGH DRYWELL PRESSURE TRANSMITTER WAS ISOLATED FROM SERVICE FOR RESPONSE TIME SURVEILLANCE TESTING. WHEN THE TWO-HOUR TIME LIMIT ALLOWED BY TECHNICAL SPECIFICATIONS FOR SURVEILLANCE TESTING EXPIRED, THE CONTAINMENT ISOLATION FUNCTION FOR THE HIGH PRESSURE CORE SPRAY (HPCS) FULL FLOW TEST VALVE BECAME INOPERABLE. THE CONDITION WAS NOT IDENTIFIED UNTIL AFTER APPROXIMATELY 39.5 HOURS OF INOPERABILITY. THE ISOLATION INSTRUMENTATION WAS THEN DECLARED INOPERABLE. SUBSEQUENTLY, THE DIVISION 3 HIGH DRYWELL PRESSURE CHANNEL WAS PLACED IN A TRIPPED CONDITION. THE CAUSE OF THE EVENT IS ATTRIBUTED TO STAFF ASSISTANT SHIFT SUPERVISOR (SASS) OVERSIGHT OF THE CONTAINMENT ISOLATION IMPACT DURING REVIEW OF THE RESPONSE TIME TEST IMPACT MATRIX. THE SASS WAS COUNSELLED FOR THE OVERSIGHT. THIS EVENT WAS DETERMINED NOT TO BE SAFETY SIGNIFICANT SINCE REDUNDANT INSTRUMENT CHANNELS WERE AVAILABLE AND SINCE THE HPCS FULL FLOW TEST VALVE WAS IN THE CLOSED POSITION THROUGHOUT THE EVENT. THE EVENT IS REPORTABLE UNDER THE PROVISIONS OF 10CFR50.73(A)(2)(I)(B) DUE TO AN OPERATION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[46] CLINTON 1 DOCKET 50-461 LER 87-070
INADEQUATE RESEARCH INTO SURVEILLANCE INSTRUMENTATION DESIGN BASIS RESULTS IN INOPERABLE DRYWELL HIGH PRESSURE TRANSMITTERS DUE TO UNQUALIFIED MATERIAL INSTALLATION.
EVENT DATE: 120787 REPORT DATE: 011588 NSSS: GE TYPE: BWR

(NSIC 207922) ON DECEMBER 7, 1987 A TEMPORARY PLANT MODIFICATION RESULTED IN AN UNQUALIFIED PRESSURE GAUGE AND TUBING BEING INSTALLED ON A SAFETY- RELATED DRYWELL PRESSURE TRANSMITTER SENSING LINE. THE INSTALLATION RESULTED IN DRYWELL HIGH PRESSURE TRANSMITTERS BEING CONSIDERED INOPERABLE. THE MODIFICATION, INITIATED PRIOR TO COMPLETION OF THE 10CFR50.59 SAFETY EVALUATION, PROVIDED AN ABSOLUTE PRESSURE GAUGE FOR DETERMINING DRYWELL-TO-CONTAINMENT DIFFERENTIAL PRESSURE TO MEET TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS. THE CONDITION WAS IDENTIFIED DURING A REVIEW OF THE 10CFR50.59 SAFETY EVALUATION ON DECEMBER 16, 1987. THE PRESSURE TRANSMITTERS PROVIDE SIGNALS FOR VARIOUS TRIP FUNCTIONS ASSOCIATED WITH A NUMBER OF SAFETY SYSTEMS WITHIN A SINGLE DIVISION. ON DECEMBER 16, THE SHIFT SUPERVISOR DIRECTED THE ISOLATION VALVE UPSTREAM OF THE UNQUALIFIED COMPONENTS TO BE TAGGED CLOSED. THE CAUSE OF THIS EVENT IS PERSONNEL ERROR DUE TO INSUFFICIENT RESEARCH OF THE DESIGN BASIS PRIOR TO IMPLEMENTING CHANGES. THE TEMPORARY MODIFICATION PROCEDURE WAS REVISED TO DELETE THE OPTION OF INSTALLING SAFETY-RELATED TEMPORARY MODIFICATIONS PRIOR TO COMPLETION OF SAFETY REVIEWS. APPROPRIATE ENGINEERING STAFF WILL BE TRAINED ON LESSONS LEARNED FROM THIS EVENT.

[47] CLINTON 1 DOCKET 50-461 LER 87-068
ERROR BY INDETERMINABLE PERSON RESULTS IN INOPERABLE STANDBY GAS TREATMENT SYSTEM HIGH RANGE RADIOACTIVITY MONITOR DUE TO MISSING PARTICULATE FILTER PAPER.
EVENT DATE: 121187 REPORT DATE: 011388 NSSS: GE TYPE: BWR

(NSIC 207921) ON DECEMBER 11, 1987, WITH THE PLANT IN MODE 1 (POWER OPERATION), RADIATION PROTECTION PERSONNEL DISCOVERED THAT THE PARTICULATE FILTER PAPER MEDIA REQUIRED TO FILTER PARTICULATE FROM THE EXHAUST SAMPLE STREAM DURING AN ACCIDENT SCENARIO WAS MISSING FROM THE STANDBY GAS TREATMENT SYSTEM HIGH RANGE RADIOACTIVITY MONITOR. THE FILTER PAPER WAS IMMEDIATELY INSTALLED. THE CAUSE OF THE EVENT IS ATTRIBUTED TO PERSONNEL ERROR; HOWEVER, THE RESPONSIBLE ORGANIZATION OR INDIVIDUAL AND THE DATE WHEN THE EVENT OCCURRED COULD NOT BE DETERMINED. THE LAST DOCUMENTED VERIFICATION OF FILTER PRESENCE IN THE MONITOR WAS JULY 11, 1987. THE MONTHLY CHANNEL CHECK SURVEILLANCE PROCEDURE FOR THE MONITOR HAS BEEN REVISED TO REQUIRE SIGNATURE VERIFICATION OF FILTER MEDIA PRESENCE TO PREVENT A RECURRENCE FROM GOING UNDETECTED FOR LONGER THAN A MONTH. THE EVENT WILL BE REVIEWED WITH ALL APPROPRIATE CONTROL AND INSTRUMENTATION, CHEMISTRY, AND RADIATION PROTECTION PERSONNEL. THE EVENT WAS ASSESSED AS NOT SAFETY SIGNIFICANT SINCE THE CHARCOAL CARTRIDGE (WHICH WAS IN PLACE) OF THE MONITOR IS ANALYZED TO DETERMINE OFFSITE DOSE ASSESSMENT TO THE PUBLIC DURING OR AFTER A SEVERE ACCIDENT. THIS EVENT IS REPORTABLE UNDER THE PROVISIONS OF 10CFR50.73(A)(2)(I)(B).

[48]   CLINTON 1   DOCKET 50-461   LER 88-001
ISOLATION OF REACTOR WATER CLEANUP SYSTEM DURING TRENDING OF MAIN STEAM LINE TUNNEL TEMPERATURES DUE TO RANDOM FAILURE OF A TEMPERATURE MODULE.
EVENT DATE: 010188   REPORT DATE: 012788   NSSS: GE   TYPE: BWR
VENDOR: RILEY-BEAIRD, INC.

(NSIC 207937) ON JANUARY 1, 1988, WITH THE PLANT IN MODE 1 (POWER OPERATION), A SPURIOUS PARTIAL ISOLATION OF THE REACTOR WATER CLEANUP SYSTEM (RWCU) OCCURRED WHILE TRENDING MAIN STEAM LINE (MSL) TUNNEL TEMPERATURES DURING POWER ASCENSION FOLLOWING MAINTENANCE. WHEN THE CONTROL ROOM OPERATOR PLACED THE READ/SET SWITCH ON THE MSL TUNNEL HEATING, VENTILATING AND AIR CONDITIONING DIFFERENTIAL TEMPERATURE MODULE TO THE READ POSITION, TWO CONTAINMENT ISOLATION VALVES CLOSED AND THE RWCU PUMPS TRIPPED. ONE OTHER RWCU CONTAINMENT ISOLATION VALVE WAS ALREADY CLOSED DUE TO THE SYSTEM LINEUP. AN ADDITIONAL RWCU ISOLATION VALVE WHICH DID NOT CLOSE SINCE THE SEAL-IN TIME DELAY WAS NOT ACTUATED DUE TO THE SHORT DURATION OF THE INITIATION SIGNAL WAS CLOSED BY OPERATORS. OPERATORS PERFORMED THE AUTOMATIC ISOLATION CHECKLIST. THE MAINTENANCE GROUP WAS REQUESTED TO INVESTIGATE THE PROBLEM. THE CAUSE OF THE EVENT IS ATTRIBUTED TO RANDOM FAILURE OF THE TEMPERATURE MODULE. THE TEMPERATURE MODULE WAS REPLACED. THE EVENT WAS ASSESSED AS NOT SAFETY SIGNIFICANT SINCE THE RWCU SYSTEM RESPONDED AS DESIGNED AND SINCE A SHORT DURATION LOSS OF CLEANUP CAPABILITIES DOES NOT AFFECT SAFETY. THIS EVENT IS REPORTABLE UNDER THE PROVISIONS OF 10CFR50.73(A)(2)(IV).

[49]   CONNECTICUT YANKEE   DOCKET 50-213   LER 87-018
PERSONNEL ERROR RESULTS IN DECLARATION OF EEQ MOVS AS INOPERABLE.
EVENT DATE: 112087   REPORT DATE: 121787   NSSS: WE   TYPE: PWR
VENDOR: LIMITORQUE CORP.
        RELIANCE ELECTRIC COMPANY

(NSIC 207652) ON NOVEMBER 17, 1987 WITH THE PLANT SHUTDOWN IN MODE 6, INVESTIGATIONS DETERMINED THAT GREASE HAD BEEN INSTALLED IN THE MOTOR HOUSINGS OF 20 EEQ LIMITORQUE MOVS AND, SEVERAL OF THE MOVS WERE NOT IN THEIR QUALIFIED CONFIGURATION. ASSESSMENT OF THESE CONDITIONS RESULTED IN DECLARATION OF THE AFFECTED MOVS AS INOPERABLE ON NOVEMBER 20, 1987. THE ROOT CAUSE OF THIS EVENT IS UNDER EVALUATION VIA THE HPES AND MORS REVIEW SYSTEMS. CORRECTIVE ACTIONS INCLUDE REWORK OF ALL MOVS AFFECTED BY GREASE AND RESTORATION OF ALL MOVS TO QUALIFIED CONFIGURATION. ADDITIONAL INFORMATION ON ROOT CAUSE AND LONG TERM CORRECTIVE ACTIONS WILL BE FORWARDED IN A SUPPLEMENTAL REPORT. THIS EVENT IS REPORTABLE PER 10CFR50.73(A)(2)(V) AS IT COULD HAVE PREVENTED FULFILLMENT OF A SAFETY FUNCTION. THIS EVENT IS ALSO REPORTABLE PER 10CFR50.73(A)(2)(VII) IN

THAT, AS A SINGLE EVENT, IT COULD HAVE CAUSED INDEPENDENT CHANNELS IN MULTIPLE SYSTEMS TO BECOME INOPERABLE.

[50] COOK 1 DOCKET 50-315 LER 87-022
POTENTIAL VIOLATION OF ESF INSTRUMENTATION LIMITING CONDITIONS FOR OPERATION TOLERANCES DUE TO CALIBRATION SHIFT.
EVENT DATE: 100787 REPORT DATE: 123187 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)
VENDOR: FOXBORO CO., THE

(NSIC 207637) AS A RESULT OF ROUTINE CALIBRATIONS PERFORMED DURING THE 1987 UNIT ONE REFUELING OUTAGE, 16 TRANSMITTERS WERE FOUND TO HAVE EXPERIENCED A RELATIVELY SMALL IN MAGNITUDE BUT SIGNIFICANT CALIBRATION SHIFT. A SIMILAR CALIBRATION SHIFT WAS EXHIBITED BY 20 UNIT TWO TRANSMITTERS WHICH WERE CALIBRATED DURING THE SAME TIME PERIOD WHILE UNIT TWO WAS IN A MAINTENANCE OUTAGE. DUE TO THE NUMBER OF TRANSMITTERS INVOLVED, AN INVESTIGATION WAS INITIATED TO DETERMINE IF A GENERIC PROBLEM EXISTED WITH TRANSMITTERS MANUFACTURERED BY FOXBORO CORPORATION. CURRENT AND HISTORICAL DATA HAVE BEEN REVIEWED BY CORPORATE INSTRUMENTATION AND CONTROL (I&C) ENGINEERS WHO HAVE DETERMINED (ON NOVEMBER 30, 1987) THAT THE TRANSMITTERS WERE EXHIBITING NORMAL CHARACTERISTICS AND A GENERIC PROBLEM DID NOT EXIST. THE TRANSMITTERS IN QUESTION WERE REPAIRED AND RECALIBRATED AS NECESSARY UPON DISCOVERY. THE DATA IS CURRENTLY BEING EVALUATED BY CORPORATE NUCLEAR SAFETY AND LICENSING FOR POSSIBLE ADVERSE SAFETY IMPLICATIONS (E.G. NON-CONSERVATIVE SET POINTS). SINCE THE LIMITING CONDITION FOR OPERATIONS (LCO) TOLERANCE REQUIREMENTS MAY HAVE BEEN EXCEEDED, THIS INTERIM REPORT IS BEING SUBMITTED. A COMPLETE FOLLOW UP REPORT WILL BE SUBMITTED UPON COMPLETION OF THE EVALUATION (TENTATIVELY SCHEDULED FOR JANUARY 29, 1988).

[51] COOK 1 DOCKET 50-315 LER 87-023
DEFICIENT DESIGN RESULTS IN FAILURE TO PROVIDE ELECTRICAL ISOLATION BETWEEN LOCAL SHUTDOWN AND INDICATION PANELS.
EVENT DATE: 110987 REPORT DATE: 012188 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)

(NSIC 207961) THIS EVENT WAS DETERMINED TO BE REPORTABLE ON DECEMBER 22, 1987. ON NOVEMBER 9, 1987, DURING A REVIEW OF AN INVESTIGATION CONCERNING REGULATORY GUIDE 1.97 COMPLIANCE, IT WAS DISCOVERED THAT THE FUSES REQUIRED FOR ISOLATION BETWEEN THE VARIOUS LOCAL SHUTDOWN AND INDICATION (LSI) PANELS WERE IMPROPERLY LOCATED ON UNIT 2 AND NOT INCORPORATED INTO THE EXISTING DESIGN ON UNIT 1. THEREFORE, A CONDITION EXISTED THAT, IN THE EVENT OF A FIRE LOCAL TO A LSI PANEL, POWER (BOTH NORMAL AND ALTERNATE) TO SOME OR ALL OF THE SAME UNIT'S REMAINING PANELS COULD HAVE BEEN LOST. THE CAUSE OF THE EVENT WAS AN OVERSIGHT (COGNITIVE PERSONNEL ERROR) BY DESIGN ENGINEERS IN THE DESIGN AND VERIFICATION PROCESS ASSOCIATED WITH THE INITIAL APPENDIX R MODIFICATIONS. FIRE WATCHES WERE ASSIGNED TO TOUR THE AFFECTED AREAS. DESIGN CHANGES HAVE BEEN IMPLEMENTED WHICH PROVIDE THE NECESSARY ISOLATION IN THE EVENT OF A FIRE. TO PREVENT RECURRENCE THE APPROPRIATE ENGINEERING PROCEDURES HAVE BEEN PREFACED TO ADDRESS THIS SPECIFIC ELECTRICAL ISOLATION CONCERN.

[52] COOK 2 DOCKET 50-316 LER 87-015
ENTRY INTO TECHNICAL SPECIFICATION 3.0.3 DUE TO CORRECTIVE MAINTENANCE.
EVENT DATE: 112087 REPORT DATE: 011488 NSSS: WE TYPE: PWR
VENDOR: AMPHENOL

(NSIC 207890) THIS EVENT WAS DETERMINED TO BE REPORTABLE PER 10CFR 50.73 ON DECEMBER 29, 1987 DURING ROUTINE EVALUATION OF THE EVENT. ON NOVEMBER 20, 1987, UNIT TWO ENTERED TECHNICAL SPECIFICATION PARAGRAPH 3.0.3 FOR A PERIOD OF SEVENTEEN (17) MINUTES DUE TO A CORRECTIVE MAINTENANCE EVOLUTION WHICH EXCEEDED

THE ASSOCIATED ACTION STATEMENT. FOLLOWING A ROUTINE CALIBRATION OF POWER RANGE NUCLEAR INSTRUMENTATION CHANNEL N-43, AT APPROXIMATELY EIGHT (8) HOURS INTO THE ASSOCIATED TWELVE (12) HOUR ACTION STATEMENT, THE B DETECTOR HIGH VOLTAGE CABLE CONNECTOR SEPARATED FROM THE CABLE. THE REPAIR AND SUBSEQUENT CHANNEL FUNCTIONAL TEST EXCEEDED THE ASSOCIATED ACTION STATEMENT TWELVE HOUR TIME LIMIT CALCULATED FROM THE START OF THE ORIGINAL CALIBRATION. THIS REQUIRED UNIT TWO TO ENTER TECHNICAL SPECIFICATION 3.0.3 FOR SEVENTEEN MINUTES UNTIL CHANNEL N-43 COULD BE DECLARED OPERABLE. DURING THE ENTIRE CALIBRATION AND REPAIR EVOLUTION (12 HOURS AND 17 MINUTES), THE BISTABLES ASSOCIATED WITH CHANNEL N-43 REMAINED IN THE TRIPPED CONDITION. THE TECHNICAL SPECIFICATION INVOLVED WITH CHANNEL N-43 IS PARAGRAPH 3.3.1.1, TABLE 3.3-1 ITEM NUMBERS 2, 3 AND 4 AND ASSOCIATED ACTION STATEMENT NUMBER 2.

[53] COOK 2 DOCKET 50-316 LER 87-014
MISSED SURVEILLANCE DUE TO PERSONNEL ERROR IN PROCESS COMPUTER SOFTWARE PROGRAMMING.
EVENT DATE: 121587 REPORT DATE: 011488 NSSS: WE TYPE: PWR

(NSIC 207889) ON DECEMBER 15, 1987, AT 1409 HOURS, WHILE INVESTIGATING THE CAUSE OF A FAILURE OF THE P-250 PROCESS COMPUTER, A PROGRAMMING ERROR WAS DISCOVERED THAT HAD CAUSED THE QUADRANT POWER TILT RATIO (QPTR) ALARM TO BE INOPERABLE. THE CONDITION HAD EXISTED SINCE DECEMBER 7, 1987, WHEN THE ANNUNCIATOR PROGRAM WAS INADVERTENTLY WRITTEN OVER DURING SOFTWARE TROUBLESHOOTING. SURVEILLANCE REQUIREMENT 4.2.4.B WAS NOT PERFORMED DURING THIS EIGHT DAY PERIOD BECAUSE IT WAS NOT KNOWN THAT THE ALARM FUNCTION WAS INOPERABLE. SHOULD AN ACTUAL QPTR HAVE OCCURRED THE POTENTIAL EXISTED THAT THE ACTION STATEMENT TIME LIMIT FOR TECHNICAL SPECIFICATION 3.2.4 MIGHT HAVE BEEN MISSED. THE COMPUTER CONTINUED TO CALCULATE THE QPTR DURING THIS PERIOD. THE RATIO DID NOT EXCEED 1.02. THE CAUSE OF THE ANNUNCIATOR PROGRAM BEING INOPERABLE WAS PERSONNEL ERROR. THE SPACE ALLOCATION FOR THE SOFTWARE PROGRAM WAS NOT CHECKED PRIOR TO LOADING INTO THE SYSTEM DISK. ALL COMPUTER ANALYSTS WHO HAVE THE RESPONSIBILITY OF PERFORMING THIS TASK HAVE BEEN COUNSELLED IN THE PROPER METHOD OF P-250 COMPUTER SOFTWARE TROUBLESHOOTING AND PROGRAM INSTALLATION.

[54] COOPER DOCKET 50-298 LER 87-025
DEFICIENCY IN A DESIGN ENGINEERING EFFORT PERFORMED IN RESPONSE TO A TMI-2 LESSONS LEARNED NRC CONCERN.
EVENT DATE: 111287 REPORT DATE: 121487 NSSS: GE TYPE: BWR

(NSIC 207437) DURING A DESIGN REVIEW MEETING FOR A PROPOSED POWER SUPPLY CHANGE TO PC-MOV-306MV, A INCH MOTOR OPERATED VALVE INSTALLED TO BYPASS 24 INCH PC-MOV-231MV, DRYWELL EXHAUST INBOARD ISOLATION VALVE, THE PARTICIPANTS DETERMINED THAT THE EXISTING POWER SUPPLY CONFIGURATION MIGHT NOT FULLY CONFORM TO ALL APPROPRIATE DESIGN CRITERIA. SPECIFICALLY, UPON LOSS OF POWER, THE VALVE WOULD REMAIN IN ITS EXISTING POSITION SINCE ITS POWER SOURCE IS NOT AUTOMATICALLY RE-ENERGIZED BY A DIESEL GENERATOR. THIS SITUATION CONSTITUTES AN APPARENT NONCOMPLIANCE WITH SAFETY DESIGN BASIS REQUIREMENTS PRESCRIBED IN THE CNS USAR IN SECTIONS I-5.2.6.2.5, V- 2.2.6, AND II-3.3.7.A. PC-MOV-306MV WAS INSTALLED DURING THE 1981 REFUELING OUTAGE IN RESPONSE TO AN NRC CONCERN REGARDING POST-ACCIDENT VENTING FOR COMBUSTIBLE GAS CONTROL PURPOSES. WHILE THE DESIGN CHANGE WAS RESPONSIVE TO THIS CONCERN (NUREG 0578, ITEM 2.1.5.A), THE ENGINEERING EFFORT APPARENTLY FAILED TO FULLY CONSIDER ALL OF THE REQUIREMENTS ASSOCIATED WITH MOTOR OPERATED VALVES WHICH MUST FUNCTION TO ACHIEVE CONTAINMENT ISOLATION. DURING THE 1988 REFUELING OUTAGE, A DESIGN CHANGE WILL BE IMPLEMENTED TO POWER C-MOV-306MV FROM A NON-LOAD-SHEDDING MOTOR CONTROL CENTER (MCC).

[55] COOPER DOCKET 50-298 LER 88-001
HUMAN FACTOR DISCREPANCY WHICH CAUSED AN UNPLANNED DIESEL GENERATOR ACTUATION WHILE VERIFYING ELECTRICAL SCHEMATIC DRAWINGS.
EVENT DATE: 010588 REPORT DATE: 012788 NSSS: GE TYPE: BWR

(NSIC 207972) AT 1:34 PM ON 1/5/88, AN UNPLANNED AUTOMATIC ACTUATION OF DIESEL GENERATOR #1 OCCURRED WHEN THE NORMAL POWER SUPPLY BREAKER FOR THE 4160V AC 1F CRITICAL BUS INADVERTENTLY TRIPPED OPEN. THE BREAKER ACTUATION OCCURRED WHEN THE DOOR TO THE CABINET HOUSING THE POTENTIAL TRANSFORMERS FOR THE BREAKER WAS OPENED BY PERSONNEL VERIFYING ELECTRICAL SCHEMATIC DRAWINGS TO AS-BUILT CONDITIONS. AT THE TIME OF THIS EVENT, THE PLANT WAS IN OPERATION AT APPROXIMATELY 95% OF RATED POWER. THE BUS WAS AUTOMATICALLY RE-ENERGIZED FROM THE EMERGENCY TRANSFORMER AND, DUE TO THE SPEED OF THE TRANSFER, PLANT OPERATION WAS UNAFFECTED. DIESEL GENERATOR #1 STARTED, BUT WAS NOT LOADED. FOLLOWING CONFIRMATION OF PLANT STATUS BY LICENSED CONTROL ROOM PERSONNEL, THE NORMAL POWER SUPPLY ALIGNMENT FOR THE 4160V AC 1F CRITICAL BUS WAS RE-ESTABLISHED AND THE DIESEL GENERATOR WAS SECURED. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO A HUMAN FACTOR DISCREPANCY IN THAT THE POTENTIAL TRANSFORMER CABINET DOOR WAS NOT ADEQUATELY LABELED WITH PRECAUTIONARY INFORMATION. CORRECTIVE ACTIONS TAKEN OR IN PROGRESS INCLUDE, INSURING CAUTION LABELS ARE INSTALLED ON ALL POTENTIAL TRANSFORMER CABINET DOORS, COUNSELLING OF THE EMPLOYEES INVOLVED, INCORPORATION OF INFORMATION REGARDING THESE CABINETS IN TRAINING PROGRAMS, AND DISSEMINATING INFORMATION REGARDING THIS EVENT TO APPROPRIATE PERSONNEL.

[56] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-009 REV 02
UPDATE ON INVERTER FAILURE AND REACTOR TRIP.
EVENT DATE: 070287 REPORT DATE: 123087 NSSS: BW TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.
SOLID STATE CONTROLS, INC.

(NSIC 207679) ON JULY 2, 1987, CRYSTAL RIVER UNIT 3 WAS OPERATING AT 88.4% RATED THERMAL POWER, GENERATING 771 MWE. SURVEILLANCE TESTING ON CHANNEL B OF THE REACTOR PROTECTION SYSTEM WAS IN PROGRESS. THE B CONTROL ROD DRIVE MECHANISM (CRDM) CIRCUIT BREAKER OPENED NORMALLY WHEN THE SHUNT TRIP DEVICE WAS TESTED, BUT WOULD NOT RECLOSE AFTER TESTING. WHILE INVESTIGATION INTO THE B CRDM BREAKER PROBLEM WAS IN PROGRESS, THE INVERTER SUPPLYING POWER TO THE A VITAL BUS FAILED. THE A VITAL BUS AUTOMATICALLY TRANSFERRED TO ITS ALTERNATE POWER SOURCE. DURING THE TRANSFER THE BUS EXPERIENCED A SLIGHT VOLTAGE DIP WHICH CAUSED THE A CRDM BREAKER TO OPEN. SINCE BOTH BREAKERS WERE OPEN SIMULTANEOUSLY, ALL POWER WAS REMOVED FROM THE CONTROL ROD DRIVE MECHANISMS. THIS CAUSED ALL CONTROL RODS TO FALL TO THE FULLY INSERTED POSITION AND SHUT DOWN THE REACTOR. THE CRDM BREAKER THAT WOULD NOT CLOSE HAS BEEN REPLACED. THE INVERTER HAS BEEN REPAIRED AND A MODIFICATION HAS BEEN PREPARED TO LOWER THE UNDERVOLTAGE SETPOINT.

[57] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-029
PERSONNEL ERROR LEADS TO EXCEEDING RADIATION MONITOR SURVEILLANCE INTERVAL AND FAILURE TO MEET TECHNICAL SPECIFICATION REQUIREMENTS FOR LIQUID RADWASTE RELEASE.
EVENT DATE: 111887 REPORT DATE: 121887 NSSS: BW TYPE: PWR

(NSIC 207523) ON NOVEMBER 18, 1987, CRYSTAL RIVER UNIT 3 WAS SHUT DOWN IN A REFUELING OUTAGE. AT 1020, IT WAS DETERMINED THAT THE MONTHLY SURVEILLANCE FOR LIQUID PROCESS AND EFFLUENT RADIATION MONITORS HAD NOT BEEN PERFORMED AND THE ALLOWABLE LIMIT FOR THREE CONSECUTIVE SURVEILLANCES HAD BEEN EXCEEDED BY ONE DAY. THE SHIFT SUPERVISOR WAS NOTIFIED AND DECLARED THE MONITORS INOPERABLE. A REVIEW OF THE SHIFT SUPERVISOR'S LOG INDICATED THAT ONE RADIOACTIVE LIQUID EFFLUENT RELEASE WAS INOPERABLE. THE TECH SPEC REQUIREMENTS FOR MAKING A RELEASE WITH AN INOPERABLE MONITOR WERE NOT MET. THESE EVENTS WERE THE RESULT OF PERSONNEL ERROR. THE SURVEILLANCE OF THESE MONITORS WAS DELAYED WHILE AWAITING A REVISION TO THE SURVEILLANCE PROCEDURE. THE SURVEILLANCE WAS DELAYED BEYOND THE

ALLOWABLE DATE BECAUSE THE REQUIRED INTERVAL WAS NOT VERIFIED. THE LIQUID RELEASE WAS MADE WITHOUT PERFORMING THE TECH SPEC REQUIRED ANALYSES AND VERIFICATIONS BECAUSE IT WAS NOT RECOGNIZED THAT THE MONITOR WAS INOPERABLE UNTIL AFTER THE RELEASE WAS MADE. RESPONSIBLE PERSONNEL HAVE BEEN INSTRUCTED TO PERFORM A REVIEW OF THE MASTER SURVEILLANCE PLAN FOR SURVEILLANCES THAT HAVE BEEN DELAYED. THE LATEST ALLOWABLE COMPLETION DATES ARE CALCULATED FOR THESE SURVEILLANCES AND REPORTED TO APPROPRIATE SUPERVISORS.

[58] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-028
INADEQUATE OPERATOR TRAINING LEADS TO IMPROPER VITAL BUS SWITCHING SEQUENCE AND ENGINEERED SAFEGUARDS SYSTEM ACTUATION.
EVENT DATE: 120587 REPORT DATE: 010488 NSSS: BW TYPE: PWR

(NSIC 207713) ON DECEMBER 5, 1987, CRYSTAL RIVER UNIT 3 WAS SHUT DOWN IN A REFUELING OUTAGE. POWER TO VITAL BUSES 4 AND 6 ("B" ENGINEERED SAFEGUARDS TRAIN VITAL BUSES) WAS BEING SUPPLIED BY THE ALTERNATE AC SOURCES VIA MANUAL BYPASS SWITCHES, RATHER THAN NORMAL ALIGNMENT TO THE INVERTERS. AT 1230, OPERATIONS TO POWER DOWN THE "B" INVERTER FOR MAINTENANCE WERE IN PROGRESS. A "B" TRAIN ENGINEERED SAFEGUARDS ACTUATION OCCURRED DURING THESE OPERATIONS. THE IMMEDIATE CAUSE OF THIS EVENT WAS THE MOMENTARY LINEUP OF THE VITAL BUSES TO DE-ENERGIZED POWER SUPPLIES BECAUSE OF AN IMPROPER SWITCHING SEQUENCE. THE ROOT CAUSE OF THE EVENT WAS INADEQUATE OPERATOR TRAINING ON THE FUNCTION OF THE INVERTER STATIC TRANSFER SWITCHES AND MANUAL BYPASS SWITCHES. FOLLOWING THE ENGINEERED SAFEGUARDS SYSTEM ACTUATION, THE TRIPPED CHANNELS WERE RESET AND THE PLANT WAS RETURNED TO ITS PRE-EVENT STATUS. THE OPERATORS INVOLVED IN THE EVENT HAVE REVIEWED THEIR ACTIONS AND ARE AWARE THAT THE IMPROPER SWITCHING SEQUENCE RESULTED IN THE ENGINEERED SAFEGUARDS ACTUATION. ENHANCED TRAINING IN THE FUNCTION OF THE INVERTER STATIC TRANSFER AND MANUAL BYPASS SWITCHES WILL BE GIVEN TO ALL LICENSED OPERATORS. ADDITIONALLY, A SCHEMATIC DIAGRAM OF THE VITAL BUS POWER SUPPLIES AND SWITCHING ARRANGEMENTS WILL BE PROVIDED AS OPERATOR AIDS IN EACH OF THE INVERTER ROOMS.

[59] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-031
INADEQUATE SAFETY EVALUATION REVIEW OF PROCEDURE CHANGE RESULTS IN THE USE OF SEISMIC MONITORS WITH MEASUREMENT RANGES NOT IN COMPLIANCE WITH THE TECH SPECS.
EVENT DATE: 121887 REPORT DATE: 011888 NSSS: BW TYPE: PWR

(NSIC 208005) ON DECEMBER 18, 1987 CRYSTAL RIVER UNIT 3 WAS IN THE COLD SHUTDOWN CONDITION (MODE 5). THE MEASUREMENT RANGE OF THE INSTALLED TRIAXIAL PEAK ACCELOGRAPH SEISMIC MONITORS WAS DISCOVERED TO BE +/- 1.0 G. THE PLANT TECH SPECS REQUIRE THAT THIS RANGE BE +/- 2.0 G. THE CALIBRATION RANGE OF THE DETECTORS WAS CHANGED AS A RESULT OF A PROCEDURE CHANGE MADE IN JUNE, 1979. THE CAUSE OF THIS EVENT IS PERSONNEL ERROR IN THE PERFORMANCE OF A SAFETY EVALUATION FOR THE PROCEDURE CHANGE MADE IN 1979. THE SAFETY EVALUATION INCORRECTLY IDENTIFIED THIS CHANGE AS COMPLYING WITH THE TECH SPECS. ALL THREE AFFECTED ACCELOGRAPHS HAVE BEEN REPLACED WITH DETECTORS OF THE PROPER RANGE. SINCE 1979, THERE HAVE BEEN SEVERAL CHANGES IN THE PROCEDURE REVIEW PROCESS WHICH HAVE REDUCED THE PROBABILITY OF SIMILAR OCCURRENCES. A TRAINING MODULE ON THE REVIEW OF 10CFR50.59 SAFETY EVALUATIONS HAS BEEN DEVELOPED.

[60] DAVIS-BESSE 1 DOCKET 50-346 LER 87-003 REV 01
UPDATE ON SERVICE WATER VALVE INCORRECTLY DECLARED INOPERABLE DURING MODE CHANGE.
EVENT DATE: 010887 REPORT DATE: 012688 NSSS: BW TYPE: PWR

(NSIC 208031) ON JANUARY 8, 1987 IT WAS DISCOVERED THAT SEVEN MODE CHANGES HAD OCCURRED WITH A CONTAINMENT AIR COOLER SERVICE WATER OUTLET VALVE (SW1357) DECLARED INOPERABLE. MAINTENANCE HAD BEEN PERFORMED ON THE NON-SAFETY RELATED PORTION OF THE VALVE ACTUATOR ONLY. ALTHOUGH THIS VALVE HAD BEEN LISTED AS

INOPERABLE, IT HAD NEVER `TUALLY BEEN RENDERED INOPERABLE AS A CONTAINMENT ISOLATION VALVE AND WAS A \LL TIMES CAPABLE OF PERFORMING ITS SAFETY FUNCTION. ALTHOUGH NO VIOLATION OF 1 `HNICAL SPECIFICATIONS RESULTED, PROCEDURAL VIOLATIONS OCCURRED WHICH COULD HAVE LED TO A VIOLATION OF TECHNICAL SPECIFICATIONS. ONCE DISCOVERED THE MAINTENANCE WORK ORDER FOR THE SERVICE WATER VALVE WAS COMPLETED AND SIGNED OFF WITH AN ENTRY IN THE UNIT LOG THAT SW 1357 WAS OPERABLE.

[61] DAVIS-BESSE 1 DOCKET 50-346 LER 87-008 REV 01
UPDATE ON INOPERABLE SAFETY FEATURES ACTUATION SYSTEM SEQUENCER DUE TO DEGRADED LOGIC POWER SUPPLY.
EVENT DATE: 033087 REPORT DATE: 012688 NSSS: BW TYPE: PWR
VENDOR: LAMBDA ELECTRONICS

(NSIC 208047) WITH THE UNIT OPERATING AT 100% POWER THE SAFETY FEATURES ACTUATION SYSTEM (SFA CHANNEL 4 SEQUENCER FAILED A MONTHLY SURVEILLANCE TEST (ST). THE SFAS CHANNEL 4 SEQUENCER WAS DECLARED INOPERABLE AND THE STATION ENTERED TECHNICAL SPECIFICATION (TS) 3.3.2.1 ACTION 9. BECAUSE OF AN OVERLY CONSERVATIVE REQUIREMENT IN TS 3.3.2.1 TO HAVE ALL FOUR SEQUENCES OPERABLE, ENTRY INTO TS 3.0.3 WAS REQUIRED, A UNIT SHUTDOWN WAS BEGUN AND AN UNUSUAL EVENT WAS DECLARED AS DEFINED BY DAVIS-BESSE EMERGENCY PREPAREDNESS PLAN, EP.15 MAINTENANCE ACTIVITIES RESTORED THE SFAS CHANNEL, ENDING THE EVENT WITH REACTOR POWER AT 94.5%. THE ROOT CAUSE WAS DETERMINED TO BE A DEGRADED SFAS LOGIC POWER SUPPLY. IN ADDITION, A TS CHANGE WAS MADE TO PREVENT FUTURE FORCED ENTRY INTO TS 3.0.3 DURING OPERATION WITH O SFAS SEQUENCER INOPERABLE.

[62] DAVIS-BESSE 1 DOCKET 50-346 LER 87-015
UNIT TRIP DUE TO LOSS OF INSTRUMENT AIR PRESSURE.
EVENT DATE: 120787 REPORT DATE: 010688 NSSS: BW TYPE: PWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)
MERCOIL CORP.
PALL TRINITY MICRO CORP.
POWELL, WILLIAM COMPANY, THE

(NSIC 207685) ON 12/7/87 THE UNIT EXPERIENCED A REACTOR TRIP AT 0656 HOURS FROM 81% REACTOR THERMAL POWER. THE INITIATING EVENT WAS A LOSS OF INSTRUMENT AIR PRESSURE WHICH CAUSED SEVERAL SECONDARY SYSTEM VALVES TO GO TO THEIR FAILED POSITION REACTOR POWER INCREASED TO THE INTEGRATED CONTROL SYSTEM (ICS) HIGH DEMAND LIMITER FEEDWATER FLOW INCREASED CAUSING REACTOR COOLANT SYSTEM TAVE TO DECREASE. DUE TO A LARGE MODERATOR TEMPERATURE COEFFICIENT, REACTOR POWER INCREASED TO THE REACTOR PROTECTION SYSTEM HIGH FLUX TRIP SETPOINT. THE POST-TRIP PLANT RESPONSE WAS NORMAL EXCEPT THAT STEAM GENERATOR 1.1 PRESSURE WAS SLIGHTLY LOWER AND STEAM GENERATOR 1.2 LEVEL WAS HIGHER THAN THE EXPECTED POST TRIP RANGES. OPERATOR ACTIONS WERE REQUIRED TO CLOSE THE MOISTURE SEPARATOR REHEATER (MSR) SECOND STAGE REHEAT STEAM SOURCE VALVES AND TO STABILIZE STEAM GENERATOR 1.1 WATER LEVELS. THE LOSS OF INSTRUMENT AIR PRESSURE WAS CAUSED BY DIRECT VENTING OF THE INSTRUMENT AIR HEADER TO ATMOSPHERE WHEN A SOLENOID VALVE FAILED ON INSTRUMENT AIR DRYERS 1-1 AND 1-2. THIS SOLENOID VALVE WAS REPAIRED. THE MSR SECOND STAGE REHEAT STEAM SOURCE VALVES DID NOT CLOSE DUE TO A PRESSURE SWITCH FAILURE. THIS SWITCH WAS REPLACED. THE ICS DID NOT RESPOND FAST ENOUGH TO STEAM GENERATOR 1-1 DECREASING WATER LEVEL. A MODIFICATION TO THE ICS IS SCHEDULED.

[63] DIABLO CANYON 1 DOCKET 50-275 LER 87-023
LOW-LOW STEAM GENERATOR LEVEL REACTOR TRIP FOLLOWING A MAIN FEEDWATER PUMP TRIP DURING A SURVEILLANCE TEST DUE TO A FAILED "PUSH TO "TEST" LAMP SOCKET.
EVENT DATE: 121387 REPORT DATE: 011288 NSSS: WE TYPE: PWR
VENDOR: DIALCO

(NSIC 207879) AT 0325 PST, DECEMBER 13, 1987, WITH THE UNIT IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER, A LOW-LOW STEAM GENERATOR LEVEL REACTOR TRIP OCCURRED FOLLOWING A TRIP OF MAIN FEEDWATER PUMP (MFHP) 1-1. ALL AUTOMATIC SAFETY FUNCTIONS RESPONDED AS REQUIRED. THE UNIT WAS STABILIZED IN MODE 3 (HOT STANDBY) BY 0500 PST. THE 4-HOUR NONEMERGENCY REPORT REQUIRED BY 10 CFR 50.72 WAS MADE AT 0411 PST. THE MFHP TRIP OCCURRED DURING CONTINUITY TESTS IN THE SAFEGUARDS TEST PANEL FOR THE SOLID STATE PROTECTION SYSTEM (SSPS) MFHP TRIP CIRCUIT. ALTHOUGH A SPECIFIC ROOT CAUSE FOR THE MFHP TRIP COULD NOT BE DETERMINED, THE MOST LIKELY CAUSE WAS A MOMENTARY SHORT IN THE "PUSH TO TEST" LIGHT SOCKET. THE LAMP SOCKET WAS REPLACED. AN EVALUATION HAS BEEN INITIATED TO ENSURE AN ADEQUATE MARGIN BETWEEN THE CURRENT THROUGH THE SOLENOID IN THE TEST MODE AND THE CURRENT REQUIRED TO ACTUATE THE SOLENOID. UNTIL THE EVALUATION IS COMPLETE, A TEMPORARY CHANGE TO SURVEILLANCE TEST PROCEDURE M-16P2, "CONTINUITY TESTING OF SLAVE RELAY K601, K620, K636, AND K621," HAS BEEN WRITTEN TO REQUIRE A CHECK TO ENSURE THAT THE BLOCKING RELAY HAS FUNCTIONED BEFORE THE RELAY BEING TESTED IS ACTUATED. A SUPPLEMENTAL REPORT WILL BE ISSUED UPON COMPLETION OF THIS EVALUATION.

[64] DIABLO CANYON 1 DOCKET 50-275 LER 87-024
REACTOR TRIP IN MODE 3 (HOT STANDBY) WHEN SOURCE RANGE CHANNEL N-32 DETECTOR VOLTAGE SUPPLY FAILED HIGH DUE TO AN UNKNOWN CAUSE.
EVENT DATE: 121387 REPORT DATE: 011288 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 207880) AT 1407 PST, DECEMBER 13, 1987, WITH THE UNIT IN MODE 3 (HOT STANDBY), A REACTOR TRIP AND SUBSEQUENT TURBINE TRIP OCCURRED DUE TO THE FAILURE OF THE DETECTOR VOLTAGE SUPPLY FOR SOURCE RANGE CHANNEL N-32. ALL AUTOMATIC SAFETY FUNCTIONS RESPONDED AS REQUIRED. THE UNIT WAS ALREADY STABLE IN MODE 3. THE 4-HOUR NONEMERGENCY REPORT REQUIRED BY 10 CFR 50.72 WAS MADE AT 1515 PST. THE HIGH-VOLTAGE POWER SUPPLY FOR SOURCE RANGE CHANNEL N-32 FAILED HIGH, CAUSING A HIGH SOURCE RANGE FLUX SIGNAL WHICH DAMAGED THE DETECTOR SIGNAL PRE-AMP. THE HIGH-VOLTAGE POWER SUPPLY AND THE DETECTOR SIGNAL PRE-AMP WERE REPLACED. THE FAILED HIGH VOLTAGE POWER SUPPLY IS BEING EXAMINED TO DETERMINE A CAUSE FOR ITS FAILURE. A SUPPLEMENTAL REPORT WILL BE PROVIDED TO REPORT THE RESULTS OF THIS EXAMINATION AND ANY CORRECTIVE ACTIONS NECESSARY TO PREVENT RECURRENCE.

[65] DIABLO CANYON 1 DOCKET 50-275 LER 87-025
HIGH STEAM GENERATOR WATER LEVEL MAIN TURBINE TRIP AND MAIN FEEDWATER ISOLATION DURING STARTUP DUE TO LACK OF GUIDANCE FOR OPERATORS ON PROPORTIONAL INTEGRAL CONTROLLERS.
EVENT DATE: 121787 REPORT DATE: 011988 NSSS: WE TYPE: PWR

(NSIC 207881) AT 0449 PST, DECEMBER 17, 1987, WITH THE UNIT IN MODE 1 (POWER OPERATION), A MAIN TURBINE TRIP AND FEEDWATER ISOLATION VALVE CLOSURE OCCURRED BECAUSE THE STEAM GENERATORS REACHED THE HIGH LEVEL SETPOINT (P-14). LICENSED OPERATORS WERE INCREASING THE POWER LEVEL WHEN WATER LEVEL IN THE STEAM GENERATORS EXCEEDED THE P-14 HIGH LEVEL SETPOINT. THE UNIT WAS STABILIZED IN MODE 2 (STARTUP). THE 4-HOUR NONEMERGENCY REPORT REQUIRED BY 10 CFR 50.72 (B)(2)(II) WAS COMPLETED AT 0509 PST, DECEMBER 17, 1987. AN INVESTIGATION OF THIS EVENT DID NOT REVEAL ANY SIGNIFICANT MALFUNCTION OF EQUIPMENT OR PERSONNEL ERROR THAT CAUSED THE STEAM GENERATOR WATER LEVEL TO EXCEED THE P-14 SETPOINT. THE COMBINATION OF STEAM GENERATOR WATER LEVEL CONTROL IN MANUAL AND PERTURBATIONS CAUSED BY THE RAPID CHANGES IN STEAM DUMP DEMAND DURING SWITCHING FROM MANUAL TO AUTOMATIC CONTROL WERE THE MAIN CONTRIBUTORS TO THE STEAM GENERATORS' HIGH LEVEL AND SUBSEQUENT TURBINE TRIP. OPERATORS DID NOT HAVE SUFFICIENT GUIDANCE TO EXECUTE THIS OPERATION IN AN OPTIMUM MANNER. AN OPERATIONS DEPARTMENT MEMORANDUM WILL BE ISSUED TO OPERATORS, PROVIDING GUIDANCE ON THE CORRECT METHOD FOR TRANSFERRING STEAM DUMP CONTROLLERS FROM MANUAL TO AUTOMATIC CONTROL.

[66]   DIABLO CANYON 1   DOCKET 50-275   LER 87-026
CHEMISTRY ANALYSIS FOR CAUSTIC SPRAY ADDITIVE FAILS TO MEET TECH SPEC REQUIREMENTS DUE TO LACK OF PROCEDURAL GUIDANCE.
EVENT DATE: 121787   REPORT DATE: 011988   NSSS: WE   TYPE: PWR

(NSIC 207940) ON DECEMBER 17, 1987, AT 0430 PST, UNIT 1 ENTERED MODE 1 (POWER OPERATION) IN VIOLATION OF TECH SPEC 3.0.4, WHICH PROHIBITS THE ENTRY INTO AN OPERATIONAL MODE UNLESS THE LIMITING CONDITIONS FOR OPERATION ARE MET WITHOUT RELIANCE ON PROVISIONS OF THE ACTION STATEMENT. ON DECEMBER 17, 1987, AT 0830 PST, IT WAS DETERMINED THAT THE CHEMISTRY ANALYSIS FOR THE CAUSTIC SPRAY ADDITIVE TANK (CSAT) TAKEN PRIOR TO THE MODE CHANGE DID NOT MEET TECH SPEC 3.6.2.2.A REQUIREMENTS, THEREFORE THE UNIT WAS IN THE ACTION STATEMENT AT THE TIME OF THE MODE CHANGE. ON DECEMBER 18, 1987, AT 1140 PST, THE CSAT MET THE TECH SPEC 3.6.2.2.A REQUIREMENTS AND THE ACTION STATEMENT WAS EXITED. THE CAUSE OF THE EVENT WAS LACK OF PROCEDURAL GUIDANCE IN THE REPORTING OF SIGNIFICANT DIGITS FOR CHEMICAL ANALYSIS RESULTS. TO PREVENT RECURRENCE SURVEILLANCE TEST PROCEDURE (STP) C-1, "SPRAY ADDITIVE SYSTEM CHEMICAL INVENTORY", WILL BE REVISED TO PROVIDE GUIDANCE CONCERNING THE REPORTING OF ANALYSIS RESULTS.

[67]   DIABLO CANYON 1   DOCKET 50-275   LER 87-027
RADIATION MONITOR ALARM AND HOT PARTICLE CAUSED FUEL HANDLING BUILDING VENTILATION SYSTEM CHANGE (ACTUATION OF ENGINEERED SAFETY FEATURE) DUE TO FAILURE TO PERFORM SURVEY.
EVENT DATE: 122087   REPORT DATE: 011988   NSSS: WE   TYPE: PWR

(NSIC 208004) AT 1446 PST, DECEMBER 20, 1987, WITH THE UNIT IN MODE 1 (POWER OPERATION) AT 68% POWER, THE FUEL HANDLING BUILDING (FHB) VENTILATION SYSTEM SHIFTED INTO THE IODINE REMOVAL MODE. THIS OCCURRED DURING UNIT 1 SPENT FUEL POOL RERACKING OPERATIONS WHEN THE INADVERTENT REMOVAL OF A PARTICLE OF CO-60 (HOT PARTICLE) FROM THE SPENT FUEL POOL CAUSED SUFFICIENT RADIATION DOSE RATE TO ALARM THE FHB RADIATION MONITOR (RE-58) RESULTING IN THE VENTILATION SHIFT. THE FHB WAS EVACUATED, AND THE PARTICLE WAS LOCATED AND REMOVED. THE FOUR-HOUR NONEMERGENCY REPORT REQUIRED BY 10 CFR 50.72 WAS COMPLETED BY 1545 PST. THE PARTICLE WAS NOT DETECTED WHEN REMOVED FROM THE POOL ON A HOSE THAT WAS NOT BEING MONITORED FOR RADIATION. CORRECTIVE ACTIONS TAKEN IN RESPONSE TO THIS EVENT WERE TO REVISE THE SPECIAL WORK PERMIT FOR RERACKING AND THE DIVER COVERAGE PROCEDURE, ADD AN ADDITIONAL RADIATION PROTECTION TECHNICIAN TO EACH SHIFT, AND REVIEW THE EVENT WITH WORKERS AND RADIATION PROTECTION PERSONNEL. NO EXPOSURE LIMITS WERE EXCEEDED; THEREFORE, THERE WERE NO ADVERSE SAFETY CONSEQUENCES OR IMPLICATIONS RESULTING FROM THE EVENT.

[68]   DIABLO CANYON 1   DOCKET 50-275   LER 87-028
BOTH TRAINS OF AUXILIARY BUILDING VENTILATION INOPERABLE DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 123087   REPORT DATE: 012988   NSSS: WE   TYPE: PWR

(NSIC 207987) ON DECEMBER 30, 1987, AT 1323 PST, WITH THE UNIT IN MODE 1 (POWER OPERATION) AT 93 PERCENT POWER, BOTH TRAINS OF THE AUXILIARY BUILDING VENTILATION SYSTEM WERE INOPERABLE, RESULTING IN VIOLATION OF TECH SPEC 3.7.6.1 AND ENTRY INTO TECH SPEC 3.0.3. AUXILIARY BUILDING EXHAUST FAN E-2 WAS CLEARED TO SUPPORT MAINTENANCE ON THE VENTILATION SYSTEM. THE ALTERNATE FAN, E-1, TRIPPED ON THERMAL OVERLOAD WHEN AN IMPROPER CLEARANCE RESULTED IN DAMPERS REPOSITIONING, IMPAIRING THE FLOWPATH FOR THE FAN. FAN E-1 WAS RESTARTED AT 1338 PST. THE ROOT CAUSE IS ATTRIBUTED TO (1) PERSONNEL ERROR (COGNITIVE), IN THAT AN INADEQUATE REVIEW OF THE CLEARANCE WAS PERFORMED, AND (2) PROCEDURAL DEFICIENCY, IN THAT NO PROCEDURE EXISTED FOR ESTABLISHING CONDITIONS FOR PERFORMING THE MAINTENANCE. AN INCIDENT SUMMARY REPORT WAS ISSUED FOR REVIEW BY ALL OPERATIONS AND WORK PLANNING CLEARANCE PERSONNEL; AND SURVEILLANCE TEST PROCEDURE M-4, "ROUTINE SURVEILLANCE

TEST OF THE AUXILIARY BUILDING SAFEGUARDS AIR FILTRATION SYSTEM," WILL BE REVISED TO ADDRESS INITIAL CONDITIONS FOR THIS MAINTENANCE.

[69] DIABLO CANYON 2 DOCKET 50-323 LER 87-002 REV 01
UPDATE ON CLOSURE OF RHR CROSSTIE VALVE 8716B FOR MAINTENANCE JEOPARDIZES SYSTEM OPERABILITY.
EVENT DATE: 031787 REPORT DATE: 020588 NSSS: WE TYPE: PWR

(NSIC 207973) ON MARCH 17, 1987, AT 0625 PST, WITH UNIT 2 IN MODE 1 (POWER OPERATION) AT 100% POWER, RESIDUAL HEAT REMOVAL (RHR) CROSSTIE VALVE 8716B WAS CLOSED AND REMOVED FROM SERVICE FOR MAINTENANCE. THIS ACTION WAS NOT CONSISTENT WITH THE SAFETY ANALYSIS ASSUMPTION THAT RHR INJECTION INTO ALL FOUR REACTOR COOLANT SYSTEM (RCS) COLD LEGS WOULD BE AVAILABLE, ASSUMING THE SINGLE ACTIVE FAILURE OF ONE RHR PUMP. THE VALVE CLOSURE VIOLATED TECHNICAL SPECIFICATION 3.5.2, IN THAT IF ONLY ONE RHR PUMP WERE OPERABLE, INJECTION FLOW WOULD BE PROVIDED TO ONLY TWO RCS COLD LEGS. HOWEVER, SUBSEQUENT EVALUATIONS SHOW THAT DURING AN ACCIDENT FLOW THROUGH TWO RCS COLD LEGS IS SUFFICIENT TO MAINTAIN COOLING WITHIN DESIGN AND REGULATORY LIMITS. WHEN VALVE 8716B WAS CLOSED, BOTH RHR PUMPS WERE OPERABLE AND CAPABLE OF INJECTING FLOW INTO ALL FOUR RCS COLD LEGS. UPON IDENTIFICATION OF THE CONCERN, THE VALVE WAS OPENED AND RETURNED TO SERVICE. ADDITIONAL GUIDANCE HAS BEEN PROVIDED TO OPERATIONS PERSONNEL ON THE REPOSITIONING OR REMOVAL FROM SERVICE OF SYSTEM-RELATED EMERGENCY CORE COOLING SYSTEM (ECCS) VALVES. PLANT ENGINEERING HAS REVIEWED ALL APPLICABLE TEST PROCEDURES RELATIVE TO THIS GUIDANCE.

[70] DRESDEN 2 DOCKET 50-237 LER 82-058 REV 01
UPDATE ON LEAK IN REACTOR WATER CLEANUP LINE.
EVENT DATE: 122682 REPORT DATE: 121587 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 207520) DURING NORMAL UNIT OPERATION, WHILE PERFORMING ROUTINE INSPECTIONS, A SMALL LEAK WAS FOUND ON REACTOR WATER CLEANUP LINE 2-1201B 8 INCH -A BETWEEN 2-1201B 8 INCH -A BETWEEN 2-1201-135B AND 2-1201-98B VALVES. LINE IS LOCATED OUTSIDE OF PRIMARY CONTAINMENT AND ALL LEAKAGE WAS CONTAINED IN THE REACTOR BUILDING FLOOR DRAIN SYSTEM. THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT ENDANGERED. A SIMILAR EVENT WAS REPORTED BY R.O. 81-21/03L ON DOCKET NO. 50-249. THE CAUSE OF THE PIPE CRACK WAS DETERMINED TO BE INTERGRANULAR STRESS CORROSION CRACKING (IGSCC) AT THE HEAT AFFECTED ZONE NEAR A WELD. DURING THE 1983 REFUELING OUTAGE, A SECTION OF PIPE APPROXIMATELY 24 INCHES LONG WAS REPLACED WITH THE TYPE 304 DUAL GRADE STAINLESS STEEL IN ACCORDANCE WITH WORK PACKAGE D24780. THIS MATERIAL IS LESS SUSCEPTIBLE TO IGSCC.

[71] DRESDEN 2 DOCKET 50-237 LER 87-035
INSUFFICIENT NUMBER OF OPERABLE REACTOR PROTECTION SYSTEM (RPS) CHANNELS DUE TO PERSONNEL ERROR.
EVENT DATE: 121987 REPORT DATE: 010888 NSSS: GE TYPE: BWR

(NSIC 207883) ON DECEMBER 19, 1987, WITH UNIT 2 OPERATING IN THE RUN MODE AT 89% POWER, AN AVERAGE POWER RANGE MONITOR (APRM) 4 HI HI ALARM WAS RECEIVED AT 0920. WHEN THE UNIT 2 OPERATOR RESPONDED TO THE ALARM, HE NOTICED APRM CHANNEL 4 WAS IN BYPASS ALONG WITH INTERMEDIATE RANGE MONITOR (IRM) 16. THIS CONFIGURATION IS NOT ALLOWED BY TECH SPECS BECAUSE IT REDUCES THE NUMBER OF OPERABLE APRM DOWNSCALE TRIPS IN REACTOR PROTECTION SYSTEM (RPS) CHANNEL B TO ONE. APRM 4 HAD BEEN BYPASSED THE DAY BEFORE AT 1505 DURING A ROUTINE APRM CALIBRATION. IRM 16 HAD BEEN PLACED IN BYPASS ON NOVEMBER 13, 1987, DUE TO ERRATIC BEHAVIOR. THE ROOT CAUSE OF THE EVENT WAS ATTRIBUTED TO PERSONNEL ERROR. THE UNIT 2 OPERATOR INVOLVED WITH THE APRM CALIBRATION THE PREVIOUS DAY FAILED TO CHECK THE ALLOWABLE-NONALLOWABLE APRM/IRM BYPASS CONFIGURATIONS PRIOR TO BYPASSING APRM/IRM

BYPASS CONFIGURATIONS PRIOR TO BYPASSING APRM 4. IMMEDIATE CORRECTIVE ACTIONS INCLUDED PROCEDURE CHANGES REQUIRING THE UNIT OPERATOR TO CHECK THE APRM/IRM BYPASS SWITCHES DURING SHIFT TURNOVER. A MEMO WAS ISSUED TO LICENSED PERSONNEL REMINDING THEM OF THE TECHNICAL SPECIFICATION REQUIREMENT.

[72] DRESDEN 3 DOCKET 50-249 LER 87-005 REV 02
UPDATE ON PRIMARY CONTAINMENT STRUCTURAL STEEL CONNECTIONS OUTSIDE FINAL SAFETY ANALYSIS REPORT DESIGN CRITERIA DUE TO APPARENT ORIGINAL CONSTRUCTION OVERSIGHT.
EVENT DATE: 031687 REPORT DATE: 120987 NSSS: GE TYPE: BWR
VENDOR: BETHLEHEM STEEL

(NSIC 207521) ON MARCH 16, 1987 WITH UNIT 3 IN THE STARTUP MODE, STATION PERSONNEL WERE NOTIFIED BY THE STATION NUCLEAR ENGINEERING DEPARTMENT (SNED) THAT UNIT 3 PRIMARY CONTAINMENT DRYWELL STRUCTURAL STEEL DID NOT MEET THE FINAL SAFETY ANALYSIS REPORT (FSAR) DESIGN REQUIREMENTS DUE TO INADEQUATE CONNECTIONS FOUND BETWEEN RADIAL AND TANGENTIAL BEAMS. A REVIEW OF THE UNIT 3 DRYWELL STRUCTURAL STEEL HAD BEEN INITIATED AS A RESULT OF SIMILAR DISCREPANCIES FOUND ON DRESDEN UNIT 2, WHICH WERE PREVIOUSLY REPORTED UNDER LICENSEE EVENT REPORT NO. 87-003 ON DOCKET 50-237. AN INSPECTION DATA ASSESSMENT COMPLETED ON MARCH 16, 1987 REVEALED THAT FOUR TANGENTIAL BEAM CONNECTIONS DID NOT MEET FSAR DESIGN CRITERIA. FURTHER INSPECTIONS WERE SUBSEQUENTLY PERFORMED DURING A UNIT 3 SHUTDOWN DURING AUGUST 1987. AN INSPECTION DATA ASSESSMENT COMPLETED ON OCTOBER 29, 1987 INDICATES THAT 11 ADDITIONAL CONNECTIONS ARE IN EXCESS OF THE FSAR DESIGN CRITERIA. IT IS BELIEVED THAT THE AS-BUILT CONDITION WAS NOT ADEQUATELY VERIFIED WITH THE DESIGN PRINTS DURING ORIGINAL CONSTRUCTION. THE SAFETY SIGNIFICANCE OF THIS EVENT HAS BEEN CONSIDERED MINIMAL SINCE THE AS-FOUND CONDITION OF THE STRUCTURAL STEEL CONNECTIONS WAS ADEQUATE TO MEET OPERABILITY REQUIREMENTS UNDER ALL DESIGN BASIS EVENTS. THESE DEFICIENCIES WILL BE REPAIRED.

[73] FARLEY 2 DOCKET 50-364 LER 87-010
UNIT SHUT DOWN DUE TO PRESSURE BOUNDARY LEAKAGE.
EVENT DATE: 120987 REPORT DATE: 010688 NSSS: WE TYPE: PWR

(NSIC 207689) AT 2255 ON 12-8-87, WITH THE UNIT AT 33% POWER FOLLOWING A REFUELING OUTAGE, IT WAS OBSERVED THAT THE CONTAINMENT COOLER DRAIN POT LEVELS WERE ABNORMALLY HIGH. A REACTOR COOLANT SYSTEM (RCS) LEAKAGE CALCULATION CONFIRMED THAT THE RCS UNIDENTIFIED LEAKAGE HAD INCREASED. A CONTAINMENT ENTRY WAS MADE AND A LEAK WAS IDENTIFIED IN THE VICINITY OF THE B LOOP RESISTANCE TEMPERATURE DETECTOR (RTD) MANIFOLD. A UNIT SHUTDOWN WAS MADE TO REPAIR THE LEAK. AFTER SHUTDOWN AND UPON CLOSER EXAMINATION, THE PRESSURE BOUNDARY LEAKAGE WAS IDENTIFIED TO BE FROM THE RCS LOOP B COLD LEG SAFETY INJECTION LINE BETWEEN A CHECK VALVE AND THE RCS LOOP. THE LEAK RESULTED FROM A THROUGH WALL DEFECT IN A WELDED JOINT BETWEEN A LONG RADIUS ELBOW AND A STRAIGHT SECTION OF PIPE. THE SECTION OF PIPING CONTAINING THE DEFECT HAS BEEN REPLACED. PRELIMINARY RESULTS OF THE METALLURGICAL EVALUATION OF THE FAILED JOINT HAVE IDENTIFIED A FATIGUE MECHANISM AS THE CAUSE FOR CRACK INITIATION AND PROPAGATION. FURTHER INVESTIGATION AND ANALYSIS ARE BEING PERFORMED TO DETERMINE THE SOURCE OF THE FATIGUE. A REVIEW OF CONSTRUCTION RADIOGRAPHS AND NONDESTRUCTIVE EXAMINATIONS (ULTRASONIC TESTS AND RADIOGRAPHIC TESTS) PERFORMED AS A RESULT OF THIS EVENT REVEALED THAT NO PROBLEMS EXIST ON SIMILAR PIPING WELDS IN THE COLD LEG INJECTION LINES OF EITHER UNIT.

[74] FERMI 2 DOCKET 50-341 LER 87-055
ISOLATION OF REACTOR CORE ISOLATION COOLING DURING RETURN TO SERVICE DUE TO PROCEDURAL INADEQUACIES.
EVENT DATE: 121087 REPORT DATE: 010988 NSSS: GE TYPE: BWR

(NSIC 207684) ON DECEMBER 10, 1987 AT 1414 HOURS, AN ISOLATION OF THE REACTOR

CORE ISOLATION COOLING (RCIC) SYSTEM STEAM LINE OCCURRED. THE ISOLATION VALVES CLOSED ON HIGH DIFFERENTIAL STEAM LINE PRESSURE. THE RCIC STEAM LINE WAS BEING WARMED AT THE TIME OF THIS EVENT. INVESTIGATION DETERMINED THAT THE ISOLATION WAS DUE TO AN INADEQUACY IN THE SYSTEM OPERATING PROCEDURE. WHEN AN ISOLATION VALVE WAS OPENED, HIGH DIFFERENTIAL PRESSURE CONDITION OCCURRED IN THE STEAM LINE. THE OPERATORS RESTORED RCIC TO STANDBY CONDITION. A REVISION WAS MADE TO THE SYSTEM OPERATING PROCEDURE WHICH DIRECTS THE OPERATOR TO THROTTLE THE INBOARD AND OUTBOARD ISOLATION VALVES TOGETHER. THIS WILL PROVIDE GREATER CONTROL WHEN WARMING THE RCIC STEAM LINE.

[75] FERMI 2 DOCKET 50-341 LER 87-054
ISOLATION OF REACTOR BUILDING VENTILATION DURING PERFORMANCE OF A SURVEILLANCE DUE TO POOR EQUIPMENT ACCESSIBILITY.
EVENT DATE: 121787 REPORT DATE: 011688 NSSS: GE TYPE: BWR

(NSIC 207899) DURING THE PERFORMANCE OF THE "REACTOR BUILDING VENTILATION EXHAUST RADIATION MONITOR, DIVISION I CALIBRATION" PROCEDURE, AN UPSCALE RADIATION SIGNAL WAS RECEIVED FROM THE REACTOR BUILDING VENTILATION EXHAUST RADIATION MONITOR. THIS RESULTED IN A DIVISION I STANDBY GAS TREATMENT SYSTEM ACTUATION AND THE ISOLATION OF THE REACTOR BUILDING HEATING VENTILATION AND AIR CONDITIONING SYSTEM. THE CAUSE OF THIS EVENT WAS EQUIPMENT ACCESSIBILITY. THE LOCATION AND DESIGN CONFIGURATION OF THE TERMINALS MADE IT DIFFICULT TO INSTALL AND REMOVE JUMPERS. THE BLOWN FUSE WAS REPLACED AND THE SYSTEM ACTUATIONS RESET. THE CALIBRATION WAS THEN COMPLETED. A DESIGN CHANGE WILL EVALUATE THE USE OF STAR LUGS IN ORDER TO IMPROVE EQUIPMENT ACCESSIBILITY FOR TESTING. THIS EVENT WILL BE ADDED TO THE REQUIRED READING LIST FOR INSTRUMENT AND CONTROL TECHNICIANS.

[76] FERMI 2 DOCKET 50-341 LER 87-056
REACTOR SCRAM DUE TO PERSONNEL ERROR AND SUBSEQUENT REACTOR WATER CLEANUP SYSTEM ISOLATION.
EVENT DATE: 123187 REPORT DATE: 013088 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 208030) ON DECEMBER 31, 1987 DURING THE INSTALLATION OF A TEMPORARY MONITOR AND PRINTER, AN INCREASING DEMAND SIGNAL WAS RECEIVED BY THE FEEDWATER CONTROL SYSTEM DUE TO THE ACCIDENTAL GROUNDING OF A CABLE. FEEDWATER FLOW INCREASED UNTIL HIGH REACTOR VESSEL WATER LEVEL CONDITION OCCURRED. A REACTOR SCRAM OCCURRED APPROXIMATELY 30 SECONDS LATER. IN THE PROCESS OF PLANT COOLDOWN, THE REACTOR WATER CLEANUP SYSTEM (RWCU) ISOLATED DUE TO A DIFFERENTIAL FLOW CONDITION. THE PERSONNEL INVOLVED HAD ALTERED A PLANT INPUT TO THE UNIT WITHOUT PROPERLY ASSESSING THE CONSEQUENCES. APPROPRIATE LEVELS OF DISCIPLINE WERE ADMINISTERED. THE TEMPORARY MODIFICATION PROCEDURE IS BEING REVISED TO ADD ADDITIONAL INSTALLATION GUIDANCE. THE POSITION INDICATION AND RELAYS FOR THE VENT VALVE WHICH CONTRIBUTED TO THE RWCU ISOLATION WAS REPLACED. A DESIGN CHANGE WHICH WILL PREVENT RECURRENCE OF THE ISOLATION IS SCHEDULED TO BE IMPLEMENTED DURING THE LOCAL LEAK RATE TESTING OUTAGE THIS SPRING.

[77] FITZPATRICK DOCKET 50-333 LER 87-011 REV 01
UPDATE ON FIRE BARRIER ELECTRICAL PENETRATION SEALS NOT INSTALLED DUE TO PERSONNEL ERROR.
EVENT DATE: 072887 REPORT DATE: 012888 NSSS: GE TYPE: BWR

(NSIC 208027) DURING PERFORMANCE OF MAINTENANCE PROCEDURE MP-76.11 (ELECTRICAL PENETRATION SEAL SURVEILLANCE INSPECTION) ON 7/28/87 WITH THE PLANT IN NORMAL OPERATION AND AT 98 PERCENT REACTOR POWER, IT WAS DISCOVERED THAT NUMEROUS ELECTRICAL CONDUIT AND FLOOR/WALL PENETRATIONS IN 10CFR50, APPENDIX R, FIRE BARRIERS WERE NOT SEALED WITH A THREE-HOUR FIRE RATED SEAL AS REQUIRED BY TECHNICAL SPECIFICATION SECTION 3.12.F. CONTINUOUS FIRE WATCHES WERE POSTED AT

THE AFFECTED FIRE BARRIERS PER TECHNICAL SPECIFICATION SECTION 3.12.F.1.B. CORRECTIVE ACTIONS WERE INITIATED TO INSTALL THREE-HOUR FIRE SEALS IN THE UNSEALED PENETRATIONS. ALL UNSEALED PENETRATIONS WERE SEALED PRIOR TO COMPLETION OF THE SURVEILLANCE TEST ON 8/14/87 EXCEPT FOR THREE UNSCHEDULED LIGHTING CONDUITS LOCATED IN THE REACTOR BUILDING. A CONTINUOUS FIRE WATCH WAS POSTED AND SEALING OF THESE PENETRATIONS WAS COMPLETED ON 9/3/87. THE CAUSE OF THE UNSEALED PENETRATIONS IS ATTRIBUTED TO USE OF AN INCOMPLETE LIST OF FIRE BARRIER PENETRATIONS ORIGINALLY GENERATED FOR PERFORMANCE OF THE SUBJECT SURVEILLANCE TEST AND 10CFR50, APPENDIX R, COMPLIANCE REVIEWS. THERE ARE NO RELATED LER EVENTS IN WHICH FIRE BARRIER PENETRATION SEALS WERE NOT PROPERLY INSPECTED OR SEALED.

[78] FITZPATRICK DOCKET 50-333 LER 87-019
FAILURE TO PERFORM SURVEILLANCE TEST AS A RESULT OF TECHNICAL SPECIFICATION MISINTERPRETATION.
EVENT DATE: 120787 REPORT DATE: 010688 NSSS: GE TYPE: BWR

(NSIC 207682) ON DECMEBER 7, 1987, WHILE OPERATING AT 100% POWER, A REVIEW OF PLANT PROCEDURES INDICATED THAT A REQUIRED TEST OF THE STANDBY GAS TREATMENT SYSTEM (SGTS) (BH) WAS NOT PERFORMED AS REQUIRED AND HAD NOT BEEN PREVIOUSLY PERFORMED. THIS TEST INVOLVES DETERMINING FILTER EFFICIENCY OF THE HIGH EFFICIENCY PARTICULATE AIR (HEPA) AND CHARCOAL FILTERS DURING EACH SCHEDULED SECONDARY CONTAINMENT (NH) LEAK RATE TEST. THE FAILURE TO PERFORM THIS SURVEILLANCE TEST WAS THE RESULT OF A TECHNICAL SPECIFICATION MISINTERPRETATION. SATISFACTORY PERFORMANCE OF REGULARLY SCHEDULED TESTS PRIOR TO AND FOLLOWING SECONDARY CONTAINMENT LEAK TESTS INDICATE THAT THE EVENT IS NOT SAFETY SIGNIFICANT. CORRECTIVE ACTIONS FOR THIS EVENT WILL BE A CLARIFICATION OF THE SURVEILLANCE TEST REQUIREMENT OR A TECHNICAL SPECIFICATION CHANGE. LER-86-009-00 IS A RELATED EVENT.

[79] FITZPATRICK DOCKET 50-333 LER 87-020
REACTOR TRIP FROM LOW WATER LEVEL ACTUATION DUE TO PERSONNEL ERROR DURING SURVEILLANCE TESTING.
EVENT DATE: 120987 REPORT DATE: 010888 NSSS: GE TYPE: BWR
VENDOR: DRAGON VALVE, INC.

(NSIC 207714) AT 0913 HOURS ON 12/9/87 WITH THE PLANT OPERATING AT 100% POWER, AND WHILE MAKING PREPARATIONS TO PERFORM A TECH SPEC TABLE 4.2-2 SURVEILLANCE TEST ON THE HIGH PRESSURE INJECTION COOLANT (HPCI) SYSTEM (BJ) REACTOR HIGH WATER LEVEL SWITCH (02-3-LIS-101D) A REACTOR TRIP WAS INADVERTENT INITIATED BY AN INSTRUMENT AND CONTROLS (I&C) TECHNICIAN. THE CAUSE OF THE REACTOR TRIP WAS NOT FULLY CLOSING INSTRUMENT PROCESS ISOLATION VALVES BEFORE VALVING IN TEST EQUIPMENT TO THE HPCI REACTOR HIGH WATER LEVEL SWITCH. THIS RESULTED IN A FALSE REACTOR WATER LEVEL TRANSIENT OCCURRING BETWEEN THE SWITCH AND REACTOR PROTECTION SYSTEM (JC) LOW WATER LEVEL TRIP INSTRUMENTATION ON THE COMMON LEVEL HEADER. CORRECTIVE ACTION WAS TO COUNSEL THE I&C TECHNICIAN ON ENSURING INSTRUMENT ISOLATION VALVES ARE FULLY CLOSED BEFORE PRECEDING INTO THE INSTRUMENT TESTING MODE. IN ADDITION INSTRUMENT ISOLATION VALVES WILL BE REPLACED. SIMILAR EVENTS HAVE BEEN REPORTED IN LER 85-007-00, 85-012-00 AND 86-016-00.

[80] FITZPATRICK DOCKET 50-333 LER 87-022
FAILURE TO PERFORM PORTION OF SURVEILLANCE TEST IN REQUIRED FREQUENCY DUE TO OPERATOR OVERSIGHT.
EVENT DATE: 121087 REPORT DATE: 010988 NSSS: GE TYPE: BWR

(NSIC 207895) ON 12/20/87 AT APPROXIMATELY 2000, WHILE CONDUCTING A REACTOR STARTUP, THE SHIFT SUPERVISOR WAS INFORMED THAT PERIODIC COMPUTATION OF DRYWELL LEAKAGE (IJ) RATES REQUIRED BY TECHNICAL SPECIFICATION 4.6.D HAD BEEN

INADVERTENTLY MISSED. THE CAUSE OF THIS EVENT WAS OPERATOR ERROR. THE COLLECTION OF DATA NECESSARY FOR THESE READINGS WAS DUE AT 1600. INVESTIGATION OF THE EVENT INDICATED THAT ATTENTION TO THE NUMEROUS REQUIREMENTS OF REACTOR STARTUP CAUSED THE OPERATORS, WHO NORMALLY PERFORM THIS SURVEILLANCE TO OVERLOOK IT. DRYWELL LEAKAGE RATES WERE FORMALLY CALCULATED UPON DISCOVERY OF THE MISSED READINGS AND WERE FOUND TO BE WELL WITHIN TECHNICAL SPECIFICATION LIMITS. CONTINUOUS CONTROL ROOM RECORDERS MONITORING DRYWELL LEAKAGE PARAMETERS HAD BEEN MONITORED BY THE CONTROL ROOM OPERATOR AT APPROXIMATELY 1600. THE OPERATORS INVOLVED IN THIS EVENT HAVE BEEN COUNSELLED ON THE IMPORTANCE OF PERFORMING SURVEILLANCE AT THE REQUIRED FREQUENCY. IN ADDITION, MODIFICATION PROPOSALS HAVE BEEN SUBMITTED TO PARTIALLY AUTOMATE THE EVOLUTION TO REDUCE MANPOWER REQUIREMENTS AND TO USE THE PLANT COMPUTER TO ALERT THE CONTROL ROOM OPERATOR TO COLLECT THE DATA FOR DRYWELL LEAKAGE RATE COMPUTATION. LER-86-001-00, 86-002-00, 86-009-00 AND 87-019 ARE RELATED EVENTS.

[81] FITZPATRICK DOCKET 50-333 LER 87-021
REACTOR WATER CLEANUP ISOLATION ON HIGH TEMPERATURE DUE TO INADEQUATE PROCEDURE.
EVENT DATE: 121387 REPORT DATE: 011288 NSSS: GE TYPE: BWR

(NSIC 207894) THE REACTOR WATER CLEANUP (RWC) SYSTEM (CE) ISOLATED PER TECH SPEC, TABLE 3.2-1, ON 12/13/87 AT 0330 HOURS DURING NORMAL OPERATION. A 1/2" PIPE FLANGE, LOCATED AT THE RWC PUMP B DISCHARGE, FAILED TO SEAT THE GASKET. REACTOR COOLANT AT APPROX. 520F AND 1200 PSIG FLASHED BETWEEN THE FLANGES. THE CONTROL ROOM RECEIVED AN ALARM ON RWC PUMP B ROOM HIGH TEMPERATURE AND WITHIN TWO MINUTES THE RWC SYSTEM AUTOMATICALLY ISOLATED ON A RWC PUMP B ROOM HIGH TEMPERATURE SIGNAL (JE) WHICH IS PART OF AN ENGINEERED SAFETY FEATURE. THE EVENT CAUSE WAS AN INADEQUATE PROCEDURE. THE PIPE FLANGE BOLT TORQUING PROCEDURE DID NOT ACCOUNT FOR SECONDARY EFFECTS (I.E., ALIGNMENT, THERMAL EXPANSION, AND NORMAL VIBRATION). THE CORRECTIVE ACTION IS TO CHANGE THIS PROCEDURE TO INCLUDE THESE SECONDARY EFFECTS. PURSUANT TO FINAL SAFETY ANALYSIS REPORT, SECTION 7.3 (PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM PCRVICS), SAFETY WAS NOT AFFECTED AND THE SAFETY FUNCTION OF THE PCRVIC FUNCTIONED AS DESIGNED TO ISOLATE THE LEAK IN A TIMELY MANNER. LER 85-019, 85-020, 85-022, 86-004, 87-005 AND 87-007 ARE RELATED.

[82] FT. CALHOUN 1 DOCKET 50-285 LER 87-028 REV 01
UPDATE ON FAILURE TO ISSUE SPECIAL REPORT ON FIRE BARRIER INOPERABILITY.
EVENT DATE: 031987 REPORT DATE: 123187 NSSS: CE TYPE: PWR

(NSIC 207511) ON OCTOBER 8, 1987, IT WAS DISCOVERED THAT A SPECIAL REPORT ON FIRE BARRIER OPERABILITY, REQUIRED BY TECHNICAL SPECIFICATION 5.9.3, HAD NOT BEEN SUBMITTED. THE SPECIAL REPORT PERTAINED TO A FIRE BARRIER SEPARATING FIRE AREAS 41 AND 42 RENDERED INOPERABLE ON MARCH 19, 1987. IN ACCORDANCE WITH TECHNICAL SPECIFICATION 2.19 (7), WITHIN ONE HOUR OPPD PERSONNEL DID VERIFY THE OPERABILITY OF FIRE DETECTION AND ESTABLISHED AN HOURLY FIREWATCH PATROL FOR THE FIRE AREAS AFFECTED. A MAINTENANCE ORDER WAS ISSUED TO REPAIR THE FIRE BARRIER. HOWEVER, THE REPAIRS WERE NOT COMPLETED WITHIN SEVEN DAYS. THEREFORE, A SPECIAL REPORT WAS REQUIRED TO BE SUBMITTED. THE SPECIAL REPORT WAS PREPARED. HOWEVER, DUE TO AN ADMINISTRATIVE OVERSIGHT, IT WAS NOT SUBMITTED. REVISION 1 TO THIS REPORT IS BEING SUBMITTED DUE TO THE FACT THAT THE ORIGINAL LER WAS NOT DESIGNATED AS "OTHER" TO SERVE AS THE ABOVE MENTIONED SPECIAL REPORT IN ADDITION TO FULFILLING THE 10 CFR 50.73 REPORTING REQUIREMENTS.

[83] FT. CALHOUN 1 DOCKET 50-285 LER 87-038
FAILURE OF CONTAINMENT ISOLATION VALVE TO MEET TECHNICAL SPECIFICATION REQUIREMENTS.
EVENT DATE: 113087 REPORT DATE: 123087 NSSS: CE TYPE: PWR

(NSIC 207660) ON NOVEMBER 30, 1987, AT 1758 HOURS CST, A LEAK RATE IN EXCESS OF THE 18,000 SCCM TECHNICAL SPECIFICATION LIMIT WAS FOUND WHILE CONDUCTING A LEAK RATE TEST ON THE CONTAINMENT PURGE ISOLATION VALVES ASSOCIATED WITH MECHANICAL PENETRATION M-87. FORT CALHOUN STATION UNIT 1 WAS OPERATING IN MODE 1 AT APPROXIMATELY 100% POWER AT THE TIME OF THE OCCURRENCE. UPON DISCOVERY OF THE CONDITION, FORT CALHOUN STATION HAD 48 HOURS TO MAKE REPAIRS TO VERIFY THAT LEAKAGE WAS WITHIN ACCEPTABLE LIMITS, PURSUANT TO THE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.5(4)C. THE FOUR-HOUR REPORT REQUIRED BY 10CFR 50.72(B)(2) WAS COMPLETED AT 1935 HOURS. PRECAUTIONS FOR A POSSIBLE LOSS OF CONTAINMENT INTEGRITY (AOP-12) WERE IMPLEMENTED. THE PURGE ISOLATION VALVE OUTSIDE CONTAINMENT WAS FOUND TO BE THE SOURCE OF THE LEAK. THE VALVE POSITION WAS DISCOVERED TO BE SLIGHTLY PAST ITS FULLY CLOSED POSITION. THE PROBLEM WAS CORRECTED BY AN ADJUSTMENT OF THE MECHANICAL STOPS FOR THE ACTUATOR.

[84] FT. ST. VRAIN DOCKET 50-267 LER 87-027
REACTOR SCRAM ACTUATION ON NEUTRON FLUX RATE OF CHANGE HIGH.
EVENT DATE: 111187 REPORT DATE: 121187 NSSS: GA TYPE: HTGR

(NSIC 208070) CAUSE - ELECTRICAL NOISE. ON NOVEMBER 11, 1987, THE REACTOR WAS SHUT DOWN WITH ALL THIRTY-SEVEN CONTROL RODS (AA) FULLY INSERTED IN THE CORE. THE PRESTRESSED CONCRETE REACTOR VESSEL (PCRV) WAS PRESSURIZED TO APPROXIMATELY 43 PSIA. REACTOR CORE COOLING WAS TEMPORARILY SHUTDOWN WHILE ELECTRICAL MAINTENANCE PERSONNEL INVESTIGATED PROBLEMS ASSOCIATED WITH THE INSTRUMENT BUS NO. 2 INVERTER (SEE LER 87-026 FOR DETAILS ON INVERTER PROBLEMS). AT 1932 HOURS, THE "RATE OF CHANGE HIGH" REACTOR SCRAM FUNCTION (4.4 DPM SETPOINT) (JC) WAS ACTUATED BY WIDE RANGE CHANNELS IV AND V (IG). CONTROL ROOM OPERATORS REVIEWED ALL NUCLEAR INSTRUMENTATION AND CONFIRMED THAT AN ERRONEOUS NOISE SIGNAL CAUSED THE ACTUATION AND NO ACTUAL CORE FLUX CHANGES OCCURRED.) THE CAUSE OF THE ELECTRICAL NOISE IN WIDE RANGE CHANNELS IV AND V HAS NOT BEEN IDENTIFIED. A REVIEW OF PLANT ACTIVITIES AT THE TIME OF THE ACTUATION WAS CONDUCTED BUT NO DEFINITIVE CAUSES WERE IDENTIFIED.

[85] FT. ST. VRAIN DOCKET 50-267 LER 87-028
DEFICIENT TEST PROCEDURE CAUSED RESERVE AUXILIARY TRANSFORMER TRIP AND LOSS OF OUTSIDE ELECTRICAL POWER.
EVENT DATE: 120787 REPORT DATE: 010688 NSSS: GA TYPE: HTGR

(NSIC 208072) CAUSE - DEFICIENT PROCEDURES AND ADMINISTRATIVE CONTROL DEFICIENCY. ON DECEMBER 7, 1987, THE REACTOR WAS SHUT DOWN, WITH PRIMARY COOLANT FLOW BEING PROVIDED BY THE "1B" AND "1D" HELIUM CIRCULATORS OPERATING ON THEIR WATER DRIVES, AND SECONDARY COOLANT FLOW BEING PROVIDED BY THE "A" BOILER FEED PUMP SUPPLYING FEEDWATER TO THE LOOP II ECONOMIZER-EVAPORATORSUPERHEATER. THE RESERVE AUXILIARY TRANSFORMER (RAT) WAS SUPPLYING PLANT HOUSE POWER AND THE UNIT AUXILIARY TRANSFORMER WAS CONNECTED TO THE MAIN TURBINE GENERATOR IN PREPARATION OF REACTOR STARTUP. AT 1745 HOURS, DURING PERFORMANCE OF A POST MAINTENANCE TEST ON FIREWATER DELUGE CONTROL RELAY CR-4505, THE RAT FEEDBREAKERS WERE AUTOMATICALLY OPENED DUE TO AN ERROR IN THE TEST PROCEDURE. THE OPENING OF THE RAT FEEDBREAKERS CAUSED A LOSS OF OUTSIDE ELECTRICAL POWER (LOEP), TRIP OF THE "1B" AND "1D" HELIUM CIRCULATORS, LOOP I SHUTDOWN, AND AN AUTOMATIC START OF THE EMERGENCY DIESEL GENERATORS. AT 1847 HOURS, THE RAT WAS RESTORED AND HOUSE POWER WAS RETURNED TO NORMAL. AT 2015 HOURS, REACTOR CORE COOLING WAS RESTORED. THIS EVENT IS BEING REPORTED HEREIN PURSUANT TO THE REQUIREMENTS OF 10 CFR 50.73(A)(2)(IV). DUE TO PERSONNEL ERROR, THE TEST PROCEDURE WAS INCOMPLETE AND FAILED TO DISABLE THE RAT TRIP FUNCTION AS INTENDED.

[86] FT. ST. VRAIN DOCKET 50-267 LER 87-029
REACTOR SCRAM ACTUATION ON NEUTRON FLUX RATE OF CHANGE HIGH.
EVENT DATE: 120787 REPORT DATE: 010688 NSSS: GA TYPE: HTGR

(NSIC 208071) CAUSE - MAINTENANCE PERSONNEL ERROR. ON DECEMBER 7, 1987, AT APPROXIMATELY 2248 HOURS, AND AGAIN ON DECEMBER 8, 1987, AT APPROXIMATELY 1320 HOURS, REACTOR SCRAM ACTUATIONS WERE RECEIVED ON ALL THREE WIDE RANGE CHANNELS NEUTRON FLUX RATE OF CHANGE HIGH. THE CAUSE OF THE REACTOR SCRAM ACTUATIONS WAS DUE TO ELECTRICAL NOISE INDUCED FROM THE INCORRECT LANDING OF A 120V AC LEAD IN A 15V DC PLANT PROTECTIVE SYSTEM (PPS) CIRCUIT. THIS LEAD WAS CONNECTED TO THE INCORRECT TERMINAL DURING THE RETURN OF A CLEARANCE. THE SUBSEQUENT INDEPENDENT VERIFICATION AND FUNCTIONAL TEST DID NOT IDENTIFY THE MISWIRED LEAD PRIOR TO ENERGIZING THE CIRCUIT AND CAUSING THE REACTOR SCRAM ACTUATIONS. SINCE THE REACTOR REMAINED SHUTDOWN WITH ALL 37 CONTROL ROD PAIRS FULLY INSERTED IN THE CORE AND THEIR POWER SUPPLY BREAKERS OPEN DURING THESE ACTUATIONS, THE ACTUATIONS AFFECTED ALARM CIRCUITRY ONLY. NO CONTROL ROD MOVEMENT OCCURRED. THE SUPERINTENDENT OF MAINTENANCE DISCUSSED THIS EVENT IN DETAIL WITH THE INDIVIDUALS WHO RETURNED AND VERIFIED THE CLEARANCE. THE MISPOSITIONED LEAD WAS RETURNED TO THE PROPER TERMINAL LOCATION AND RELAY XCR-93159-A AND LOGIC MODULE CT-1AL1 WERE REPLACED.

[87] GINNA DOCKET 50-244 LER 87-006
INADVERTENT ATTENDANT COOLING UNIT INOPERABILITY DUE TO OPEN BREAKER CAUSES 1D STANDBY AUXILIARY FEEDWATER PUMP TO BE DEEMED INOPERABLE BEYOND THE TECH SPEC LIMIT.
EVENT DATE: 113087 REPORT DATE: 123087 NSSS: WE TYPE: PWR

(NSIC 207712) ON NOVEMBER 30, 1987 AT 1330 EST WITH THE UNIT AT 100% REACTOR POWER THE 1D STANDBY AUXILIARY FEEDWATER PUMP (SAFWP) WAS DEEMED TO HAVE BEEN INOPERABLE FOR LONGER THAN THE TECH SPECS ALLOWED. THIS WAS DUE TO THE ATTENDANT SAFWP ROOM COOLING UNIT 1B AC ELECTRICAL BREAKER BEING FOUND IN THE OFF POSITION. IMMEDIATE CORRECTIVE ACTION TAKEN WAS TO RETURN THE AFFECTED BREAKER TO THE ON POSITION. THE INTERMEDIATE CAUSE OF THE EVENT WAS INDETERMINATE AS NO REASON OR ACTION WAS FOUND FOR THE BREAKER TO BE IN THE OFF POSITION. THE ROOT CAUSE OF THE EVENT WAS DETERMINED TO BE THE LEVEL OF AWARENESS CONCERNING OPERATOR AND MAINTENANCE ACTIONS AS THEY RELATE TO THE RELATIVELY NEW OPERABILITY TECH SPECS. CORRECTIVE ACTION TO PREVENT RECURRENCE WAS TO LOCK THE AFFECTED BREAKER IN THE ON POSITION AND TO REQUEST TRAINING TO DESIGN AND IMPLEMENT MODULES FOCUSING ON THE RELATIVELY NEW OPERABILITY TECH SPECS.

[88] GINNA DOCKET 50-244 LER 87-007
DISCOVERY OF APPARENT DESIGN INADEQUACY CAUSES POTENTIAL FOR LOSS OF CORE COOLING DURING THE HIGH HEAD RECIRCULATION PHASE.
EVENT DATE: 121887 REPORT DATE: 011788 NSSS: WE TYPE: PWR

(NSIC 207938) ON DECEMBER 18, 1987 DURING THE REVIEW OF A WESTINGHOUSE CORPORATION LETTER ENTITLED "OPERATING PLANT FEEDBACK - NON-VITAL POWER SUPPLY USED IN VALVE INTERLOCK LOGIC," IT WAS DISCOVERED THAT THE POTENTIAL EXISTED FOR A LOSS OF CORE COOLING DURING THE HIGH HEAD RECIRCULATION PHASE. EVEN THOUGH THE REVIEW DETERMINED NO SUSCEPTABILITY TO THE CONDITION AS DESCRIBED IN THE REFERENCED LETTER, FURTHER EVALUATION REVEALED THE BELOW DESCRIBED DEFICIENCY. THE APPARENT ROOT CAUSE OF THE EVENT WAS IDENTIFIED AS A DESIGN FLAW, IN THAT A COMMON POWER SUPPLY WAS UTILIZED TO POWER A MOTOR OPERATED VALVE ON EACH TRAIN OF THE HIGH HEAD RECIRCULATION SYSTEM. A POSTULATED FAILURE OF THE ELECTRICAL POWER SUPPLY PRIOR TO OPENING OF THE SUBJECT VALVES WOULD RESULT IN BOTH FLOW PATHS LEADING TO THE SAFETY INJECTION AND CONTAINMENT SPRAY PUMPS BEING BLOCKED, CREATING POTENTIAL LOSS OF CORE COOLING. (SEE GINNA USFAR FOR CONFIGURATION DESCRIPTION.) CORRECTIVE ACTION TO PREVENT RECURRENCE WAS TO POSITION THE

AFFECTED VALVE IN THE ONE-OF-TWO SERIES ARRANGEMENT TO THE OPEN POSITION, THEREBY ELIMINATING THE POTENTIAL FOR COMMON MODE FAILURE.

[89] GINNA DOCKET 50-244 LER 87-008
INOPERABLE SAFEGUARDS CIRCUIT BREAKERS DUE TO ZERO CLEARANCE BETWEEN AMPTECTOR ACTUATOR ARM AND BREAKER TRIP BAR CAUSES POSSIBILITY OF COMMON MODE FAILURE.
EVENT DATE: 122387 REPORT DATE: 012288 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 207952) ON DECEMBER 21, 1987 AT 1411 EST WITH THE UNIT AT 100% REACTOR POWER THE 1B RHR PUMP FAILED TO START FOR TESTING DUE TO ZERO CLEARANCE BETWEEN ITS BREAKERS AMPTECTOR ACTUATOR ARM AND THE TRIPPER BAR. FOLLOW UP TESTING OF SELECTED SAFEGUARDS BREAKERS REVEALED A SECOND FAILURE ON DECEMBER 23, 1987 OF THE 1B SAFETY INJECTION PUMP. BECAUSE A MAJORITY OF THE SAFETY RELATED BREAKERS ARE OF THIS SAME DESIGN, A POSSIBILITY OF COMMON MODE FAILURE EXISTED. IMMEDIATE CORRECTIVE ACTION WAS TO INSPECT, ADJUST AND TEST THE AFFECTED BREAKERS AND RETURN THEM TO OPERABLE STATUS. THE ROOT CAUSE OF THE EVENT WAS DETERMINED TO BE INADEQUATE DESIGN INFORMATION PROVIDED BY THE VENDOR DURING MODIFICATION INSTALLATION OF THE AMPTECTOR DEVICES. CORRECTIVE ACTION TO PREVENT RECURRENCE INCLUDES CHECKING AND SETTING, IF NECESSARY, THE CLEARANCE BETWEEN THE AMPTECTOR ACTUATOR ARM AND BREAKER TRIPPER BAR DURING PREVENTIVE MAINTENANCE ON THESE BREAKERS.

[90] GRAND GULF 1 DOCKET 50-416 LER 87-018
LATE SURVEILLANCE DURING CORE ALTERATIONS.
EVENT DATE: 111487 REPORT DATE: 121487 NSSS: GE TYPE: BWR

(NSIC 207474) CORE ALTERATIONS BEGAN ON NOVEMBER 14, 1987. ON NOVEMBER 18, 1987 IT WAS DISCOVERED THAT THE SURVEILLANCE PROGRAM TRACKING SYSTEM (SPTS) REPORT DID NOT INCLUDE ADEQUATE INFORMATION ON SURVEILLANCES FOR THE OPERATIONAL CONDITION "DURING CORE ALTERATION AND OPERATIONS WITH A POTENTIAL FOR DRAINING THE REACTOR VESSEL". PRIOR TO THIS DATE THE SPTS REPORT WAS BEING USED TO TRACK THOSE SURVEILLANCES REQUIRED FOR OPERATIONAL CONDITIONS 4 AND 5. ONE PROCEDURE, 06-IC-IB21-M-1004, REQUIRED FOR CORE ALTERATIONS EXCEEDED THE NRC LATE DATE ON NOVEMBER 15, 1987. CORE ALTERATIONS WERE IMMEDIATELY STOPPED UPON DISCOVERY OF THE MISSED SURVEILLANCE AND THE LATE PROCEDURE WAS STARTED. OPERATIONS ISSUED A LIMITING CONDITION FOR OPERATIONS (LCO) REPORT AND PERFORMED THE REQUIRED ACTIONS OF TECHNICAL SPECIFICATIONS. THE SURVEILLANCE PROCEDURE WAS PERFORMED SUCCESSFULLY AND CORE ALTERATIONS RESUMED. THE CORRECTED SPTS REPORT WAS REVIEWED TO DETERMINE IF OTHER LATE SURVEILLANCES EXISTED. EACH DISCIPLINE SUPERVISOR WAS NOTIFIED OF THE OCCURRENCE AND IT WAS VERIFIED THAT EACH HAD A CORRECT AND CURRENT SPTS REPORT. TO PREVENT FUTURE OCCURRENCES, THE INTEGRATED OPERATING INSTRUCTIONS (IOI) HAS BEEN REVISED TO REQUIRE TECHNICAL SUPPORT SUPERINTENDENT VERIFICATION PRIOR TO ENTERING OPERATIONAL CONDITION.

[91] GRAND GULF 1 DOCKET 50-416 LER 87-019
HEAVY LOAD TRANSPORTED OVER NEW FUEL IN UPPER CONTAINMENT POOL FUEL STORAGE RACKS.
EVENT DATE: 111787 REPORT DATE: 121787 NSSS: GE TYPE: BWR

(NSIC 207475) DURING REFUELING OPERATIONS ON NOVEMBER 17, 1987 THE CONTAINMENT POLAR CRANE OPERATOR MOVED A REACTOR VESSEL HEAD STUD TENSIONER OVER THE SOUTHWEST CORNER OF THE UPPER CONTAINMENT POOL FUEL STORAGE RACKS. NO SPENT FUEL WAS STORED IN THIS AREA OF THE RACKS; HOWEVER, NEW FUEL WAS. THE WEIGHT OF THE REACTOR VESSEL HEAD STUD TENSIONER EXCEEDS THE 1140 POUND LOAD LIMIT OF TECHNICAL SPECIFICATION 3.9.7. PERSONNEL ON THE REFUELING FLOOR AT THE TIME OBSERVED THE EVENT, STOPPED THE MOVEMENT, AND HAD THE LOAD REROUTED TO A SAFE LOCATION USING A CORRECT PATHWAY. FUEL MOVEMENTS AND CRANE OPERATIONS WERE SUSPENDED UNTIL AN EVALUATION OF CRANE OPERATIONS COULD BE PERFORMED AND CORRECTIVE ACTION TAKEN TO

PREVENT RECURRENCE. THIS EVENT IS ATTRIBUTABLE TO INADEQUATE CONTROL OF CERTAIN ACTIVITIES PERFORMED BY THE REFUELING FLOOR CONTRACTOR, GENERAL ELECTRIC COMPANY (GE) AND SYSTEM ENERGY RESOURCES, INC. (SERI). GE WAS REQUIRED TO FORMALIZE CONTROL OVER REFUELING FLOOR ACTIVITIES. THIS CONTROL INCLUDED STRICTER CONTROLS ON REFUELING FLOOR CRANE ACTIVITIES. ADDITIONAL LICENSEE PERSONNEL WERE PLACED ON SHIFT WITH RESPONSIBILITY TO OVERSEE GE'S PERFORMANCE. THESE LICENSEE PERSONNEL REMAINED ON SHIFT LONG ENOUGH TO ENSURE THAT THE CORRECTIVE ACTIONS WERE EFFECTIVE.

[92] GRAND GULF 1 DOCKET 50-416 LER 87-020
SHUTDOWN COOLING ISOLATION DUE TO BLOWN FUSE.
EVENT DATE: 111987 REPORT DATE: 121787 NSSS: GE TYPE: BWR

(NSIC 207476) ON NOVEMBER 19, 1987 DURING A PLANT REFUELING OUTAGE, A 5 AMP FUSE IN THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM ISOLATION LOGIC BLEW CAUSING AN ISOLATION OF RESIDUAL HEAT REMOVAL (RHR) CONTAINMENT ISOLATION VALVE E12-F009 AND A TRIP OF THE OPERATING RHR SHUTDOWN COOLING PUMP. SHUTDOWN COOLING WAS OUT OF OPERATION FOR ONE HOUR AND TWO MINUTES FOR INVESTIGATION AND THE REPLACEMENT OF THE FUSE. THE REACTOR RECIRCULATION SYSTEM WAS OPERATED AS THE ALTERNATE METHOD OF REACTOR COOLANT CIRCULATION. THE LACK OF SHUTDOWN COOLING OPERATION CAUSED NO ADVERSE SAFETY CONSEQUENCES. THE CAUSE FOR THE BLOWN FUSE IS INDETERMINATE. THE INVESTIGATION COULD NOT DETERMINE WHETHER THE FUSE BLEW AT LESS THAN NOMINAL DESIGN CURRENT OR BLEW DUE TO A POWER TRANSIENT. ALTHOUGH A SURVEILLANCE WAS IN PROGRESS AT THE TIME OF THE EVENT, NO ACTIONS ASSOCIATED WITH ITS PERFORMANCE COULD BE ATTRIBUTED TO THE BLOWN FUSE.

[93] GRAND GULF 1 DOCKET 50-416 LER 87-021
SHUTDOWN COOLING SUCTION VALVE ISOLATED DUE TO PERSONNEL ERROR.
EVENT DATE: 113087 REPORT DATE: 123087 NSSS: GE TYPE: BWR

(NSIC 207695) ON 11/30/87 CONTAINMENT ISOLATION VALVE E12-FO09 ISOLATED WHEN OPERATORS RESTORED POWER TO THE DIVISION 2 ESF BUS FOLLOWING A SCHEDULED POWER OUTAGE. THE VALVE IS LOCATED ON THE COMMON SUCTION PIPE OF BOTH SHUTDOWN COOLING LOOPS. SHUTDOWN COOLING PUMP A TRIPPED WHEN THE VALVE CLOSED. OPERATORS PERFORMED A PANEL WALKDOWN AND DETERMINED THAT THE BREAKER SUPPLYING POWER TO VALVE E12-FO09 WAS CLOSED AND THE BREAKER SUPPLYING POWER TO THE ISOLATION LOGIC WAS OPEN. THE VALVE ISOLATED WHEN POWER WAS RESTORED TO ITS MOTOR OPERATOR BECAUSE OF THE DE-ENERGIZED ISOLATION LOGIC. IT WAS RECOGNIZED THAT THE ISOLATION LOGIC WAS DE-ENERGIZED PRIOR TO RESTORING BUS POWER. OPERATORS WERE PROVIDED A LIST OF ACTIONS TO PERFORM PRIOR TO RESTORING BUS POWER. ONE OF THE ACTIONS WAS TO OPEN THE BREAKER TO VALVE E12-F009 WHICH WOULD ENSURE THE VALVE DID NOT ISOLATE. THE OPERATOR BELIEVED THE BREAKER TO ALREADY BE OPENED BECAUSE THE VALVE POSITION INDICATOR LIGHTS WERE DE-ENERGIZED. HOWEVER, THE BREAKER WAS ACTUALLY CLOSED AND THE ABSENCE OF POWER TO THE VALVE INDICATOR LIGHTS WAS DUE TO THE BUS OUTAGE. LONG TERM CORRECTIVE ACTION WILL BE TO PROCEDURALIZE MAJOR POWER OUTAGES. FOR THE REMAINDER OF THE PRESENT REFUELING OUTAGE, A LICENSED PERSON ON THE OPERATION'S SUPPORT STAFF WILL DEVELOP CLEARLY WRITTEN INSTRUCTIONS FOR ENERGIZING AND DE-ENERGIZING EQUIPMENT FOR MAJOR ELECTRICAL OUTAGES.

[94] GRAND GULF 1 DOCKET 50-416 LER 87-022
RWCU ISOLATION DUE TO BLOWN FUSE CAUSED BY WORKING CONDITIONS.
EVENT DATE: 120787 REPORT DATE: 010688 NSSS: GE TYPE: BWR

(NSIC 207696) ON DECEMBER 6, 1987 AT 1815 REACTOR WATER CLEANUP (RWCU) ISOLATION VALVE 1G33-F004 CLOSED DUE TO A HIGH TEMPERATURE SIGNAL RECEIVED FROM TEMPERATURE SWITCH N1G33-N008. THIS TEMPERATURE SIGNAL IS NOT AN ESF ACTUATION. THE SWITCH TRIPPED PRIOR TO REACHING ITS CALIBRATED SETPOINT OF 140 DEGREES F AND IS BELIEVED TO HAVE BEEN TRIPPED BY RADIO FRFQUENCY INTERFERENCE (RFI). RWCU WAS

RESTORED TO SERVICE IN FIVE MINUTES. A MAINTENANCE WORK ORDER WAS INITIATED. INSTRUMENTATION AND CONTROL TECHNICIANS INVESTIGATED THE TEMPERATURE SWITCH SINCE RWCU WAS BEING USED AS AN ALTERNATE SHUTDOWN COOLING METHOD. ON DECEMBER 7, 1987 AT 0130 RWCU ISOLATION VALVES 1G33-F004, F054, AND F039 CLOSED DUE TO AN ESF ACTUATION CAUSED BY A BLOWN FUSE. THE FUSE BLEW WHILE TROUBLESHOOTING TEMPERATURE SWITCH N1G33-N008 WHICH HAD TRIPPED EARLIER DURING THE SHIFT AS DISCUSSED ABOVE. THE FUSE WAS REPLACED. HOWEVER, THE FUSE BLEW A SECOND TIME AT 0200 DURING THE TROUBLESHOOTING EFFORT. THE SWITCH IS LOCATED IN AN AREA THAT REQUIRES FULL ANTI-CONTAMINATION CLOTHING AND IS MOUNTED IN A SMALL METAL ENCLOSURE ON PANEL 1H22-P004 WHICH MAKES ACCESS AND TROUBLESHOOTING DIFFICULT. THESE CONDITIONS CONTRIBUTED TO THE INCIDENT. A LIMITING CONDITION FOR OPERATIONS WAS ENTERED AT 0130. THE SECOND FUSE WAS REPLACED AND THE SYSTEM RESTORED TO NORMAL AT 0220.

[95] GRAND GULF 1 DOCKET 50-416 LER 87-023
INADVERTENT DIVISION II LOCA SIGNAL GENERATED DURING RESTORATION FROM SURVEILLANCE PROCEDURE.
EVENT DATE: 120887 REPORT DATE: 010788 NSSS: GE TYPE: BWR

(NSIC 207648) ON DECEMBER 8, 1987 WHILE PERFORMING THE RESTORATION SECTION FROM SURVEILLANCE PROCEDURE "RHR CONTAINMENT SPRAY INITIATION LOGIC SYSTEM FUNCTIONAL TEST," A LICENSED REACTOR OPERATOR DIRECTED INSTRUMENTATION AND CONTROL (I&C) TECHNICIANS TO REMOVE A TEMPORARY EMERGENCY CORE COOLING SYSTEM (ECCS) TEST SWITCH PRIOR TO RESETTING THE RESIDUAL HEAT REMOVAL (RHR) "B"/RHR "C" INITIATION LOGIC. THIS WAS NOT THE CORRECT SEQUENCE SPECIFIED IN THE APPROVED PROCEDURE. THIS PERSONNEL ERROR CAUSED AN INADVERTENT DIVISION II LOSS OF COOLANT ACCIDENT (LOCA) SIGNAL AND ACTUATED THE COMBUSTIBLE GAS CONTROL SYSTEM "B" TRAIN.

[96] GRAND GULF 1 DOCKET 50-416 LER 87-024
RPS ACTUATION CAUSED BY PROCEDURAL ERROR.
EVENT DATE: 121287 REPORT DATE: 011188 NSSS: GE TYPE: BWR

(NSIC 207697) DURING THE 1987 REFUELING OUTAGE, SYSTEM ENERGY RESOURCES, INC. (SERI) IMPLEMENTED CERTAIN DESIGN MODIFICATIONS TO MITIGATE THE CONSEQUENCES OF AN ANTICIPATED TRANSIENT WITHOUT A SCRAM (ATWS) AS REQUIRED BY 10CFR50.62. THE DESIGN CHANGES INCLUDED THE INSTALLATION OF THE ALTERNATE ROD INSERTION (ARI) SYSTEM AND MODIFICATIONS TO THE EXISTING RECIRCULATION PUMP TRIP (RPT) SYSTEM. A SPECIAL TEST INSTRUCTION WAS WRITTEN TO VERIFY PROPER OPERATION OF THE DESIGN CHANGES. THE SPECIAL TEST ISOLATED THE SCRAM VALVE PILOT AIR HEADER. ON DECEMBER 12, 1987 DURING PERFORMANCE OF THE SPECIAL TEST, THE HEADER AIR PRESSURE DIMINISHED FASTER THAN ANTICIPATED CAUSING THE SCRAM VALVES TO DRIFT OPEN AND ALLOWING WATER TO FILL THE SCRAM DISCHARGE VOLUME (SDV). WHEN SDV LEVEL REACHED THE SCRAM SETPOINT, A REACTOR PROTECTION SYSTEM (RPS) SCRAM OCCURRED. THE RPS ACTUATION COULD HAVE BEEN PREVENTED IF THE SPECIAL TEST INSTRUCTION HAD REQUIRED THE SDV BYPASS SWITCHES TO BE PLACED IN THE BYPASS POSITION. TEST ENGINEERS WERE AWARE THAT THE AIR HEADER WOULD BE ISOLATED BUT BELIEVED THE HEADER PRESSURE COULD BE RESTORED PRIOR TO THE RPS TRIP. THERE WERE NO ADVERSE SAFETY CONSEQUENCES SINCE ALL CONTROL RODS WERE ALREADY INSERTED AND THE SPECIAL TEST CONFIGURATION WAS NOT INTENDED FOR USE DURING POWER OPERATION.

[97] GRAND GULF 1 DOCKET 50-416 LER 87-025
RPS ACTUATION CAUSED BY INCORRECT INSTALLATION OF FUSE DURING SURVEILLANCE.
EVENT DATE: 121987 REPORT DATE: 011888 NSSS: GE TYPE: BWR

(NSIC 207915) ON DECEMBER 19, 1987 WITH THE REACTOR IN COLD SHUTDOWN, A REACTOR PROTECTION SYSTEM (RPS) ACTUATION OCCURRED DURING PERFORMANCE OF SURVEILLANCE PROCEDURE "MAIN STEAM LINE ISOLATION VALVE CLOSURE CALIBRATION." THIS ACTUATION WAS NOT PART OF THE PLANNED SEQUENCE OF THE SURVEILLANCE TEST. THE

INSTRUMENTATION & CONTROL (I&C) TECHNICIANS PERFORMING THE SURVEILLANCE HAD REMOVED ONE FUSE AND TESTED FOR A HALF SCRAM IN RPS CHANNEL A. THE FUSE WAS TO BE REPLACED AND ANOTHER FUSE REMOVED TO TEST FOR A HALF SCRAM IN RPS CHANNEL B. THE FUSE FOR RPS CHANNEL A WAS REPLACED AND TESTING PROCEEDED TO RPS CHANNEL B. AFTER REMOVING THE FUSE FOR RPS CHANNEL B, MAIN STEAM ISOLATION VALVE (MSIV) 1B21-F022A WAS SLOW CLOSED. A HALF SCRAM IN RPS CHANNEL B WAS EXPECTED; HOWEVER, A FULL SCRAM OCCURRED. AN INSPECTION REVEALED THAT THE FUSE REINSTALLED FOR RPS CHANNEL A WAS NOT MAKING GOOD CONTACT THEREFORE A HALF SCRAM WAS ALSO RECEIVED IN RPS CHANNEL A. SCRAM SIGNALS TO BOTH RPS CHANNELS CAUSED THE FULL SCRAM. THE TWO INDIVIDUALS WERE REPRIMANDED FOR FAILING TO INSTALL THE FUSE CORRECTLY.

[98] HATCH 1 DOCKET 50-321 LER 87-016
PROCEDURE INADEQUACY RESULTS IN FALSE CHLORINE SIGNAL CAUSING CONTROL ROOM ISOLATION.
EVENT DATE: 121487 REPORT DATE: 011388 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 2 (BWR)
VENDOR: WALLACE & TIERMAN, INC.

(NSIC 207891) ON 12/14/87 AT APPROXIMATELY 183, CST, UNIT 1 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2436 MWT (APPROXIMATELY 100 PERCENT OF RATED THERMAL POWER). UNIT 2 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 1830 MWT (APPROXIMATELY 75 PERCENT OF RATED THERMAL POWER). AT THAT TIME, THE MAIN CONTROL ROOM ENVIRONMENTAL CONTROL (MCREC EIIS CODE VI) SYSTEM WENT INTO THE ISOLATION MODE OF OPERATION AS A RESULT OF A SENSED HIGH CHLORINE SIGNAL. THE ROOT CAUSE OF THIS EVENT IS A PROCEDURE INADEQUACY. SPECIFICALLY, THE SURVEILLANCE PROCEDURE DOES NOT SPECIFY WHEN ELECTROLYTE LEVELS MUST BE REPLENISHED. AS A RESULT, THE CHLORINE DETECTOR'S ELECTROLYTE RESERVOIR RAN DRY AND A FALSE HIGH CHLORINE SIGNAL WAS GENERATED. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED: 1) VERIFYING A HIGH CHLORINE CONDITION DID NOT EXIST, 2) LEAVING MCREC IN THE ISOLATION MODE UNTIL THE EVENT WAS THOROUGHLY INVESTIGATED, 3) REPAIRING, REPLACING, AND RECALIBRATING THE CHLORINE DETECTOR, 4) CHECKING THE ELECTROLYTE LEVEL WEEKLY UNTIL PROCEDURE REVISIONS ARE IN PLACE, AND 5) INITIATING A PROCEDURE REVISION.

[99] HATCH 2 DOCKET 50-366 LER 87-009 REV 01
UPDATE ON CIRCUIT BREAKER FAILS CAUSING POWER FAILURE RESULTING IN REACTOR SCRAM.
EVENT DATE: 080387 REPORT DATE: 011888 NSSS: GE TYPE: BWR
VENDOR: HEINEMANN ELECTRIC CO.

(NSIC 207902) ON 8/3/87 AT APPROXIMATELY 1152 CDT, UNIT 2 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2193 MWT (APPROXIMATELY 90 PERCENT OF RATED THERMAL POWER). AT THAT TIME, VITAL AC (EIIS CODE EE) POWER WAS LOST. THIS RESULTED IN A DECREASE IN THE REACTOR FEEDWATER PUMPS FLOW AND A DECREASE IN REACTOR WATER LEVEL. THE REACTOR WATER LEVEL DECREASED TO THE REACTOR PROTECTION SYSTEM (RPS EIIS CODE JC) ACTUATION SETPOINT AND A REACTOR SCRAM OCCURRED. THE ROOT CAUSE OF THIS EVENT IS ELECTRICAL EQUIPMENT FAILURE. SPECIFICALLY, CIRCUIT BREAKER CB-4 WOULD OPEN UNDER UNDULY LOW FORCE CONDITIONS. IT WAS CONCLUDED AFTER FIELD TESTING AND CONSULTATION WITH THE MANUFACTURER THAT THE TRIPPING MECHANISM WAS WEAK. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED: 1) INSTALLING JUMPERS AND REMOVING EQUIPMENT FROM SERVICE, 2) DESIGNING AND INSTALLING BARRIER BOXES, 3) VERIFYING TRIP INSTRUMENTATION AND LEVEL TRANSMITTERS IN CALIBRATION, 4) VENTING INSTRUMENT LINES AND TRANSMITTERS, 5) PERFORMING EVALUATIONS OF AIR ENTRAINMENT AND SPIKING IN INSTRUMENT LINES, 6) INITIATING PROCEDURE REVISIONS, AND 7) VERIFYING CERTAIN OTHER SYSTEMS DO NOT HAVE LOW SUCTION TRIPS.

[100] HATCH 2 DOCKET 50-366 LER 87-017
EQUIPMENT AGING CAUSES DEFECTIVE AMPLIFIER RESULTING IN LOSS OF AUTOMATIC SAFETY FUNCTION.
EVENT DATE: 111987 REPORT DATE: 121587 NSSS: GE TYPE: BWR
VENDOR: BAILEY METER COMPANY

(NSIC 207456) ON 11/19/87 AT APPROXIMATELY 0900 CST, UNIT 2 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2066 MWT (APPROXIMATELY 85 PERCENT OF RATED THERMAL POWER). AT THAT TIME, PLANT OPERATIONS PERSONNEL WERE PERFORMING A SURVEILLANCE PROCEDURE AND DETERMINED THAT THE HIGH PRESSURE COOLANT INJECTION (HPCI EIIS CODE BJ) SYSTEM WOULD NOT FUNCTION CORRECTLY IN THE AUTOMATIC MODE OF OPERATION. THIS WAS A CONDITION THAT COULD HAVE PREVENTED THE AUTOMATIC FULFILLMENT OF THE SAFETY FUNCTION OF HPCI TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT. THE HPCI SYSTEM WOULD FUNCTION CORRECTLY IN THE MANUAL MODE OF OPERATION. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO NORMAL EQUIPMENT AGING. SPECIFICALLY, A DEFECTIVE AMPLIFIER CARD AND A DEFECTIVE SOLDER JOINT WERE FOUND IN SUB-MODULES OF THE CONTROLLER AMPLIFIER. THE CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED: 1) REPLACING THE DEFECTIVE AMPLIFIER CARD, 2) REPAIRING THE SOLDER JOINT, 3) CALIBRATING THE CONTROLLER AMPLIFIER, AND 4) DEMONSTRATING THAT THE HPCI SYSTEM WAS OPERABLE IN THE AUTOMATIC MODE OF OPERATION.

[101] HATCH 2 DOCKET 50-366 LER 87-016
SUPERVISOR ERROR RESULTS IN MISSED BATTERY SURVEILLANCE TEST.
EVENT DATE: 123187 REPORT DATE: 011888 NSSS: GE TYPE: BWR

(NSIC 207903) ON 12/31/87 AT APPROXIMATELY 0815 CST, UNIT 2 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 1828 MWT (APPROXIMATELY 75 PERCENT OF RATED THERMAL POWER). AT THAT TIME, PLANT MAINTENANCE PERSONNEL DETERMINED THAT A BATTERY SURVEILLANCE HAD NOT BEEN PERFORMED WITHIN THE ALLOWABLE TIME LIMITS OF THE PLANT'S TECHNICAL SPECIFICATIONS. THE SURVEILLANCE PERIOD WAS MISSED BY APPROXIMATELY ONE DAY. THE ROOT CAUSE OF THIS EVENT IS COGNITIVE PERSONNEL ERROR ON THE PART OF A PRIMARY SURVEILLANCE FOREMAN. THE FOREMAN BELIEVED THAT THE SURVEILLANCE HAD BEEN PERFORMED WHEN, IN FACT, IT HAD NOT. BASED ON THIS ASSUMPTION, THE FOREMAN SIGNED THE SURVEILLANCE SCHEDULE TO INDICATE THE SURVEILLANCE WAS COMPLETED. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED: 1) PERFORMING THE SURVEILLANCE, 2) VERIFYING THAT THE SURVEILLANCE RESULTS WERE SATISFACTORY, 3) MODIFYING THE WEEKLY SURVEILLANCE SCHEDULE SHEETS TO INCLUDE A TRANSMITTAL NUMBER, 4) LISTING WEEKLY SURVEILLANCES AND OUTSTANDING SURVEILLANCE ON THE SURVEILLANCE SCHEDULE SHEETS, 5) LISTING THE SURVEILLANCES ON THE PLAN OF THE DAY TRACKING SHEETS, AND 6) COUNSELING THE INVOLVED SUPERVISOR RELATIVE TO THIS EVENT.

[102] HOPE CREEK 1 DOCKET 50-354 LER 87-048
TRIP OF "B" AND "D" SAFETY AUXILIARIES COOLING SYSTEM PUMPS AND AUTO START OF "A" SACS PUMP DUE TO PROCEDURAL AND DESIGN DEFICIENCIES.
EVENT DATE: 112587 REPORT DATE: 122487 NSSS: GE TYPE: BWR

(NSIC 207548) ON 11/25/87 AT APPROXIMATELY 0800, THE "B" AND "D" SAFETY AUXILIARIES COOLING SYSTEM (SACS) PUMPS TRIPPED ON LOW DIFFERENTIAL PRESSURE DURING ROUTINE VENTING OF THE "B" AND "D" EMERGENCY DIESEL GENERATOR ROOM COOLERS. PRIOR TO THE PUMPS TRIPPING, SACS LOOP "B" WAS SUPPLYING THE TURBINE AUXILIARIES COOLING SYSTEM (TACS). WHEN THE PUMPS TRIPPED, TACS SUPPLY AUTOMATICALLY SWAPPED TO SACS LOOP "A", AND "A" SACS PUMP AUTOMATICALLY STARTED. INVESTIGATION SUBSEQUENT TO THE INCIDENT DETERMINED THAT THE LOW DIFFERENTIAL PRESSURE TRIP OF "B" AND "D" SACS PUMPS OCCURRED DUE TO A HIGH SYSTEM FLOW CONDITION THAT WAS CAUSED BY SIMULTANEOUS VENTING OF THE "B" AND "D" EDG ROOM COOLERS. PAST OPERATIONAL EXPERIENCE INDICATES THAT NITROGEN BUBBLES IN THE SACS SYSTEM MAY HAVE ALSO CONTRIBUTED TO THE LOW DIFFERENTIAL PRESSURE CONDITION. CORRECTIVE ACTIONS INCLUDED REVISING THE SACS VENTING PROCEDURES TO VENT ONLY ONE

EDG ROOM COOLER AT A TIME, AND EXPEDITING A PREVIOUSLY IDENTIFIED DESIGN CHANGE TO PREVENT PROPAGATION OF NITROGEN FROM THE SACS HYDROPNEUMATIC ACCUMULATORS TO THE SACS LOOPS.

[103] HOPE CREEK 1 DOCKET 50-354 LER 87-050
MISSED SURVEILLANCE OF A MSIV OUTBOARD STEAM SEALING GAS TEST LINE ISOLATION VALVE DUE TO PERSONNEL ERROR.
EVENT DATE: 120487 REPORT DATE: 122187 NSSS: GE TYPE: BWR

(NSIC 207556) ON DECEMBER 4, 1987 AT 1500 HOURS THE PLANT WAS IN OPERATIONAL CONDITION 1 (POWER OPERATION) AT 100% POWER GENERATING 1100 MWE. AT THIS TIME IT WAS DETERMINED THAT THE OUTBOARD MSIV STEAM SEALING GAS TEST LINE ISOLATION VALVE SURVEILLANCE WHICH WAS DUE ON NOVEMBER 11, 1987 TEST HAD BEEN MISSED AND THE OUTBOARD MSIV STEAM SEALING SYSTEM HAD THEREFORE BEEN INOPERABLE FROM NOVEMBER 11, 1987 TO DECEMBER 4, 1987 - A TECH SPEC VIOLATION. THE ISOLATION VALVE WAS TESTED SATISFACTORY AND THE OUTBOARD MSIV STEAM SEALING SYSTEM WAS DECLARED OPERABLE. THE ROOT CAUSE OF THIS OCCURRENCE WAS AN OVERSIGHT IN THE PERFORMANCE OF A SCHEDULED SURVEILLANCE TEST - A PERSONNEL ERROR. THE PROCEDURE FOR SCHEDULING AND TRACKING OF OPERATIONS DEPARTMENT SURVEILLANCE TESTING WILL BE REINFORCED WITH ALL OPERATIONS PERSONNEL. IN ADDITION, THE OPERATIONS DEPARTMENT COORDINATOR WILL BE COUNSELLED ON THE IMPORTANCE OF TRACKING SURVEILLANCE PROCEDURES, AND A REVIEW OF HOPE CREEK STATION SURVEILLANCE TRACKING PROCEDURES WILL BE PERFORMED TO PRECLUDE REOCCURRENCE OF THIS TYPE OF EVENT.

[104] HOPE CREEK 1 DOCKET 50-354 LER 87-051
REACTOR SCRAM CAUSED BY A SPURIOUS SPIKE IN A MAIN STEAM LINE (MSL) RADIATION MONITOR DUE TO EQUIPMENT DEFICIENCY.
EVENT DATE: 120887 REPORT DATE: 010688 NSSS: GE TYPE: BWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 207900) ON DECEMBER 8, 1987 AT 1405 HOURS, THE PLANT WAS IN OPERATIONAL CONDITION 1 (POWER OPERATION) AT 100% POWER GENERATING 1106 MWE WHEN AN AUTOMATIC REACTOR SCRAM OCCURRED. VESSEL LEVEL DECREASED TO -18 IN. FOLLOWING THE SCRAM AND INCREASED RAPIDLY UNTIL ALL REACTOR FEED PUMPS TRIPPED AT +54 IN. THE REACTOR SCRAM WAS RESET AT 1409 HOURS AT WHICH TIME THE REACTOR WAS STABLE. THE ROOT CAUSE OF THIS EVENT WAS THE COMMON GROUNDING OF THE "C" AND "D" CABINETS WHICH PROVIDED A PATH FOR THE SPURIOUS SIGNAL WHICH RESULTED IN THE SCRAM. CONTRIBUTING CAUSES WERE THE FAULTY CABLE ATTACHMENT AT THE "D"MSL RAD MONITOR DRAWER CONNECTOR WHICH CREATED THE SHORT TO GROUND AND THE DEGRADED PHYSICAL CONDITION OF THE CABLE AND DRAWER. CORRECTIVE ACTIONS INCLUDE AN ANALYSIS OF THE MSL RAD MONITOR GROUNDING CONFIGURATION FOR POSSIBLE IMPROVEMENT AND INSTRUCTION OF PERSONNEL IN THE IMPORTANCE OF RESTORING INSTRUMENT DRAWERS TO THEIR ORIGINAL CONFIGURATION.

[105] HOPE CREEK 1 DOCKET 50-354 LER 87-052
REACTOR WATER CLEANUP SYSTEM ISOLATION WHEN PRESSURIZATING THE "B" FILTER/DEMINERALIZER DUE TO INLET DESIGN DEFICIENCY.
EVENT DATE: 121087 REPORT DATE: 011188 NSSS: GE TYPE: BWR

(NSIC 207901) ON DECEMBER 12, 1987 AN ISOLATION OF THE REACTOR WATER CLEANUP SYSTEM (RWCU) PRIMARY CONTAINMENT INBOARD ISOLATION VALVE (HV-F001) OCCURRED WHILE PRESSURIZING THE "B" RWCU FILTER/DEMINERALIZER (F/D). THE SHIFT CHEMISTRY TECHNICIAN WAS ATTEMPTING TO PLACE THE "B" RWCU F/D IN THE "HOLD" MODE FOLLOWING PRECOATING WHEN THE RWCU STEAM LEAK ISOLATION TIMER INITIATED ON HIGH DIFFERENTIAL FLOW. AFTER 45 SECONDS, THE RELAY TIMED OUT, RESULTING IN AN AUTOMATIC CLOSURE OF HV-F001 AND TRIPPING BOTH OPERATING RWCU PUMPS. THE ISOLATION WAS RESET AND "A" AND "B" RWCU PUMPS WERE RESTARTED APPROXIMATELY 10 MINUTES AFTER THE ISOLATION OCCURRED. INVESTIGATIONS OF PREVIOUS SIMILAR RWCU

ISOLATIONS HAVE DETERMINED THAT A DESIGN DEFICIENCY EXISTS IN THE SYSTEM WHICH CAN LEAD TO A HIGH DIFFERENTIAL FLOW CONDITION WHEN PRESSURIZING AN F/D FOLLOWING PRECOATING. CORRECTIVE ACTIONS INCLUDE MINOR PROCEDURAL CHANGES AND THE SCHEDULING OF A PREVIOUSLY IDENTIFIED DESIGN CHANGE FOR THE 1988 REFUELING OUTAGE.

[106] HUMBOLDT BAY DOCKET 50-133 LER 87-002
ACTIVATION OF GAS TREATMENT SYSTEM.
EVENT DATE: 112287 REPORT DATE: 121787 NSSS: GE TYPE: BWR

(NSIC 207412) ON NOVEMBER 22, 1987, WHILE THE UNIT WAS IN MODE N (SHUTDOWN) IN THE PROCESS OF BEING DECOMMISSIONED, THE GAS TREATMENT SYSTEM WAS ACTIVATED WHEN AN AREA RADIATION MONITOR IN THE REFUELING BUILDING SPIKED TRANSFERRING THE REFUELING BUILDING VENTILATION SYSTEM TO THE "BUILDING ABOVE NORMAL RADIATION" MODE OF OPERATION. NO ABNORMAL PLANT RADIATION LEVELS EXISTED AT THE TIME OF THE ACTIVATION. OPERATING PERSONNEL VERIFIED THAT PLANT CONDITIONS WERE NORMAL AND RESTORED THE GAS TREATMENT SYSTEM TO ITS STANDBY MODE. THE ROOT CAUSE OF THE EVENT WAS AN ELECTRICAL DISTURBANCE ON THE 60 KV TRANSMISSION SYSTEM EXTERNAL TO HUMBOLDT BAY POWER PLANT. DUE TO THE RANDOM NATURE OF THE EVENTS NO CORRECTIVE ACTION WAS DEEMED NECESSARY.

[107] HUMBOLDT BAY DOCKET 50-133 LER 87-003
ACTIVATION OF GAS TREATMENT SYSTEM DUE TO INADVERTENT GROUNDING OF AN ISOLATION MONITOR.
EVENT DATE: 120787 REPORT DATE: 010688 NSSS: GE TYPE: BWR

(NSIC 207667) ON DECEMBER 7, 1987, WHILE THE UNIT WAS IN MODE N (SHUTDOWN) IN THE PROCESS OF BEING DECOMMISSIONED, THE GAS TREATMENT SYSTEM WAS ACTIVATED WHEN A TECHNICIAN GROUNDED THE ISOLATION MONITOR WHILE REPAIRING A BROKEN INPUT LEAD FROM THE ISOLATION MONITOR TO THE AREA MONITOR RECORDER. NO ABNORMAL PLANT RADIATION LEVELS EXISTED AT THE TIME OF THE ACTIVATION. OPERATING PERSONNEL VERIFIED THAT PLANT CONDITIONS WERE NORMAL AND RESTORED THE GAS TREATMENT SYSTEM TO THE STANDBY MODE. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR (COGNITIVE). THE TECHNICIAN WAS NOT AWARE THAT THE SOLDERING IRON USED IN THE REPAIR WOULD GROUND THE MONITOR. CORRECTIVE ACTION TO PREVENT RECURRENCE INCLUDED INSTRUCTION OF ALL INSTRUMENTATION AND CONTROL (I&C) PERSONNEL IN THE PROPER USE OF SOLDERING IRONS AND THE ASSOCIATED HAZARDS.

[108] INDIAN POINT 2 DOCKET 50-247 LER 87-017
ENVIRONMENTAL QUALIFICATION OF ELECTRICAL SPLICES.
EVENT DATE: 120187 REPORT DATE: 123187 NSSS: WE TYPE: PWR
VENDOR: CROUSE-HINDS
RAYCHEM CORP.

(NSIC 207670) IN RESPONSE TO IE NOTICE 86-53, AN INSPECTION PROGRAM WAS INITIATED DURING THE CURRENT REFUELING OUTAGE TO EVALUATE THE SAFETY RELATED ELECTRICAL SPLICES LOCATED IN THOSE AREAS OF INDIAN POINT UNIT 2 (IP2) WHICH WOULD BE SUBJECT TO A HARSH ENVIRONMENT DURING ACCIDENT CONDITIONS. THE PURPOSE OF THE INSPECTION WAS TO VERIFY THAT THE FIELD INSTALLATION MATCHED THE TESTED (QUALIFIED) CONFIGURATION. THE RESULTS OF THIS PROGRAM INDICATED THAT CERTAIN SPLICES DID NOT MEET THE ACCEPTANCE CRITERIA, AND HENCE THE AFFECTED ELECTRICAL DEVICES WERE POTENTIALLY UNAVAILABLE IN A HIGH ENERGY LINE BREAK ENVIRONMENT. THE AFFECTED ELECTRICAL SPLICES WERE REPLACED WITH QUALIFIED SPLICES DURING THE 1987 REFUELING OUTAGE. PROCEDURAL MEASURES, INCLUDING QA/QC COVERAGE OF REPLACEMENT SPLICE INSTALLATIONS, WERE INSTITUTED. A MEETING WAS HELD WITH THE NRC AT THE REGION I, KING OF PRUSSIA, OFFICE ON DECEMBER 1, 1987. A COMPREHENSIVE PRESENTATION WAS MADE TO THEM REGARDING THE RESULTS OF OUR INSPECTION PROGRAM AND ADDITIONAL QUALIFICATION TESTING, OUR CORRECTIVE ACTIONS, AND REPORTABILITY OF THE EVENT. THE PUBLIC HEALTH AND SAFETY WERE NOT AFFECTED.

[109] INDIAN POINT 2 DOCKET 50-247 LER 87-018
COMMON RHR RECIRCULATION LINE CAN LEAD TO PUMP FAILURE.
EVENT DATE: 120487 REPORT DATE: 010488 NSSS: WE TYPE: PWR

(NSIC 207631) WHILE IN COLD SHUTDOWN FOR A REFUELING OUTAGE, WESTINGHOUSE ELECTRIC CORPORATION PROVIDED NOTIFICATION OF A RESIDUAL HEAT REMOVAL SYSTEM (RHRS) PUMP MINIMUM FLOW (MINIFLOW) CONFIGURATION DESIGN CONCERN APPLICABLE TO INDIAN POINT UNIT NO. 2 (IP-2). THE CONCERN INVOLVES THE POTENTIAL OF DEAD-HEADING ONE OF TWO PUMPS (P) OPERATING IN PARALLEL IN A SYSTEM DESIGN THAT PROVIDES A MINIFLOW RECIRCULATION LINE COMMON TO BOTH PUMPS (P). BASED UPON AN EVALUATION PERFORMED BY THE PUMP MANUFACTURER, DAMAGE, AND ULTIMATELY PUMP (P) FAILURE, CAN BE EXPECTED TO OCCUR WHEN OPERATED IN A DEAD-HEADED CONDITION. AT IP-2, FOR ACCIDENTS WHICH DO NOT INJECT BORATED WATER INTO CONTAINMENT (E.G., STEAMLINE BREAK), THE RHR PUMPS (P) WILL BE OPERATING IN PARALLEL WITH RCS PRESSURE AT OR ABOVE THEIR SHUTOFF HEAD AND WILL THUS BE SUBJECTED TO THE POTENTIAL DEAD-HEADING FAILURE MODE. CURRENT PLANT EMERGENCY OPERATING PROCEDURES DO NOT PROVIDE FOR THE PROTECTION OF A DEAD-HEADED RHR PUMP (P) IN THE EARLY STAGES OF SUCH AN EVENT. THE EOPS ARE BEING REVISED AND WILL BE EFFECTIVE PRIOR TO CRITICALITY TO MAINTAIN THE AVAILABILITY OF BOTH RHR PUMPS (P) FOR ACCIDENT MITIGATION. THE HEALTH AND SAFETY OF THE PUBLIC ARE NOT AFFECTED.

[110] INDIAN POINT 2 DOCKET 50-247 LER 87-019
ACCUMULATOR TANK LEVEL INSTRUMENT CALIBRATION LEVEL ERROR.
EVENT DATE: 120887 REPORT DATE: 010788 NSSS: WE TYPE: PWR
VENDOR: DELTA SOUTHERN CO.

(NSIC 207875) INDIAN POINT 2 WAS AT COLD SHUTDOWN FOR A REFUELING OUTAGE. THE SAFETY INJECTION ACCUMULATOR TANK LEVEL TRANSMITTERS WERE BEING REPLACED WITH UPGRADED EQUIPMENT. DURING THE ENGINEERING REVIEW OF THE CALIBRATION FOR THE NEW TRANSMITTERS IT WAS NOTED THAT THE CALIBRATION OF THE OLD TRANSMITTERS WAS IN ERROR. AS A RESULT THE PLANT HAS BEEN OPERATED WITHIN AN ACTUAL ACCUMULATOR WATER VOLUME RANGE THAT IS 30.3 CUBIC FEET HIGHER THAN THE TECHNICAL SPECIFICATION ALLOWABLE RANGE. HOWEVER, PARAMETRIC STUDIES OF ACCUMULATOR WATER VOLUME DONE IN SUPPORT OF THE PRESENT EMERGENCY CORE COOLING SYSTEM (ECCS) ANALYSIS SHOWED THAT THE ACCEPTANCE CRITERIA OF 10CFR50.46(B) ARE MET EVEN WITH THE GREATER ACCUMULATOR WATER VOLUME. THE NEW TRANSMITTERS ARE BEING CALIBRATED TO THE PROPER LEVEL VOLUME, CORRELATION. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED.

[111] INDIAN POINT 2 DOCKET 50-247 LER 87-016
PRESSURIZER SAFETY VALVES ACTUATION SETPOINT OUT OF TOLERANCE.
EVENT DATE: 121187 REPORT DATE: 122187 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE & GAGE CO.

(NSIC 207418) WHILE INDIAN POINT 2 WAS AT COLD SHUTDOWN FOR A REFUELING OUTAGE, A TEST OF THE THREE PRESSURIZER SAFETY RELIEF VALVES WAS PERFORMED ON OCTOBER 21 AND 22, 1987 . THE AS-FOUND SETPOINT PRESSURES OF TWO OF THE THREE VALVES WERE DETERMINED TO BE SLIGHTLY OUTSIDE THE RANGE PERMITTED BY THE APPLICABLE TECHNICAL SPECIFICATION BY A SMALL MARGIN. SETPOINT PRESSURE VARIATION OF THE MAGNITUDE FOUND IN THE TEST ARE COMMON IN INDUSTRY EXPERIENCE AND ARE TYPICALLY CAUSED BY A COMBINATION OF EROSION OF THE VALVE DISC, THERMAL CYCLING AND SETPOINT DRIFT. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED.

[112] INDIAN POINT 3 DOCKET 50-286 LER 87-011
DISCREPANCIES OF ENVIRONMENTALLY QUALIFIED EQUIPMENT.
EVENT DATE: 113087 REPORT DATE: 123087 NSSS: WE TYPE: PWR
VENDOR: BRAND REX CO.

(NSIC 207677) DURING THE CYCLE 5/6 REFUELING OUTAGE, WITH A FEW EXCEPTIONS, AN INSPECTION OF ALL EQUIPMENT REQUIRED BY THE ENVIRONMENTAL QUALIFICATION (EQ) MASTER LIST WAS CONDUCTED. THE PURPOSE OF THIS INSPECTION WAS TO ENSURE COMPLIANCE WITH THE ENVIRONMENTAL QUALIFICATION REQUIREMENTS. THE FIELD INSPECTION IDENTIFIED VARIOUS DISCREPANCIES RELATED TO ELECTRICAL EQ EQUIPMENT WITHIN THE CONTAINMENT BUILDING. ON NOVEMBER 30, 1987, THE NEW YORK POWER AUTHORITY DETERMINED THAT SEVERAL DISCREPANCIES CONSTITUTED NONCONFORMANCES WITH 10CFR50.49. THE AFFECTED EQUIPMENT WERE: 1) FIVE PRESSURIZER RELIEF AND SAFETY VALVE ACOUSTIC MONITORS, 2) FOUR MOTOR OPERATORS FOR REMOTE ISOLATION VALVES, AND 3) SIX CONTAINMENT BUILDING WATER LEVEL TRANSMITTERS. THE CAUSE FOR THE NONCONFORMANCES WAS A COMBINATION OF INSUFFICIENT PROCEDURAL INSTRUCTIONS AND IMPROPER WORK PRACTICES. SINCE THE PLANT OPERATED IN CYCLE 5 IN EXCESS OF THE ALLOWED TECHNICAL SPECIFICATION OUT OF SERVICE TIME FOR THESE COMPONENTS, THE NONCONFORMANCES ARE REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B). ALL REPORTABLE NONCONFORMANCES WERE CORRECTED PRIOR TO PLANT HEAT-UP FROM THE CYCLE 5/6 REFUELING OUTAGE. IN ORDER TO PREVENT RECURRENCE, PROCEDURAL CHANGES AND TRAINING OF APPLICABLE PERSONNEL HAVE BEEN CONDUCTED.

[113] INDIAN POINT 3 DOCKET 50-286 LER 87-0.2
UNIT TRIP CAUSED BY MALFUNCTIONING RELAY.
EVENT DATE: 122287 REPORT DATE: 012188 NSSS: WE TYPE: PWR
VENDOR: COMBUSTION ENGINEERING, INC.
I.T.E IMPERIAL POWER EQUIPMENT CO.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 207958) ON DECEMBER 22, 1987, WITH THE REACTOR AT 100 PERCENT POWER, AN AUTOMATIC UNIT TRIP OCCURRED WHILE PLANT PERSONNEL WERE PERFORMING A MONTHLY SURVEILLANCE TEST ON THE REACTOR COOLANT LOW FLOW PROTECTION SYSTEM. ALL PLANT SYSTEMS FUNCTIONED PROPERLY FOLLOWING THE TRIP WITH TWO EXCEPTIONS: 1) MOTOR OPERATED DISCONNECT FL-3 DID NOT AUTOMATICALLY OPEN FOLLOWING THE OPENING OF GENERATOR OUTPUT BREAKERS AND WOULD NOT OPEN VIA ITS CONTROL ROOM FLIGHT PANEL SWITCH, AND 2) THE PLANT COMPUTER PROVIDED AN INCOMPLETE SEQUENCE OF EVENTS REPORT. IT WAS DETERMINED THAT, THE TRIPPING OF A BISTABLE SWITCH ASSOCIATED WITH A LOW REACTOR COOLANT FLOW LOGIC RELAY DURING THE TEST (LOOP 4, CHANNEL I) IN CONJUNCTION WITH A HIGH RESISTANCE CONTACT ON A SIMILAR RELAY IN THE SAME LOOP (LOOP 4, CHANNEL III) RESULTED IN THE DE-ENERGIZATION OF A LOW REACTOR COOLANT FLOW MATRIX OUTPUT RELAY AND CAUSED THE REACTOR TRIP. REACTOR COOLANT FLOW WAS ACTUALLY NORMAL AT THE TIME OF THE TRIP. THE RELAY FOR LOOP 4, CHANNEL III WAS FOUND TO BE FAULTY DUE TO TARNISH BUILD-UP ON A RELAY CONTACT. AFTER REPLACING THE FAULTY RELAY, THE REACTOR WAS BROUGHT CRITICAL ON DECEMBER 23, 1987. THE UNIT WAS SYNCHRONIZED TO THE BUS ON DECEMBER 24, 1987.

[114] KEWAUNEE DOCKET 50-305 LER 87-012
POTENTIAL FOR CONTROL VALVE FAILURE IN THE NON-SAFE MODE DUE TO OVERPRESSURIZED SOLENOID VALVES.
EVENT DATE: 112887 REPORT DATE: 020888 NSSS: WE TYPE: PWR
VENDOR: ASCO VALVES
FISHER CONTROLS CO.

(NSIC 208022) ON NOVEMBER 28, 1987, WITH THE PLANT AT FULL POWER, TWO INDEPENDENT, CONTAINMENT ISOLATION CONTROL VALVES FAILED TO CLOSE FROM THE CONTROL ROOM DURING PERFORMANCE OF THEIR INSERVICE TESTING (IST). THE TWO VALVES, MU-1010-1 AND RC-507, FAILED TO CLOSE BECAUSE THEIR RESPECTIVE SOLENOID VALVES WERE UNABLE TO BLOCK THE INSTRUMENT AIR FLOW TO THE CONTROL VALVES WHEN DEENERGIZED. THE SOLENOID VALVES, WHICH HAVE A MAXIMUM OPERATING PRESSURE DIFFERENTIAL (MOPD) OF 70 PSI, WERE FOUND TO HAVE THEIR CORRESPONDING AIR LINE REGULATORS SET AT 80 PSI. THIS RESULTED IN THE SOLENOID VALVES BEING OVERPRESSURIZED. THESE SOLENOID VALVES HAD BEEN REPLACED AS A PART OF A RECENT DESIGN CHANGE DURING WHICH TIME THE INCORRECT MODELS, WITH TOO LOW AN MOPD, WERE

PURCHASED AND INSTALLED. A FIELD WALKDOWN OF OTHER SAFETY RELATED SOLENOID VALVES IN THE PLANT AND THEIR AIR REGULATOR SETTINGS WAS PERFORMED. NO OTHER CASES WERE FOUND WHERE THE AIR REGULATOR SETTING WOULD HAVE PREVENTED THE SOLENOID FROM OPERATING PROPERLY. HOWEVER, OTHER CASES WERE FOUND WHERE THE SOLENOID VALVES HAD AN MOPD LESS THAN INSTRUMENT AIR PRESSURE. PLANNED CORRECTIVE ACTIONS INCLUDE REPLACING SOLENOID VALVES AND ESTABLISHING A PROGRAM TO FORMALIZE INSTRUMENT AIR REGULATOR SETTINGS. THIS EVENT IS BEING REPORTED IN THE "OTHER" CATEGORY BECAUSE OF ITS POTENTIAL INTEREST TO OTHER LICENSEES.

[115] KEWAUNEE DOCKET 50-305 LER 87-011
POTENTIAL FAILURE OF MODEL 62H CONTROLLER INTERNAL RELAYS DUE TO OVERVOLTAGE CONDITION AS DESCRIBED BY FOXBORO IN A 10CFR21 REPORT.
EVENT DATE: 122187 REPORT DATE: 012288 NSSS: WE TYPE: PWR
VENDOR: FOXBORO CO., THE

(NSIC 207959) ON 12/8/87, WHILE THE PLANT WAS OPERATING AT 100% POWER, WISCONSIN PUBLIC SERVICE CORPORATION (WPSC) WAS ADVISED BY ANOTHER UTILITY OF A CONCERN REGARDING THE PREMATURE FAILURES OF CERTAIN RELAYS IN FOXBORO 62H CONTROLLERS. THESE SAME RELAYS WERE INSTALLED AT KNPP AS PART OF AN OVERALL CONTROLLER REFURBISHMENT PROJECT FOR BOTH THE REACTOR PROTECTION (RPS) AND PLANT PROCESS CONTROL SYSTEMS. INCLUDED IN THE SCOPE OF REFURBISHMENT IS THE CHANGEOUT OF A NON-HERMETICALLY SEALED RELAY TO A HERMETICALLY SEALED UNIT TO ELIMINATE FAILURES CAUSED BY CONTACT CORROSION. THE INVESTIGATION INTO THE CAUSE OF THE RELAY FAILURES IS NOT COMPLETE; HOWEVER, INITIAL INVESTIGATIONS STRONGLY SUGGEST THAT THE RELAYS FAILED DUE TO AN OVERVOLTAGE CONDITION. PRESENTLY, THE VOLTAGE ACROSS BOTH THE ORIGINALLY INSTALLED AND THE RECENTLY INSTALLED REPLACEMENT RELAYS IS HIGHER THAN THEIR RATED VOLTAGE. HOWEVER, THE ORIGINAL RELAYS DO NOT APPEAR AS SUSCEPTIBLE TO PREMATURE FAILURES. ALTHOUGH THE REPLACEMENT RELAYS ARE SUSCEPTIBLE TO PREMATURE FAILURE, THEY ARE OPERABLE AND CAPABLE OF PERFORMING THEIR SAFETY RELATED FUNCTION. PERIODIC SURVEILLANCE TESTING PROVIDES CONTINUED ASSURANCE OF RELAY OPERABILITY. IN ADDITION, A LETTER WHICH IDENTIFIES THE SYSTEMS WHICH USE THE 62H CONTROLLER AND PROVIDES CONTINGENT INSTRUCTIONS HAS BEEN MADE REQUIRED READING FOR ALL OPERATORS.

[116] LA SALLE 1 DOCKET 50-373 LER 87-035
SPURIOUS AMMONIA DETECTOR TRIP DUE TO FAILURE OF THE DETECTOR'S FRONT OPTICS INDICATOR LAMP.
EVENT DATE: 110887 REPORT DATE: 120387 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 2 (BWR)
VENDOR: M D A SCIENTIFIC, INC.

(NSIC 207406) AT 2050 HOURS ON NOVEMBER 8, 1987, WITH UNITS 1 AND 2 IN OPERATIONAL CONDITION 1 (RUN) AT 88% AND 83% POWER RESPECTIVELY, THE "B" CONTROL ROOM HVAC SYSTEM (VC) "B" AMMONIA DETECTOR (OXY-VC165B) TRIPPED. PER DESIGN, AN ENGINEERED SAFETY FEATURE (ESF) DAMPER ACTUATION OCCURRED WHICH ISOLATED THE "B" VC TRAIN FROM THE OUTSIDE AIR AND RECIRCULATED THE AIR FLOW THROUGH THE "ODOR EATER" (CHARCOAL ADSORBER). THE INSTRUMENT MAINTENANCE DEPARTMENT INVESTIGATED THE EVENT AND FOUND THAT THE FRONT OPTICS INDICATOR LAMP HAD FAILED. THE LAMP WAS REPLACED AND THE DETECTOR WAS RETURNED TO SERVICE AT 1400 HOURS ON NOVEMBER 9, 1987. THIS WAS THE FIRST OCCURRENCE OF A FAILED OPTICS LAMP CAUSING AN AMMONIA DETECTOR TRIP. THE SAFETY CONSEQUENCES OF THIS EVENT WERE MINIMAL SINCE THE "B" VC SYSTEM RESPONDED TO THE AMMONIA DETECTOR TRIP PER DESIGN. THIS EVENT IS REPORTABLE PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV) DUE TO THE ACTUATION OF AN ESF SYSTEM.

[117] LA SALLE 1 DOCKET 50-373 LER 87-037
REACTOR WATER CLEANUP SUCTION ISOLATION VALVE CLOSURE ON HI FILTER/DEMINERALIZER INLET TEMPERATURE DUE TO FAILED TEMPERATURE SWITCH.
EVENT DATE: 121387 REPORT DATE: 011188 NSSS: GE TYPE: BWR
VENDOR: FENWALL, INC.

(NSIC 207722) AT 0046 HOURS ON DECEMBER 13, 1987, WITH UNIT 1 IN OPERATIONAL CONDITION 1 (RUN) AT 83% POWER, THE UNIT 1 REACTOR WATER CLEANUP (RWCU) FILTER/DEMINERALIZER INLET TEMPERATURE SWITCH (TIS-1G33-N008), FAILED IN THE TRIPPED (HIGH TEMPERATURE) CONDITION, CAUSING THE OUTBOARD CONTAINMENT ISOLATION VALVE (1G33-F004) TO AUTOMATICALLY CLOSE VIA THE PRIMARY CONTAINMENT ISOLATION SYSTEM GROUP V OUTBOARD ISOLATION LOGIC. UPON VERIFICATION THAT A HIGH TEMPERATURE CONDITION HAD NEVER EXISTED, A JUMPER WAS INSTALLED TO BYPASS THE HIGH TEMPERATURE TRIP AND THE RWCU SYSTEM WAS RESTARTED AT 0350 HOURS ON THE SAME DAY. THE CAUSE OF THIS EVENT WAS AN INOPERABLE CAPACITOR IN THE TEMPERATURE SWITCH (TIS-1G33-N008) TRIP CIRCUIT. THE SAFETY CONSEQUENCES OF THIS EVENT WERE MINIMAL SINCE THE OUTBOARD ISOLATION VALVE OF THE RWCU SYSTEM CLOSED AS DESIGNED. THE SWITCH WAS REPAIRED AND PLACED BACK IN SERVICE BY DECEMBER 24, 1987, AND NO FURTHER PROBLEMS HAVE BEEN EXPERIENCED. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV) DUE TO AN ENGINEERED SAFETY FEATURE ACTUATION.

[118] LA SALLE 1 DOCKET 50-373 LER 87-038
REACTOR SCRAM ON LOW REACTOR WATER LEVEL CAUSED BY FEEDWATER HEATER STRING ISOLATIONS DUE TO DESIGN DEFICIENCY.
EVENT DATE: 121687 REPORT DATE: 011588 NSSS: GE TYPE: BWR

(NSIC 207945) AT 0955 HOURS ON DECEMBER 16, 1987, WITH UNIT 1 IN OPERATIONAL CONDITION 1 (RUN) AT 77.5% POWER, UNIT 1 SCRAMMED ON LOW REACTOR WATER LEVEL (LEVEL 3, +12.5 INCHES) DUE TO A LEVEL TRANSIENT CAUSED BY FEEDWATER HEATER STRING ISOLATIONS. THE STRING ISOLATIONS OCCURRED WHILE OPERATING PERSONNEL WERE ATTEMPTING TO RESTORE THE "12C" LOW PRESSURE FEEDWATER HEATER TO A NORMAL OPERATING CONDITION. THE ROOT CAUSE OF THE EVENT WAS A DESIGN DEFICIENCY IN THE "11C" FEEDWATER HEATER WHICH CAUSED IT TO BECOME "AIR-BOUND". THIS HEATER WAS ORIGINALLY INTENDED TO HAVE A CONTINUOUS OPERATING VENT, BUT THE VENT WAS NEVER INSTALLED. WITH THE "11C" HEATER "AIR-BOUND," THE "12C" HEATER BECAME OVERLOADED AND A HIGH LEVEL SITUATION OCCURRED (IN THE "12C" HEATER) WHICH INITIATED A STRING ISOLATION. THE SAFETY CONSEQUENCES OF THIS EVENT WERE MINIMAL. ALL REQUIRED ACTUATIONS OCCURRED AS EXPECTED. MODIFICATIONS ARE SCHEDULED DURING THE NEXT UNIT 1 AND UNIT 2 REFUELING OUTAGES TO INSTALL A CONTINUOUS OPERATING VENT ON THE "11/21" AND "12/22" FEEDWATER HEATERS. THIS EVENT IS REPORTABLE PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV) DUE TO THE AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM.

[119] LA SALLE 1 DOCKET 50-373 LER 87-039
REACTOR CORE ISOLATION COOLING WATER LEG PUMP FAILURE.
EVENT DATE: 121687 REPORT DATE: 010688 NSSS: GE TYPE: BWR
VENDOR: CRANE DEMING PUMPS

(NSIC 207723) ON DECEMBER 16, 1987 AT 1720 HOURS, WHILE UNIT 1 WAS IN HOT SHUTDOWN (MODE 3), THE REACTOR CORE ISOLATION COOLING (RCIC) WATER LEG PUMP SEIZED. AT THAT TIME, THE UNIT 1 RCIC SYSTEM WAS DECLARED INOPERABLE, WITH TECH SPEC 3/4.7.3 BEING APPLICABLE. UNIT 1 ENTERED COLD SHUTDOWN (MODE 4) AT 0215 HOURS ON DECEMBER 17, 1987 IN ORDER TO PERFORM CERTAIN MAINTENANCE ACTIVITIES THAT WERE REQUIRED AT THE TIME. THE CAUSE OF THE WATER LEG PUMP FAILURE IS UNKNOWN AT THIS TIME. THE SAFETY CONSEQUENCES OF THIS EVENT WERE MINIMAL DUE TO ADHERENCE TO TECH SPEC 3/4.7.3. THE HIGH PRESSURE CORE SPRAY (HPCS) SYSTEM WAS OPERABLE THROUGHOUT THE EVENT. BY 2046 HOURS ON DECEMBER 17, 1987 THE UNIT 1 RCIC WATER LEG PUMP WAS REPLACED, SATISFACTORILY TESTED, AND DECLARED OPERABLE.

UNIT 1 REMAINED IN COLD SHUTDOWN UNTIL DECEMBER 19, 1987. THIS EVENT IS REPORTABLE IN ACCORDANCE WITH 10CFR50.73(A)(2)(VII) DUE TO AN EVENT THAT CAUSED AN INOPERABILITY OF A SYSTEM THAT REMOVES RESIDUAL HEAT.

[120] LA SALLE 1 DOCKET 50-373 LER 87-040
1A DIESEL GENERATOR OUTPUT BREAKER FAILURE TO CLOSE DUE TO BAD CONTACT IN CLOSING CIRCUIT.
EVENT DATE: 121887 REPORT DATE: 011388 NSSS: GE TYPE: BWR
VENDOR: POTTER & BRUMFIELD

(NSIC 207946) ON DECEMBER 18, 1987 AT APPROXIMATELY 1314 HOURS, WITH UNIT 1 IN COLD SHUTDOWN (OPERATIONAL CONDITION 4) AT 0% POWER, LASALLE OPERATING SURVEILLANCE LOS-DG-M2, "1A DIESEL GENERATOR OPERABILITY TEST," WAS BEING PERFORMED. THE UNIT 1 NUCLEAR STATION OPERATOR (NSO, LICENSED REACTOR OPERATOR) HAD STARTED THE "1A" DIESEL GENERATOR (DG) AND ATTEMPTED TO SYNCHRONIZE IT TO BUS 142Y. THE "1A" DG OUTPUT BREAKER WOULD NOT CLOSE. SEVERAL ATTEMPTS WERE MADE AND ALL WERE UNSUCCESSFUL. TROUBLESHOOTING DETERMINED THAT VOLTAGE PERMISSIVE RELAY 27X DID NOT OPERATE AS DESIGNED. THE CAUSE OF THIS EVENT WAS DUE TO THE FAILURE OF A CONTACT ON THE 27X RELAY, LOCATED IN THE CLOSING CIRCUIT OF THE "1A" DG OUTPUT BREAKER, TO CLOSE AND MAKE ELECTRICAL CONTACT. THE SAFETY CONSEQUENCES OF THIS EVENT WERE MINIMAL. THE "1B" AND "0" DIESEL GENERATORS AND ALL UNIT 1 EMERGENCY CORE COOLING SYSTEMS, WERE OPERABLE AT THE TIME OF THIS EVENT. ALL TECH SPEC REQUIREMENTS WERE MET. THE 27X RELAY WAS REPLACED, AND LOS-DG-M2 WAS RE-PERFORMED. THE OUTPUT BREAKER SUCCESSFULLY CLOSED ONTO BUS 142Y AND THE REST OF THE SURVEILLANCE WAS PERFORMED SATISFACTORILY. THIS IS A VALID TEST FAILURE PER REGULATORY GUIDE 1.108 AND IS REQUIRED TO BE REPORTED AS A SPECIAL REPORT PER TECH SPEC 4.8.1.1.3.

[121] LA SALLE 1 DOCKET 50-373 LER 87-041
AMMONIA DETECTOR TRIP DUE TO CHEMCASSETTE OUT OF TAPE CAUSED BY PROCEDURE WEAKNESS.
EVENT DATE: 122287 REPORT DATE: 012188 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 2 (BWR)
VENDOR: M D A SCIENTIFIC, INC.

(NSIC 207993) AT 1633 HOURS ON 12/22/87, WITH UNITS 1 AND 2 IN OPERATIONAL CONDITION 1 (RUN) AT 85% AND 84% POWER, RESPECTIVELY, THE "B" CONTROL ROOM HVAC SYSTEM (VC) "B" AMMONIA DETECTOR (OXY-VC165B) TRIPPED. PER DESIGN, AN ENGINEERED SAFETY FEATURE (ESF) DAMPER ACTUATION OCCURRED WHICH ISOLATED THE "B" VC SYSTEM FROM OUTSIDE AIR AND RECIRCULATED AIR THROUGH THE "ODOR EATER" (CHARCOAL ADSORBER). AT THE TIME OF THIS EVENT, THE "B" VC SYSTEM WAS OPERATING IN THE RECIRCULATION MODE WITH MINIMUM OUTSIDE AIR DAMPERS OPEN AND FLOW BYPASSING THE "ODOR EATER". THE INSTRUMENT MAINTENANCE DEPARTMENT INVESTIGATED THE EVENT AND FOUND THAT THE AMMONIA DETECTOR CHEMCASSETTE WAS OUT OF TAPE. THE CHEMCASSETTE WAS REPLACED AND THE DETECTOR WAS DECLARED OPERABLE AT 1730 HOURS ON DECEMBER 22, 1987. THE EVENT RESULTED FROM TOO MUCH TAPE BEING USED (WOUND ON THE TAKE-UP REEL) DURING THE CHEMCASSETTE LOADING, THUS REDUCING THE LENGTH OF TIME FOR AMMONIA DETECTION OPERATION UNTIL THE NEXT CASSETTE REPLACEMENT. THE ROOT CAUSE OF THE EVENT WAS THE FACT THAT THE CASSETTE REPLACEMENT PROCEDURE DID NOT PROVIDE GUIDANCE TO THE TECHNICIAN IN THE EVENT A LARGE AMOUNT OF TAPE IS USED DURING THE LOADING OF THE CASSETTE. PROCEDURE REVISIONS WILL BE IMPLEMENTED TO PROVIDE THE TECHNICIAN GUIDANCE.

[122] LA SALLE 2 DOCKET 50-374 LER 87-020
FAILURE OF SEVERAL SOR DIFFERENTIAL PRESSURE SWITCHES DUE TO DIAPHRAGM FAILURES.
EVENT DATE: 111987 REPORT DATE: 121887 NSSS: GE TYPE: BWR
VENDOR: STATIC-O-RING

(NSIC 207558) DURING THE PERIOD OF NOVEMBER 19, 1987 TO NOVEMBER 24, 1987, FOUR (4) UNIT 2 REACTOR CORE ISOLATION COOLING (RCIC)/RESIDUAL HEAT REMOVAL (RHR) STEAM LINE ISOLATION SWITCHES WERE FOUND TO HAVE INTERNAL LEAKAGE FROM ONE SIDE OF THEIR DIAPHRAGM TO THE OTHER. THESE DISCOVERIES WERE MADE DURING THE PERFORMANCE OF LASALLE INSTRUMENT SURVEILLANCE LIS-RI-401, "UNIT 2 STEAM LINE HIGH FLOW RCIC ISOLATION FUNCTIONAL TEST." THE UNIT 2 REACTOR WAS IN OPERATIONAL CONDITION 1 (RUN) AT POWER LEVELS IN EXCESS OF 75% DURING THIS TIME PERIOD. TWO OF THE FOUR SWITCHES WHICH WERE FOUND TO BE FAULTY, FUNCTION TO PROVIDE INBOARD AND OUTBOARD HIGH FLOW ISOLATIONS OF THE STEAM LINE ASSOCIATED WITH THE RCIC/RHR SYSTEMS. THE OTHER TWO SWITCHES PROVIDE THE SAME ISOLATIONS UPON DETECTING A FAILURE OF THE HIGH SIDE INSTRUMENT LINE TO ONE OF THE HIGH FLOW SWITCHES. AT THIS TIME, THE ROOT CAUSE OF THE DIAPHRAGM LEAKS HAS NOT BEEN IDENTIFIED. ALL FOUR FAULTY SWITCHES WERE SENT TO THEIR MANUFACTURER, SOR INC., FOR DISASSEMBLY AND INSPECTION. COMMONWEALTH EDISON'S MATERIALS ANALYSIS DEPARTMENT WILL ALSO EXAMINE THESE DIAPHRAGMS. WHEN THE RESULTS OF THESE INSPECTIONS ARE KNOWN, THIS LICENSEE EVENT REPORT WILL BE SUPPLEMENTED.

[123] LIMERICK 1 DOCKET 50-352 LER 85-074 REV 01
UPDATE ON NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM ACTUATION AND STANDBY GAS TREATMENT SYSTEM INITIATION DUE TO A BLOWN FUSE.
EVENT DATE: 091985 REPORT DATE: 011588 NSSS: GE TYPE: BWR

(NSIC 2[illegible]728) DURING SURVEILLANCE TESTING ON SEPTEMBER 19, 1985, WITH THE UNIT AT 22 PERCENT POWER, AN I&C TECHNICIAN INADVERTENTLY CAUSED A FUSE TO BLOW WHICH RESULTED IN THE AUTOMATIC ISOLATION OF NUMEROUS NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) COMPONENTS AND INITIATION OF STANDBY GAS TREATMENT (SBGT) SYSTEM. ISOLATIONS WERE VERIFIED AND RESET PER PROCEDURE AND SYSTEMS WERE RETURNED TO NORMAL AFTER THE FUSE WAS REPLACED. SPECIAL METERING AND SECTIONALIZING FUSES WHICH WERE INSTALLED AS A RESULT OF LER 85-048 HAVE SHOWN NO ABNORMALITIES. THE INCIDENT HAS NOT RECURRED IN OVER TWO YEARS AND IS CONSIDERED AN ISOLATED EVENT. NO FURTHER INVESTIGATION OR CORRECTIVE ACTIONS ARE PLANNED.

[124] LIMERICK 1 DOCKET 50-352 LER 87-043 REV 01
UPDATE ON MISSED SNUBBER SURVEILLANCE REQUIREMENT DUE TO A PERSONNEL ERROR WHICH RESULTED IN A PROCEDURE DEFICIENCY.
EVENT DATE: 082487 REPORT DATE: 012588 NSSS: GE TYPE: BWR

(NSIC 207965) ON 8/24/87 AT 1500 HOURS IT WAS DETERMINED THAT A SNUBBER ON THE HIGH PRESSURE COOLANT INJECTION SYSTEM WAS NOT VISUALLY INSPECTED WITHIN TEN MONTHS OF COMMENCING POWER OPERATION AS REQUIRED BY TECH SPEC SURVEILLANCE REQUIREMENT 4.7.4. DURING A VISUAL INSPECTION OF ALL SNUBBERS IN THE MAY 1986 SURVEILLANCE TEST OUTAGE SNUBBER EBD-106-E6-H31 (SNUBBER H31) WAS NOT INSPECTED. SNUBBER EBD-105-E6-H12 (SNUBBER H12) WAS VISUALLY INSPECTED TWICE, ONCE MISTAKENLY FOR SNUBBER H31. THE CAUSE OF THIS OVERSIGHT WAS A PERSONNEL ERROR WHICH RESULTED IN A PROCEDURAL DEFICIENCY IN THAT THE SERIAL NUMBER LISTED IN THE SURVEILLANCE TEST FOR SNUBBER H31 WAS INCORRECTLY LISTED AS THE SERIAL NUMBER FOR SNUBBER H12. THIS NUMBER WAS INCORRECTLY ASSIGNED TO SNUBBER H31 PRIOR TO COMPLETION OF THE STARTUP TEST PROGRAM AT LIMERICK GENERATING STATION. SNUBBER H31 WAS VISUALLY INSPECTED AND FUNCTIONALLY TESTED DURING JULY OF THE 1987 REFUEL OUTAGE AND WAS DETERMINED TO BE OPERABLE. SURVEILLANCE TEST ST-4-103-098-1 WHICH PROVIDES FOR VISUAL INSPECTION OF SNUBBER H31 HAS BEEN REVISED TO INCLUDE THE CORRECT SERIAL NUMBER FOR THIS SNUBBER. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THIS EVENT. THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT.

[125] LIMERICK 1 DOCKET 50-352 LER 87-048 REV 01
UPDATE ON REACTOR SCRAM RESULTING FROM A MAIN TURBINE TRIP DUE TO LOW ELECTRO-HYDRAULIC CONTROL PRESSURE.
EVENT DATE: 091987 REPORT DATE: 010588 NSSS: GE TYPE: BWR

(NSIC 207686) ON 9/19/87 AT 0910 HOURS, THE REACTOR PROTECTION SYSTEM INITIATED A REACTOR SCRAM FROM 90% POWER AND A RECIRCULATION PUMP TRIP, FOLLOWING A MAIN TURBINE TRIP ON LOW ELECTRO-HYDRAULIC CONTROL (EHC) SYSTEM OIL PRESSURE. FOLLOWING THE TURBINE TRIP, OPERATION OF THE TURBINE BYPASS VALVES WAS MAINTAINED UNTIL THEIR EHC ACCUMULATOR PRESSURE BLED DOWN. REACTOR PRESSURE REACHED A PEAK VALUE OF 1093 PSIG AND REACTOR VESSEL WATER LEVEL REACHED A MINIMUM LEVEL OF MINUS 2 INCHES DURING THE EVENT. THERE WERE NO ADVERSE CONSEQUENCES AND THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL RESULTING FROM THIS EVENT. THE ROOT CAUSE OF THE EVENT WAS THE FAILURE OF A TUBING SOCKET WELD IN THE EHC FLUID ACTUATING SUPPLY (FAS) LINE TO THE #3 MAIN TURBINE CONTROL VALVE (MTCV). THE SECTION OF EHC PIPE CONTAINING THE FAILED WELD WAS REMOVED FOR INSPECTION AND A NEW SECTION OF PIPE WAS WELDED INTO THE LINE. ANALYSIS OF THE MEASURED PIPING MOVEMENT AND VIBRATION LEVELS INDICATES THAT A PROPERLY BONDED WELD WOULD NOT HAVE FAILED AS A RESULT OF THE VIBRATIONS PRESENT. AS SUCH, THIS EVENT IS CONSIDERED AN ISOLATED INCIDENT. ADJUSTMENTS WERE MADE TO THE STEAM LINE RESONANCE COMPENSATOR (SLRC) AND A SECOND SLRC WAS INSTALLED IN SERIES WITH THE FIRST TO REDUCE THE CONTROL SIGNAL OSCILLATIONS. FULL POWER OPERATION WAS ACHIEVED ON 11/21/87.

[126] LIMERICK 1 DOCKET 50-352 LER 87-061
OPERATION OUTSIDE OF TECHNICAL SPECIFICATIONS DUE TO A PERSONNEL ERROR WHICH RENDERED THE CONTROL ROOM EMERGENCY FRESH AIR SUPPLY SYSTEM INOPERABLE.
EVENT DATE: 110787 REPORT DATE: 121687 NSSS: GE TYPE: BWR

(NSIC 207547) ON 11/7/87 BETWEEN 0212 AND 0535 HOURS BOTH CONTROL ROOM VENTILATION RETURN FANS WERE NOT OPERATING, RENDERING BOTH SUBSYSTEMS OF THE CONTROL ROOM EMERGENCY FRESH AIR SUPPLY (CREFAS) SYSTEM INOPERABLE. THIS PLACED THE UNIT IN AN OPERATING CONDITION WHICH IS OUTSIDE OF THE BOUNDS OF TECH SPECS AND AS SUCH IS REPORTABLE. AT 0212 HOURS, THE RETURN FANS TRIPPED DURING SIMULTANEOUS PURGING OF THE CONTROL ROOM AND THE AUXILIARY EQUIPMENT ROOM FOR TEMPERATURE CONTROL. THE RETURN FAN BREAKERS WERE RESET, BUT THE FAN HAND SWITCHES WERE MAINTAINED IN THE 'OFF' POSITION, TO PREVENT SUBSEQUENT RETURN FAN TRIPS. CREFAS WAS BELIEVED TO BE INOPERABLE ONLY DURING THE TIME THAT THE RETURN FAN BREAKERS WERE TRIPPED; THEREFORE, NO ACTION WAS TAKEN WITHIN THE ONE HOUR REQUIREMENT OF TECH SPECS 3.0.3 TO PLACE THE UNIT IN THE STARTUP MODE. AT 0535 HOURS SHIFT SUPERVISION DETERMINED THAT IT WOULD BE PRUDENT TO PLACE THE 'A' RETURN FAN IN SERVICE IN ORDER TO NORMALIZE SYSTEM ALIGNMENT. PURGING OF THE CONTROL ROOM AND THE AUXILIARY EQUIPMENT ROOM WAS THEN COMMENCED ON AN ALTERNATING BASIS TO PREVENT RETURN FAN TRIPS. THERE WERE NO ADVERSE CONSEQUENCES AND NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT. A MEMO WAS WRITTEN TO ALL SHIFT PERSONNEL INSTRUCTING THEM TO REVIEW THE REQUIREMENTS FOR CREFAS OPERABILITY.

[127] LIMERICK 1 DOCKET 50-352 LER 87-062
REACTOR ENCLOSURE ISOLATION ON LOW DIFFERENTIAL PRESSURE DUE TO PERSONNEL ERROR.
EVENT DATE: 111487 REPORT DATE: 122187 NSSS: GE TYPE: BWR

(NSIC 207448) ON NOVEMBER 14, 1987 AT 1621 HOURS, AN ISOLATION OF THE REACTOR ENCLOSURE OCCURRED ON LOW DIFFERENTIAL PRESSURE AFTER A NONLICENSED OPERATOR CHANGED THE OPERATING LINEUP OF THE REACTOR ENCLOSURE VENTILATION EXHAUST FANS. FOLLOWING THE ISOLATION, THE STANDBY GAS TREATMENT SYSTEM (SGTS) AND THE REACTOR ENCLOSURE RECIRCULATION SYSTEM (RERS) INITIATED AS DESIGNED. THERE WERE NO ADVERSE CONSEQUENCES AND THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT. THE ISOLATION WAS RESET AT 1628 HOURS AND OPERATION OF SGTS AND RERS WAS TERMINATED. THE ROOT CAUSE OF THE EVENT WAS THE FAILURE OF AN OPERATOR

TO INFORM THE CONTROL ROOM OF HIS INTENDED ACTIONS, PRIOR TO CHANGING THE OPERATING STATUS OF THE EXHAUST FANS. A CONTRIBUTING FACTOR WAS THE ABSENCE OF A TAG DESCRIBING THE OPERATING LIMITATION. FOLLOWING THE ISOLATION AN OPERATOR AID WAS POSTED STATING THE INABILITY OF THE 'A' AND 'C' LINEUP TO MAINTAIN REACTOR ENCLOSURE DIFFERENTIAL PRESSURE. THE OPERATOR INVOLVED IN THE EVENT WAS COUNSELED ON THE IMPORTANCE OF NOTIFYING THE CONTROL ROOM PRIOR TO CHANGING THE OPERATING STATUS OF PLANT EQUIPMENT. A LETTER FROM THE OPERATIONS ENGINEER TO ALL SHIFT PERSONNEL WAS DISTRIBUTED AND WAS DISCUSSED WITH EACH SHIFT STRESSING THE IMPORTANCE OF NOTIFYING THE CONTROL ROOM PRIOR TO CHANGING THE STATUS OF PLANT EQUIPMENT.

[128] LIMERICK 1 DOCKET 50-352 LER 87-063
FIRE SUPPRESSION WATER SYSTEM TECHNICAL SPECIFICATION VIOLATION DUE TO PERSONNEL ERROR.
EVENT DATE: 111687 REPORT DATE: 122487 NSSS: GE TYPE: BWR

(NSIC 207449) ON NOVEMBER 16, 1987, A TECHNICAL SPECIFICATIONS VIOLATION OCCURRED WHEN AN UNAPPROVED BLOCKING PERMIT WAS MISTAKENLY APPLIED, REMOVING FROM SERVICE SEVERAL REQUIRED FIRE SUPPRESSION SYSTEMS. TWO COGNITIVE PERSONNEL ERRORS BY A LICENSED AND A NONLICENSED OPERATOR LED TO THE VIOLATION. THE OPERATORS FAILED TO FOLLOW TWO ADMINISTRATIVE PROCEDURES, AND CONSEQUENTLY SHIFT SUPERVISION WAS UNAWARE THAT EQUIPMENT WAS INOPERABLE AND THEREFORE DID NOT IMPLEMENT THE APPROPRIATE TECHNICAL SPECIFICATIONS REQUIREMENTS. ONCE THE CONTROL SUPERVISOR WAS MADE AWARE OF THE CLOSED VALVES, HE IMMEDIATELY DIRECTED REMOVAL OF THE PERMIT BLOCKING. THE FIRE SUPPRESSION SYSTEMS WERE OUT OF SERVICE FOR 4 HOURS AND 15 MINUTES. THERE WERE NO ADVERSE CONSEQUENCES AND NO RELEASE OF RADIOACTIVE MATERIAL TO THE ENVIRONMENT AS A RESULT OF THIS EVENT. THE PERSONNEL INVOLVED WERE COUNSELED ABOUT ADHERENCE TO ADMINISTRATIVE PROCEDURES AND A MEMO TO ALL SHIFT PERSONNEL WAS ISSUED STRESSING ATTENTION TO DETAIL. IN ADDITION, THIS AND OTHER RECENT EVENTS WERE REVIEWED FOR SIMILAR CAUSES, THE RESULTS OF WHICH WILL BE INCORPORATED INTO FUTURE REQUALIFICATION TRAINING.

[129] LIMERICK 1 DOCKET 50-352 LER 87-064
REACTOR ENCLOSURE VENTILATION ISOLATION AND NSSSS ISOLATION DUE TO LOW DIFFERENTIAL PRESSURE RESULTING FROM THE LOSS OF AUXILIARY STEAM HEAT.
EVENT DATE: 112187 REPORT DATE: 122187 NSSS: GE TYPE: BWR

(NSIC 207450) ON 11/21/87, AT 1541 HOURS WITH THE UNIT AT 90.3% POWER, A REACTOR ENCLOSURE VENTILATION ISOLATION AND A NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) GROUPS VIA AND VIB (ENGINEERED SAFETY FEATURES) ISOLATION OCCURRED RESULTING FROM A LOW DIFFERENTIAL PRESSURE. AUTOMATIC INITIATION OF THE STANDBY GAS TREATMENT SYSTEM (SGTS) AND REACTOR ENCLOSURE RECIRCULATION SYSTEM (RERS) FOLLOWED WITH THE SYSTEMS PERFORMING AS DESIGNED. THE LOW DIFFERENTIAL PRESSURE CONDITION WAS CREATED BY THE LOSS OF AUXILIARY STEAM TO THE REACTOR ENCLOSURE SUPPLY AIR STEAM HEATING COILS, ALLOWING COOL OUTSIDE AIR TO ENTER, WARM, AND EXPAND IN THE REACTOR ENCLOSURE. THIS OVERWHELMED THE EXHAUST FAN'S ABILITY TO CONTROL THE REACTOR ENCLOSURE PRESSURE. THE SUBSEQUENT DECREASE IN DIFFERENTIAL PRESSURE RESULTED IN THE ISOLATION. THE AUXILIARY STEAM WAS RESTORED, THE NSSSS AND REACTOR ENCLOSURE ISOLATIONS WERE RESET BY 1705, AND NORMAL REACTOR ENCLOSURE VENTILATION WAS RESTORED. THE DURATION OF THE ISOLATION WAS 1 HOUR AND 24 MINUTES. THERE WERE NO ADVERSE CONSEQUENCES AND NO RELEASE OF RADIOACTIVE MATERIAL TO THE ENVIRONMENT AS A RESULT OF THIS EVENT. ENGINEERING IS EVALUATING METHODS OF IMPROVING THE RELIABILITY OF THE AUXILIARY BOILERS. AN OPERATIONAL AID (21-12) WAS POSTED TO AID OPERATORS.

[130] LIMERICK 1 DOCKET 50-352 LER 87-065
REACTOR ENCLOSURE VENTILATION ISOLATION DUE TO LOW DIFFERENTIAL PRESSURE CAUSED BY THE TRIPPING OF THE REACTOR ENCLOSURE SUPPLY AIR FANS.
EVENT DATE: 112187 REPORT DATE: 122187 NSSS: GE TYPE: BWR

(NSIC 207451) ON NOVEMBER 21, 1987, AT 0816 HOURS WITH THE UNIT AT 85% POWER, A REACTOR ENCLOSURE VENTILATION ISOLATION (AN ENGINEERED SAFETY FEATURE) OCCURRED RESULTING FROM LOW DIFFERENTIAL PRESSURE. AUTOMATIC INITIATION OF THE STANDBY GAS TREATMENT SYSTEM (SGTS) AND THE REACTOR ENCLOSURE RECIRCULATION SYSTEM (RERS) FOLLOWED; HOWEVER, THE INDICATED SECONDARY CONTAINMENT FLOW RATE EXCEEDED THE DESIGNED 1250 CFM. IT WAS THEN DISCOVERED THAT THE REFUEL FLOOR TO STANDBY GAS TREATMENT FLOW TRANSMITTER (FT-76-073A) HAD DRIFTED OUT OF CALIBRATION CAUSING A FALSE HIGH SGTS FLOW READING IN THE CONTROL ROOM. FOLLOWING RECALIBRATION, INDICATED SGTS FLOW DECREASED BELOW THE TECHNICAL SPECIFICATION LIMITS. THE LOW REACTOR ENCLOSURE DIFFERENTIAL PRESSURE CONDITION WAS CREATED WHEN THE REACTOR ENCLOSURE SUPPLY AIR FANS TRIPPED ON LOW SUPPLY INLET AIR TEMPERATURE, THE CAUSE OF WHICH IS UNKNOWN, BUT IS ATTRIBUTED TO LOW AUXILIARY STEAM FLOW TO THE FAN HEATING COILS. WITH THE LOSS OF SUPPLY AIR, EXCESSIVE NEGATIVE PRESSURE WAS CREATED IN THE REACTOR ENCLOSURE AUTOMATICALLY TRIPPING THE EXHAUST AIR FANS. WITHOUT THE REQUIRED VENTILATION, REACTOR ENCLOSURE DIFFERENTIAL PRESSURE DECREASED RESULTING IN THE LOW DIFFERENTIAL PRESSURE ISOLATION AND SUBSEQUENT ESF ACTUATIONS.

[131] LIMERICK 1 DOCKET 50-352 LER 87-066
HPCI INOPERABILITY DUE TO MALFUNCTION OF HYDRAULIC-MECHANICAL OVERSPEED TRIP MECHANISM.
EVENT DATE: 120887 REPORT DATE: 010788 NSSS: GE TYPE: BWR
VENDOR: TERRY CORP.

(NSIC 207687) ON 12/8/87 AT 1447 HOURS, DURING THE PERFORMANCE OF SURVEILLANCE TEST ST-6-055-230-1, "HPCI PUMP, VALVE AND FLOW TEST", THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM (AN EMERGENCY CORE COOLING SYSTEM) WAS DECLARED INOPERABLE DUE TO THE ERRATIC OPERATION OF THE HPCI TURBINE STOP VALVE. APPROXIMATELY 20 MINUTES INTO THE TEST, WHILE IN ITS FULL FLOW TEST CONFIGURATION, THE HPCI STOP VALVE UNEXPECTEDLY FULLY CLOSED AND REOPENED. THE STOP VALVE CONTINUED TO CLOSE AND REOPEN SEVERAL TIMES. AN IMMEDIATE INVESTIGATION INTO THE CAUSE OF THE PROBLEM REVEALED THAT ALTHOUGH A HPCI TURBINE OVERSPEED CONDITION WAS NOT PRESENT, THE HYDRAULIC-MECHANICAL OVERSPEED TRIP MECHANISM WAS CYCLING BETWEEN THE TRIPPED AND NORMAL POSITIONS. THE CAUSE OF THE MALFUNCTION IS BELIEVED TO HAVE BEEN A BLOCKAGE OF A SMALL INTERNAL DRAIN PORT IN THE OVERSPEED TRIP AND AUTOMATIC RESET PISTON ASSEMBLY. THERE WERE NO ADVERSE CONSEQUENCES OF THIS EVENT. NO RADIOACTIVE MATERIAL WAS RELEASED AS A RESULT OF THE EVENT. THE AUTOMATIC DEPRESSURIZATION SYSTEM (ADS), REACTOR CORE ISOLATION CONTROL (RCIC) SYSTEM AND THE OTHER EMERGENCY CORE COOLING SYSTEMS WERE OPERABLE DURING THE EVENT. THE HPCI SYSTEM WAS DECLARED OPERABLE AT 1830 HOURS ON DECEMBER 9, 1987 AFTER BEING INOPERABLE FOR 27 HOURS AND 43 MINUTES. INCREASED SURVEILLANCE OF THE OVERSPEED TRIP MECHANISM HAS BEEN INITIATED.

[132] LIMERICK 1 DOCKET 50-352 LER 87-067
REACTOR ENCLOSURE ISOLATIONS ON LOW DIFFERENTIAL PRESSURE DUE TO REACTOR ENCLOSURE SUPPLY FANS TRIPPING ON LOW SUPPLY AIR TEMPERATURE.
EVENT DATE: 122987 REPORT DATE: 012888 NSSS: GE TYPE: BWR

(NSIC 208046) ON 12/29/87 AT 1636 HOURS AND 1705 HOURS WITH THE UNIT AT 99% POWER, THE REACTOR ENCLOSURE SECONDARY CONTAINMENT ISOLATED ON LOW REACTOR ENCLOSURE TO OUTSIDE ATMOSPHERE DIFFERENTIAL PRESSURE. FOLLOWING EACH ISOLATION, THE STANDBY GAS TREATMENT SYSTEM (SGTS) AND REACTOR ENCLOSURE RECIRCULATION SYSTEM (RERS), ENGINEERED SAFETY FEATURES, INITIATED AS DESIGNED AND NUCLEAR STEAM SUPPLY SHUTDOWN SYSTEM (NSSSS) GROUP VI A AND B ISOLATION SIGNALS WERE

RECEIVED. STRATIFICATION OF COLD AIR AROUND THE TEMPERATURE SENSING ELEMENT IN THE REACTOR ENCLOSURE VENTILATION INTAKE PLENUM DUE TO LEAKAGE OF AIR PAST THE BYPASS DAMPERS AND LEAKS IN THE HEATING COILS CAUSED TEMPERATURE SWITCH (TSL-076-105) TO TRIP THE REACTOR ENCLOSURE SUPPLY FANS. AN AUTOMATIC SEQUENCE OF EVENTS FOLLOWING THE LOSS OF SUPPLY FANS RESULTED IN THE ISOLATIONS ON LOW DIFFERENTIAL PRESSURE. THE ISOLATIONS WERE RESET AT 1546 HOURS AND 2225 HOURS. THERE WERE NO ADVERSE CONSEQUENCES AND THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT. A THIRD ISOLATION OCCURRED ON DECEMBER 31, 1987 AT 0021 HOURS (LER 87-070) AND THE SUPPLY FAN TRIPPING LOGIC ASSOCIATED WITH TSL-076- 105 WAS BYPASSED TO PREVENT FURTHER ISOLATIONS. MAINTENANCE REQUEST FORMS WILL BE IMPLEMENTED DURING THE NEXT OUTAGE TO REPLACE THE POSITIONER.

[133] LIMERICK 1 DOCKET 50-352 LER 87-068
TECHNICAL SPECIFICATION VIOLATION DUE TO INADEQUATE TRAINING AND AN EQUIPMENT FAILURE.
EVENT DATE: 123087 REPORT DATE: 012988 NSSS: GE TYPE: BWR
VENDOR: BALKSDALE VALVE COMPANY

(NSIC 207930) ON DECEMBER 30, 1987 AT APPROXIMATELY 1000 HOURS THE RADIATION MONITOR SAMPLING SYSTEM FOR THE HOT MAINTENANCE SHOP WAS FOUND OUT OF SERVICE WHILE THE SHOP VENTILATION SYSTEM WAS OPERATING. THIS VIOLATES TECHNICAL SPECIFICATION 3.3.7.12 WHICH REQUIRES CONTINUOUS SAMPLING WHILE THE SHOP VENTILATION IS IN SERVICE. THE MONITORING SYSTEM WAS OUT OF SERVICE FOR APPROXIMATELY TWENTY-FOUR HOURS WHILE THE SHOP VENTILATION WAS OPERATING. THE HOT SHOP VENTILATION WAS SECURED AT APPROXIMATELY 1000 HOURS ON THE 30TH AND THE SYSTEM WAS BLOCKED OUT OF SERVICE UNTIL CORRECTIVE ACTIONS COULD BE TAKEN TO PRECLUDE THE POSSIBILITY OF FURTHER OPERATION OF THE SYSTEM IN VIOLATION OF TECHNICAL SPECIFICATIONS. THE CAUSE OF THE EVENT WAS COMBINATION OF INADEQUATE TRAINING AND EQUIPMENT FAILURE. TO PREVENT RECURRENCE TRAINING WILL BE PROVIDED TO ALL LICENSED PERSONNEL BY JUNE 2, 1988. THE TRAINING WILL STRESS THE INTERDEPENDENCE OF SYSTEM INDICATIONS AND WHAT INDICATIONS ARE VALID FOR DETERMINING SYSTEM OPERABILITY. IN ADDITION, THE SCOPE OF REVIEW OF A PREVIOUSLY REQUESTED MODIFICATION HAS EXPANDED AND IS CONTINUING. THERE WERE NO CONSEQUENCES AND NO RELEASE OF RADIATION OCCURRED AS A RESULT OF THIS EVENT.

[134] LIMERICK 1 DOCKET 50-352 LER 87-069
CONTROL ROOM EMERGENCY FRESH AIR SUPPLY ACTUATION RESULTING FROM A PROCEDURAL ERROR DURING A CHLORINE ANALYZER FUNCTIONAL TEST.
EVENT DATE: 123087 REPORT DATE: 012988 NSSS: GE TYPE: BWR

(NSIC 208048) ON 12/30/87 AT 1855 HOURS, THE CONTROL ROOM VENTILATION SYSTEM ISOLATED AND THE 'A' TRAIN OF THE CONTROL ROOM EMERGENCY FRESH AIR SUPPLY (CREFAS) SYSTEM, AN ENGINEERED SAFETY FEATURE, INITIATED AS DESIGNED. THE 'C' CHLORINE ANALYZER, WHICH FUNCTIONS TO ISOLATE THE CONTROL ROOM VENTILATION SYSTEM IN THE EVENT THE CHLORINE CONCENTRATION IN THE CONTROL ENCLOSURE INTAKE PLENUM EXCEEDS 0.4 PPM, MOMENTARILY SPIKED TO 0.5 PPM. THE REMAINING TWO OPERABLE CHLORINE ANALYZERS SHOWED NO INCREASE IN CHLORINE CONCENTRATION. THE 'C' ANALYZER WAS INSPECTED AND A SMALL AMOUNT OF WHITE POWDERY SUBSTANCE WAS FOUND ON THE ANALYZER PROBE'S PLASTIC CASING. THE ANALYZER PROBE WAS REPLACED WITH A SPARE PROBE AT 1730 HOURS ON DECEMBER 31, 1987. THE WHITE POWDER WAS ANALYZED AND DETERMINED TO CONTAIN SODIUM CHLORIDE. THE CAUSE OF THIS EVENT WAS A PROCEDURAL DEFICIENCY WHICH RESULTED IN AN ERROR DURING THE PERFORMANCE OF A PREVIOUS ROUTINE FUNCTIONAL TEST. THE TEST REQUIRED THE PLACEMENT OF A BOTTLE OF SODIUM HYPOCHLORITE SOLUTION UNDER THE ANALYZER PROBE AND THE ERROR ALLOWED SOME TEST SOLUTION TO DEPOSIT ON THE PROBE CASING. THE TEST PROCEDURE WILL BE REVISED TO REQUIRE THE USE OF A LARGE MOUTH SOLUTION CONTAINER TO ENSURE SOLUTION DOES NOT CONTACT THE PROBE. THERE WERE NO ADVERSE CONSEQUENCES RESULTING FROM THIS EVENT.

[135] LIMERICK 1 DOCKET 50-352 LER 87-070
REACTOR ENCLOSURE ISOLATION ON LOW DIFFERENTIAL PRESSURE DUE TO REACTOR ENCLOSURE SUPPLY FANS TRIPPING ON LOW SUPPLY AIR TEMPERATURE.
EVENT DATE: 123187 REPORT DATE: 012988 NSSS: GE TYPE: BWR

(NSIC 208049) ON 12/31/87 AT 0021 HOURS, AN ISOLATION OF THE REACTOR ENCLOSURE SECONDARY CONTAINMENT OCCURRED ON LOW REACTOR ENCLOSURE TO OUTSIDE ATMOSPHERE DIFFERENTIAL PRESSURE. THE STANDBY GAS TREATMENT SYSTEM (SGTS) AND REACTOR ENCLOSURE RECIRCULATION SYSTEM (RERS), ENGINEERED SAFETY FEATURES, INITIATED AS DESIGNED AND NUCLEAR STEAM SUPPLY SHUTDOWN SYSTEM (NSSSS) GROUP VIA AND B ISOLATION SIGNALS WERE RECEIVED. FOLLOWING INITIATION OF THE SGTS, THE 'B' SGTS FAN TRIPPED DUE TO ITS FLOW MONITORING SWITCH DRIFTING OUT OF CALIBRATION AND THE 'A' FAN AUTOMATICALLY STARTED AND CONTINUED TO OPERATE AS DESIGNED. THE ISOLATION WAS RESET AT 0132 HOURS. THERE WERE NO ADVERSE CONSEQUENCES AND THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT. STRATIFICATION OF COLD AIR AROUND THE TEMPERATURE SENSING ELEMENT IN THE REACTOR ENCLOSURE SUPPLY INTAKE PLENUM CAUSED TEMPERATURE SWITCH (TSL-076-105) TO TRIP THE REACTOR ENCLOSURE SUPPLY FANS WHICH RESULTED IN THE ISOLATION ON LOW DIFFERENTIAL PRESSURE. A TEMPORARY CIRCUIT ALTERATION WAS IMPLEMENTED ON DECEMBER 31, 1987 TO BYPASS THE TEMPERATURE SWITCH'S REACTOR ENCLOSURE SUPPLY FAN TRIPPING LOGIC TO PREVENT FURTHER ISOLATIONS. MAINTENANCE REQUEST FORMS WILL BE IMPLEMENTED DURING THE NEXT OUTAGE TO REPLACE THE POSITIONER.

[136] MCGUIRE 1 DOCKET 50-369 LER 87-035
OPERATION ABOVE RATED THERMAL POWER BECAUSE THE NUMBER OF DECIMAL PLACES USED TO ADJUST PLANT OUTPUT WERE ROUNDED OFF.
EVENT DATE: 061187 REPORT DATE: 010588 NSSS: WE TYPE: PWR

(NSIC 207690) UNIT 1 COMPUTER LOGS FOR 05/20/87 - 06/11/87 INDICATE THAT UNIT 1 MAY HAVE OPERATED SLIGHTLY ABOVE 100% OF RATED THERMAL POWER AS DEFINED IN TECHNICAL SPECIFICATION INTERPRETATION (TS) 1.25 - DEFINITION OF RATED THERMAL POWER. THE DISCREPANCY WAS DISCOVERED WHILE COMPUTER PRINTOUTS OF UNIT 1 POWER HISTORY WERE BEING REVIEWED. IN EACH INSTANCE THE AVERAGE POWER OVER AN EIGHT HOUR SHIFT WAS GREATER THAN 100%; IN THE WORST CASE THE AVERAGE WAS 100.1847%. A CAUSE OF OTHER HAS BEEN ASSIGNED TO THIS EVENT BECAUSE THE NUMBER OF DECIMAL PLACES TO USE IN MEASURING OR DETERMINING RATED THERMAL POWER AND/OR A RULING ON ROUNDING OFF VALUES HAD NOT BEEN ESTABLISHED. ALSO, THE PARTICULAR COMPUTER LOG USED IN THE DETERMINATION DID NOT RECORD A REPRESENTATIVE NUMBER OF SAMPLES, WAS NOT INTENDED FOR USE IN DETERMINING COMPLIANCE WITH TS 1.25, AND THE STATISTICAL AVERAGING OF ITS DATA WOULD NEITHER PROVE NOR DISPROVE CONCLUSIVELY WHETHER THE UNIT ACTUALLY EXCEEDED 100% AVERAGE POWER DURING PARTICULAR EIGHT HOUR PERIODS. A COMPUTER PROGRAM HAS BEEN INSTALLED WHICH MAINTAINS A CONTINUOUS DISPLAY OF EIGHT HOUR AVERAGE POWER TO PRECLUDE FUTURE PROBLEMS OF THIS TYPE.

[137] MCGUIRE 1 DOCKET 50-369 LER 87-030
A D/G FUEL OIL ISOLATION VALVE WAS MISTAKENLY CLOSED DUE TO BEING MISLABELED CAUSING D/G INOPERABILITY AND DEFECTIVE PROCEDURE.
EVENT DATE: 090887 REPORT DATE: 122187 NSSS: WE TYPE: PWR

(NSIC 207461) ON 09/11/87, QUALITY ASSURANCE PERFORMING PNEUMATIC TESTING ON UNIT 1 DIESEL FUEL OIL (FD) PIPING NOTICED THE VALVE NUMBER LABELS ATTACHED TO THE REACH RODS OF TWO FD VALVES WERE REVERSED WITH RESPECT TO THE OLDER METAL TAGS ATTACHED TO THE VALVE BODIES. OPERATIONS (OPS) VERIFIED THE METAL TAGS WERE CORRECT AND DETERMINED THAT DUE TO THE MISLABELING OF THE REACH RODS, THE SUPPLY VALVE TO DIESEL GENERATOR (D/G) 1B FROM FD STORAGE TANK 1B HAD BEEN MISTAKENLY CLOSED ON 09/08/87 AT 2350 AS PART OF A BLOCK TAGOUT. D/G 1B WAS THEREFORE INOPERABLE, AND DIESEL GENERATOR 1A HAD PREVIOUSLY BEEN DECLARED INOPERABLE. OPS OPENED THE SUPPLY VALVE TO D/G 1B UPON INITIAL CONFIRMATION OF THE MISLABELING AT 1730 ON 09/11/87 AND VERIFIED CLOSED AND HUNG RED TAGS ON THE SUPPLY VALVE TO D/G

1A FROM FD STORAGE TANK 1A. A CAUSE OF DEFECTIVE PROCEDURE HAS BEEN ASSIGNED TO THIS EVENT SINCE THE MISLABELING OF THE VALVES WAS DUE TO REVERSED VALVE LOCATION DESCRIPTIONS IN THE A PROCEDURE VALVE CHECKLIST. THE CHECKLIST WAS CORRECTED AND PROJECTS HAS CLARIFIED INSTRUCTIONS TO PERSONNEL WHO ARE PERFORMING LABELING.

[138] MCGUIRE 1 DOCKET 50-369 LER 87-034 REV 01
UPDATE ON TWO FIRE DOORS WERE BLOCKED OPEN AND A FIRE WATCH WAS MISSED DUE TO PERSONNEL ERROR.
EVENT DATE: 110987 REPORT DATE: 012688 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 207975) ON 11/09/87 AT ABOUT 1010, WHILE PERFORMING ROUTINE PLANT SURVEILLANCE, QUALITY ASSURANCE (QA) NOTICED THAT TECHNICAL SPECIFICATIONS (TS) REQUIRED FIRE DOOR 819A WAS BLOCKED OPEN, AND THE "FIRE BARRIER WATCH" TAG INDICATED THAT AN HOURLY FIRE WATCH HAD BEEN PERFORMED. QA CONTINUED SURVEILLANCE AND NOTICED THAT FIRE DOOR 621A WAS ALSO BLOCKED OPEN, AND DISCOVERED THE REQUIRED FIRE WATCH HAD NOT BEEN PERFORMED ON AN HOURLY BASIS. QA NOTIFIED THE CONTROL ROOM ABOUT THE OPENED FIRE DOOR. THE CONTROL ROOM NOTIFIED MECHANICAL MAINTENANCE (MNT) RESPONSIBLE FOR THE FIRE WATCH. MNT CLOSED TS FIRE DOORS 819A AND 621A AND HAD THE DOORS CLEARED FROM THE UNIT 2 TECHNICAL SPECIFICATION ACTION ITEM LOGBOOK AT 1200. THIS EVENT IS ASSIGNED A CAUSE OF PERSONNEL ERROR BECAUSE THE MNT SUPERVISORS INVOLVED DID NOT ADEQUATELY COMMUNICATE PERTINENT INFORMATION ABOUT THE TS FIRE DOORS TO APPROPRIATE MNT PERSONNEL DURING SHIFT TURNOVER. A CONTRIBUTORY CAUSE OF MANAGEMENT DEFICIENCY IS ALSO ASSIGNED BECAUSE MNT SHIFT DOES NOT MAINTAIN A LOGBOOK OR TURNOVER SHEET. MNT WILL ESTABLISH A LOGBOOK FOR TS ITEMS THEY ARE RESPONSIBLE FOR AND WILL REVIEW THIS REPORT WITH MNT SHIFT PERSONNEL.

[139] MCGUIRE 1 DOCKET 50-369 LER 87-033
FOUR MAIN STEAM TO AUXILIARY EQUIPMENT VALVES WERE OMITTED FROM THE INSERVICE VALVE TESTING PROGRAM DUE TO PERSONNEL ERROR.
EVENT DATE: 120887 REPORT DATE: 012988 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 208051) ON 12/08/87, AN EVALUATION OF MCGUIRE WAS BEING PERFORMED BY NRC ANALYSIS AND EVALUATION OF OPERATIONAL DATA PERSONNEL (NRC). NRC FOUND THAT VALVES 1SA-5 AND 2SA-5, MAIN STEAM 1C AND 2C TO AUXILIARY FEEDWATER PUMP TURBINE CHECK, AND 1SA-6 AND 2SA-6, MAIN STEAM 1B AND 2B TO AUXILIARY FEEDWATER PUMP TURBINE CHECK, HAD BEEN OMITTED FROM THE INSERVICE VALVE TESTING PROGRAM. ON 12/09/87 AT 1600, OPERATIONS (OPS) DECLARED UNIT 1 AND 2 TURBINE DRIVEN AUXILIARY FEEDWATER (CA) PUMPS INOPERABLE. WORK REQUESTS WERE WRITTEN BY OPS TO VERIFY THAT THE FOUR VALVES WERE IN THE FULLY CLOSED POSITION. ON 12/09/87, THE FREEDOM OF TRAVEL WAS MEASURED ON ALL FOUR VALVES BY OPERATIONS AND DETERMINED TO BE WITHIN THE VALVE MANUFACTURERS ACCEPTABLE TOLERANCE. THE TURBINE DRIVEN CA PUMPS WERE RETURNED TO OPERABLE STATUS AT 1900. A CAUSE OF PERSONNEL ERROR IS ASSIGNED TO THIS EVENT BECAUSE THE VALVES WHICH SHOULD HAVE BEEN INCLUDED IN THE INSERVICE VALVE TESTING PROGRAM WERE OMITTED WHEN ENGINEERS, WHO INITIALLY DEVELOPED THE PROGRAM, MISINTERPRETED THE DESIGN FUNCTION OF THE VALVES AND FAILED TO RECOGNIZE THE VALVES PERFORMED A FUNCTION REQUIRED TO ACHIEVE A SAFE SHUTDOWN CONDITION. TEST REQUIREMENTS WILL BE DETERMINED AND NECESSARY PROCEDURE CHANGES WILL BE MADE FOR THESE VALVES.

[140] MCGUIRE 1 DOCKET 50-369 LER 87-036
REACTOR TRIP DUE TO ERROR ON A SCHEMATIC DIAGRAM WHICH DIRECTED PERSONNEL TO WRONG CABINET DUE TO DESIGN DEFICIENCY.
EVENT DATE: 122887 REPORT DATE: 012788 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 208052) ON 12/28/87 AT 1322, DURING PERIODIC TESTING, UNIT 1 REACTOR TRIP OCCURRED WHEN ELECTRICAL MAINTENANCE (EM) PERSONNEL WERE DIRECTED TO THE WRONG REACTOR PROTECTIVE SYSTEM (7300 RPS) CABINET BY AN ERROR IN A SCHEMATIC DIAGRAM. THE ERROR IN THE SCHEMATIC DIAGRAM CAUSED THE PERSONNEL TO COMPLETE A 2 OUT OF 4 LOGIC REACTOR TRIP SIGNAL. A LOW-LOW LEVEL IN STEAM GENERATOR 1B DID NOT ACTUALLY EXIST. OPERATIONS IMPLEMENTED THE REACTOR TRIP RECOVERY PROCEDURE TO RECOVER FROM THE TRANSIENT. AT 2214, EM INITIATED A REACTOR TRIP SIGNAL WHEN THEY DID NOT FOLLOW PROCEDURE AND CAUSED A GENERAL WARNING ON TRAIN A OF THE SOLID STATE PROTECTION SYSTEM (SSPS) WHILE A GENERAL WARNING EXISTED ON TRAIN B OF THE SSPS. UNIT 1 WAS RETURNED TO MODE 1, POWER OPERATION, ON DECEMBER 29, AT 1249. THE REACTOR TRIP IS ASSIGNED A CAUSE OF DESIGN DEFICIENCY BECAUSE THE SCHEMATIC DIAGRAM WAS INCORRECT. ALSO PERSONNEL ERROR BECAUSE PERSONNEL DID NOT PURSUE ADEQUATE CHECKS ON THE DIAGRAM. THE SECOND REACTOR TRIP SIGNAL IS ASSIGNED A PERSONNEL ERROR, BECAUSE PERSONNEL INVOLVED DID NOT FOLLOW PROCEDURE. ALSO DEFECTIVE PROCEDURE BECAUSE NO PRECAUTIONS WERE IN THE PROCEDURE TO PREVENT EVENTS OF THIS TYPE. THE DIAGRAMS WERE MARKED AS INCORRECT AND CHANGES TO CORRECT UNITS 1 AND 2 DIAGRAMS WERE IMPLEMENTED. ALL OTHER 7300 DIAGRAMS WERE REVIEWED.

[141] MCGUIRE 1 DOCKET 50-369 LER 87-037
WASTE GAS SURVEILLANCE SAMPLE WAS NOT OBTAINED WITHIN TECH SPEC TIME LIMIT DUE TO PERSONNEL ERROR.
EVENT DATE: 123187 REPORT DATE: 020188 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 208053) ON 12/31/87 AT APPROXIMATELY 2200, RADWASTE CHEMISTRY (RDW) REVIEWING THE RDW DAILY STATUS BOARD REALIZED THAT A SAMPLE REQUIRED BY TECHNICAL SPECIFICATION (TS) 3/4.11.2.5 HAD NOT BEEN OBTAINED ON DAY SHIFT WITHIN THE 24 HOUR SURVEILLANCE INTERVAL. RDW INFORMED HEALTH PHYSICS (HP) OF THE OMITTED SAMPLE OF WASTE GAS (WG) SHUTDOWN TANK B, AND HP OBTAINED A SAMPLE FROM THE TANK AT 2247. THE SAMPLE VERIFIED THAT THE RADIOACTIVE NOBLE GAS IN WG SHUTDOWN TANK B WAS LESS THAN THE TS LIMIT. THE SAMPLE WAS NOT OBTAINED WITHIN THE TIME LIMIT BECAUSE DAY SHIFT RDW PERSONNEL HAD ERRONEOUSLY DETERMINED THAT A SAMPLE FROM WG SHUTDOWN TANK B WAS NOT NECESSARY TO SATISFY THE TS SAMPLING REQUIREMENT. THIS EVENT IS ASSIGNED A CAUSE OF PERSONNEL ERROR BECAUSE RDW PERSONNEL INCORRECTLY DETERMINED THAT A SAMPLE WAS NOT REQUIRED ON WG SHUTDOWN TANK B ON THE AFTERNOON OF 12/31/87. THE GOVERNING PROCEDURE WILL BE CHANGED TO REQUIRE 2 RDW PERSONNEL TO CHECK THE WG LOGBOOK DAILY AND DOCUMENT WHICH TANKS ARE TO BE SAMPLED PRIOR TO CONTACTING HP. THE DAILY WG TURNOVER SHEET WILL BE ENHANCED AND RDW PERSONNEL WILL BE REQUALIFIED ON WG TANK SAMPLING.

[142] MCGUIRE 2 DOCKET 50-370 LER 87-016 REV 01
UPDATE ON REACTOR TRIP DUE TO OVERCURRENT FAULTS IN AN INSTRUMENT AIR COMPRESSOR MOTOR CAUSED BY LOSS OF POWER TO A MAIN TURBINE CONTROL SYSTEM RELAY.
EVENT DATE: 090687 REPORT DATE: 121687 NSSS: WE TYPE: PWR
VENDOR: RELIANCE ELECTRIC COMPANY

(NSIC 207557) AT 100% POWER ON SEPTEMBER 6, 1987 AT 1035, A UNIT 2 REACTOR/TURBINE TRIP OCCURRED DUE TO HIGH PRESSURIZER PRESSURE WHEN MAIN TURBINE GOVERNOR AND INTERCEPT VALVES CLOSED AS DIRECTED BY THE DIGITAL ELECTRO-HYDRAULIC (DEH) TURBINE CONTROL SYSTEM. THE GOVERNOR AND INTERCEPT VALVE CLOSE SIGNAL WAS GENERATED BY LOSS OF POWER TO A DEH TURBINE CONTROL SYSTEM RELAY WHEN POWER WAS LOST TO KXB. POWER WAS LOST ON AUXILIARY POWER PANELBOARD KXB DUE TO AN OVERCURRENT FAULT BREAKER TRIP CAUSED BY A GROUNDED MOTOR LEAD CONNECTOR (INSULATING TAPE HAD WORN ALLOWING CONNECTING LUG TO GROUND TO MOTOR FRAME) ON INSTRUMENT AIR (VI) COMPRESSOR A. OPERATIONS IMPLEMENTED THE REACTOR TRIP PROCEDURE. POWER WAS RESTORED TO AUXILIARY POWER PANELBOARD KXB FROM STATIC INVERTER KXB. UNIT 2 RETURNED TO MODE 1, POWER OPERATION, ON SEPTEMBER 7, AT 2110. THE CONNECTING LUG WAS REINSULATED IN THE CONNECTION BOX AND THE COMPRESSOR

WAS RETURNED TO SERVICE. VI COMPRESSOR MOTORS B&C WILL BE INSPECTED FOR SIMILAR CONDITION. SIMILAR MOTORS IN OTHER APPLICATIONS WILL BE INSPECTED AND RETAPED AS NECESSARY.

[143] MCGUIRE 2 DOCKET 50-370 LER 87-021
REACTOR TRIP OCCURRED DUE TO REVERSED WIRING ON GENERATOR STATOR COOLING D/P SWITCH CIRCUITRY COMBINED WITH CALIBRATION DRIFT OF D/P SWITCH AND GAUGE.
EVENT DATE: 113087 REPORT DATE: 123087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 1 (PWR)
VENDOR: ITT-BARTON
UNITED ELECTRIC CONTROLS COMPANY
WESTINGHOUSE ELECTRIC CORP.

(NSIC 207691) ON 11/30/87 AT APPROXIMATELY 1730, THE UNIT 2 MAIN GENERATOR BREAKERS OPENED, SHORTLY AFTER A UNIT 2 GENERATOR STATOR COOLING (KG) FLOW DIFFERENTIAL PRESSURE (D/P) LOW ALARM WAS DISPLAYED ON THE OPERATOR AID COMPUTER (OAC); THE TURBINE THEN TRIPPED ON OVERSPEED, AND A REACTOR TRIP RESULTED. THE REACTOR TRIP WAS DUE TO REVERSED WIRING ON THE LOW AND EMERGENCY LOW GENERATOR KG D/P SWITCH CIRCUITRY, COMBINED WITH CALIBRATION DRIFT ON THE LOW GENERATOR KG D/P SWITCH AND D/P GAUGE. UNIT 2 STABILIZED FROM THE REACTOR TRIP BY 1830. INSTRUMENT AND ELECTRICAL (IAE) CORRECTED THE WIRING PROBLEM ON DECEMBER 1, 1987, AND UNIT 2 WAS RETURNED TO POWER OPERATION. THIS EVENT IS ASSIGNED A CAUSE OF OTHER BECAUSE THE ORIGIN OF THE INCORRECT WIRING CONFIGURATION ON THE LOW AND EMERGENCY LOW GENERATOR KG FLOW D/P SWITCH CIRCUITS COULD NOT BE DETERMINED AND BECAUSE THE CALIBRATION DRIFT OF THE LOW GENERATOR KG FLOW D/P SWITCH AND PRESSURE GAUGE WAS DUE TO EQUIPMENT MALFUNCTION. UNIT 1 WILL BE CHECKED AND WIRING CORRECTION WILL BE MADE AS NECESSARY. PREVIOUSLY PLANNED KG SYSTEM MODIFICATION WILL BE REVIEWED TO ENSURE CRITICAL KG PRESSURE SWITCHES ARE REPLACED. LOOP CALIBRATIONS FOR THE GENERATOR KG FLOW D/P LOW AND EMERGENCY LOW CIRCUITRY WILL BE DEVELOPED.

[144] MCGUIRE 2 DOCKET 50-370 LER 87-022
SURVEILLANCE PROGRAM INADEQUATE TO DEMONSTRATE OPERABILITY OF COMPONENT COOLING SYSTEM HEAT EXCHANGERS.
EVENT DATE: 120287 REPORT DATE: 011388 NSSS: WE TYPE: PWR
VENDOR: DELTA SOUTHERN CO.

(NSIC 207932) ON 12/02/87, MCGUIRE RECOGNIZED THAT PAST TEST DATA ON THE UNIT 2 COMPONENT COOLING (KC) HEAT EXCHANGERS (HX) COULD BE INTERPRETED TO INDICATE THAT BOTH UNIT 2 KC HXS HAD BEEN INOPERABLE AT THE SAME TIME IN THE FALL OF 1987. THE PERFORMANCE TESTS MEASURED DIFFERENTIAL PRESSURE (DP) ACROSS THE HXS AS AN INDICATOR OF HEAT TRANSFER CAPABILITY. WHILE AVAILABLE DATA DOES NOT UNEQUIVOCALLY SUPPORT THE CONCLUSION THAT BOTH UNIT 2 KC HXS WERE INOPERABLE AT THE SAME TIME, THE TESTING PROGRAM DID FAIL TO POSITIVELY DEMONSTRATE OPERABILITY. SUBSEQUENT SUCCESSIVE TESTS HAVE DEMONSTRATED OPERABILITY. THE INDETERMINATE OPERABILITY OF THE KC HXS IS ASSIGNED A CAUSE OF MANAGEMENT DEFICIENCY BECAUSE THE TESTING PROGRAM WAS INADEQUATE TO POSITIVELY VERIFY OPERABILITY. A DESIGN STUDY WAS INITIATED FOR THE INSTALLATION OF A CONTINUOUS KC HX DP MONITORING SYSTEM, AND FOR ENHANCEMENTS TO THE NUCLEAR SERVICE WATER SYSTEM TO REDUCE HX FOULING PROBLEMS. TEMPORARY DP TRANSMITTERS WILL BE INSTALLED ON KC HXS AND MONITORED DAILY. A PROGRAM WILL BE DEVELOPED TO ENSURE THAT SURVEILLANCE TEST AFFECTING EQUIPMENT OPERABILITY WHICH FAILS WILL BE FOLLOWED BY A RE-EVALUATION OF SURVEILLANCE INTERVAL.

[145] MCGUIRE 2 DOCKET 50-370 LER 87-020
REACTOR COOLANT SYSTEM AND PRESSURIZER EXCEEDED TECH SPEC HEATUP/COOLDOWN RATES.
EVENT DATE: 121187 REPORT DATE: 012988 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 1 (PWR)

(NSIC 208054) DURING DECEMBER 1987, ANALYSIS AND EVALUATION OF OPERATIONAL DATA (NRC) CONDUCTED AN EVALUATION OF QUALITY ASSURANCE (QA) SURVEILLANCE. THIS EVALUATION INDICATED SEVERAL EVENTS WHERE PLANT COOLDOWN AND HEATUP RATES EXCEEDED ADMINISTRATIVE LIMITS AND ONE VIOLATION OF TECHNICAL SPECIFICATIONS (TS). SUBSEQUENTLY, OPERATIONS (OPS) DISCOVERED OTHER PAST INCIDENTS THAT EXCEEDED TS LIMITS. THE INCIDENTS OCCURRED DURING NORMAL SHUTDOWNS. ON 12/11 AND 16/87, WESTINGHOUSE PERFORMED A PRELIMINARY EVALUATION OF EFFECTS OF SIMILAR TRANSIENTS ON THE REACTOR COOLANT (NC) SYSTEM INTEGRITY AND CONCLUDED THAT EXCEEDING THE ALLOWABLE TS LIMITS FOR THE PRESSURIZER (PZR) AND THE NC SYSTEM DID NOT COMPROMISE STRUCTURAL INTEGRITY OF THE PZR AND NC SYSTEM. THIS EVENT IS ASSIGNED A CAUSE OF PERSONNEL ERROR BECAUSE NO DISCREPANCIES WERE NOTED BY OPS ON 3 OUT OF 4 PROCEDURES USED DURING THE SHUTDOWN AND STARTUP MODES OF OPERATION, AND OPS VERIFIED THE PROCEDURES COMPLETE WITH ALL ACCEPTANCE CRITERIA BEING MET. THIS EVENT IS ALSO ASSIGNED A CAUSE OF MANAGEMENT DEFICIENCY BECAUSE ONE REASON FOR THE TS COOLDOWN LIMITS BEING EXCEEDED WAS DUE TO NC SURGES INTO AND OUT OF THE PZR. SINCE THESE SURGES ARE NOT UNCOMMON DURING THIS MODE OF OPERATION; OPS SHOULD HAVE ADDRESSED THIS PROBLEM WITH DESIGN ENGINEERING AND WESTINGHOUSE. WESTINGHOUSE WILL PROVIDE

[146] MILLSTONE 1 DOCKET 50-245 LER 87-043
HYDRAULIC SNUBBER FAILURE CAUSED BY INADEQUATE DESIGN.
EVENT DATE: 111687 REPORT DATE: 121687 NSSS: GE TYPE: BWR
VENDOR: BERGEN-PATTERSON PIPE SUPPORT CORPORATION

(NSIC 207415) ON NOVEMBER 16, 1987 WHILE THE PLANT WAS IN COLD SHUTDOWN TO INVESTIGATE THE CAUSE FOR THE INCREASE IN THE UNIDENTIFIED LEAKAGE IN CONTAINMENT, IT WAS NOTICED THAT ONE OF THE HYDRAULIC SNUBBERS WAS LEAKING FLUID AND ITS RESERVOIR LEVEL WAS LOW. AS A RESULT OF THIS FINDING, A VISUAL INSPECTION OF ALL SAFETY RELATED HYDRAULIC SNUBBERS IN CONTAINMENT WAS PERFORMED TO ENSURE THAT THIS CONDITION DID NOT EXIST ON ANY OF THE REMAINING SNUBBERS. A TOTAL OF TWO SNUBBERS WERE FOUND TO BE LOW ON HYDRAULIC FLUID (HSS-2; HSS-95). IN ORDER TO DETERMINE THE OPERABILITY OF THE SNUBBERS, THE SNUBBERS WERE REMOVED AND FUNCTIONALLY TESTED. BOTH SNUBBERS FAILED TO MEET THE REQUIRED LOCKUP RATE IN COMPRESSION AND WERE DECLARED INOPERABLE. THE SNUBBERS WERE DISASSEMBLED AND INSPECTED. BOTH SNUBBERS WERE FOUND TO HAVE THE "O" RINGS DISPLACED FROM THEIR RETAINER ASSEMBLY. THE SNUBBERS WERE REBUILT, FUNCTIONALLY TESTED FOR ACCEPTABILITY, AND REINSTALLED.

[147] MILLSTONE 2 DOCKET 50-336 LER 87-004 REV 01
UPDATE ON CONTAINMENT PURGE ISOLATION DUE TO ACTUATION OF ESAS.
EVENT DATE: 020287 REPORT DATE: 010488 NSSS: CE TYPE: PWR
VENDOR: NUCLEAR MEASUREMENTS CORP.

(NSIC 207683) THIS LER ADDRESSES TWO (2) EVENTS CONCERNING EMERGENCY SAFEGUARDS ACTUATION SYSTEM (ESAS) ACTUATION, RESULTING IN A CONTAINMENT PURGE VALVE ISOLATION. AT 2101 ON FEBRUARY 2, 1987 WITH THE PLANT IN MODE 5 AN AUTOMATIC ACTUATION BY ESAS RESULTED IN A CONTAINMENT PURGE VALVE ISOLATION. THE ACTUATION WAS DUE TO A HIGH RADIATION SIGNAL FROM RADIATION MONITOR RM8123A. THE RADIATION MONITOR WAS RESET AND NO OTHER RADIATION MONITORS INDICATED ANY CHANGE DURING THE EVENT. THE ESAS WAS RESET AND CONTAINMENT PURGE REINSTATED. AT 2208 ON FEBRUARY 8, 1987 WITH THE PLANT IN MODE 5 AN AUTOMATIC ACTUATION BY ESAS RESULTED IN A CONTAINMENT PURGE VALVE ISOLATION. THE ACTUATION WAS DUE TO A HIGH RADIATION SIGNAL FROM RADIATION MONITOR RM8123A. THE RADIATION MONITOR WAS RESET AND NO OTHER RADIATION MONITORS INDICATED ANY CHANGE DURING THE EVENT. THE ESAS WAS RESET AND CONTAINMENT PURGE REINSTATED. INVESTIGATION DISCOVERED A BROKEN MYLAR WINDOW ON THE BETA SCINTILLATION DETECTOR. THIS WAS SUBSEQUENTLY DETERMINED TO BE THE CAUSE FOR THE FIRST OCCURRENCE. HEALTH PHYSICS PROCEDURE HP 908/2908/3908D AND ITS ASSOCIATED FORM HP 908/2908/3908D-3 WERE REVISED JUNE 30,

1987. THE REVISION SPECIFIES THAT THE RADIATION MONITOR RM8123A WILL BE REMOVED FROM SERVICE DURING FILTER CHANGEOUT.

[148] MILLSTONE 2 DOCKET 50-336 LER 87-013 REV 01
UPDATE ON FIRE WATCH VIOLATIONS UNDER LIMITING CONDITIONS FOR OPERATION.
EVENT DATE: 121987 REPORT DATE: 012988 NSSS: CE TYPE: PWR

(NSIC 208028) ON DECEMBER 19, 1987 AT 0845 MILLSTONE UNIT 2 WAS OPERATING IN MODE 1 AT 95% POWER, AND UNDER THE TECHNICAL SPECIFICATION ACTION STATEMENT FOR LIMITING CONDITION FOR OPERATION (LCO) 3.7.10.A.1. AS REQUIRED BY LCO 3.7.10.A.1, A FIRE WATCH PATROL HAD BEEN ESTABLISHED INITIALLY TO INSPECT A CABLE VAULT AREA CONTAINING A NON-QUALIFIED CABLE TRAY ENCLOSURE. THE FIRE WATCH PATROL IS REQUIRED TO INSPECT AFFECTED AREAS ONCE PER HOUR AND RECORD THE INSPECTION IN A FIRE WATCH ROUNDS LOG BOOK LOCATED IN THE MAIN CONTROL ROOM. THIS FIRE WATCH PATROL HAD ORIGINALLY COMMENCED ON JULY 13, 1987 AT 1830 HOURS. THIS LER IS SUBMITTED TO REPORT NON-COMPLIANCE WITH LCO 3.7.10.A.1. THIS OCCURRED WHEN THE FIRE WATCH PATROL FAILED TO CONDUCT AN HOURLY INSPECTION OF AFFECTED AREAS ON DECEMBER 19, 1987 BETWEEN 0745 AND 0945. THE MISSED INSPECTION WAS DETECTED SHORTLY AFTER 0900 BY THE CONTROL ROOM OPERATOR UPON REVIEW OF THE FIRE WATCH ROUNDS LOG BOOK. AN OPERATOR WAS PROMPTLY DISPATCHED TO CONDUCT THE FIRE WATCH PATROL AND THE FIRE WATCH ROUNDS LOG BOOK WAS SIGNED AT 0945. ADDITIONAL CORRECTIVE ACTION INVOLVED REPLACING THE FIRE WATCH CONTRACTOR, AND RELOCATING THE LOG BOOK TO A MORE PROMINENT LOCATION WHERE MISSED ENTRIES WILL BE PICKED UP MORE QUICKLY BY THE SENIOR CONTROL ROOM OPERATOR.

[149] MILLSTONE 2 DOCKET 50-336 LER 87-014
MAIN STEAM VALVE SETPOINT DRIFT.
EVENT DATE: 123187 REPORT DATE: 012588 NSSS: CE TYPE: PWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV

(NSIC 207964) THIS LICENSEE EVENT REPORT (LER) IS SUBMITTED FOR INFORMATION ONLY. ON DECEMBER 31, 1987 AT 1200, THE PLANT CONDUCTED ROUTINE MAIN STEAM SAFETY VALVE SIMMER TESTING AS REQUIRED BY TECHNICAL SPECIFICATION SURVEILLANCE 4.7.1.1. PLANT CONDITIONS WERE AS FOLLOWS: HOT STANDBY (MODE 3), 0% POWER, 515 DEGREES FAHRENHEIT, AND 2010 PSI. TWELVE OF THE SIXTEEN VALVES TESTED FAILED THE INITIAL SIMMER TEST; TWO WITH SETPOINTS BELOW THE REQUIRED RANGE AND TEN WITH SETPOINTS ABOVE. THE IMMEDIATE CAUSE OF THE SETPOINT DRIFT IS NOT KNOWN. EACH VALVE SETPOINT WAS ADJUSTED AS IT WAS DETERMINED TO BE OUT OF SPECIFICATION, EXCEPT FOR THE VALVES SENT OUT TO BE REBUILT. THE PLANT COMPLIED WITH THE REQUIREMENTS OF TECHNICAL SPECIFICATION ACTION STATEMENT 3.7.1.1.A AT ALL TIMES. SIMILAR EVENTS: 86-008, 83-021

[150] MILLSTONE 3 DOCKET 50-423 LER 87-010 REV 01
UPDATE ON LOOSE PART DETECTION SYSTEM INOPERABLE CHANNEL FOR UNKNOWN REASONS.
EVENT DATE: 022287 REPORT DATE: 021088 NSSS: WE TYPE: PWR

(NSIC 207980) THIS SUPPLEMENTAL SPECIAL REPORT IS BEING SUBMITTED TO REPORT THE CAUSE OF AN INOPERABLE CHANNEL IN THE LOOSE PARTS MONITORING SYSTEM (LPMS). ON 2/22/87 WHILE AT 100% POWER IN MODE 1, CHANNEL 4 OF THE LPMS FAILED. THE PLANT ENTERED THE APPLICABLE LIMITING CONDITION FOR OPERATION ACTION STATEMENT WHICH PERMITS CONTINUED OPERATION. DUE TO INACCESSIBILITY OF THE EQUIPMENT WHILE IN MODEL 1, A PLANT OUTAGE WAS REQUIRED TO DETERMINE EXACT CAUSE OF THE CHANNEL FAILURE. IT HAS BEEN DETERMINED THAT THE CAUSE OF THE CHANNEL FAILURE WAS A LOOSE ELECTRICAL CONNECTION. THE PLUG CONNECTING THE SENSOR CO-AXIAL CABLE TO THE PRE-AMPLIFIER INSIDE CONTAINMENT WAS FOUND LOOSE. IT IS UNKNOWN HOW OR WHY THE CONNECTING PLUGS WORKED LOOSE. AS PART OF A RECENT SYSTEM UPGRADE, THE PRE-AMPLIFIERS INSIDE CONTAINMENT FOR THE EIGHT (8) SYSTEM CHANNELS WERE REMOVED. THE PLUG CONNECTORS WERE REPLACED WITH A DIFFERENT TYPE CONNECTOR WHICH TIES THE

CO-AXIAL CABLE TO A TERMINAL BLOCK. THE UTILIZATION OF THESE NEW TYPE CONNECTORS IS EXPECTED TO PREVENT RECURRENCES.

[151] MILLSTONE 3 DOCKET 50-423 LER 87-009 REV 01
UPDATE ON EARLY LIFTING OF PRESSURIZER SAFETY VALVES FOR UNDETERMINED REASONS.
EVENT DATE: 031187 REPORT DATE: 012288 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE

(NSIC 207979) ON MARCH 3, 1987, WYLE LABORATORIES NOTIFIED NORTHEAST UTILITIES OF AN ANOMALY IN THE TEST RESULTS ON THREE PREVIOUSLY INSTALLED PRESSURIZER SAFETY VALVES. THE SUBJECT VALVES ARE REQUIRED BY TECHNICAL SPECIFICATIONS TO BE OPERABLE WITH A SETTING OF 2500 PSIA +/- 1% DURING MODES 1, 2, AND 3. BENCH TESTING AT WYLE LABORATORIES REVEALED ONE VALVE WAS LIFTING AT 2448 PSIA, ANOTHER WAS LIFTING AT 2417 PSIA, AND THE THIRD WAS LIFTING AT 2402 PSIA. THE EXACT CAUSE OF THE SAFETY VALVES DRIFTING FROM THEIR SETPOINTS IS NOT KNOWN. DIFFERENCES IN VENDOR PROCEDURES MAY BE A CONTRIBUTING FACTOR, IN THAT THE VALVES WERE ORIGINALLY SET BY ONE VENDOR, AND LATER TESTED BY ANOTHER. IT IS ALSO THOUGHT THAT LEAKAGE OF NON-CONDENSIBLE GASES PAST THE VALVE SEATS MAY HAVE CONTRIBUTED TO THE SETPOINT DRIFT. THE VALVES WERE SUBSEQUENTLY RESET TO THEIR PROPER PRESSURE, AND REWORKED SUCH THAT NO LEAKAGE WAS EXPERIENCED WHEN LEAK TESTED WITH STEAM AND AIR. THE SAFETIES CURRENTLY INSTALLED ON THE PRESSURIZER WERE REMOVED DURING THE FALL 1987 REFUELING OUTAGE. THESE VALVES WILL BE TESTED, AND THE RESULTS FORWARDED AS SUPPLEMENT 2 TO THIS LER.

[152] MILLSTONE 3 DOCKET 50-423 LER 87-023 REV 01
UPDATE ON PRESSURIZER CUBICLE EXCEEDING AREA TEMPERATURE LIMITS.
EVENT DATE: 042587 REPORT DATE: 012288 NSSS: WE TYPE: PWR

(NSIC 207981) THIS SPECIAL REPORT IS BEING SUBMITTED PURSUANT TO PLANT TECHNICAL SPECIFICATIONS 3.7.14B AND 6.9.2 WHICH REQUIRE THAT A SPECIAL REPORT BE SUBMITTED TO THE NRC IF ONE OR MORE AREAS EXCEED THE SPECIFIED TEMPERATURE LIMIT BY LESS THAN 20 DEGREES FAHRENHEIT FOR MORE THAN 8 HOURS. CONTAINMENT AREA CS-01 (INSIDE CRANE WALL) HAS EXCEEDED THE 120 DEGREES FAHRENHEIT SPECIFIED LIMIT. THE PRESSURIZER CUBICLE IS THE ONLY AREA EXCEEDING THE LIMITS. SPECIAL REPORT LER 87-023-00 (DATED 5/26/87) STATED THAT ON 4/25/87 THE PRESSURIZER CUBICLE TEMPERATURE EXCEEDED THE 120 DEGREE FAHRENHEIT LIMITS AND THAT THE TEMPERATURE HAD BEEN VARYING BETWEEN APPROXIMATELY 115 TO 125 DEGREES SINCE, REACHING A HIGH OF 127 DEGREES (FOR A TWO HOUR DURATION) DURING TESTING. THE ROOT CAUSE HAS BEEN DETERMINED TO BE INHERENT TO THE DESIGN OF THE PRESSURIZER CUBICLE. A REVISION TO THE PLANT'S TECHNICAL SPECIFICATIONS WILL BE SUBMITTED. THERMAL LIFE CALCULATIONS WILL BE REVISED ACCORDINGLY.

[153] MILLSTONE 3 DOCKET 50-423 LER 87-041
INADEQUATE TESTING OF CONTAINMENT PENETRATION CIRCUIT BREAKERS.
EVENT DATE: 111687 REPORT DATE: 121687 NSSS: WE TYPE: PWR

(NSIC 207477) ON NOVEMBER 16, 1987 AT 1830 HOURS, WITH THE PLANT AT 89 DEGREES FAHRENHEIT AND ATMOSPHERIC PRESSURE DURING REFUEL OPERATIONS, IT WAS DISCOVERED THAT THE 480 AND 120VAC CIRCUIT BREAKERS SERVING AS CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES HAD NOT BEEN TESTED IN ACCORDANCE WITH PLANT TECHNICAL SPECIFICATION 4.8.4.1(A)(2). THE CAUSE WAS A FAILURE TO PERFORM A PRE-LICENSING REVIEW OF INITIAL CIRCUIT BREAKER TEST DATA AGAINST THE REQUIREMENTS OF THE FINALIZED PLANT TECHNICAL SPECIFICATIONS. THE BREAKERS WERE INITIALLY TESTED USING PROCEDURES WHICH WERE APPROVED FOR USE APPROXIMATELY THREE YEARS BEFORE FINALIZED TECHNICAL SPECIFICATIONS BECAME AVAILABLE. A TECHNICAL SPECIFICATION CHANGE REQUEST WAS SUBMITTED TO PERMIT TESTING IN ACCORDANCE WITH THE ACCEPTED INDUSTRY STANDARD. THE CIRCUIT BREAKERS WILL BE RETESTED TO TECHNICAL SPECIFICATION REQUIREMENTS PRIOR TO PLANT HEAT-UP. IN ADDITION, THE

CONTAINMENT PENETRATION BREAKER SURVEILLANCE PROCEDURES WERE REVIEWED TO VERIFY TECHNICAL SPECIFICATION COMPLIANCE.

[154] MILLSTONE 3 DOCKET 50-423 LER 87-042
MISSED INTERMEDIATE RANGE/POWER RANGE SURVEILLANCE DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 111787 REPORT DATE: 121687 NSSS: WE TYPE: PWR

(NSIC 207478) ON NOVEMBER 17, 1987 AT 0930, WHILE IN MODE 6 (0% POWER, 89F, 14.7 PSIA) IT WAS DISCOVERED THAT A PLANT TECHNICAL SPECIFICATION SURVEILLANCE, REQUIRING MEASURING A DETECTOR PLATEAU CURVE FOR EACH INTERMEDIATE RANGE (IR) AND POWER RANGE (PR) NEUTRON DETECTOR, HAD NOT BEEN PERFORMED. IMMEDIATE ACTION WAS TO INFORM PLANT MANAGEMENT. THE SURVEILLANCES WERE MISSED DUE TO PROCEDURAL INADEQUACY. WHEN THE IR AND PR SURVEILLANCES WERE WRITTEN, THE REQUIREMENT TO PERFORM THESE PLATEAU CURVES WAS OMITTED. IT COULD NOT BE DETERMINED WHY THE IR AND PR PROCEDURES WERE DEFICIENT. AS A CORRECTIVE ACTION, A NEW IR/PR SURVEILLANCE PROCEDURE IS BEING WRITTEN TO PERFORM DETECTOR PLATEAU CURVE DETERMINATION. THIS WILL BE PERFORMED DURING THE NEXT POWER ASCENSION. FURTHERMORE, A COMPLETE REVIEW OF ALL TECHNICAL SPECIFICATIONS AND THEIR ASSOCIATED SURVEILLANCE PROCEDURES WILL BE INITIATED. A SUPPLEMENTAL LER WILL BE ISSUED ON OR BEFORE DECEMBER 15, 1988 ON THE RESULTS OF THIS REVIEW.

[155] MILLSTONE 3 DOCKET 50-423 LER 87-043
BYPASS LEAKAGE IN EXCESS OF TECHNICAL SPECIFICATION LIMITS.
EVENT DATE: 111887 REPORT DATE: 121787 NSSS: WE TYPE: PWR
VENDOR: FISHER CONTROLS CO.

(NSIC 207479) ON NOVEMBER 18, 1987 AT APPROXIMATELY 1800 WHILE AT 0% POWER, ATMOSPHERIC PRESSURE, AND 95 DEGREES (MODE 6 - REFUELING), DURING THE PERFORMANCE OF LOCAL LEAK RATE TESTING (LLRT) ON 3CDS-CTV91B, THE AS- FOUND VALVE LEAKAGE EXCEEDED THE TECHNICAL SPECIFICATION LIMIT (PER 3.6.1.2.C - ENCLOSURE BUILDING BYPASS LEAKAGE PATHS) OF 0.01 LA (13,690 SCCM). THIS VALVE IS THE "B" TRAIN CHILLED WATER SYSTEM INSIDE CONTAINMENT SUPPLY ISOLATION VALVE. AFTER REPLACEMENT OF THE T-RING, THE LLRT WAS REPERFORMED AND A SATISFACTORY LEAKAGE OF 10 SCCM WAS RECORDED. SINCE THIS FAILURE IS THE FIRST OF THIS TYPE, IT IS CONSIDERED TO BE AN ISOLATED CASE.

[156] MILLSTONE 3 DOCKET 50-423 LER 87-044
VENTILATION RADIATION MONITOR SURVEILLANCE PERFORMED LATE.
EVENT DATE: 112087 REPORT DATE: 122187 NSSS: WE TYPE: PWR

(NSIC 207480) AT 1200 ON NOVEMBER 20, 1987, ADMINISTRATIVE REVIEW REVEALED THE SURVEILLANCE INTERVAL FOR CHANNEL CALIBRATION OF UNIT 3 VENTILATION VENT STACK (TURBINE BUILDING) RADIATION MONITOR HAD BEEN EXCEEDED. THE UNIT 3 VENTILATION VENT STACK (TURBINE BUILDING) RADIATION MONITOR WAS DECLARED INOPERABLE, AND THE CHANNEL CALIBRATION SURVEILLANCE BEGUN. IN ACCORDANCE WITH PLANT TECHNICAL SPECIFICATIONS, AUXILIARY MEANS FOR COLLECTING SAMPLES WERE ESTABLISHED: CONTINUOUS SAMPLING FOR IODINE AND PARTICULATES WAS INITIATED, GAS GRAB SAMPLES WERE OBTAINED EVERY 12 HOURS, AND FLOW THROUGH THE AUXILIARY SAMPLE RIG WAS VERIFIED EVERY 4 HOURS. UNIT 3 VENTILATION VENT STACK (TURBINE BUILDING) RADIATION MONITOR CALIBRATION WAS COMPLETED ON NOVEMBER 28, 1987. ALL I&C (INSTRUMENTATION AND CONTROLS) 18-MONTH SURVEILLANCES WERE VERIFIED TO BE SCHEDULED WITHIN REQUIRED INTERVALS. ADMINISTRATIVE PROCEDURES WERE ESTABLISHED REQUIRING DEPARTMENT HEAD APPROVAL TO RESCHEDULE SURVEILLANCES WHICH WERE NOT PERFORMED ON THE SCHEDULED DATE. SURVEILLANCE TRACKING PROGRAMS WERE EXPANDED TO INCLUDE ALL 18-MONTH SURVEILLANCES.

[157] MILLSTONE 3 DOCKET 50-423 LER 87-045
FAILURE TO SAMPLE DIESEL FUEL OIL FOR KINEMATIC VISCOSITY PRIOR TO ADDITION TO STORAGE TANKS.
EVENT DATE: 112187 REPORT DATE: 121887 NSSS: WE TYPE: PWR

(NSIC 207481) ON NOVEMBER 21, 1987 AT 1954 HOURS IT WAS DISCOVERED THAT EMERGENCY DIESEL GENERATOR (EDG) FUEL OIL WAS NOT BEING ANALYZED FOR KINEMATIC VISCOSITY BEFORE ADDITION TO A STORAGE TANK. THE ANALYSIS, REQUIRED BY TECHNICAL SPECIFICATIONS, WAS PERFORMED AS PART OF THE 30-DAY ANALYSIS, BUT IT WAS NOT PERFORMED BEFORE ADDITION TO A STORAGE TANK DURING THE FIRST CYCLE OF PLANT OPERATION. THERE HAVE BEEN 45 FUEL OIL DELIVERIES FROM THE BEGINNING OF COMMERCIAL OPERATION IN MAY, 1986 UNTIL THE DISCOVERY OF MISSED SURVEILLANCES. THE PLANT WAS IN MODE 6 (REFUELING) AT 0% POWER AT THE TIME OF DISCOVERY. THE ROOT CAUSE FOR MISSING THE SURVEILLANCES WAS INCORRECT COMPARISON OF ACCEPTANCE CRITERIA BETWEEN TWO FORMS. THE DEFICIENT FORM HAS BEEN UPDATED TO INCLUDE KINEMATIC VISCOSITY AS WELL AS THE REMAINING ANALYSES REQUIRED BY TECHNICAL SPECIFICATIONS.

[158] MILLSTONE 3 DOCKET 50-423 LER 87-047
CORE ALTERATION PERFORMED WITHOUT PROPER COMMUNICATIONS OR SRO COVERAGE DUE TO PROCEDURAL ERROR.
EVENT DATE: 113087 REPORT DATE: 122887 NSSS: WE TYPE: PWR

(NSIC 207649) ON NOVEMBER 30, 1987 AT 1530 WITH THE PLANT IN REFUELING MODE 6 (0% POWER, 98F, ATMOSPHERIC PRESSURE) A CORE ALTERATION (PLACING OF THE UPPER INTERNALS INTO THE REACTOR VESSEL) WAS MADE WITHOUT A LICENSED SENIOR REACTOR OPERATOR (SRO) PRESENT IN THE CONTAINMENT, OR PROPER COMMUNICATION ESTABLISHED BETWEEN THE CONTAINMENT AND THE CONTROL ROOM AS REQUIRED BY TECHNICAL SPECIFICATIONS. IMMEDIATE ACTION WAS TO STOP ALL CORE ALTERATIONS UNTIL SUCH TIME AS PROPER COMMUNICATIONS AND SRO COVERAGE IN THE CONTAINMENT COULD BE ESTABLISHED. THE EVENT WAS CAUSED BY BOTH PERSONNEL AND PROCEDURAL ERRORS. THE MAINTENANCE PROCEDURE AND OPERATIONS PROCEDURE THAT CONTROLLED PLACING THE UPPER INTERNALS IN THE REACTOR VESSEL DID NOT SPECIFY THE PROPER CONTROLS TO ENSURE THAT ALL TECHNICAL SPECIFICATIONS WERE MET. AS AN ACTION TO PREVENT RECURRENCE BOTH THE MAINTENANCE AND OPERATIONS PROCEDURES HAVE BEEN REVISED TO ENSURE THAT ALL TECHNICAL SPECIFICATIONS WILL BE MET PRIOR TO MAKING ANY CORE ALTERATIONS.

[159] MILLSTONE 3 DOCKET 50-423 LER 87-048
FAILURE TO MONITOR INOPERABLE FIRE ASSEMBLIES.
EVENT DATE: 120387 REPORT DATE: 010488 NSSS: WE TYPE: PWR

(NSIC 207650) THIS REPORT DOCUMENTS TWO OCCURRENCES OF BREACHED FIRE BARRIERS. THE PLANT CONDITIONS AT THE TIME OF THESE INSTANCES WAS 0% POWER, APPROXIMATELY 110 DEGREES, AND ATMOSPHERIC PRESSURE WHILE SHUTDOWN FOR REFUELING. THE FIRST OCCURRENCE WAS DISCOVERED AT 0600 ON DECEMBER 3, 1987. DRAIN HOSES WERE STRUNG THROUGH TWO LEVELS OF THE B TRAIN RECIRCULATION SPRAY SYSTEM CUBICLE AT 2030 ON DECEMBER 2, 1987. THERE WAS AN OPERATOR IN ATTENDANCE UNTIL THE MIDNIGHT SHIFT TURNOVER, BUT NOT FROM MIDNIGHT TO 0600 DECEMBER 3, WHEN THE PROBLEM WAS DISCOVERED AND CORRECTED. THE IMMEDIATE CORRECTIVE ACTION WAS TO POST A FIRE WATCH. THE SECOND OCCURRENCE WAS DISCOVERED AT 1600 ON DECEMBER 5, 1987. AN AIR HOSE WAS STRUNG THROUGH THE DOOR TO THE CONTROL BUILDING MECHANICAL EQUIPMENT ROOM AT 1300 ON DECEMBER 4, 1987. THE HOSE WAS IN USE AND MONITORED UNTIL 1400, WHEN THE WORK WAS COMPLETED. THE CLEANUP AFTER THE JOB DID NOT REMOVE THE AIR HOSE, AND NO FURTHER ACTION WAS TAKEN UNTIL AN UNLICENSED OPERATOR FOUND THE DOOR BLOCKED OPEN AT 1600 ON DECEMBER 5. IMMEDIATE CORRECTIVE ACTION WAS TO REMOVE THE HOSE AND REESTABLISH THE FIRE BOUNDARY. THE ROOT CAUSE OF BOTH EVENTS WAS HUMAN ERROR. BOTH OCCURRENCES WOULD HAVE BEEN AVOIDED HAD THE REQUIRED ACTION STATEMENT BEEN ENTERED AT THE TIME THAT THE DOORS WERE OPENED.

[160] MILLSTONE 3 DOCKET 50-423 LER 87-049
MISSED ENGINEERING EVALUATION DUE TO MISINTERPRETATION OF TECHNICAL SPECIFICATIONS.
EVENT DATE: 121687 REPORT DATE: 011388 NSSS: WE TYPE: PWR

(NSIC 207871) ON DECEMBER 16, 1987 AT 1155 HOURS WHILE AT 0% POWER (98 DEGREES, ATMOSPHERIC PRESSURE) IN MODE 6 (REFUELING) THE UNIT SUPERINTENDENT, WHILE REVIEWING A NON-CONFORMANCE REPORT, DISCOVERED THAT AN ENGINEERING EVALUATION REQUIRED BY PLANT TECHNICAL SPECIFICATIONS HAD NOT BEEN PERFORMED WITHIN ITS ALLOTTED TIME FRAME. ON BOTH NOVEMBER 20, 1987 (MODE 6, 85 DEGREES, ATMOSPHERIC PRESSURE) AND NOVEMBER 27, 1987 (MODE 6, 98 DEGREES, ATMOSPHERIC PRESSURE) WHILE FUNCTIONALLY TESTING SNUBBERS A FAILURE WAS DISCOVERED. EACH SNUBBER WAS REPLACED WITHIN 72 HOURS, HOWEVER AN ENGINEERING REVIEW WAS NOT COMPLETED WITHIN THE ALLOTTED TIME. IMMEDIATE CORRECTIVE ACTION WAS TO PERFORM AN ENGINEERING EVALUATION TO DETERMINE IF THE COMPONENTS WERE ADVERSELY AFFECTED. THE RESULTS CONFIRMED THE COMPONENTS WERE CAPABLE OF MEETING THE DESIGN SERVICE. THE ROOT CAUSE OF THE MISSED ENGINEERING EVALUATION WAS A MISINTERPRETATION OF THE ACTION STATEMENT OF TECHNICAL SPECIFICATION 3.7.10. AS ACTION TO PREVENT RECURRENCE, THIS EVENT HAS BEEN REVIEWED WITH ALL LICENSED SENIOR OPERATORS.

[161] MILLSTONE 3 DOCKET 50-423 LER 87-050
MISSED AREA TEMPERATURE MONITORING SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 122187 REPORT DATE: 012088 NSSS: WE TYPE: PWR

(NSIC 207916) ON DECEMBER 21, 1987 AT 1800 HOURS WHILE IN COLD SHUTDOWN, 0% POWER, 107 DEGREES FAHRENHEIT, ATMOSPHERIC PRESSURE, IT WAS DISCOVERED THAT THE TEMPERATURES FOR VARIOUS AREAS HOUSING SAFETY-RELATED AND ENVIRONMENTALLY QUALIFIED EQUIPMENT HAD NOT BEEN DETERMINED TO BE WITHIN TECHNICAL SPECIFICATIONS LIMITS WITHIN THE SPECIFIED INTERVAL. THE ROOT CAUSE OF THE INCIDENT WAS THE LACK OF DETAILED REVIEWS OF THE CALCULATING DATALOGGER PRINTOUTS BY SHIFT PERSONNEL. POOR CALCULATING DATALOGGER PRINTOUT FORMAT CONTRIBUTED TO THE INCIDENT. IMMEDIATE CORRECTIVE ACTION WAS TO MANUALLY OBTAIN TEMPERATURES OF THE AREAS IN QUESTION. BASED ON THE TEMPERATURE HISTORY OF THE AREAS AND AMBIENT TEMPERATURE TRENDS SURROUNDING THE TIME OF THE EVENT, NO ADVERSE SAFETY CONSEQUENCES TO THE PLANT OR TO THE HEALTH AND SAFETY OF THE PUBLIC RESULTED. DEPARTMENT PERSONNEL HAVE BEEN INSTRUCTED TO REVIEW INDIVIDUAL POINTS FOR VALIDITY WHEN REVIEWING THE CALCULATING DATALOGGER PRINTOUTS AND TO INITIATE CORRECTIVE ACTION FOR INVALID POINTS AS NECESSARY. THE APPLICABLE PROCEDURES HAVE BEEN UPDATED TO REQUIRE THIS REVIEW. METHODS TO IMPROVE THE HUMAN FACTORS DESIGN OF THE AREA TEMPERATURE MONITORING SYSTEM ARE IN PROGRESS AND WILL BE COMPLETED BY JUNE 15, 1988.

[162] MILLSTONE 3 DOCKET 50-423 LER 87-051
MISSED SURVEILLANCE ON FIRE RATED DOORS DUE TO PROCEDURAL DEFECT.
EVENT DATE: 122987 REPORT DATE: 012888 NSSS: WE TYPE: PWR

(NSIC 208042) ON 12/29/87 AT 0400 WHILE THE PLANT WAS SHUTDOWN FOR REFUELING (100 DEGREES, ATMOSPHERIC PRESSURE), IT WAS NOTED THAT SEVERAL DOORS THAT WERE BEING CHECKED WEEKLY AS FIRE RATED ASSEMBLIES SHOULD HAVE BEEN CHECKED DAILY PER TECH SPECS. FURTHER INVESTIGATION REVEALED THAT CERTAIN DOORS HAD NOT BEEN INCLUDED IN THE SURVEILLANCE EITHER DAILY OR WEEKLY. THE IMMEDIATE CAUSE OF THE EVENT WAS AN INCORRECT LISTING SUPPLIED TO THE OPERATORS PERFORMING THE INSPECTION. THE LISTING CONSISTS OF DOORS WHICH ARE LOCKED, REQUIRING WEEKLY CHECKS, AND DOORS WHICH ARE NOT LOCKED, REQUIRING DAILY CHECKS. AN OPERATOR REVIEWING HIS LISTING DISCOVERED THAT SOME OF THE DOORS HE WAS SCHEDULED TO CHECK WEEKLY WERE UNLOCKED, AND SHOULD HAVE BEEN CHECKED DAILY. IMMEDIATE CORRECTIVE ACTION CONSISTED OF INSPECTING ALL DOORS LISTED AS LOCKED TO DETERMINE THEIR STATUS. FOLLOW-ON ACTION CONSISTED OF A COMPREHENSIVE REVIEW OF ALL DOORS LISTED IN DESIGN DOCUMENTS AS FIRE RATED ASSEMBLIES. SURVEILLANCES HAVE BEEN MODIFIED TO ACCOUNT

FOR ALL FIRE DOORS. AS ACTION TO PREVENT RECURRENCE ALL DOORS LISTED AS LOCKED WILL BE CHECKED LOCKED DURING THE INSPECTION TO ENSURE THAT THERE IS NO CHANGE IN STATUS. A CHANGE WILL BE MADE TO THE ADMINISTRATIVE PROCEDURE CONTROLLING LOCKS TO DIRECTLY LINK LOCK CHANGES TO CHANGING THE SURVEILLANCE PROCEDURE.

[163] MONTICELLO DOCKET 50-263 LER 87-022
LATCHES ON TWO FIRE DOORS MADE INOPERABLE DUE TO INADEQUACIES IN PROCEDURES AND TRAINING.
EVENT DATE: 111287 REPORT DATE: 121487 NSSS: GE TYPE: BWR

(NSIC 207424) DURING PLANT INSPECTIONS, TWO FIRE DOORS LOCATED IN APPENDIX R FIRE BARRIERS WERE FOUND INOPERABLE DUE TO LATCHES HELD OPEN WITH TAPE. THESE FIRE DOORS ARE REQUIRED TO BE OPERABLE PER TECH. SPEC. 3.13.G.1. UPON DISCOVERY, THE TAPE WAS REMOVED AND THE DOORS MADE OPERABLE. THE ROOT CAUSE OF THESE TWO EVENTS WAS DETERMINED TO BE INADEQUACIES IN PROCEDURES AND TRAINING. ONE OF THE LATCHES WAS TAPED OPEN TO PREVENT THE DOOR FROM LOCKING, WHILE PERFORMING A RADIATION PROTECTION PROCEDURE WHICH REQUIRED THIS DOOR TO BE UNLOCKED IN PREPARATION FOR THE REFUELING OUTAGE. THE OTHER DOOR IS BELIEVED TO HAVE BEEN TAPED BY CONSTRUCTION PERSONNEL WHO WERE NOT AWARE THAT TAPING THE LATCH OPEN MAKES THE FIRE DOOR INOPERABLE. THE RADIATION PROTECTION PROCEDURE WILL BE REVISED TO INSTRUCT PERSONNEL NOT TO TAPE OPEN THE DOOR LATCHES. THE TRAINING PROGRAM WILL BE UPGRADED TO PROVIDE TRAINING ON WHAT WILL MAKE A FIRE DOOR INOPERABLE. ALSO, THE DAILY FIRE DOOR INSPECTION PROCEDURE WILL BE REVISED TO PROVIDE A MORE THOROUGH DAILY INSPECTION.

[164] MONTICELLO DOCKET 50-263 LER 87-020
CHECK VALVE DISC NUT TACK WELD FAILURE RESULTS IN POTENTIAL HPCI DEGRADATION.
EVENT DATE: 112087 REPORT DATE: 122187 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 207422) DURING A REFUELING OUTAGE THE REACTOR WATER CLEANUP (RWCU) RETURN CHECK VALVE, RC 6-1, WAS DISASSEMBLED FOR INSPECTION AS PART OF ASME BOILER AND PRESSURE VESSEL CODE SECTION XI VALVE TESTING PROGRAM. IT WAS FOUND THAT THE DISC HAD BECOME DETACHED FROM THE HINGE. IT APPEARED THAT THE DISC NUT TACK WELD FAILED, ALLOWING THE NUT TO WORK ITS WAY OFF THE DISC STEM. THIS COULD HAVE RESULTED IN DEGRADED HIGH PRESSURE COOLANT INJECTION SYSTEM OPERATION ON SYSTEM INITIATION DUE TO FLOW LOSS INTO NON-SAFETY RELATED, NON-SEISMIC PORTIONS OF THE RWCU SYSTEM THROUGH COMMON LINES. THE VALVE WAS REPAIRED ACCORDING TO THE VALVE MANUFACTURER'S RECOMMENDATIONS. THE SIZE OF THE WELD WAS INCREASED TO PREVENT RECURRENCE. THE VALVE WAS TESTED AND RETURNED TO SERVICE PRIOR TO PLANT STARTUP. THE LONG TERM SUITABILITY OF THIS VALVE IN THIS APPLICATION WILL BE EVALUATED.

[165] MONTICELLO DOCKET 50-263 LER 87-024
FALSE ASSUMPTION OF TEST COMPLETION CAUSES MISSED SAMPLING FOR AN INOPERATIVE PROCESS MONITOR.
EVENT DATE: 112187 REPORT DATE: 122187 NSSS: GE TYPE: BWR

(NSIC 207426) DURING A ROUTINE PLANT SHUTDOWN AND REFUELING OUTAGE, THE SERVICE WATER DISCHARGE PIPE GROSS RADIOACTIVITY MONITOR (SWDPGRM) WAS FOUND TO HAVE BEEN IN AN INOPERABLE CONDITION WITHOUT GRAB SAMPLES BEING COLLECTED FOR LONGER THAN 8 HOURS TIME. THE MISSED SAMPLE WAS DUE TO PERSONNEL ERROR WHEN THE SHIFT CHEMISTRY RADIATION PROTECTION SPECIALIST (RPS) REQUESTED COMPLETION OF THE MONITOR OPERABILITY TEST, BUT DID NOT VERIFY THAT A FUNCTIONAL TEST REQUIRED TO CLEAR AN INOPERABLE CONDITION ON THE SERVICE WATER MONITOR HAD BEEN COMPLETED, BEFORE HE SECURED HIS SAMPLING ROUTINE. AFTER INVESTIGATION OF THE CIRCUMSTANCES, THE RPS REALIZED THAT INFORMAL COMMUNICATIONS HAD CAUSED REQUIRED SAMPLES TO BE MISSED. HE THEN RESUMED 8 HOUR SAMPLING. THE SITUATION WAS DISCUSSED IN DETAIL WITH HIS SUPERVISORS. HE WAS COUNSELED CONCERNING PROPER SHIFT TURNOVER AND SURVEILLANCE

VERIFICATION. APPLICABLE LOGS AND PROCEDURES ARE BEING REVIEWED FOR POSSIBLE REVISION TO ASSIST IN PREVENTING FUTURE OCCURRENCES OF THIS NATURE.

[166] MONTICELLO DOCKET 50-263 LER 87-021
ESF ACTUATIONS DUE TO IMPROPER JUMPER PLACEMENT DURING PREOPERATIONAL TESTING.
EVENT DATE: 112487 REPORT DATE: 122387 NSSS: GE TYPE: BWR

(NSIC 207423) WHILE PERFORMING A MODIFICATION PREOPERATIONAL TEST PROCEDURE, A JUMPER WAS INADVERTENTLY INSTALLED ACROSS AN ISOLATION LOGIC RELAY COIL INSTEAD OF THE INTENDED RELAY CONTACT. THIS CAUSED THE FUSE IN AN UPSTREAM DISTRIBUTION PANEL FEEDING THE ENTIRE PRIMARY CONTAINMENT ISOLATION LOGIC PANEL TO BLOW. A SHUTDOWN COOLING TRIP, STANDBY GAS TREATMENT SYSTEM INITIATION AND AN ISOLATION OF REACTOR WATER CLEAN-UP AND REACTOR BUILDING AND DRYWELL VENTILATION SYSTEMS RESULTED. THE PLANT WAS FIVE WEEKS INTO A REFUELING OUTAGE WHEN THE EVENT OCCURRED. THE BLOWN FUSE WAS REPLACED AND SHUTDOWN COOLING WAS RESTORED. THE CAUSE OF THE EVENT WAS FAILURE TO RECOGNIZE THE CORRECT RELAY TERMINAL DESIGNATIONS AND IS THEREFORE ATTRIBUTED TO PERSONNEL ERROR. IMPROPER FUSE COORDINATION AFFECTED THE EXTENT OF THE EVENT. TO PRECLUDE SIMILAR EVENTS, ADDITIONAL TRAINING WILL BE CONDUCTED AND RELAY DRAWINGS WILL BE MADE AVAILABLE IN KEY AREAS. FUSES HAVE BEEN CHANGED TO IMPROVE FUSE BREAKER COORDINATION.

[167] MONTICELLO DOCKET 50-263 LER 87-023
REACTOR RECIRCULATION SYSTEM DECONTAMINATION CONNECTION DUE TO HIGH CYCLE FATIGUE.
EVENT DATE: 112587 REPORT DATE: 122887 NSSS: GE TYPE: BWR
VENDOR: SANDVIK STEEL INC.

(NSIC 207425) WHILE PERFORMING THE REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE TEST, A CRACK WAS DISCOVERED IN A TWO INCH DECONTAMINATION CONNECTION ON THE REACTOR RECIRCULATION SYSTEM A LOOP. THE PLANT WAS SHUTDOWN FOR A REFUELING OUTAGE. METALLOGRAPHIC EXAMINATION REVEALED THAT THE CRACK WAS CAUSED BY HIGH CYCLE FATIGUE. IT IS BELIEVED THAT THE FATIGUE FAILURE RESULTED FROM CYCLIC STRESSES CAUSED BY FORCED RESONANT VIBRATION IN THE DECONTAMINATION CONNECTION. THE DECONTAMINATION CONNECTION WAS MODIFIED SO THAT ITS NATURAL FREQUENCY DOES NOT COINCIDE WITH THE FREQUENCY OF ANY EXPECTED RECIRCULATION SYSTEM FORCING FUNCTIONS. OTHER RECIRCULATION SYSTEM BRANCH CONNECTIONS WERE EVALUATED. TWO OTHER DECONTAMINATION CONNECTIONS WERE IN THE RANGE OF CONCERN AND WERE MODIFIED. IT IS BELIEVED THAT NO OTHER CONNECTIONS ARE SUSCEPTIBLE TO THIS FAILURE MODE.

[168] NINE MILE POINT 1 DOCKET 50-220 LER 87-023
REACTOR BUILDING EMERGENCY VENTILATION INITIATION BECAUSE OF FUSE FAILURE AND FAILURE TO REPORT THE EVENT WITH RESPECT TO 10 CFR 50.73 AS A RESULT OF PERSONNEL ERRORS.
EVENT DATE: 042287 REPORT DATE: 122287 NSSS: GE TYPE: BWR
VENDOR: MCGRAW EDISON CO., POWER SYSTEMS DIV

(NSIC 207414) THIS LICENSEE EVENT REPORT REPORTS TWO RELATED EVENTS. THE FIRST EVENT INVOLVED AN ENGINEERED SAFETY FEATURE ACTUATION. ON APRIL 22, 1987, NINE MILE POINT UNIT 1 (NMP1) WAS OPERATING AT FULL POWER. AT 1434 HOURS, THE UNIT EXPERIENCED A TRIP OF THE NORMAL REACTOR BUILDING VENTILATION AND AN INITIATION OF REACTOR BUILDING EMERGENCY VENTILATION (RBEV). THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR INVOLVING INADVERTANT SHORTING OF A FUSE TO GROUND IN THE INITIATION CIRCUIT OF THE RBEV. IMMEDIATE CORRECTIVE ACTION INVOLVED REPLACING THE FUSE UNDER A STATION WORK REQUEST. THE RBEV WAS SECURED AT 1645 HOURS ON APRIL 22 AND THE NORMAL REACTOR BUILDING VENTILATION WAS RETURNED TO SERVICE. ON NOVEMBER 23, 1987, THE NMP1 TECHNICAL SUPPORT DEPARTMENT BECAME AWARE THAT THE 10 CFR 50.73 REPORT HAD NOT YET BEEN SUBMITTED FOR THE APRIL 22 EVENT. THE CAUSE OF THIS EVENT IS PERSONNEL ERROR. IMMEDIATE CORRECTIVE ACTION INVOLVED A DISCUSSION WITH THE PERSONNEL INVOLVED AS TO HOW AND WHY THIS EVENT OCCURRED. SUBSEQUENT

CORRECTIVE ACTION WILL INVOLVE PREPARING A LESSONS LEARNED TRANSMITTAL TO INFORM THE PERSONNEL OF THE AFFECTED DEPARTMENTS OF THE NEED FOR ATTENTION TO DETAIL DURING WORK ACTIVITIES.

[169] NINE MILE POINT 1 DOCKET 50-220 LER 87-026
FAILURE TO IDENTIFY PIPING SUPPORT AS ASME COMPONENT RESULTED IN RETURNING COMPONENT SUPPORT TO SERVICE WITHOUT ASME SECTION XI REQUIRED INSPECTION.
EVENT DATE: 103087 REPORT DATE: 010488 NSSS: GE TYPE: BWR

(NSIC 207731) ON DECEMBER 2, 1987, WHILE NINE MILE POINT UNIT 1 WAS AT 90 PERCENT POWER, INSERVICE INSPECTION DEFICIENCY/CORRECTIVE ACTION NOTICE (DCA) N1-87-019 WAS INITIATED. DCA N1-87-019 IDENTIFIED THAT ASME COMPONENT RESTRAINTS 93-R12-A/B AND 93-R13-A/B WERE RETURNED TO SERVICE WITHOUT PROPER PRESERVICE EXAMINATIONS BEING PERFORMED OR NOTIFICATION OF THE AUTHORIZED NUCLEAR INSERVICE INSPECTOR (ANII). THIS WAS A VIOLATION OF TECHNICAL SPECIFICATION (TECH SPEC) 3.2.6.A.(1), WHICH REQUIRES THAT FOR COMPONENTS TO BE CONSIDERED OPERABLE, THEY SHALL SATISFY THE REQUIREMENTS CONTAINED IN SECTION XI OF THE ASME BOILER AND PRESSURE VESSEL CODE. THE ROOT CAUSE HAS BEEN DETERMINED TO BE PERSONNEL ERROR BY ENGINEERING, DUE TO LACK OF IDENTIFYING THAT A MODIFICATION TO RESTRAINTS 93-R12-A/B AND 93-R13-A/B WOULD HAVE AN ASME SECTION XI IMPACT. IMMEDIATE CORRECTIVE ACTIONS INCLUDED PERFORMING EXAMINATIONS IN ACCORDANCE WITH ASME SECTION XI AND NOTIFICATION OF THE ANII. A LESSONS LEARNED TRANSMITTAL HAS BEEN ISSUED TO ENGINEERING, SITE TECHNICAL SUPPORT AND QUALITY ASSURANCE FOR REVIEW. ADMINISTRATIVE PROCEDURE FOR MODIFICATIONS (AP-6) IS BEING REVISED AND WILL REQUIRE REVIEW OF THE MODIFICATION WORK REQUEST (MWR) BY THE INSERVICE INSPECTION DEPARTMENT PRIOR TO INSTALLATION.

[170] NINE MILE POINT 1 DOCKET 50-220 LER 87-022
TECHNICAL SPECIFICATION VIOLATION DUE TO FAILURE TO REDUCE ROD BLOCK AND SCRAM SETPOINTS.
EVENT DATE: 111587 REPORT DATE: 121587 NSSS: GE TYPE: BWR

(NSIC 207413) ON NOVEMBER 16, 1987, AT APPROXIMATELY 1530, WITH NINE MILE POINT UNIT 1 OPERATING AT 98% POWER, IT WAS DISCOVERED THAT THE REACTOR HAD BEEN OPERATED FOR APPROXIMATELY 19.4 HOURS IN VIOLATION OF THE PLANT TECHNICAL SPECIFICATIONS (TECH. SPECS.) DURING THE PREVIOUS WEEKEND. THE REVIEW OF THE DAILY FUEL SURVEILLANCE PROCEDURE FOR NOVEMBER 15 REVEALED THAT THE MAXIMUM TOTAL PEAKING FACTOR (MTPF) HAD BEEN IN EXCESS OF THE SPECIFIED LIMIT (3.00) AND THAT THE REQUIRED CORRECTIVE ACTION OF REDUCING THE AVERAGE POWER RANGE MONITOR SCRAM AND ROD BLOCK SETTINGS HAD NOT BEEN PERFORMED. AT THE TIME, THE REACTOR WAS OPERATING AT REDUCED POWER TO CONDUCT A ROD PATTERN ADJUSTMENT AND TO PERFORM PLANT REPAIRS. MTPF RETURNED TO WITHIN THE TECH. SPEC. LIMIT WITHOUT DIRECT OPERATOR ACTION DUE TO CHANGING XENON INVENTORY IN THE CORE AND THE PROCESS OF RAISING REACTOR POWER. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE PERSONNEL ERROR BY THE REACTOR PHYSICS TECHNICIAN WHO PERFORMED THE DAILY FUEL SURVEILLANCE. A CONTRIBUTING CAUSE OF THIS EVENT IS A DEFICIENCY NOTED IN THE SURVEILLANCE PROCEDURE. ALL REACTOR PHYSICS TECHNICIANS WERE BRIEFED ON THIS EVENT AND THE TECH. SPEC. REQUIREMENTS OF THE SURVEILLANCE PROCEDURE WERE REVIEWED. DISCIPLINARY ACTION IS BEING TAKEN AGAINST THE TECHNICIAN.

[171] NINE MILE POINT 1 DOCKET 50-220 LER 87-027
TECHNICAL SPECIFICATION VIOLATION DUE TO ISI PROGRAM DEFICIENCIES.
EVENT DATE: 120987 REPORT DATE: 010888 NSSS: GE TYPE: BWR

(NSIC 207669) ON DECEMBER 9, 1987, WITH NINE MILE POINT 1 (NMPL) AT 0% POWER DURING A FORCED OUTAGE, A SITE OPERATIONS REVIEW COMMITTEE REVIEW OF OCCURRENCE REPORT (OR) 87-272 DETERMINED THAT A VIOLATION OF TECHNICAL SPECIFICATION 3.2.6 OCCURRED DUE TO INSERVICE INSPECTION (ISI) PROGRAM DEFICIENCIES. THE

DEFICIENCIES, AS IDENTIFIED IN OR 87-272, INVOLVE THE FAILURE TO PROPERLY EVALUATE AND CLOSE OUT FIVE ISI DEPARTMENT DEFICIENCY/CORRECTIVE ACTION (DCA) NOTICES IN THE CORE SPRAY AND EMERGENCY COOLING SYSTEMS PRIOR TO RETURNING THESE PLANT SYSTEMS TO OPERABLE STATUS FOLLOWING THE 1986 REFUELING OUTAGE. THE INOPERABILITY OF PLANT COMPONENTS IS A VIOLATION OF SECTION XI OF THE ASME BOILER AND PRESSURE VESSEL CODE. TECHNICAL SPECIFICATION 3.2.6 REQUIRES COMPLIANCE TO SECTION XI OF THE CODE. A HUMAN PERFORMANCE EVALUATION SYSTEM (HPES) REVIEW IS BEING CONDUCTED FOR THIS EVENT AS WELL AS A ROOT CAUSE INVESTIGATION. THE ROOT CAUSE HAS BEEN DETERMINED TO BE MANAGEMENT INEFFECTIVENESS IN IMPLEMENTING THE ISI PROGRAM, SPECIFICALLY IN VERIFYING THE PROPER REQUIREMENTS ARE FULFILLED IN ORDER TO CLOSE OUT PAPERWORK. INITIAL CORRECTIVE ACTIONS WERE THE DISPOSITION OF THE FIVE OPEN DCA'S IDENTIFIED ON O.R. 87-272, AND A REVIEW TO DETERMINE IF OTHER OPEN DCA'S EXIST AND DISPOSITION OF THEM.

[172] NINE MILE POINT 1 DOCKET 50-220 LER 87-025
REACTOR SCRAM DUE TO SPURIOUS TRIP OF NEUTRON MONITOR CAUSED BY NOISE.
EVENT DATE: 121087 REPORT DATE: 010888 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 207668) ON DECEMBER 10, 1987, AT 0956 HOURS NINE MILE POINT UNIT ONE EXPERIENCED A FULL REACTOR SCRAM WHILE IN A COLD SHUTDOWN CONDITION WITH THE MODE SWITCH IN THE "REFUEL" POSITION. THE SCRAM WAS THE ONLY SAFETY SYSTEM ACTUATION WHICH OCCURRED DURING THE EVENT. PRIOR TO THE SCRAM, ALL CONTROL RODS WERE ALREADY FULLY INSERTED TO THE "00" POSITION. THE SCRAM WAS CAUSED BY SPURIOUS ACTUATION OF AN INTERMEDIATE RANGE MONITOR IN BOTH OF THE REACTOR PROTECTION SYSTEM CHANNELS DUE TO NOISE. A PROBLEM REPORT HAS BEEN INITIATED TO DETERMINE THE SOURCE(S) OF THE NOISE DISTURBANCE AND TO PREVENT RECURRENCE OF THE PROBLEM. THIS EVALUATION WILL BE STARTED IN EARLY 1988.

[173] NINE MILE POINT 1 DOCKET 50-220 LER 87-028
MANUAL REACTOR SCRAM INITIATED DUE TO FEEDWATER PIPING VIBRATION.
EVENT DATE: 121987 REPORT DATE: 011888 NSSS: GE TYPE: BWR
VENDOR: FISHER CONTROLS CO.
ITT GRINNELL
WORTHINGTON PUMP CORP.

(NSIC 207874) AT 18:13:56 ON DECEMBER 19, 1987, NINE MILE POINT UNIT 1 (NMP1) WAS MANUALLY SCRAMMED DUE TO EXCESSIVE VIBRATION OBSERVED IN THE HIGH PRESSURE FEEDWATER PIPING. IT IS BELIEVED THAT THE FAILURE OF THE STEM/VALVE PLUG ASSEMBLY ? FLOW CONTROL VALVE 13A CAUSED PRESSURE AND FLOW SURGES IN THE FEEDWATER SYSTEM AND WAS THE SOURCE OF THE PIPE VIBRATION. THE PRESSURE-RETAINING BOUNDARY OF THE FEEDWATER SYSTEM WAS NOT DAMAGED BY THIS TRANSIENT. A DETAILED VISUAL INSPECTION OF THE FEEDWATER SYSTEM PIPING AND SUPPORTS WAS CONDUCTED. NON-DESTRUCTIVE EXAMINATION OF SELECTED PIPE WELDS IN THE FEEDWATER SYSTEM ARE CONTINUING AT THE TIME OF THIS REPORT. THE RESULTS OF THESE INSPECTIONS AND THE ROOT CAUSE ANALYSIS OF THIS EVENT WILL BE PRESENTED IN A SUPPLEMENTAL REPORT.

[174] NINE MILE POINT 2 DOCKET 50-410 LER 87-035
FIRE WATCH INAPPROPRIATELY SUSPENDED WHICH RESULTS IN A TECHNICAL SPECIFICATION VIOLATION DUE TO PERSONNEL ERROR.
EVENT DATE: 061887 REPORT DATE: 071387 NSSS: GE TYPE: BWR

(NSIC 207727) ON JUNE 18, 1987 AT 1150 WITH THE REACTOR IN COLD SHUTDOWN (OPERATIONAL CONDITION 4), A FIRE WATCH WAS INAPPROPRIATELY TERMINATED WHILE THE DETECTION IN TWO FIRE ZONES WAS INOPERABLE. THIS ACTION RESULTED IN A VIOLATION OF THE NINE MILE POINT UNIT 2 (NMP2) TECHNICAL SPECIFICATION SECTIONS 3.3.7.8 AND 3.7.7.2. THE FIRE DETECTION IN THESE FIRE ZONES WAS RESTORED TO AN OPERABLE

STATUS AT 1345 WHICH ENDED THIS EVENT. THE ROOT CAUSE FOR THIS EVENT IS PERSONNEL ERROR. THE CORRECTIVE ACTIONS TAKEN SUBSEQUENT TO THIS EVENT ARE: 1. LOG ENTRY REQUIREMENTS HAVE BEEN REVISED FOR THE FIRE CHIEFS' LOG BOOK. 2. AN IMPROVED STATUS BOARD WILL BE PROCURED AND SHALL BE LOCATED IN THE FIRE DEPARTMENT OFFICE. THE ANTICIPATED IMPLEMENTATION DATE IS SEPTEMBER 15, 1987. 3. A LESSONS LEARNED PROGRAM, BEING PREPARED BY THE FIRE DEPARTMENT SUPERVISION, WILL BE IMPLEMENTED FOR THE NMP2 FIRE DEPARTMENT. THE ANTICIPATED IMPLEMENTATION DATE IS AUGUST 1, 1987. 4. MEETINGS DISCUSSING FIREMAN RESPONSIBILITY AND CONDUCT HAVE BEEN HELD WITH THE FIRE DEPARTMENT PERSONNEL INVOLVED IN THIS EVENT. 5. THIS REPORT WILL BE DISCUSSED IN FIRE DEPARTMENT TRAINING.

[175] NINE MILE POINT 2 DOCKET 50-410 LER 87-040 REV 01
UPDATE ON SECONDARY CONTAINMENT INTEGRITY NOT MAINTAINED DUE TO PLANT CONDITIONS NOT CONSISTENT WITH ASSUMPTIONS FOR THE STANDBY GAS DRAW DOWN CALCULATION.
EVENT DATE: 070387 REPORT DATE: 012888 NSSS: GE TYPE: BWR

(NSIC 208037) ON 7/3/87, IN SUPPORT OF AN EFFORT TO FILE A REQUEST TO INCREASE THE NINE MILE POINT UNIT 2 TECH SPEC (TS) ALLOWABLE SERVICE WATER TEMPERATURE, IT WAS DETERMINED THAT SOME OF THE ASSUMPTIONS USED IN THE CALCULATION FOR THE STANDBY GAS TREATMENT (SBGT) SYSTEM DRAW DOWN TIME FOR SECONDARY CONTAINMENT INTEGRITY WERE NOT CONSISTENT WITH THE CURRENT PLANT CONDITIONS. THIS COULD HAVE RESULTED IN DRAW DOWN TIMES IN EXCESS OF THAT REVIEWED AND APPROVED IN THE SAFETY EVALUATION REPORT, NUREG-1047 SUPPLEMENT 3. IMMEDIATE CORRECTIVE ACTIONS WERE TO REEVALUATE THE CALCULATION AND TO IMPOSE ADMINISTRATIVE LIMITS ON PLANT OPERATION. ON 7/13/87, A POTENTIALLY MORE LIMITING SCENARIO FOR THE SBGT DRAW DOWN TIME WAS IDENTIFIED AND NEW ADMINISTRATIVE LIMITS WERE IMPOSED. CORRECTIVE ACTIONS HAVE BEEN TAKEN TO ADDRESS THE ASSUMPTIONS OF THE CALCULATION, BY ESTABLISHING A MINIMUM TEMPERATURE DIFFERENTIAL BETWEEN REACTOR BUILDING AIR AND SERVICE WATER DISCHARGE HEADER TEMPERATURE. ALSO, BY INSTALLATION OF A MODIFICATION TO AUTOMATICALLY START THE UNIT COOLERS ON A LOSS-OF-COOLANT-ACCIDENT (LOCA) SIGNAL AND TO MONITOR THIS DIFFERENTIAL TEMPERATURE. ANALYSIS IS SCHEDULED TO CONTINUE ON THIS SUBJECT IN AN EFFORT TO ELIMINATE THE NEED FOR MAINTAINING THE DIFFERENTIAL TEMPERATURE REQUIREMENT.

[176] NINE MILE POINT 2 DOCKET 50-410 LER 87-073
REACTOR WATER CLEANUP ISOLATION DUE TO PERSONNEL ERROR AND INATTENTION TO DETAIL.
EVENT DATE: 112487 REPORT DATE: 122187 NSSS: GE TYPE: BWR

(NSIC 207470) ON NOVEMBER 24, 1987 AT 2053 HOURS, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED THE ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF), SPECIFICALLY, ISOLATION OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM. AT THE TIME OF THE EVENT, THE PLANT WAS IN THE COLD SHUTDOWN CONDITION WITH THE REACTOR MODE SWITCH IN THE "SHUTDOWN" POSITION. REACTOR PRESSURE AND COOLANT TEMPERATURE WERE ATMOSPHERIC AND 102F, RESPECTIVELY. THE ROOT CAUSE OF THIS EVENT WAS COGNITIVE PERSONNEL ERROR; INATTENTION TO DETAIL. IMMEDIATE CORRECTIVE ACTIONS TAKEN BY THE OPERATORS WERE TO VERIFY THE AUTOMATIC RESPONSE OF THE RWCU SYSTEM, VERIFY THE PLANT STATUS AS NORMAL, RESET THE ISOLATION SIGNAL AND RESTORE THE RWCU SYSTEM TO SERVICE. ADDITIONAL CORRECTIVE ACTIONS FOR THIS EVENT ARE: 1. THE OPERATOR INVOLVED HAS BEEN COUNSELED. 2. A TRAINING MODIFICATION RECOMMENDATION HAS BEEN INITIATED TO ADDRESS THIS ISSUE DURING OPERATIONS CONTINUING TRAINING 3. THIS EVENT HAS BEEN NOTED IN THE OPERATIONS' DEPARTMENT LESSONS LEARNED PROGRAM.

[177] NINE MILE POINT 2 DOCKET 50-410 LER 87-076
POTENTIAL RADIOACTIVE RELEASE RESULTING FROM A DESIGN DEFICIENCY IN THE FLOOR DRAIN SYSTEM.
EVENT DATE: 121287 REPORT DATE: 011188 NSSS: GE TYPE: BWR

(NSIC 207910) ON DECEMBER 12, 1987 DURING A 10CFR21 (F87-014) REVIEW, IT WAS

DISCOVERED THAT THE DRAIN LINES FROM THE CONTROL ROOM CHARCOAL FILTER TRAIN CUBICLES, THE MAIN STEAM TUNNEL, AND THE AUXILIARY SERVICE BUILDING JOINED TOGETHER IN THE AUXILIARY SERVICE BUILDING AND CONTINUED INTO THE REACTOR BUILDING AND TERMINATED IN THE REACTOR BUILDING FLOOR DRAIN SUMP. SPECIFICALLY, THERE WAS AN AIR COMMUNICATION PATH BETWEEN THE CONTROL ROOM PRESSURE BOUNDARY, THE MAIN STEAM TUNNEL, AND SECONDARY CONTAINMENT THROUGH THE EQUIPMENT AND FLOOR DRAIN PIPING. THIS PARTICULAR DESIGN POSED A THREAT TO PLANT PERSONNEL AND EQUIPMENT SAFETY IN THE EVENT OF A HIGH ENERGY LINE BREAK OR A FEEDWATER LINE BREAK ACCIDENT IN THE MAIN STEAM TUNNEL. AT THE TIME OF DISCOVERY THE REACTOR WAS IN COLD SHUTDOWN WITH THE REACTOR MODE SWITCH IN THE "SHUTDOWN" POSITION. THE CAUSE OF THIS EVENT WAS A DESIGN DEFICIENCY. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED THE INCORPORATION OF NUMEROUS MODIFICATIONS INTO THE NINE MILE POINT UNIT 2 PLANT DESIGN TO ELIMINATE THE POTENTIAL PROBLEM THAT EXISTED DUE TO THE INTERCOMMUNICATING DRAIN LINES.

[178] NINE MILE POINT 2 DOCKET 50-410 LER 87-077
EMERGENCY CORE COOLING SYSTEM ACTUATION AND COOLANT INJECTION DUE TO A PERSONNEL ERROR.
EVENT DATE: 121787 REPORT DATE: 011488 NSSS: GE TYPE: BWR

(NSIC 207911) ON DECEMBER 17, 1987 AT 1114 WITH THE REACTOR IN COLD SHUTDOWN (OPERATIONAL CONDITION 4), NINE MILE POINT UNIT 2 EXPERIENCED AN AUTOMATIC ACTUATION OF THE DIVISION 1 EMERGENCY CORE COOLING SYSTEM (ECCS) WITH A SUBSEQUENT COOLANT INJECTION, AND AN AUTOMATIC STARTUP OF THE DIVISION 1 DIESEL GENERATOR AND ITS ASSOCIATED VENTILATION SYSTEM. THESE ACTIONS OCCURRED AS A RESULT OF REPAIR ACTIVITIES ON A FLOW TRANSMITTER INSTRUMENT LINE. THE ECCS SYSTEMS WERE SECURED AND THE INITIATION SIGNAL WAS RESET BY 1116 THAT SAME DAY. THE ROOT CAUSE FOR THIS EVENT IS PERSONNEL ERROR DUE TO INATTENTION TO DETAIL. THE CORRECTIVE ACTIONS FOR THIS EVENT ARE: 1. A TRAINING MODIFICATION RECOMMENDATION HAS BEEN INITIATED REQUESTING INSTRUMENT AND CONTROL (I&C) TECHNICIAN TRAINING. 2. THE TECHNICIAN INVOLVED HAS BEEN COUNSELED. 3. A SUMMARY OF THE EVENT WILL BE INCLUDED IN THE I&C DEPARTMENT LESSONS LEARNED BOOK. 4. IDENTIFICATION TAGS FOR CERTAIN UNLABELED INSTRUMENT VALVES WILL BE PROVIDED.

[179] NINE MILE POINT 2 DOCKET 50-410 LER 87-074
INOPERABLE FIRE BARRIER DUE TO AN UNSATISFACTORY FLOOR PLUG INSTALLATION DUE TO DESIGN DEFICIENCY.
EVENT DATE: 121987 REPORT DATE: 011888 NSSS: GE TYPE: BWR

(NSIC 207909) ON DECEMBER 19, 1987 AT 1100 WITH THE REACTOR IN COLD SHUTDOWN (OPERATIONAL CONDITION 4), AN UNSATISFACTORY FLOOR PLUG INSTALLATION WAS DISCOVERED IN A FIRE RATED FLOOR IN THE DIVISION 2 VENTILATION ROOM LOCATED ON CONTROL BUILDING ELEVATION 306. THE UNSATISFACTORY INSTALLATION, DISCOVERED BY THE NINE MILE POINT UNIT 2 (NMP2) FIRE DEPARTMENT, CONSTITUTES AN APPENDIX R VIOLATION. AS A RESULT, THE FIRE RATED FLOOR WAS DECLARED INOPERABLE AS DEFINED BY THE NMP2 FINAL SAFETY ANALYSIS REPORT SECTION 9A.3.5.1.1. ON JANUARY 14, 1988 FOUR OTHER UNSATISFACTORY FLOOR PLUG INSTALLATIONS WERE IDENTIFIED IN FIRE RATED BARRIERS. A DETAILED DISCUSSION ON THESE INSTALLATIONS SHALL BE SUBMITTED IN A SUPPLEMENT TO THIS REPORT. THE ROOT CAUSE FOR THIS EVENT IS A DESIGN DEFICIENCY. HOWEVER, PERSONNEL ERROR WAS A CONTRIBUTING FACTOR. THE CORRECTIVE ACTIONS FOR THIS EVENT ARE: 1. FIRE WATCH PATROLS WERE ESTABLISHED IN THE AFFECTED FIRE AREAS. 2. OTHER FLOOR PLUG INSTALLATIONS WILL BE EVALUATED. 3. NIAGARA MOHAWK ENGINEERING IS EVALUATING SUITABLE SEALING METHODS FOR THESE FLOOR PLUG INSTALLATIONS.

[180] NINE MILE POINT 2 DOCKET 50-410 LER 87-078
SECONDARY CONTAINMENT ISOLATION AND STANDBY GAS TREATMENT INITIATION DUE TO DESIGN DEFICIENCIES.
EVENT DATE: 122087 REPORT DATE: 011988 NSSS: GE TYPE: BWR

(NSIC 207912) WHILE RESTORING THE NORMAL REACTOR BUILDING VENTILATION (HVR) SYSTEM ON DECEMBER 20, 1987 AT 1335 HOURS AND ON DECEMBER 28, 1987 AT 1030 HOURS, A HVR SUPPLY FAN DID NOT START AND RESULTED IN SEVERAL ENGINEERED SAFETY FEATURE (ESF) ACTUATIONS. FOR THE FIRST EVENT, A NIAGARA MOHAWK LICENSED OPERATOR MISPOSITIONED A CONTROL SWITCH WHILE ATTEMPTING TO SIMULTANEOUSLY START ONE EXHAUST AND ONE SUPPLY FAN. SINCE THE SUPPLY FAN DID NOT START, THE OPERATOR MANUALLY TRIPPED THE RUNNING EXHAUST FAN. THIS GENERATED A LOW FLOW SIGNAL AND SUBSEQUENT SECONDARY CONTAINMENT ISOLATION AND THE AUTO START OF THE EMERGENCY HVR SYSTEM. FOR THE SECOND EVENT, AN OPERATOR ATTEMPTED TO SIMULTANEOUSLY START ONE EXHAUST AND ONE SUPPLY FAN PER THE APPROVED PROCEDURE. THE SUPPLY FAN NEVER STARTED AND RESULTED IN AN EXHAUST FAN TRIP AND SUBSEQUENT SECONDARY CONTAINMENT ISOLATION, STANDBY GAS TREATMENT (GTS) AND EMERGENCY HVR SYSTEM INITIATIONS. THE ROOT CAUSE FOR THESE EVENTS IS DESIGN DEFICIENCY. IMMEDIATE CORRECTIVE ACTIONS WERE TO RESET THE ISOLATIONS, TO SECURE THE SAFETY SYSTEMS AND TO RESTORE NORMAL HVR. FURTHER CORRECTIVE ACTIONS INCLUDE DISCHARGE DAMPER STROKE TIME CHANGES, THE ADDITION OF A LOW FLOW OVERRIDE SWITCH, AND PROCEDURE REVISIONS TO INCORPORATE THE DESIGN CHANGES.

[181] NINE MILE POINT 2 DOCKET 50-410 LER 87-079
EMERGENCY SAFETY FEATURE ACTUATION CAUSED BY A DROPPED LEAD DUE TO DESIGN DEFICIENCY.
EVENT DATE: 122287 REPORT DATE: 012188 NSSS: GE TYPE: BWR

(NSIC 207913) ON DECEMBER 22, 1987 AT 1048 HOURS WITH THE REACTOR AT APPROXIMATELY 1% POWER AND THE MODE SWITCH IN THE "STARTUP" POSITION, NINE MILE POINT UNIT 2 EXPERIENCED AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION. AT THE TIME OF THE EVENT REACTOR TEMPERATURE AND PRESSURE WERE APPROXIMATELY 326F AND 85 POUNDS PER SQUARE INCH GAUGE, RESPECTIVELY. THIS EVENT CONSISTED OF A SECONDARY CONTAINMENT ISOLATION AND THE INITIATION OF THE STANDBY GAS TREATMENT (GTS) SYSTEM, REACTOR BUILDING (HVR) UNIT COOLERS, AND EMERGENCY RECIRCULATION SYSTEM. PER PROCEDURE, AN INSTRUMENT AND CONTROL (I&C) TECHNICIAN WAS PLACING A JUMPER (MAINTAINING POSITIVE CONTROL WITH BOTH HANDS) WHEN THE OPEN END OF THE LEAD SLIPPED FROM HIS HAND AND SHORTED TO GROUND. THIS SHORT TO GROUND CAUSED A REACTOR BUILDING VENTILATION SUPPLY DAMPER 2HVR*AODLA TO CLOSE. THE CLOSED DAMPER PROHIBITED FLOW INITIATING THE SECONDARY CONTAINMENT ISOLATION AND ACTUATING THE HVR UNIT COOLERS, TRAIN B OF BOTH GTS, AND THE EMERGENCY RECIRCULATION SYSTEM. THE ROOT CAUSE WAS DIFFICULTY IN ESTABLISHING JUMPER CONNECTIONS AT THE PANEL. CORRECTIVE ACTIONS INCLUDE RESTORE NORMAL REACTOR BUILDING VENTILATION. 2. A MODIFICATION REGUEST (I20205) HAS BEEN INITIATED TO INSTALL HARD WIRE TEST SWITCHES, WHICH WILL ELIMINATE THE USE OF JUMPERS IN THIS SURVEILLANCE TEST.

[182] NINE MILE POINT 2 DOCKET 50-410 LER 87-081
REACTOR SCRAM DUE TO LOSS OF CONDENSER VACUUM CAUSED BY EQUIPMENT FAILURE.
EVENT DATE: 122587 REPORT DATE: 012588 NSSS: GE TYPE: BWR
VENDOR: MASONEILAN INTERNATIONAL, INC.

(NSIC 207976) ON DECEMBER 26, 1987 AT 1218 HOURS, NINE MILE POINT UNIT 2 EXPERIENCED A REACTOR SCRAM FROM APPROXIMATELY 26% OF RATED THERMAL POWER AS A RESULT OF A TURBINE TRIP CAUSED BY LOW CONDENSER VACUUM. AT APPROXIMATELY 0800 HOURS CONTROL ROOM PERSONNEL NOTED THAT CONDENSER VACUUM WAS DECREASING AND THAT OFF GAS SYSTEM FLOW RATES WERE INCREASING. ACTIONS TAKEN BY OPERATORS TO IDENTIFY AND STOP THE SOURCE OF INLEAKAGE TO THE CONDENSER WERE UNSUCCESSFUL AND THE REACTOR SUBSEQUENTLY SCRAMMED AT 1218 HOURS ON THE TURBINE TRIP. THE ROOT

CAUSE OF THIS EVENT WAS MALFUNCTION OF THE NORMAL (2HDL-FV5A) AND HIGH (2HDL-FV25A) LEVEL CONTROL VALVES OF THE FIFTH POINT FEEDWATER HEATER (2HDL-E5A). WATER LEVEL IN 2HDL-E5A WAS FLUCTUATING FROM HIGH-HIGH TO LOW LEVEL WHICH RESULTED IN A LARGE VOLUME OF WATER BEING PERIODICALLY DISCHARGED TO THE CONDENSER. THESE HYDRAULIC TRANSIENTS CAUSED THE FAILURE OF CONDENSER NOZZLE WELDS WHICH ALLOWED AIR INLEAKAGE TO THE CONDENSER. IMMEDIATE CORRECTIVE ACTIONS WERE TO RESET THE SCRAM AT 1225 HOURS, TO FOLLOW THE SCRAM RECOVERY PROCEDURE FOR SAFE SHUTDOWN OF THE PLANT, AND TO CONTINUE TO INVESTIGATE FOR THE CAUSE OF THE LOSS OF VACUUM. WORK REQUESTS HAVE BEEN COMPLETED TO REPAIR THE CRACKED WELDS AND THE LEVEL CONTROL VALVES.

[183] NINE MILE POINT 2 DOCKET 50-410 LER 87-082
MAIN STEAM LINE ISOLATION AND SUBSEQUENT REACTOR SCRAM DUE TO DESIGN DEFICIENCY REACTOR MODE SWITCH.
EVENT DATE: 122987 REPORT DATE: 012888 NSSS: GE TYPE: BWR

(NSIC 208038) ON 12/29/87 AT 2228 HOURS, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED THE ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF), SPECIFICALLY, ISOLATION OF THE MAIN STEAM LINES. CLOSURE OF THE MAIN STEAM ISOLATION VALVES (MSIV) RESULTED IN A SUBSEQUENT REACTOR SCRAM. AT THE TIME OF THE EVENT, A PLANT START-UP WAS BEING INITIATED. REACTOR PRESSURE AND COOLANT TEMPERATURE WERE APPROXIMATELY ATMOSPHERIC AND 125F, RESPECTIVELY. THE ROOT CAUSE OF THIS EVENT IS THE DEFICIENT DESIGN OF THE REACTOR MODE SWITCH. A LARGE AMOUNT OF TORQUE IS NEEDED TO OPERATE THE SWITCH. AWARE OF THIS TORQUE REQUIREMENT, AS THE OPERATOR ATTEMPTED TO PLACE THE SWITCH IN THE "STARTUP/HOT STANDBY" POSITION HE APPLIED MORE FORCE THAN WAS NECESSARY AND THE SWITCH OVERTRAVELLED TO THE "RUN" POSITION. PLACING THE REACTOR MODE SWITCH IN THE "RUN" POSITION AT LOW REACTOR PRESSURE INITIATED THE MSIV ISOLATION AND SUBSEQUENT REACTOR SCRAM. IMMEDIATE CORRECTIVE ACTIONS TAKEN BY THE OPERATORS WERE TO RETURN THE REACTOR MODE SWITCH TO THE "SHUTDOWN" POSITION, RESET THE SCRAM AND VERIFY THAT ALL SIV'S WERE CLOSED. ADDITIONAL CORRECTIVE ACTIONS INCLUDE: 1. A PROBLEM REPORT HAS BEEN ISSUED TO EVALUATE POSSIBLE DESIGN IMPROVEMENTS. 2. THIS EVENT WILL BE INCORPORATED IN THE OPERATION'S DEPARTMENT LESSONS LEARNED PROGRAM.

[184] NINE MILE POINT 2 DOCKET 50-410 LER 87-083
TECH SPEC VIOLATION DUE TO PERSONNEL ERROR AND FAILURE TO FOLLOW PROCEDURE RESULTS IN MISSED SURVEILLANCE.
EVENT DATE: 123087 REPORT DATE: 012888 NSSS: GE TYPE: BWR

(NSIC 208039) 12/30/87 AT 1145 HOURS, IT WAS DISCOVERED THAT NINE MILE POINT UNIT 2 NMP2) HAD VIOLATED A TECH SPEC SURVEILLANCE REQUIREMENT (SR) FOR PRESSURE/TEMPERATURE LIMITS FOR THE REACTOR COOLANT SYSTEM. AT THE TIME OF THE DISCOVERY, THE PLANT WAS OPERATING AT 0.9% OF RATED THERMAL POWER WITH THE REACTOR MODE SWITCH IN THE "STARTUP/HOT STANDBY" POSITION. REACTOR PRESSURE AND COOLANT TEMPERATURE WERE APPROXIMATELY 201 POUNDS PER SQUARE INCH GAUGE (PSIG) AND 95F, RESPECTIVELY. THE ROOT CAUSE OF THIS EVENT WAS COGNITIVE PERSONNEL ERROR; FAILURE TO FOLLOW PROCEDURE. A CONTRIBUTING CAUSE FOR THIS EVENT WAS A PROCEDURAL DEFICIENCY IN THE SURVEILLANCE PROCEDURE BEING PERFORMED. INITIAL CORRECTIVE ACTIONS WERE FOR THE OPERATORS TO RESUME THE SURVEILLANCE AND VERIFY THAT DURING THE MISSED SURVEILLANCE TIME THAT THE TEMPERATURE AND PRESSURE VALUES REMAINED TO THE RIGHT OF THE CRITICALITY LIMIT LINE AS SPECIFIED IN THE TECHNICAL SPECIFICATIONS. ADDITIONAL CORRECTIVE ACTIONS INCLUDE: 1. OPERATIONS SURVEILLANCE PROCEDURE N2-OSP-RCS-@001, "RCS PRESSURE/TEMPERATURE VERIFICATION", WILL BE REVISED TO MAKE THE ORDER OF THE STEPS MORE LOGICAL. THIS REVISION WILL BE COMPLETED PRIOR TO THE START-UP. 2. THE OPERATOR INVOLVED IN THIS EVENT HAS BEEN COUNSELED. 3. THIS EVENT WILL BE INCORPORATED IN THE OPERATION'S DEPARTMENT LESSONS LEARNED PROGRAM.

[185] NINE MILE POINT 2 DOCKET 50-410 LER 87-084
TECH SPEC VIOLATION DUE TO FAILURE OF AUXILIARY GASEOUS EFFLUENT MONITORING EQUIPMENT AND DESIGN DEFICIENCY.
EVENT DATE: 123087 REPORT DATE: 012888 NSSS: GE TYPE: BWR

(NSIC 208040) ON DECEMBER 30, 1987 AT 1700 WITH THE REACTOR IN STARTUP (OPERATIONAL CONDITION 2), AT A TEMPERATURE AND PRESSURE OF APPROXIMATELY 370 DEGREES FAHRENHEIT AND 175 POUNDS PER SQUARE INCH GAUGE RESPECTIVELY, AND AT A POWER LEVEL OF APPROXIMATELY 1 PERCENT RATED THERMAL CAPACITY, THE AUXILIARY GASEOUS EFFLUENT MONITORING SAMPLING EQUIPMENT FOR THE RADWASTE/REACTOR BUILDING VENT EFFLUENT SYSTEM (2RMS-CAB180) WAS DISCOVERED TO BE INOPERABLE. AS A RESULT, NINE MILE POINT UNIT 2 WAS NOT IN COMPLIANCE WITH TECH SPEC 3.3.7.10 SINCE THIS EQUIPMENT WAS NOT OPERABLE AS REQUIRED BY THIS SECTION. THE AUXILIARY SAMPLING EQUIPMENT WAS RESTORED TO AN OPERABLE STATUS BY 2200 THAT SAME DAY. THE PROBABLE ROOT CAUSE FOR THIS EVENT IS A DESIGN DEFICIENCY. THE CORRECTIVE ACTIONS FOR THIS EVENT ARE: 1. & 2. THE AUXILIARY SAMPLING EQUIPMENT WAS RESTORED TO AN OPERABLE STATUS. 3. THE SAMPLE RETURN LINE FOR 2RMS-CAB180 WILL BE REROUTED.

[186] NORTH ANNA 1 DOCKET 50-338 LER 87-023 REV 01
UPDATE ON KAMAN PROCESS VENT NORMAL RANGE RADIATION MONITOR EXCEEDED TECH SPEC ACTION STATEMENT.
EVENT DATE: 112487 REPORT DATE: 011288 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)
VENDOR: KAMAN SCIENCES CORP.

(NSIC 207897) AT 1110 HOURS ON NOVEMBER 24, 1987, WITH UNIT 1 AT 0 PERCENT POWER (MODE 2) AND UNIT 2 AT 100 PERCENT POWER (MODE 1), THE KAMAN PROCESS VENT NORMAL RANGE RADIATION MONITOR, RI-GW-178-1, WAS DECLARED INOPERABLE BECAUSE THE CENTRAL PROCESSING UNIT (CPU) FAILED TO OPERATE. ACTION STATEMENT 35 OF TECHNICAL SPECIFICATION 3.3.3.1 REQUIRES THAT THE RADIATION MONITOR BE RESTORED TO OPERABLE STATUS WITHIN 72 HOURS OR INITIATE THE ALTERNATE METHOD OF MONITORING AND PREPARE A SPECIAL REPORT. SINCE THIS ACTION STATEMENT EXPIRED AT 1110 HOURS ON NOVEMBER 27, 1987 AND THE RADIATION MONITOR WAS STILL INOPERABLE, THIS EVENT IS REPORTABLE PURSUANT TO TECHNICAL SPECIFICATION 6.9.2. INVESTIGATION OF THE RADIATION MONITOR FAILURE DETERMINED THAT A CPU MEMORY LOSS WAS POSSIBLY CAUSED BY A LOOSE CONNECTION BETWEEN THE CPU CARD AND THE SYSTEM. ON NOVEMBER 26, 1987 AND NOVEMBER 30, 1987, THE CPU CARD WAS REPLACED AND THE CARD EDGE CONNECTOR SPRING TENSION WAS INCREASED. THE RADIATION MONITOR HAS NOT MALFUNCTIONED SINCE THESE ACTIONS WERE PERFORMED AND WAS RETURNED TO OPERABLE STATUS ON DECEMBER 9, 1987. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE THERE ARE BACKUP RADIATION MONITORS FOR THE RELEASE PATH WHICH REMAINED OPERABLE THROUGHOUT THIS EVENT. THE HEALTH AND SAFETY OF THE GENERAL PUBLIC WERE NOT AFFECTED.

[187] NORTH ANNA 1 DOCKET 50-338 LER 88-001
VENT STACK "A" NORMAL RANGE RADIATION MONITOR EXCEEDED TECH SPEC ACTION STATEMENT.
EVENT DATE: 010388 REPORT DATE: 011488 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)
VENDOR: KAMAN SCIENCES CORP.

(NSIC 207922) AT 0545 HOURS ON 1/3/88, WITH UNITS 1 AND 2 AT 100% 1 POWER (MODE 1), THE KAMAN VENT STACK "A" NORMAL RANGE RADIATION MONITOR, 1 RI-VG-179-1, WAS DECLARED INOPERABLE DUE TO HIGH INDICATED RADIATION LEVELS. TWO ADDITIONAL RADIATION MONITORS IN THE SAME EFFLUENT RELEASE PATH WERE INDICATING NORMAL RADIATION LEVELS DURING THIS PERIOD. ACTION STATEMENT 35 OF TECH SPEC 3.3.3.1 REQUIRES THAT THE RADIATION MONITOR BE RETURNED TO OPERABLE STATUS WITHIN 72 HOURS, OR INITIATE THE PREPLANNED ALTERNATE METHOD OF MONITORING AND PREPARE A SPECIAL REPORT. SINCE THIS ACTION STATEMENT EXPIRED AT 0545 HOURS ON 1/6/88 WITH THE RADIATION MONITOR STILL INOPERABLE, THIS EVENT IS REPORTABLE PURSUANT TO TECH SPEC 6.9.2. INVESTIGATION INTO THE CAUSE FOR THE ERRONEOUS INDICATION OF

RI-VG-179-1 REVEALED THAT THE SCINTILLATOR TUBE HAD SHORTED TO THE DETECTOR HOUSING AND CREATED A GROUND LOOP. THIS GROUND LOOP CAUSED ELECTRONIC NOISE TO BE PICKED UP AND ADDED TO THE BACKGROUND RADIATION LEVELS. ON 1/6/88 THE SCINTILLATOR TUBE WAS REPLACED. ON 1/7/88 THE SCINTILATOR TUBE WAS ELECTRICALLY INSULATED FROM THE HOUSING AND THE RADIATION MONITOR INDICATED PROPERLY. RI-VG-179-1 WAS RETURNED TO OPERABLE STATUS AT 0916 HOURS ON 1/7/88. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE THERE WERE BACKUP RADIATION MONITORS FOR THE RELEASE PATH.

[188] NORTH ANNA 1 DOCKET 50-338 LER 88-003
NUCLEAR RESEARCH CORPORATION RADIATION MONITOR EXCEEDED TECHNICAL SPECIFICATION ACTION STATEMENT.
EVENT DATE: 010788 REPORT DATE: 012188 NSSS: WE TYPE: PWR
VENDOR: NUCLEAR RESEARCH CORP.

(NSIC 207974) AT 2000 HOURS ON JANUARY 7, 1988, WITH UNIT 1 AT 100 PERCENT POWER (MODE 1) AND UNIT 2 AT 100 PERCENT POWER (MODE 1), THE NUCLEAR RESEARCH CORPORATION "1A'' MAIN STEAM LINE RADIATION MONITOR WAS DECLARED INOPERABLE BECAUSE THE MONITOR WAS SPIKING HIGH. ACTION STATEMENT 35 OF TECHNICAL SPECIFICATION 3.3.3.1 REQUIRES THE RADIATION MONITOR BE RETURNED TO OPERABLE STATUS WITHIN 72 HOURS OR INITIATE THE PREPLANNED ALTERNATE METHOD OF MONITORING AND PREPARE A SPECIAL REPORT. SINCE THIS ACTION STATEMENT EXPIRED AT 2000 HOURS ON JANUARY 10, 1988, AND THE RADIATION MONITOR WAS STILL INOPERABLE, THIS EVENT IS REPORTABLE PURSUANT TO TECHNICAL SPECIFICATION 6.9.2. INVESTIGATION INTO THE CAUSE FOR THE INOPERABILITY OF THE RADIATION MONITOR HAS REVEALED THAT THERE WAS A PROBLEM WITH THE ION CHAMBER, AS WELL AS, THE LOWER RANGE GEIGER-MULLER TUBE. THE RADIATION MONITOR HAS NOT MALFUNCTIONED SINCE THESE PROBLEMS WERE CORRECTED AND WAS RETURNED TO OPERABLE STATUS AT 1230 HOURS ON JANUARY 14, 1988. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE AN ALTERNATE METHOD OF CALCULATING AN INCREASE IN RADIOACTIVITY WAS AVAILABLE THROUGHOUT THIS EVENT. THE HEALTH AND SAFETY OF THE GENERAL PUBLIC WERE NOT AFFECTED.

[189] NORTH ANNA 1 DOCKET 50-338 LER 88-006
VENT STACK "A" RADIATION MONITOR EXCEEDED TECH SPEC ACTION STATEMENT.
EVENT DATE: 011288 REPORT DATE: 012888 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)
VENDOR: KAMAN SCIENCES CORP.

(NSIC 207998) AT 1500 HOURS ON 1/12/88, WITH UNITS 1 AND 2 IN MODE 1, AT 5% AND 100% POWER, RESPECTIVELY, THE KAMAN VENT STACK "A" NORMAL RANGE RADIATION MONITOR, RI-VG-179-1, WAS DECLARED INOPERABLE WHEN AN INCREASE IN RADIATION LEVEL INDICATION ACTUATED AN ALERT ALARM AND COULD NOT BE RESET. TWO ADDITIONAL RADIATION MONITORS IN THE SAME EFFLUENT RELEASE PATH WERE INDICATING NORMAL RADIATION LEVELS DURING THIS PERIOD. ACTION STATEMENT 35 OF TECH SPEC (T.S.) 3.3.3.1 REQUIRES THAT THE RADIATION MONITOR BE RETURNED TO OPERABLE STATUS WITHIN 72 HOURS, OR INITIATE THE PREPLANNED ALTERNATE METHOD OF MONITORING AND PREPARE A SPECIAL REPORT. SINCE THIS ACTION STATEMENT EXPIRED AT 1500 HOURS ON 1/15/88, WITH THE RADIATION MONITOR STILL INOPERABLE, THIS EVENT IS REPORTABLE PURSUANT TO TECH SPEC 6.9.2. INVESTIGATION INTO THE CAUSE FOR THE ALERT ALARM REVEALED THAT THE FLOW INVERTER BOARD HAD MALFUNCTIONED AND CAUSED ELECTRONIC NOISE TO BE TRANSMITTED THROUGH THE RADIATION DETECTOR TO THE MICROPROCESSOR. THE MICROPROCESSOR READ THE NOISE AS RADIATION SIGNALS AND CAUSED ERRONEOUSLY HIGH RADIATION SIGNALS TO BE GENERATED. THESE ERRONEOUS RADIATION SIGNALS WERE HIGHER THAN WHAT THE DETECTOR WAS ACTUALLY MEASURING AND CAUSED THE LOCAL AND REMOTE ALERT ALARMS TO ACTUATE.

[190] NORTH ANNA 2 DOCKET 50-339 LER 87-009 REV 01
UPDATE ON MAIN STEAM SAFETY VALVES SET PRESSURES NOT WITHIN TECHNICAL SPECIFICATION LIMITS.
EVENT DATE: 091587 REPORT DATE: 012188 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE & GAGE CO.

(NSIC 207992) AT 1015 HOURS ON 9/15/87, WITH UNIT 2 IN MODE 6 (REFUELING) 11 OF THE 15 MAIN STEAM LINE CODE SAFETY VALVES (MSSVS) "AS FOUND" SET PRESSURES WERE FOUND TO BE HIGHER THAN THE MAXIMUM LIFT SET PRESSURES ALLOWED BY TECH SPEC 3.7.1.1. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B). ON 9/9/87, ALL 15 MSSVS WERE SENT TO WYLE LABS TO DETERMINE THE "AS FOUND" SET PRESSURE, THE AMOUNT OF DISC TO SEAT LEAKAGE UNDER LIMITED FLOW CONDITIONS AND THE SET PRESSURE, BLOWDOWN AND THE AMOUNT OF DISC TO SEAT LEAKAGE UNDER FULL-FLOW CONDITIONS. ELEVEN OF THE MSSVS "AS FOUND" SET PRESSURES EXCEEDED THE TECH SPEC LIMITS AND FOURTEEN EXHIBITED DISC TO SEAT LEAKAGE FOLLOWING THE "AS FOUND" TESTING. NONE OF THE VALVES LEAKED DURING THE FINAL LEAK TEST FOLLOWING INSPECTION, CLEANING, AND REFURBISHMENT. THE IMPACT OF THE HIGH "AS FOUND" MSSVS SET PRESSURES HAS BEEN REVIEWED. THE EXPECTED PEAK MAIN STEAM PRESSURE RESULTING FROM THE "AS FOUND" PRESSURIZER AND MAIN STEAM SAFETY VALVES SET PRESSURE WAS FOUND TO BE LESS THAN THE MAIN STEAM DESIGN BASIS PRESSURE.

[191] NORTH ANNA 2 DOCKET 50-339 LER 87-015 REV 01
UPDATE ON INOPERABLE REDUNDANT S/G STEAM FLOW CHANNELS EXCEED TECHNICAL SPECIFICATION ACTION STATEMENT.
EVENT DATE: 110487 REPORT DATE: 020388 NSSS: WE TYPE: PWR
VENDOR: RAYCHEM CORP.

(NSIC 208029) ON 11/4/87, AT 2153 HOURS, WITH UNIT 2 IN MODE 1 AT 27% REACTOR POWER FOLLOWING A REFUELING OUTAGE, THE "A" S/G STEAM FLOW CHANNEL III AND "B" S/G STEAM FLOW CHANNEL IV WERE DECLARED INOPERABLE. THE CHANNELS HAD NOT BEEN DECLARED INOPERABLE WITHIN ONE HOUR OF THE FIRST INDICATIONS OF POTENTIAL INOPERABILITY. AT 1200 HOURS ON 11/4/87, WITH UNIT 2 IN MODE 2 AT 5%, THE SURVEILLANCE CHANNEL CHECK RECORDED THE SAME "A" AND "B" S/G STEAM FLOW CHANNELS TO BE OUTSIDE OF THE ACCEPTABLE TOLERANCE LIMITS. AT 1816, WITH UNIT 2 IN MODE 1 AT 24%, IT WAS NOTED THAT THE SAME "A" AND "B" S/G STEAM FLOW CHANNELS WERE STILL READING ZERO. THE CHANNELS HAD NOT BEEN DECLARED INOPERABLE AT 1200 HOURS OR AT 1816 HOURS BECAUSE THE OPERATORS BELIEVED THAT THE STEAM FLOW CHANNEL ACCURACY WAS UNRELIABLE AT LOW POWER DURING STARTUP FOLLOWING A UNIT OUTAGE. TECH SPEC 3.3.1.1 AND 3.3.2.1 REQUIRE THAT DURING OPERATION IN MODES 1 THROUGH 3, A STEAM FLOW CHANNEL BE PLACED IN THE TRIPPED CONDITION WITHIN ONE HOUR OF DETERMINING IT INOPERABLE. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73 (A)(2)(I)(B). NO SAFETY CONSEQUENCES RESULTED FROM THE INOPERABILITY OF THE "A" S/G STEAM FLOW CHANNEL III OR "B" S/G STEAM FLOW CHANNEL IV.

[192] NORTH ANNA 2 DOCKET 50-339 LER 87-018
INADVERTENT EMERGENCY DIESEL GENERATOR START DURING TESTING.
EVENT DATE: 122187 REPORT DATE: 011488 NSSS: WE TYPE: PWR

(NSIC 207898) AT 0854 HOURS ON DECEMBER 21, 1987, WITH UNIT 2 AT 100 PERCENT POWER (MODE 1), THE "2J" EMERGENCY DIESEL GENERATOR (EDG) WAS INADVERTENTLY STARTED DURING THE PERFORMANCE OF AN UNDERVOLTAGE PERIODIC TEST ON THE "2H" EMERGENCY BUS. SINCE THE EDG IS PART OF THE ENGINEERED SAFETY FEATURES SYSTEM AND STARTING THE EDG WAS NOT A PREPLANNED SEQUENCE DURING TESTING, THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV). A FOUR HOUR REPORT WAS MADE IN ACCORDANCE WITH 10CFR50.72(B)(2)(II). THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR AND FAILURE TO FOLLOW PROCEDURE. AS A CORRECTIVE ACTION, THE PERFORMANCE OF THE PERIODIC TEST WAS TERMINATED AND THE "2J" EDG WAS SECURED AND RETURNED TO NORMAL STANDBY STATUS. A HUMAN PERFORMANCE EVALUATION IS BEING PERFORMED WITH REGARD TO THIS EVENT. NO SIGNIFICANT SAFETY CONSEQUENCES RESULTED FROM THIS

EVENT BECAUSE THE "2J" EDG STARTED AS DESIGNED WHEN AN UNDERVOLTAGE CONDITION WAS SIMULATED. SINCE AN ACTUAL UNDERVOLTAGE CONDITION DID NOT EXIST, THE EDG DID NOT LOAD BECAUSE CLOSURE OF THE EDG OUTPUT BREAKER IS CONTROLLED BY DIFFERENT CIRCUIT AND LOGIC SCHEMES. IF AN ACTUAL UNDERVOLTAGE CONDITION HAD OCCURRED WHILE IN THIS CONDITION, THE EDG WOULD HAVE BEEN CAPABLE OF LOADING. THE HEALTH AND SAFETY OF THE GENERAL PUBLIC WERE NOT AFFECTED.

[193] OCONEE 1 DOCKET 50-269 LER 87-005 REV 01
UPDATE ON POTENTIAL TRIPPING OF HIGH PRESSURE INJECTION PUMPS DURING STARTING.
EVENT DATE: 062487 REPORT DATE: 020188 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 208017) ON JUNE 24, 1987 AN ANALYSIS WAS CONFIRMED AND REPORTED TO STATION PERSONNEL WHICH IDENTIFIED A SITUATION IN WHICH THE HIGH PRESSURE INJECTION (HPI) PUMP MOTOR ON EACH UNIT COULD TRIP ON OVERCURRENT DURING A LOSS OF COOLANT ACCIDENT (LOCA)/LOSS OF OFFSITE POWER (LOOP) SCENARIO. THE SITUATION WAS IDENTIFIED AS A PART OF A DUKE INITIATED AUXILIARY POWER SYSTEM REVIEW FOLLOWING THE OCONEE SAFETY SYSTEM FUNCTIONAL INSPECTION (SSFI) CONDUCTED BY THE NRC. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE A DESIGN DEFICIENCY WHICH RESULTED IN AN INSUFFICIENT OVERCURRENT RELAY SETTING FOR THE HPI PUMP MOTORS. WHILE THE PROTECTIVE RELAYS WERE SET TO PROTECT THE EQUIPMENT (I.E., HPI PUMP MOTOR), IT COULD NOT BE DETERMINED FROM THE AVAILABLE DOCUMENTATION THAT THE RELAY SETTINGS WERE REVIEWED IN CONJUNCTION WITH THE 4160VAC AUXILIARY POWER SYSTEM MODIFICATIONS FOR POTENTIAL CHANGES IN SYSTEM DYNAMICS. THE IMMEDIATE CORRECTIVE ACTIONS WERE 1) TO INFORM THE PLANT OPERATORS OF THE POTENTIAL PROBLEM, 2) TO CONFIRM WITH THE OPERATORS THE STEPS NECESSARY TO RESTART THE HPI PUMP MOTORS SHOULD THEY TRIP IN A LOOP EVENT, AND 3) TO INITIATE RESETTING OF THE HPI PUMP MOTOR OVERCURRENT RELAYS TO CORRECT THE SITUATION. AN HPI PUMP TRIP FOLLOWING COINCIDENT LOSS OF OFFSITE POWER AND ENGINEERED SAFEGUARDS ACTUATION POTENTIALLY IMPACTS FSAR SMALL BREAK LOCA ANALYSIS.

[194] OCONEE 1 DOCKET 50-269 LER 87-008 REV 01
UPDATE ON A DIRECT PATHWAY FROM CONTAINMENT TO THE ENVIRONMENT DURING FUEL MOVEMENT RESULTS IN A VIOLATION OF TECHNICAL SPECIFICATIONS DUE TO MANAGEMENT DEFICIENCY AND DESIGN DEFICIENCY.
EVENT DATE: 101287 REPORT DATE: 020388 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 208018) ON OCTOBER 12, 1987, WITH UNIT 1 IN REFUELING SHUTDOWN, IT WAS IDENTIFIED THAT TECHNICAL SPECIFICATION 3.8.6 WAS VIOLATED WHEN FUEL WAS MOVED IN THE REACTOR BUILDING CONCURRENT WITH THE EXISTENCE OF A DIRECT PATHWAY FROM INSIDE CONTAINMENT TO THE OUTSIDE ENVIRONMENT. THE PATHWAY EXISTED THROUGH THE EMERGENCY HATCH AND WAS ASSOCIATED WITH A TEMPORARY MODIFICATION. THE ROOT CAUSES OF THIS INCIDENT WERE DETERMINED TO BE A MANAGEMENT DEFICIENCY IN THE PLANNING PROCESS OF THE TASK AND AN UNRELATED DESIGN DEFICIENCY. SUBSEQUENTLY, SIMILAR DESIGN DEFICIENCIES WERE DETERMINED TO HAVE CAUSED PREVIOUS INCIDENTS. THE IMMEDIATE CORRECTIVE ACTION WAS TO STOP FUEL MOVEMENT. SUBSEQUENT CORRECTIVE ACTIONS WERE TO REPAIR THE DIRECT PATHWAY. PLANNED CORRECTIVE ACTIONS INCLUDE REVIEW OF THIS INCIDENT BY APPROPRIATE PERSONNEL. HAD A FUEL HANDLING ACCIDENT OCCURRED DURING THIS OR PREVIOUS SIMILAR INCIDENTS, THE CONSEQUENCES WOULD BE BOUNDED BY ACCIDENT ANALYSES INCLUDED IN THE OCONEE FINAL SAFETY ANALYSIS REPORT (FSAR).

[195] OCONEE 1 DOCKET 50-269 LER 87-013
TECHNICAL SPECIFICATION VIOLATION DUE TO LACK OF CONTAINMENT INTEGRITY RESULTING FROM A PERSONNEL ERROR.
EVENT DATE: 111987 REPORT DATE: 011888 NSSS: BW TYPE: PWR
VENDOR: VELAN ENGINEERING COMPANIES

(NSIC 207878) ON NOVEMBER 19, 1987 AT 2137, TECHNICAL SPECIFICATION 3.6.1 REGARDING CONTAINMENT INTEGRITY WAS VIOLATED ON UNIT 1 WHEN CONTAINMENT ISOLATION VALVE 1N-106 WAS OPENED. THE CONTAINMENT ISOLATION VALVE WAS DISCOVERED OPEN ON DECEMBER 19, AT 1400, BY NUCLEAR EQUIPMENT OPERATOR (NEO) 'B', WHO WAS PERFORMING AN INSPECTION IN UNIT 1'S EAST PENETRATION ROOM. UNIT 1 WAS AT 100% FULL POWER THROUGHOUT THIS INCIDENT. THE ROOT CAUSE OF THIS INCIDENT WAS DETERMINED TO BE A PERSONNEL ERROR, BECAUSE A SENIOR REACTOR OPERATOR (SRO) DID NOT GIVE ADEQUATE GUIDANCE WHEN HE ASSIGNED A RESTORATION PROCEDURE TO NEO 'A'. THE SRO ALSO DID A LESS THAN ADEQUATE REVIEW OF THE RESTORATION PROCEDURE BEFORE HE ASSIGNED IT. THE IMMEDIATE CORRECTIVE ACTION WAS TO CLOSE 1N-106, THEREBY ESTABLISHING CONTAINMENT INTEGRITY. SUBSEQUENT CORRECTIVE ACTIONS INCLUDED A REVIEW OF PROCEDURES, COUNSELING OF THE SRO AND NEO 'A', AND A REACTOR BUILDING ENTRY TO VERIFY 1N-246 WAS CLOSED. PLANNED CORRECTIVE ACTIONS INCLUDE A PROCEDURE CHANGE AND ISSUANCE OF TRAINING PACKAGES.

[196] OCONEE 1 DOCKET 50-269 LER 87-011
CABLE ROOM SPRINKLER SYSTEMS INOPERABLE DUE TO DESIGN DEFICIENCY OF PRESSURE AND FLOW RATES.
EVENT DATE: 120287 REPORT DATE: 010488 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 207672) ON OCTOBER 8, 1987 DUKE POWER'S DESIGN ENGINEERING GROUP IDENTIFIED THAT THE UNIT 3 CABLE ROOM SPRINKLER SYSTEM COULD NOT PROVIDE ITS DESIGN FLOW DUE TO A DESIGN DEFICIENCY. THIS WAS IDENTIFIED WHILE DESIGN ENGINEERING WAS RESPONDING TO A 5/87 FIRE PROTECTION SYSTEM AUDIT. IT WAS DETERMINED THAT WHEN THE SPRINKLERS WERE INSTALLED, THE ABILITY OF THE HIGH PRESSURE SERVICE WATER SYSTEM (HPSW) TO SUPPLY THE SPRINKLERS WAS NOT PROPERLY VERIFIED. IT WAS ALSO IDENTIFIED THAT A CROSS CONNECTION, WHICH COULD BE USED TO SUPPLY THE CABLE ROOMS IF THEIR NORMAL SUPPLY HEADER WAS VALVED OUT, COULD NOT PROVIDE SUFFICIENT FLOW AND PRESSURE. THIS WAS ALSO DUE TO A DESIGN DEFICIENCY BECAUSE THE POTENTIAL USE OF THE CROSS CONNECTION AS A SUPPLY TO THE CABLE ROOMS WAS NOT CONSIDERED PRIOR TO INSTALLATION. THE ROOT CAUSE OF THIS INCIDENT WAS DETERMINED TO BE A DESIGN DEFICIENCY. THE INOPERABILITY OF THE UNIT 3 CABLE ROOM SPRINKLER SYSTEM WAS ORIGINALLY MISCLASSIFIED AS NON-REPORTABLE TO THE NRC DUE TO THE FACT THAT IT WAS NOT INITIALLY IDENTIFIED AS A DESIGN DEFICIENCY. CORRECTIVE ACTIONS INCLUDED ESTABLISHING FIRE WATCHES AT THE UNIT 3 CABLE ROOM, RESTORING THE SYSTEM TO AN OPERABLE CONDITION BY INCREASING ITS SUPPLY LINE DIAMETER, AND ESTABLISHING ADMINISTRATIVE CONTROLS TO ENSURE COMPENSATORY ACTIONS WERE TAKEN.

[197] OCONEE 3 DOCKET 50-287 LER 87-009
TECHNICAL SPECIFICATION VIOLATION DUE TO INADEQUATE PROCEDURE FOR FUNCTIONAL TEST OF ENGINEERED SAFEGUARDS SYSTEM.
EVENT DATE: 120787 REPORT DATE: 010688 NSSS: BW TYPE: PWR
VENDOR: BAILEY METER COMPANY

(NSIC 207885) ON DECEMBER 3, 1987 WITH UNIT 3 AT 100% FULL POWER PERFORMANCE OF AN ENGINEERED SAFEGUARDS (ES) SYSTEM ANALOG CHANNEL B ON LINE CALIBRATION PROCEDURE IDENTIFIED A POTENTIAL PROBLEM WITH THE ES SYSTEM. AN ANALYSIS PERFORMED ON DECEMBER 7 IDENTIFIED A FAULTY ROTARY SWITCH THAT HAD RENDERED A PORTION OF ES SYSTEM DIGITAL CHANNEL 2 INOPERABLE SINCE DECEMBER 3, 1987. THUS, TECHNICAL SPECIFICATION TABLE 3.5.1-1 WAS VIOLATED. THE ROOT CAUSE OF THIS INCIDENT WAS THE FAILURE OF THE ES SYSTEM CALIBRATION PROCEDURE TO INSURE THE

PROPER OPERATION OF THE ROTARY SWITCH. A CONTRIBUTING CAUSE WAS THE LACK OF FOLLOW UP ACTION TAKEN BY INVOLVED PERSONNEL. ON DECEMBER 7, 1987, IMMEDIATE CORRECTIVE ACTION WAS TAKEN BY REPLACING AND FUNCTIONALLY CHECKING THE ES SYSTEM MODULE CONTAINING THE FAULTY ROTARY SWITCH. IN ADDITION, ANALOG CHANNELS ON ALL THREE UNITS WERE TESTED TO INSURE THE PROPER DIGITAL CHANNEL INDICATIONS WERE RECEIVED. SUPPLEMENTAL CORRECTIVE ACTIONS INVOLVED COUNSELING INVOLVED PERSONNEL AND REVISING PROCEDURES. PLANNED CORRECTIVE ACTIONS INCLUDE AN AUDIT OF ES MAINTENANCE PROCEDURES AND ISSUANCE OF A TRAINING LETTER.

[198] OYSTER CREEK DOCKET 50-219 LER 86-017 REV 02
UPDATE ON ELECTRICAL STORM INDUCED CONTAINMENT ISOLATIONS AND SGTS INITIATIONS DUE TO AUTOMATIC BUS TRANSFER TIME VS LOGIC RELAY DROPOUT TIME.
EVENT DATE: 072986 REPORT DATE: 122387 NSSS: GE TYPE: BWR

(NSIC 207711) ON JULY 29, 1986 AT APPROXIMATELY 1503 AND 1655 HOURS, AND AGAIN ON JULY 30, AT 1659 HOURS, A PRIMARY CONTAINMENT ISOLATION, SECONDARY CONTAINMENT ISOLATION AND STANDBY GAS TREATMENT SYSTEM (SBGTS) (EIIS SYSTEM CODE BH) INITIATION OCCURRED DURING AN AUTOMATIC ELECTRICAL BUS TRANSFER. AT THE TIME, THE REACTOR WAS IN THE REFUEL MODE AND DEFUELED. THE EVENTS OCCURRED WHEN LIGHTNING STRUCK 34.5KV AND 230KV DISTRIBUTION LINES OUTSIDE THE PLANT, CAUSING A VOLTAGE TRANSIENT IN THE LINE. THE VOLTAGE TRANSIENT CAUSED VITAL AC POWER PANEL 1 (VACP-1) TO TRANSFER TO ITS ALTERNATE POWER SUPPLY. THE POWER SUPPLY TRANSFER CAUSED SEVERAL REACTOR PROTECTION SYSTEM (RPS) (EIIS SYSTEM CODE JC) RELAYS TO DEENERGIZE, CAUSING THE CONTAINMENT ISOLATIONS AND SBGTS INITIATION. THE ISOLATION SIGNAL WAS RESET MANUALLY. AFTER THE SECOND STRIKE ON JULY 29 AND THE STROKE ON JULY 30, SBGTS OPERATION WAS CONTINUED IN THE MANUAL MODE TO PREVENT FURTHER CYCLING OF THE SYSTEM UNTIL THE STORM PASSED. THE SAFETY SIGNIFICANCE OF THE EVENTS IS CONSIDERABLE MINIMAL. A SUBSEQUENT EVALUATION REVEALED THAT THE SHORTEST TRANSFER TIME ACHIEVABLE WITH THE AUTOMATIC TRANSFER SWITCH EXCEEDS THE DROPOUT TIME FOR THE RPS RELAYS.

[199] OYSTER CREEK DOCKET 50-219 LER 87-014 REV 01
UPDATE ON DRYWELL ISOLATION CAUSED BY INCORRECTLY LIFTING A LEAD DUE TO USING A PLANT DRAWING WHICH HAD NOT BEEN UPDATED.
EVENT DATE: 030387 REPORT DATE: 012088 NSSS: GE TYPE: BWR

(NSIC 207951) ON 3/3/87, WITH THE REACTOR SHUTDOWN, AN ELECTRICAL LEAD TERMINATION REPAIR WAS IN PROGRESS. AN ELECTRICIAN LIFTED A LEAD SUPPLYING ELECTRICAL POWER TO THE CONTAINMENT HIGH RANGE RADIATION MONITORING (CHRM) SYSTEM CHANNEL 2. THIS CAUSED A CONTAINMENT VENT AND PURGE VALVE ISOLATION. THE CAUSE OF THE EVENT HAS BEEN ATTRIBUTED TO THE DRAWINGS UTILIZED DURING THE PREPARATION AND REVIEW OF THE DETAILED INSTRUCTIONS WHICH DID NOT REFLECT RECENT PLANT MODIFICATIONS. THE ROOT CAUSE OF THE EVENT WAS A CONFIGURATION CONTROL PROCESS DEFICIENCY CAUSED BY A BACKLOG OF DOCUMENTATION AWAITING FINAL CLOSE-OUT FOLLOWING SYSTEM MODIFICATION. THE REQUIRED INFORMATION HAD NOT BEEN FINALIZED AT THE TIME OF THE EVENT. THE LEAD WAS RELANDED AND THE ISOLATION RESET. AFTER REVIEWING THE SITUATION AND PRIOR TO RESUMING MAINTENANCE, THE CONTROL ROOM OPERATORS BYPASSED THE CHRM SYSTEM ISOLATION FUNCTION. A REVISION TO THE COMPUTERIZED DRAWING CONTROL SYSTEM HAS BEEN IMPLEMENTED WHICH WILL IDENTIFY DRAWINGS AFFECTED BY SCHEDULED MODIFICATIONS. PROCEDURES WILL BE REVISED TO REQUIRE A SUMMARY OF ALL DRAWING CHANGES RESULTING FROM A MODIFICATION TO BE AVAILABLE AT THE TIME THE MODIFICATION IS TURNED OVER TO THE PLANT DIVISION. THE SAFETY SIGNIFICANCE OF THIS EVENT IS CONSIDERED MINIMAL AS THE VENT AND PURGE VALVE ISOLATIONS HAD NO ADVERSE EFFECT ON PLANT SYSTEMS

[200] OYSTER CREEK DOCKET 50-219 LER 87-046
REACTOR SAFETY VALVE LEAKED DURING HYDRO TEST DUE TO MANUAL LIFTING NUT COTTER PIN FAILURE.
EVENT DATE: 111087 REPORT DATE: 010788 NSSS: GE TYPE: BWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV

(NSIC 207730) ON NOVEMBER 10, 1987 AT APPROXIMATELY 1709 HOURS, SAFETY VALVE V-1-162 "F" WAS FOUND TO BE LEAKING DURING A REACTOR VESSEL HYDROSTATIC PRESSURE TEST. AT THE TIME OF THIS EVENT, THE REACTOR WAS SHUTDOWN AT APPROXIMATELY 200 PSIG AND A REACTOR VESSEL HYDROSTATIC PRESSURE TEST WAS IN PROGRESS. MAINTENANCE PERSONNEL INVESTIGATED THE CAUSE OF THE SAFETY VALVE LEAKING AND DISCOVERED A COTTER PIN DESIGNED TO HOLD THE VALVE MANUAL LIFT NUT IN PLACE HAD FAILED. THE NUT HAD ROTATED DOWN THE VALVE STEM THREADS TO A POSITION WHICH INDUCED A PRELOAD TENSION WHEN THE VALVE COOLED OPPOSING THE FORCE OF THE CLOSING SPRING. THIS WOULD ALLOW VALVE SEAT LEAKAGE AT A REDUCED PRESSURE BUT WOULD NOT HAVE AFFECTED THE ACTUAL VALVE LIFT SETPOINT WHEN THE VALVE COOLED DOWN. THE NUT WAS REMOVED AND THE VESSEL HYDROSTATIC PRESSURE TEST WAS SUCCESSFULLY COMPLETED. AFTERWARD, THE SAFETY VALVE WAS REPLACED AND THE MANUAL LIFT NUTS WERE REMOVED FROM ALL INSTALLED REACTOR SAFETY VALVES TO PREVENT THIS EVENT FROM RECURRING. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL AS NEITHER NUCLEAR NOR PERSONNEL SAFETY WAS AFFECTED.

[201] OYSTER CREEK DOCKET 50-219 LER 87-045
SGTS INITIATION DUE TO WATER ACCUMULATION IN THE AOG SYSTEM.
EVENT DATE: 120187 REPORT DATE: 122887 NSSS: GE TYPE: BWR

(NSIC 207630) ON DECEMBER 1, 1987 AT 0933 HOURS A REACTOR BUILDING ISOLATION AND STANDBY GAS TREATMENT SYSTEM (SGTS) AUTO INITIATION OCCURRED. AT THE TIME THE REACTOR WAS OPERATING AT FULL POWER. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO MAIN CONDENSER OFFGAS PRESSURE OSCILLATIONS CAUSING OFFGAS FLOW THROUGH A WATER SEAL IN A DRAIN LINE. THE DRAIN INVOLVED IS NORMALLY OPEN TO ALLOW WATER ACCUMULATED IN THE OFFGAS LINE PIPING TO EXIT TO A SUMP LOCATED UNDER THE PLANT VENT STACK. DUE TO PRESSURE OSCILLATIONS IN THE OFFGAS SYSTEM, THE OFFGAS LINE BECAME MOMENTARILY PRESSURIZED ABOVE THE STATIC HEAD OF THE WATER SEAL POT. HYDROGEN GAS AND RADIOACTIVE GAS WERE RELEASED. THE REACTOR BUILDING VENT MONITOR LOCATED NEAR THE SUMP ALARMED CAUSING THE SAFETY SYSTEM ACTUATION. PLANT EMERGENCY OPERATING PROCEDURES WERE ENTERED. ALL SAFETY EQUIPMENT FUNCTIONED NORMALLY. THE SAFETY SIGNIFICANCE IS CONSIDERED MINIMAL SINCE ONLY A SMALL QUANTITY OF OFFGAS WAS RELEASED TO THE AREA. THE FOUR (4) INCH OFFGAS LINE CAUSING THE PRESSURE OSCILLATIONS WAS DRAINED AND MAINTENANCE IS SCHEDULED TO VERIFY CLEAR PASSAGE OF THE DRAIN LINE.

[202] PALISADES DOCKET 50-255 LER 87-037 REV 01
UPDATE ON FAILURE TO TAKE COMPENSATORY MEASURES DURING TEMPORARY LOSS OF FIRE SUPPRESSION SYSTEM.
EVENT DATE: 110687 REPORT DATE: 012288 NSSS: CE TYPE: PWR

(NSIC 208003) ON NOVEMBER 6, 1987 AT 1055 AN UNDERGROUND PIPE ASSOCIATED WITH THE FIRE SUPPRESSION WATER SYSTEM (KP;PSP) RUPTURED. THIS RESULTED IN ALL THREE FIRE SYSTEM WATER PUMPS (KP;P) AUTOMATICALLY AND SEQUENTIALLY ACTIVATING DUE TO LOW SYSTEM PRESSURE. ALL FIRE SYSTEM WATER PUMPS WERE SECURED BY 1100. BY 1140 THE FIRE WATER SYSTEM WAS REPRESSURIZED AND BELIEVED TO BE IN NORMAL OPERATING STATUS WITH THE EXCEPTION OF THE ISOLATION SECTION. AT 1720 AN AUXILIARY OPERATOR REPORTED THAT NO FIRE WATER PRESSURE WAS AVAILABLE TO THE SPRINKLER SYSTEM IN THE AREA OF THE VOLUME REDUCTION SYSTEM (VRS) (WB). COMPENSATORY FIRE TOURS WERE INITIATED AND BY 2320 ALL REQUIRED BACKUP FIRE SUPPRESSION EQUIPMENT HAD BEEN STAGED BY UTILIZING FIRE HOSES AS TEMPORARY CONNECTIONS. THE PIPE FAILURE HAS BEEN ATTRIBUTED TO A TEMPORARY CONSTRUCTION PIPE SUPPORT MAINTAINING THE POSITION OF THE PIPE WHILE THE SURROUNDING GROUND SETTLED. A HEAVY CRANE WAS BEING

UTILIZED IN THE AREA OF THE PIPE PRIOR TO THE FAILURE. THE FAILURE TO MEET THE REQUIRED COMPENSATORY MEASURES WITHIN ONE HOUR WAS THE RESULT OF A MISLEADING PLANT DIAGRAM. THE PIPE HAS BEEN REPLACED, TESTED AND THE SYSTEM RETURNED TO NORMAL OPERATING CONDITION. A CHANGE REQUEST HAS BEEN INITIATED TO CORRECT THE MISLEADING DIAGRAM.

[203] PALO VERDE 1 DOCKET 50-528 LER 87-016 REV 01
UPDATE ON BOTH TRAINS OF ESF PUMP ROOM AIR EXHAUST CLEANUP SYSTEM INOPERABLE DUE TO PERSONNEL ERROR IN SCHEDULING MAINTENANCE ACTIVITIES.
EVENT DATE: 060987 REPORT DATE: 010788 NSSS: CE TYPE: PWR

(NSIC 207707) ON 6/11/87 IT WAS DISCOVERED THAT BETWEEN 1700 AND 2330 MST ON 6/9/87, WITH PALO VERDE UNIT 1 IN MODE 1 (POWER OPERATION) OPERATING AT 100% POWER, BOTH TRAINS OF THE ENGINEERED SAFETY FEATURE (ESF) PUMP ROOM AIR EXHAUST CLEANUP SYSTEM (PRAECS) WERE RENDERED INOPERABLE AT THE SAME TIME. WHILE REVIEWING THE WORK COMPLETED DURING A RECENT FUEL BUILDING ESSENTIAL VENTILATION TRAIN "B" ONLINE OUTAGE, THE ON-SHIFT SHIFT SUPERVISOR DISCOVERED THAT THE COMBINATION OF TWO SEPARATE MAINTENANCE ACTIVITIES MAY HAVE RENDERED BOTH TRAINS OF THE ESF PRAECS INOPERABLE. BASED ON FURTHER EVALUATION, IF THE OPERABLE TRAIN "A" OF THE ESF PRAECS HAD BEEN STARTED FOLLOWING A SAFETY INJECTION ACTUATION SIGNAL, THE ABILITY TO EXHAUST THE TECH SPEC REQUIRED FLOWRATE FROM THE AUXILIARY BUILDING BELOW THE 100' ELEVATION WOULD HAVE BEEN IMPAIRED. THE ROOT CAUSE OF THE EVENT WAS A COGNITIVE PERSONNEL ERROR BY THE WORK CONTROL SHIFT SUPERVISOR WHO DID NOT RECOGNIZE THAT CONCURRENT MAINTENANCE ACTIVITIES SUCH AS THESE WOULD RENDER THE SYSTEM INOPERABLE. AS CORRECTIVE ACTION TO PREVENT RECURRENCE, A REPORT OF THE EVENT HAS BEEN ISSUED TO THE APPROPRIATE OPERATIONS, MAINTENANCE AND WORK CONTROL PERSONNEL, WARNING TAGS HAVE BEEN PLACED ON THE APPROPRIATE EQUIPMENT TO HELP PREVENT CROSS TIES BETWEEN VENTILATION SYSTEMS.

[204] PALO VERDE 1 DOCKET 50-528 LER 87-027
CONTAINMENT SPRAY SYSTEM "B" TRAIN PUMP INCORRECTLY DECLARED OPERABLE.
EVENT DATE: 072887 REPORT DATE: 010588 NSSS: CE TYPE: PWR

(NSIC 207873) AT APPROXIMATELY 0947 MST ON DECEMBER 7, 1987, IT WAS DISCOVERED THAT PALO VERDE UNIT 1 HAD ENTERED MODE 4 (HOT SHUTDOWN) AT APPROXIMATELY 0234 MST ON JULY 28, 1987, WITHOUT MEETING THE SURVEILLANCE REQUIREMENTS FOR THE CONTAINMENT SPRAY S STEM (BE) "B" TRAIN PUMP (P) AS SPECIFIED IN TECHNICAL SPECIFICATION (T.S.) 4.6.2.1.B. A TEST CONDUCTED ON AUGUST 5, 1987, LATER VERIFIED OPERABILITY OF THE PUMP IN ACCORDANCE WITH T.S. 4.6.2.1.B. THIS EVENT WAS CAUSED BY PERSONNEL ERROR AS A DIRECT RESULT OF AN ERROR IN AN APPROVED PROCEDURE. THE CONTAINMENT SPRAY PUMP OPERABILITY TEST PROCEDURE HAD BEEN REVISED ON JULY 8, 1987 TO INCLUDE PROVISIONS FOR PERFORMING SECTION XI TESTING OF THE PUMP IN SHUTDOWN COOLING MODE. THE REQUIREMENTS SPECIFIED IN T.S. 4.6.2.1.B REQUIRE THIS TEST TO BE PERFORMED IN RECIRCULATION MODE, HOWEVER, THE REVISED PROCEDURE PERMITTED THE TEST STEPS NECESSARY TO COMPLY WITH T.S. 4.6.2.1.B TO BE BYPASSED WITHOUT VIOLATING THE PROCEDURE. THE "B" TRAIN PUMP WAS SUBSEQUENTLY TESTED ON JULY 10, 1987 AND DETERMINED TO BE OPERABLE WITHOUT MEETING T.S. 4.6.2.1.B. AS CORRECTIVE ACTION, "A" TRAIN PUMP TESTS WERE VERIFIED TO HAVE BEEN PERFORMED CORRECTLY, AND THE UNIT 1 PROCEDURE WAS REVISED TO ENSURE PROPER VERIFICATION OF T.S. 4.6.2.1.B. NO SIMILAR EVENTS HAVE BEEN IDENTIFIED.

[205] PALO VERDE 1 DOCKET 50-528 LER 87-025 REV 01
UPDATE ON MODIFICATIONS TO STEAM TO TURBINE DRIVEN AUXILIARY FEEDWATER PUMP ISOLATION VALVES RENDER PUMP INOPERABLE.
EVENT DATE: 112787 REPORT DATE: 010788 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 2 (PWR)

(NSIC 207708) THIS IS A SUPPLEMENT TO LER 87-025-00. ON NOVEMBER 27, 1987 AT

APPROXIMATELY 0220 MST, WITH PALO VERDE UNIT 2 IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER, THE TURBINE DRIVEN AUXILIARY FEEDWATER PUMP (BA)(P) DID NOT ACHIEVE RATED SPEED DURING THE MONTHLY OPERABILITY SURVEILLANCE TEST. AN INVESTIGATION FOUND THAT THE "OPEN" LIMIT SWITCH (SB)(33) SETPOINTS FOR THE "B" AND "A" TRAIN ISOLATION VALVES (SB)(ISV) WERE ADJUSTED ON OCTOBER 14 AND 15, 1987, RESPECTIVELY, TO PREVENT THE VALVE INTERNALS FROM IMPACTING ON THE BACKSEAT. THIS WAS DONE IN ACCORDANCE WITH AN APPROVED ENGINEERING EVALUATION WHICH DID NOT PROVIDE CORRESPONDING ADJUSTMENTS TO THE "RAMP UP" LIMIT SWITCHES (SB)(33), THEREFORE PREVENTING THE PUMP FROM ACHIEVING RATED SPEED. FOLLOWING THESE ADJUSTMENTS, THE PUMP WAS RETURNED TO OPERABLE STATUS, CONTRARY TO TECHNICAL SPECIFICATION 3.7.1.2. ONGOING INVESTIGATION DETERMINED THAT VALVES HAD BEEN RETURNED TO OPERABLE STATUS CONTRARY TO T.S. 3.6.3. AS IMMEDIATE CORRECTIVE ACTION THE LIMIT SWITCHES WERE READJUSTED, OPERABILITY TESTS CONDUCTED ON NOVEMBER 27, 1987, AND THE INVESTIGATION EXPANDED TO INCLUDE UNITS 1 AND 3. PRELIMINARY EVALUATIONS IDENTIFIED THE ROOT CAUSE AS COGNITIVE PERSONNEL ERROR (UTILITY, NON-LICENSED) IN THAT THE ENGINEERING EVALUATION DID NOT ADDRESS THE FULL IMPACT OF THE APPROVED MODIFICATION.

[206] PALO VERDE 1 DOCKET 50-528 LER 87-028
PERSONNEL ERROR RESULTS IN INCOMPLETE TECHNICAL SPECIFICATION SAMPLE ANALYSIS.
EVENT DATE: 121787 REPORT DATE: 011588 NSSS: CE TYPE: PWR

(NSIC 207925) AT APPROXIMATELY 1320 MST ON DECEMBER 17, 1987, PALO VERDE UNIT 1 WAS IN MODE 5 (COLD SHUTDOWN) WHEN IT WAS DISCOVERED THAT TWO WEEKLY PLANT VENT RADIATION MONITOR (VL)(IL)(RI) SAMPLES HAD NOT BEEN RETAINED FOR INCLUSION WITH THE QUARTERLY COMPOSITE STRONTIUM - 89/90 SAMPLE. THIS RESULTED IN AN INABILITY TO MEET THE SAMPLING REQUIREMENTS OF TECHNICAL SPECIFICATION 4.11.2.1.2. THE ROOT CAUSE OF THIS EVENT WAS A COGNITIVE PERSONNEL ERROR IN THAT A CHEMISTRY TECHNICIAN (UTILITY, NON-LICENSED) DID NOT TAKE SUFFICIENT MEASURES TO ENSURE THAT THE SAMPLES WERE RETAINED. PROCEDURAL CONTROLS WERE EVALUATED AND DETERMINED TO BE ADEQUATE. AS CORRECTIVE ACTION TO PREVENT RECURRENCE, THE RESPONSIBLE INDIVIDUAL WILL RECEIVE APPROPRIATE DISCIPLINARY ACTION. UNIT CHEMISTRY DEPARTMENT PERSONNEL WILL RECEIVE APPROPRIATE TRAINING, AND ADDITIONAL CONTROLS FOR TRACKING AND STORING SAMPLES WILL BE EVALUATED. A SIMILAR EVENT WAS REPORTED IN UNIT 1 LER 86-007-00.

[207] PALO VERDE 2 DOCKET 50-529 LER 87-021
PASS INCORRECTLY DECLARED OPERABLE DUE TO IMPROPER VALVE LINEUP.
EVENT DATE: 120287 REPORT DATE: 122987 NSSS: CE TYPE: PWR

(NSIC 207542) AT APPROXIMATELY 1440 MST ON DECEMBER 2, 1987 THE POST-ACCIDENT SAMPLING SYSTEM (PASS)(IP) WAS DECLARED INOPERABLE FOLLOWING DISCOVERY OF AN IMPROPER VALVE LINEUP. THE LINEUP WAS PERFORMED ON NOVEMBER 7, 1987 TO PERMIT INSTALLATION OF TWO PASS CHECK VALVES (IP)(V). WORK WAS SUSPENDED AND THE PASS DECLARED OPERABLE AT APPROXIMATELY 1220 MST ON NOVEMBER 9, 1987. BASED ON SUBSEQUENT INVESTIGATION, THE PASS WAS DETERMINED TO HAVE BEEN INOPERABLE SINCE APPROXIMATELY 0405 MST, NOVEMBER 7, 1987, AND TO HAVE EXCEEDED THE 7 DAY LIMIT FOR INOPERABILITY PER TECH SPEC 3.3.3.1 AT 0405 MST ON NOVEMBER 14, 1987. THE PREPLANNED ALTERNATE SAMPLING PROGRAM (PAS) WAS INITIATED AT 1420 MST ON DECEMBER 4, 1987, THEREFORE PALO VERDE UNIT 2 OPERATED FOR APPROXIMATELY 20 DAYS IN A CONDITION CONTRARY TO TECH SPEC 3.3.3.1. AS IMMEDIATE CORRECTIVE ACTION THE PASS WAS RESTORED, TESTED FOR OPERABILITY, AND DECLARED OPERABLE AT 1900 MST ON DECEMBER 6, 1987. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE PERSONNEL ERROR CONTRARY TO APPROVED PROCEDURES. THIS EVENT REMAINS UNDER INVESTIGATION. NECESSARY CORRECTIVE ACTIONS REQUIRED TO PREVENT RECURRENCE WILL BE PROVIDED IN A SUPPLEMENT UPON COMPLETION OF THE INVESTIGATION AND EVALUATION OF THE RESULTS.

[208] PALO VERDE 3 DOCKET 50-530 LER 87-004
REACTOR TRIP OCCURS DUE TO CONTROL ELEMENT ASSEMBLY SUBGROUP DEVIATION.
EVENT DATE: 121787 REPORT DATE: 011588 NSSS: CE TYPE: PWR
VENDOR: ELECTRO-MECHANICS
I-T-E CIRCUIT BREAKER

(NSIC 207926) AT APPROXIMATELY 0430 MST ON DECEMBER 17, 1987, PALO VERDE UNIT 3 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 50% POWER WHEN A REACTOR (RX) TRIP OCCURRED AS A RESULT OF A DEVIATION IN POSITION OCCURRING BETWEEN PART LENGTH CONTROL ELEMENT ASSEMBLY (AA) SUBGROUPS. THE TRIP OCCURRED AS THE REACTOR WAS AT THE 50% TEST PLATEAU DURING POWER ASCENSION TESTING. THE REACTOR TRIP WAS UNCOMPLICATED AND THE PLANT WAS STABILIZED WITHIN 30 MINUTES TERMINATING THE EVENT. THERE WERE NO ENGINEERED SAFETY FEATURES (JE) ACTUATIONS. THE ROOT CAUSE OF THE TRIP WAS A MALFUNCTIONING LOGIC SEQUENCER CARD (ZC) IN THE CONTROL ELEMENT DRIVE MECHANISM CONTROL SYSTEM (CEDMCS)(AA). AS CORRECTIVE ACTION, THE CARD WAS REPLACED AND THE CEDMCS WAS VERIFIED TO BE OPERATING PROPERLY. ADDITIONAL PREVENTIVE CORRECTIVE ACTIONS ARE BEING IMPLEMENTED AS DISCUSSED IN THE TEXT OF THIS REPORT. FOLLOWING THE TRIP, A RELAY/CONTACT (94/CNTR) ASSEMBLY DID NOT OPERATE PROPERLY WHICH RESULTED IN THE CLASS 1E PRESSURIZER (PZR) HEATERS (EHTR) CONTINUING TO REMAIN ENERGIZED BELOW THE LOW LEVEL TRIP SETPOINT. AS CORRECTIVE ACTION, THE RELAY/CONTACT ASSEMBLY WAS REPLACED AND THE HEATERS WERE VERIFIED TO BE OPERATING PROPERLY. NO SIMILAR REACTOR TRIPS HAVE OCCURRED.

[209] PEACH BOTTOM 2 DOCKET 50-277 LER 87-024
GROUP IIA ISOLATION OF THE RWCU SYSTEM DUE TO THE BLOWING OF A FUSE.
EVENT DATE: 111287 REPORT DATE: 121787 NSSS: GE TYPE: BWR

(NSIC 207543) ON NOVEMBER 12, 1987 AT 2145 HOURS, AN ISOLATION OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM OCCURRED ON UNIT 2 AS THE RESULT OF THE BLOWING OF A FUSE (12A-F1) WHICH SUPPLIES POWER TO FOUR TEMPERATURE SWITCHES (TIS-2-12-99 AND TIS-2-12-89A, B AND C). THE LOSS OF POWER CAUSED THE TEMPERATURE SWITCHES TO OPEN. THE TEMPERATURE SWITCHES SENSE THE TEMPERATURE AT THE DISCHARGE OF THE RWCU NON-REGENERATIVE HEAT EXCHANGER AND SENSE THE TEMPERATURE OF THE COOLING WATER SUPPLIED TO THE RWCU RECIRCULATION PUMPS BY THE REACTOR BUILDING CLOSED COOLING WATER (RBCCW) SYSTEM. THE FUSE WAS REPLACED BUT POWER WAS NOT RESTORED DUE TO THE OPENING OF A BREAKER WHICH SUPPLIES POWER TO THE FOUR TEMPERATURE SWITCHES. THE SHIFT TECHNICAL ADVISOR VERIFIED THAT THE BREAKER HAD BEEN OPENED AS PART OF THE APPLICATION OF A BLOCKING PERMIT 15 MINUTES AFTER THE ISOLATION OCCURRED. THE OPENING OF THE BREAKER WOULD HAVE ALSO CAUSED AN UNEXPECTED ISOLATION. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THIS EVENT. THE ROOT CAUSE FOR THE OPENING OF THE BREAKER IN THE 20Y3501 PANEL WAS A PERSONNEL ERROR IN THAT THE SHIFT SUPERVISOR FAILED TO PERFORM AN ADEQUATE REVIEW OF THE BLOCKING PERMIT FOR ENGINEERING SAFETY FEATURE ACTUATIONS BEFORE ISSUING IT. AS CORRECITVE ACTIONS, THE FUSE (12A-F1) WAS REPLACED, THE BLOCKING PERMIT WAS CLEARED, THE SUPPLY BREAKER CLOSED, AND THE ISOLATION RESET.

[210] PEACH BOTTOM 2 DOCKET 50-277 LER 87-025
PRIMARY CONTAINMENT ISOLATION DUE TO PERSONNEL ERROR DURING TESTING.
EVENT DATE: 120287 REPORT DATE: 010488 NSSS: GE TYPE: BWR

(NSIC 207673) ON 12/2/87 AT 0945 HOURS WITH THE UNIT IN COLD SHUTDOWN, A GROUP III OUTBOARD ISOLATION OCCURRED. A GROUP III ISOLATION INVOLVES THE VENTILATION SYSTEM OF THE PRIMARY AND SECONDARY CONTAINMENT. THE ROOT CAUSE OF THE ISOLATION WAS PERSONNEL ERROR WHICH RESULTED IN AN ACCIDENTAL CONTACT OF A TEST JUMPER TO A NON-TEST POINT DURING PERFORMANCE OF A MODIFICATION ACCEPTANCE TEST. THE DESIGN AND LOCATION OF RELAY 16A-K41 WERE ALSO CONTRIBUTORS TO THE CAUSE. WHILE ATTEMPTING TO PLACE THE JUMPER ON A SET OF CONTACTS ON RELAY 16A-K41, THE LEAD ACCIDENTALLY TOUCHED AN ADJACENT CONTACT DUE TO LIMITED ACCESS OF THE RELAY. THIS RESULTED IN TWO OUT-OF-PHASE POWER SOURCES TO BE DRAWN THROUGH A FUSE,

CAUSING IT TO BLOW. THE BLOWN FUSE INITIATED THE GROUP III ISOLATION. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THE EVENT. ALL EXPECTED ACTIONS EITHER OCCURRED OR WERE ELECTRICALLY BLOCKED. THE FUSE WAS REPLACED AND THE ISOLATION RESET WITHIN APPROXIMATELY ONE HOUR. THE INDIVIDUAL WHO PERFORMED THE TEST WAS COUNSELED ON THE IMPORTANCE OF ATTENTION TO DETAIL DURING PERFORMANCE OF DUTIES. THE EVENT IS BELIEVED TO BE AN ISOLATED OCCURRENCE AND NO ADDITIONAL CORRECTIVE ACTIONS ARE PLANNED. THIS EVENT IS REPORTABLE DUE TO THE ACTUATION OF THE PRIMARY CONTAINMENT ISOLATION SYSTEM, AN ENGINEERED SAFETY FEATURE.

[211] PEACH BOTTOM 2 DOCKET 50-277 LER 87-029
PRIMARY CONTAINMENT ISOLATION SYSTEM GROUP III ISOLATION DUE TO PERSONNEL ERROR WHILE PULLING FUSES TO APPLY A BLOCKING PERMIT.
EVENT DATE: 122187 REPORT DATE: 012588 NSSS: GE TYPE: BWR

(NSIC 207956) ON DECEMBER 21, 1987 AT 1135 HOURS, A UNIT 2 REACTOR BUILDING VENTILATION SYSTEM (RBVS) INBOARD ISOLATION AND STANDBY GAS TREATMENT SYSTEM (SBGTS) INITIATION OCCURRED WHILE REAPPLYING A BLOCKING PERMIT FOR MODIFICATION WORK. THESE ACTUATIONS WERE THE RESULT OF AN OPERATOR ERROR IN THAT A FUSE WAS PULLED NOT IN COMPLIANCE WITH THE BLOCKING PERMIT. THE ISOLATION WAS RESET AT 1140 HOURS AND THE FUSES WERE EXCHANGED IN ACCORDANCE WITH THE PERMIT. FURTHER BLOCKING WAS THEN TERMINATED. DUE TO PREVIOUS ISOLATIONS, THIS SAME BLOCKING PERMIT HAD BEEN REVISED TO INCLUDE A CAUTION THAT AN ISOLATION WOULD OCCUR IF PRIOR ISOLATIONS WERE NOT RESET PRIOR TO REMOVING THE FUSES, DESPITE THE PROPER REAPPLICATION OF THE PERMIT. HOWEVER, THE PERMIT WAS NOT FOLLOWED PROPERLY THEREBY ALLOWING THE UNEXPECTED ISOLATION OF RBVS AND INITIATION OF SBGTS. AS A RESULT OF THIS EVENT, THE OPERATOR WAS COUNSELED TO BE ATTENTIVE TO HIS DUTIES AND TO FOLLOW INSTRUCTIONS. OPERATIONS MANAGEMENT EMPHASIZES THE NECESSITY OF ADHERING TO ALL INSTRUCTIONS AND PROCEDURES, AS WELL AS ATTENTION TO DETAIL WHEN REAPPLYING BLOCKING PERMITS. COMPLIANCE WITH THESE INSTRUCTIONS WILL MINIMIZE THE OPPORTUNITY FOR PERSONNEL ERROR.

[212] PEACH BOTTOM 3 DOCKET 50-278 LER 87-010 REV 01
UPDATE ON OPENING OF A REACTOR VESSEL LEVEL INSTRUMENT WHICH ULTIMATELY RESULTED IN A GROUP I ISOLATION SIGNAL.
EVENT DATE: 102687 REPORT DATE: 122887 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 2 (BWR)

(NSIC 207659) ON OCTOBER 26, 1987 AT 1042 HOURS, A PRIMARY CONTAINMENT GROUP I ISOLATION SIGNAL WAS GENERATED WHILE THE UNITS WERE SHUTDOWN AND OPERATING THE RESIDUAL HEAT REMOVAL SYSTEM IN THE SHUTDOWN COOLING MODE. THIS ACTION OCCURRED AS A RESULT OF A PHILADELPHIA ELECTRIC COMPANY MAINTENANCE CRAFTSMAN OPENING THE LOW SIDE INSTRUMENT DRAIN VALVE (IDV-02-3-58BL) OF REACTOR VESSEL LEVEL INSTRUMENT LIS-3-02-3-058B. THIS RESULTED IN THE DRAINING OF THE VARIABLE LEG SIDE OF THE INSTRUMENTATION IN THE 3BC65B RACK AND ULTIMATELY RESULTING IN A GROUP I ISOLATION SIGNAL AND A LOSS OF SHUTDOWN COOLING ON UNITS 2 AND 3. THE CAUSE FOR THE EVENT WAS PERSONNEL ERROR IN THAT THE CRAFTSMAN DID NOT COMPLY WITH THE PHILADELPHIA ELECTRIC COMPANY "RULES FOR PERMITS AND BLOCKING" WHEN OPENING THE IDV-02-3-58BL VALVE WHICH WAS OUTSIDE THE BLOCKING PERMIT BOUNDARY. THERE WERE NO ADVERSE CONSEQUENCES OF THIS EVENT. AS IMMEDIATE CORRECTIVE ACTION, THE VALVE WAS CLOSED, THE VARIABLE LEG WAS REFILLED, AND NORMAL INDICATION LEVEL WAS RESTORED. ALL PLANT FUNCTIONS WERE RESTORED TO NORMAL BY 1110 HOURS. AS ACTION TO PREVENT RECURRENCE, THE CRAFTSMAN WILL BE DISCIPLINED IN ACCORDANCE WITH THE DISCIPLINARY POLICY AND WILL BE REINSTRUCTED WITH REGARDS TO ENSURING THAT THE RULES FOR PERMITS AND BLOCKING ARE CORRECTLY FOLLOWED.

[213] PEACH BOTTOM 3 DOCKET 50-278 LER 87-011
REACTOR WATER CLEANUP SYSTEM ISOLATION SIGNAL DURING PIPING DECONTAMINATION PROCESS DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 112987 REPORT DATE: 122987 NSSS: GE TYPE: BWR

(NSIC 207675) ON 11/29/87 AT 1800 HOURS, REACTOR WATER CLEANUP SYSTEM (RWCU) ISOLATION AND RWCU PUMP TRIP SIGNALS WERE GENERATED DUE TO A HIGH NON-REGENERATIVE HEAT EXCHANGER OUTLET TEMPERATURE SIGNAL. THE RWCU SYSTEM WAS OUT OF SERVICE AT THE TIME AND A SPECIAL DECONTAMINATION PROCESS OF THE PIPING WAS IN PROGRESS. PROCESS FLOW TEMPERATURE REACHED THE TRIP SETPOINT AND THE TRIP LOGIC ACTUATED. NO VALVES MOVED BECAUSE THE ISOLATION VALVES WERE BLOCKED IN THE CLOSED POSITION. THE RWCU PUMPS WERE NOT AFFECTED BECAUSE THEY WERE NOT IN SERVICE. A TEMPORARILY INSTALLED PUMP WAS IN USE. THE ISOLATION SIGNAL WAS RESET APPROXIMATELY TWO HOURS LATER. THERE WERE NO ADVERSE SAFETY CONSEQUENCES AND THIS EVENT HAS NO SAFETY SIGNIFICANCE. THIS EVENT OCCURRED BECAUSE OF A DEFICIENCY IN THE PROCEDURE BEING USED FOR THE SPECIAL DECONTAMINATION PROCESS. WHEN THE PROCEDURE WAS PREPARED AND REVIEWED, IT WAS NOT RECOGNIZED THAT PRECAUTIONS WERE NEEDED TO PREVENT A HIGH TEMPERATURE ISOLATION SIGNAL. CONSEQUENTLY, THE PROCEDURE DID NOT INCLUDE STEPS TO PREVENT THE ISOLATION SIGNAL. ALTHOUGH IT IS NOT EXPECTED THAT THIS PROCEDURE WILL BE USED AGAIN, IT HAS BEEN REVISED TO PREVENT A HIGH TEMPERATURE ISOLATION DURING A DECONTAMINATION PROCESS.

[214] PERRY 1 DOCKET 50-440 LER 87-031 REV 01
UPDATE ON FAILURE TO TEST LOCA RELAY CONTACTS RESULTS IN TECH SPEC VIOLATION.
EVENT DATE: 050887 REPORT DATE: 020588 NSSS: GE TYPE: BWR

(NSIC 208043) ON 5/8/87, AT 1445 DURING A SURVEILLANCE TEST INSTRUCTION (SVI) REVIEW IT WAS DISCOVERED THAT SELECTED DIVISION 3 LOSS OF COOLANT ACCIDENT (LOCA) RELAY CONTACTS WERE NOT TESTED AS REQUIRED BY TECH SPEC LOGIC SYSTEM FUNCTIONAL TEST (LSFT) PER SECTION 4.3.3.2. WHEN OPERATIONS PERSONNEL WERE NOTIFIED OF THE DEFICIENT TEST THE AFFECTED COMPONENT (DIVISION 3 ELECTRICAL SYSTEM) WAS DECLARED INOPERABLE AND TECH SPEC SECTION 3.0.3 WAS ENTERED. TESTING OF THESE RELAY CONTACTS WAS IMMEDIATELY PERFORMED AND COMPLETED SATISFACTORILY, AT 2025. POWER OPERATION CONTINUED WITH PLANT START UP TESTING IN PROGRESS SUBSEQUENT TO THIS EVENT. THE CAUSE OF THIS EVENT WAS AN INADEQUATE SURVEILLANCE TEST INSTRUCTION. THE SVI DID NOT SATISFY THE SURVEILLANCE REQUIREMENTS OF TECH SPEC. TO PREVENT RECURRENCE SVI E22-T1192, "HIGH PRESSURE CORE SPRAY LOGIC SYSTEM FUNCTIONAL TEST," HAS BEEN REVISED TO INCLUDE TESTING OF THESE RELAY CONTACTS. ADDITIONALLY, A COMPLETE REVIEW OF ALL SVIS ASSOCIATED WITH LOGIC TESTING HAS BEEN COMPLETED. THIS REVIEW VERIFIED THAT ALL PORTIONS OF LOGIC CIRCUITS REQUIRED BY TECH SPECS, FROM SENSOR THROUGH AND INCLUDING THE ACTUATING DEVICE, ARE TESTED BY THE SVI PROGRAM. ALTHOUGH PROCEDURAL DEFICIENCIES WERE IDENTIFIED DURING THIS REVIEW, NONE RESULTED IN A VIOLATION OF TECHNICAL SPECIFICATIONS.

[215] PERRY 1 DOCKET 50-440 LER 87-076 REV 01
UPDATE ON DEFICIENT SURVEILLANCE INSTRUCTIONS RESULT IN MAIN STEAM DRAIN LINE ISOLATIONS.
EVENT DATE: 120387 REPORT DATE: 012288 NSSS: GE TYPE: BWR

(NSIC 207985) ON DECEMBER 3, 1987 AT APPROXIMATELY 2145 AND 2305, UNEXPECTED MAIN STEAM DRAIN LINE ISOLATIONS OCCURRED DUE TO NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) ISOLATION SIGNALS DURING THE PERFORMANCE OF TURBINE STOP VALVE SURVEILLANCE INSTRUCTIONS. THE CAUSE OF THIS EVENT WAS INADEQUATE SURVEILLANCE INSTRUCTIONS THAT DID NOT IDENTIFY THE TRIPPING OF THE NSSSS ISOLATION LOGIC CHANNELS NOR INCLUDE STEPS TO RESET THE ISOLATION CHANNELS DURING THE SURVEILLANCE TESTS. THESE SVIS SIMULATE THE OPENING OF THE TURBINE STOP VALVES BY JUMPERING ACROSS APPROPRIATE RELAY CONTACTS, WHICH RESULTED IN A NSSSS ISOLATION SIGNAL BEING GENERATED DUE TO LOW MAIN CONDENSER VACUUM. AS A RESULT

OF THIS EVENT, THE SURVEILLANCE INSTRUCTIONS HAVE BEEN REVISED TO PREVENT THE NSSSS ISOLATIONS DURING THEIR PERFORMANCE. ADDITIONALLY, ALL OTHER SURVEILLANCE INSTRUCTIONS INVOLVING THE TURBINE STOP VALVES HAVE BEEN REVIEWED TO ENSURE SIMILAR INADEQUACIES DID NOT EXIST. ONE ADDITIONAL SURVEILLANCE INSTRUCTION WAS REVISED AFTER IT WAS IDENTIFIED THAT UNDER SPECIAL TEST CONDITIONS THE SURVEILLANCE INSTRUCTION COULD RESULT IN A NSSSS ISOLATION. NO ADDITIONAL PROBLEMS WERE IDENTIFIED.

[216] PERRY 1 DOCKET 50-440 LER 87-077
PERSONNEL ERROR RESULTS IN FAILURE TO DEMONSTRATE OPERABILITY OF OFFSITE ELECTRICAL POWER SOURCES.
EVENT DATE: 121687 REPORT DATE: 011588 NSSS: GE TYPE: BWR

(NSIC 207920) ON DECEMBER 16, 1987 AT 1130, CONTROL ROOM OPERATORS FAILED TO DEMONSTRATE THE OPERABILITY OF THE REQUIRED OFFSITE ELECTRICAL POWER SOURCES WITHIN THE 8 HOURS AS REQUIRED BY TECHNICAL SPECIFICATIONS WITH THE DIVISION II DIESEL GENERATOR (DG) INOPERABLE. IN POWER OPERATION, TECHNICAL SPECIFICATION 3.8.1.1 ACTION B REQUIRES THAT, WITH EITHER DG DIVISION I OR DIVISION II INOPERABLE, THE OFFSITE ELECTRICAL POWER SOURCES BE DEMONSTRATED OPERABLE WITHIN 1 HOUR AND AT LEAST ONCE PER 8 HOURS THEREAFTER. ON DECEMBER 16, THE APPROPRIATE SURVEILLANCE INSTRUCTION (SVI) SHOULD HAVE BEEN COMPLETED BY 1130 TO COMPLY WITH THE 8 HOUR FREQUENCY REQUIREMENT, BUT WAS NOT COMPLETED UNTIL 1230. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THE OPERATORS RESPONSIBLE FOR PERFORMING THE SVI WERE COGNIZANT OF THE REQUIREMENT. HOWEVER, THEY BECAME INVOLVED WITH OTHER OPERATIONAL ACTIVITIES AND DID NOT COMPLETE THE SVI WITHIN THE REQUIRED TIME FRAME. THE OPERATIONS PERSONNEL INVOLVED IN THIS EVENT HAVE RECEIVED DISCIPLINARY ACTION, AND THIS EVENT WILL BE INCLUDED IN THE OPERATOR REQUALIFICATION TRAINING PROGRAM.

[217] PILGRIM 1 DOCKET 50-293 LER 87-021
AUTOMATIC START OF "A" EMERGENCY DIESEL GENERATOR DUE TO PROCEDURAL AMBIGUITY.
EVENT DATE: 042887 REPORT DATE: 020188 NSSS: GE TYPE: BWR

(NSIC 208020) ON APRIL 28, 1987, AT 1251 HOURS, AN AUTOMATIC START SIGNAL WAS RECEIVED BOTH THE "A AND B" EMERGENCY DIESEL GENERATORS (EDGS). THE "A" EDG STARTED, BUT DID NOT LOAD. THE "B" EDG DID NOT START DUE TO THE ISOLATION OF ITS AIR START SYSTEM FOR MAINTENANCE ACTIVITIES. AT THE TIME OF THIS EVENT, UTILITY ELECTRICAL MAINTENANCE PERSONNEL WERE TERMINATING ELECTRICAL LEADS FROM THE ANALOG TRIP SYSTEM (ATS) PANELS. THE EDG AUTOMATIC START SIGNAL RESULTED FROM A HIGH DRYWELL PRESSURE TRIP SIGNAL WHEN THE LEADS ASSOCIATED WITH PRESSURE TRANSMITTERS PT1001-89A & 89C WERE LANDED. THE CAUSE OF THE HIGH DRYWELL PRESSURE SIGNAL WAS DETERMINED TO BE IMPROPER VENTING OF TRANSMITTERS PT1001-89A & 89C DURING PERFORMANCE OF TEMPORARY PROCEDURE (TP) 86-52 "ATS TRANSMITTER PRESSURE TESTING". THE IMPROPER VENTING RESULTED FROM AN AMBIGUITY IN THE PROCEDURAL STEP THAT SHOULD HAVE VENTED THE TEST PRESSURE FROM THESE INSTRUMENTS. IMMEDIATE ACTION WAS TAKEN TO SUSPEND THE ELECTRICAL TERMINATION WORK AND VENT THE AFFECTED TRANSMITTERS. TP86-52 WAS REVISED TO REMOVE THE PROCEDURAL AMBIGUITY. THIS EVENT OCCURRED DURING AN EXTENDED OUTAGE WITH THE MODE SWITCH IN THE REFUEL POSITION. THE REACTOR VESSEL WAS DEFUELED AND ALL CONTROL RODS WERE FULLY INSERTED.

[218] PILGRIM 1 DOCKET 50-293 LER 87-014 REV 01
UPDATE ON LOSS OF OFFSITE POWER.
EVENT DATE: 111287 REPORT DATE: 122387 NSSS: GE TYPE: BWR
VENDOR: NUTHERM INTERNATIONAL
VIKING PUMP COMPANY
WESTINGHOUSE ELECTRIC CORP.

(NSIC 207435) ON NOVEMBER 12, 1987, AT 0206 HOURS, A LOSS OF PREFERRED OFFSITE POWER OCCURRED RESULTING IN AN AUTOMATIC START OF THE "A" AND "B" EMERGENCY DIESEL GENERATORS, A PRIMARY AND SECONDARY CONTAINMENT ISOLATION SIGNAL AND AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM (FULL SCRAM TRIP SIGNAL). THE CAUSE OF THE LOSS OF PREFERRED OFFSITE POWER WAS A SERIES OF STORM RELATED FAULTS IN THE POWER TRANSMISSION SYSTEM REMOTE TO THE PILGRIM NUCLEAR POWER STATION (PNPS). THE PNPS SWITCHYARD EQUIPMENT FUNCTIONED AS DESIGNED TO ISOLATE THE STATION AS A RESULT OF THE INCOMING POWER LOSS. HOWEVER, UNANTICIPATED EQUIPMENT INDICATIONS DID OCCUR WHICH DELAYED POWER RESTORATION UNTIL INVESTIGATION AND ASSESSMENTS WERE COMPLETED. AT THE TIME OF THIS EVENT, THE PLANT WAS IN AN EXTENDED OUTAGE AND IN A COLD STABLE CONDITION WITH FUEL LOADED. THE SECONDARY OFFSITE POWER SOURCE WAS UNAVAILABLE DUE TO ONGOING MODIFICATION WORK. THE SAFETY SIGNIFICANCE OF THIS LOSS OF PREFERRED OFFSITE POWER WAS MINIMAL DUE TO THE AUTOMATIC START OF THE 'A' AND 'B' DIESEL GENERATORS. DUE TO THE NEGLIBLE CORE DECAY HEAT LEVELS, REACTOR COOLANT TEMPERATURES REMAINED STABLE DURING THE TEMPORARY LOSS OF SHUTDOWN COOLING. ALL CONTROL RODS WERE FULLY INSERTED AND THE PLANT REMAINED IN A COLD AND STABLE SHUTDOWN CONDITION.

[219] PILGRIM 1 DOCKET 50-293 LER 87-018
AUTOMATIC ACTUATIONS OF PRIMARY CONTAINMENT SYSTEM GROUP 6 ISOLATION VALVES.
EVENT DATE: 112387 REPORT DATE: 011388 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.
TELEDYNE REPUBLIC MFG

(NSIC 207928) ON 11/23, 24, 26/87 AUTOMATIC ACTUATIONS IN THE RWCU SYSTEM PORTION OF THE PCIS OCCURRED. THE ACTUATIONS RESULTED IN THE AUTOMATIC CLOSURE OF ISOLATION VALVES IN THE RWCU SYSTEM PORTION OF THE PRIMARY CONTAINMENT SYSTEM (PCS) THEREBY CAUSING INTERRUPTIONS IN THE OPERATION OF THE RWCU SYSTEM. THE CAUSE FOR THE NOVEMBER 23 AND 26, 1987 ACTUATIONS WAS PARTIALLY CLOGGED PRESSURE GAGE SNUBBERS (HYDRAULIC DAMPENERS) LOCATED IN INSTRUMENTATION SENSING LINES OF SWITCHES THAT ARE PART OF THE RWCU/PCIS LOGIC CIRCUITRY. THE SNUBBERS WERE CLEANED AND THE SIZE OF THE SNUBBER ADJUSTMENT PINS WERE CHANGED. ADDITIONAL CORRECTIVE ACTION HAS BEEN INITIATED. A CORRECTIVE ACTION PROGRAM DOCUMENT WAS ISSUED REQUESTING ENGINEERING ANALYSIS REGARDING PRESSURE GAGE SNUBBERS INSTALLED FOR THE SAME SERVICE IN OTHER INSTRUMENTATION LINES. THE CAUSE FOR THE NOVEMBER 24, 1987 ACTUATION WAS A FAILED COIL IN A LOGIC RELAY THAT IS PART OF THE RWCU/PCIS LOGIC CIRCUITRY. THE RELAY WAS REPLACED AND THE RWCU SYSTEM RETURNED TO SERVICE. BASED ON ANALYSIS, ADDITIONAL RELAYS (OR RELAY COILS) HAVE BEEN SELECTED FOR REPLACEMENT PRIOR TO STARTUP. THE ACTUATIONS OCCURRED DURING AN EXTENDED OUTAGE WHILE IN THE COLD SHUTDOWN CONDITION WITH NEGLIGIBLE CORE DECAY HEAT AND WITH THE MODE SWITCH IN THE SHUTDOWN POSITION.

[220] PILGRIM 1 DOCKET 50-293 LER 87-016
UNPLANNED ACTUATIONS OF PRIMARY CONTAINMENT, SECONDARY CONTAINMENT AND STANDBY GAS TREATMENT SYSTEMS.
EVENT DATE: 112487 REPORT DATE: 122387 NSSS: GE TYPE: BWR

(NSIC 207436) ON NOVEMBER 24, 1987 AT 1155 HOURS UNPLANNED AUTOMATIC ACTUATIONS OF PORTIONS OF THE PRIMARY AND SECONDARY CONTAINMENT SYSTEMS OCCURRED. THE ACTUATIONS WERE THE RESULT OF PLANNED REPLACEMENT OF THE COIL IN A LOGIC RELAY IN THE PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM. THE CAUSE FOR THE ACTUATIONS HAS BEEN ATTRIBUTED TO INADEQUATE ADMINISTRATIVE CONTROLS FOR REPLACEMENT OF THE RELAY COIL. THE REPLACEMENT WAS IMMEDIATELY SUSPENDED PENDING INVESTIGATION. THE INVESTIGATION RESULTED IN THE ESTABLISHMENT OF ADDITIONAL REVIEWS AND CONTROLS PRIOR TO RESUMING REPLACEMENT OF THE COIL. THE COIL AND RELAY WERE REPLACED AND POST WORK TESTING COMPLETED SATISFACTORILY. ADDITIONAL CORRECTIVE ACTIONS HAVE BEEN IDENTIFIED AND ARE BEING TRACKED. THIS EVENT OCCURRED DURING AN EXTENDED OUTAGE WHILE IN THE COLD SHUTDOWN CONDITION WITH NEGLIGIBLE CORE DECAY HEAT AND WITH THE MODE SWITCH IN THE SHUTDOWN POSITION. THE REACTOR VESSEL

WAS REFUELED WITH THE HEAD INSTALLED AND WITH ALL CONTROL RODS FULLY INSERTED IN THE CORE. THERE WERE NO COMPONENT OR SYSTEM FAILURES THAT CAUSED THIS EVENT OR RESULTED FROM THIS EVENT. THIS EVENT POSED NO THREAT TO THE PUBLIC HEALTH AND SAFETY OR TO PLANT OPERATION.

[221] PILGRIM 1 DOCKET 50-293 LER 87-015
UNPLANNED CLOSINGS OF THE SHUTDOWN COOLING SYSTEM ISOLATION VALVES.
EVENT DATE: 120787 REPORT DATE: 010488 NSSS: GE TYPE: BWR

(NSIC 207678) ON DECEMBER 7 AND 8, 1987 ISOLATION SIGNALS TERMINATED THE RHR SHUTDOWN COOLING (SDC) SUBSYSTEM AT 1428 HOURS AND 2145 HOURS RESPECTIVELY. THE PLANT WAS IN COLD SHUTDOWN WITH THE MODE SWITCH IN "REFUEL" ON DECEMBER 7 AND "SHUTDOWN" ON DECEMBER 8, 1987. THE DECEMBER 7, 1987 ISOLATION WAS DUE TO A SPURIOUS ISOLATION SIGNAL WHICH WAS GENERATED WHEN AN INSTRUMENTATION AND CONTROL TECHNICIAN WAS INSTALLING A JUMPER AROUND THE ISOLATION LOGIC TO PERFORM THE REACTOR COOLANT SYSTEM (RCS) HYDROSTATIC TEST. THE DECEMBER 8, 1987 ISOLATION OCCURRED DURING THE HYDROSTATIC TEST WHEN RCS PRESSURE REACHED APPROXIMATELY 100 PSIG. THE INBOARD CONTAINMENT ISOLATION VALVE (MO-1001-50) IN THE SUCTION PIPING FOR THE RHR PUMPS CLOSED. THIS ISOLATION WAS NOT SIGNIFICANT SINCE THE SDC SUBSYSTEM HAD ALREADY BEEN REMOVED FROM SERVICE IN ORDER TO PERFORM THE RCS HYDROSTATIC TEST. CORRECTIVE ACTION CONSISTED OF THE RESTORATION OF SDC FOR THE DECEMBER 7, 1987 ACTUATION AND THE REVISION OF PROCEDURE NO. 2.1.8.1 "CLASS 1 SYSTEM HYDROSTATIC TEST" TO INCLUDE AN ADDITIONAL JUMPER AND CAUTIONARY NOTES FOR THE DECEMBER 8, 1987 ACTUATION. THESE ACTUATIONS WERE NOT SIGNIFICANT BECAUSE OF THE CURRENT EXTENDED COLD SHUTDOWN WHICH HAS RESULTED IN ALMOST NEGLIGIBLE CORE DECAY HEAT. THE PUBLIC HEALTH AND SAFETY WERE NOT AFFECTED BY THESE ACTUATIONS.

[222] PILGRIM 1 DOCKET 50-293 LER 87-019
AUTOMATIC ACTUATION OF PRIMARY CONTAINMENT SYSTEM GROUP 6 ISOLATION VALVE.
EVENT DATE: 121787 REPORT DATE: 011488 NSSS: GE TYPE: BWR

(NSIC 207886) ON DECEMBER 17, 1987 AT 1105 HOURS, AN ACTUATION IN THE REACTOR WATER CLEANUP (RWCU) SYSTEM PORTION OF THE PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM (PCIS) OCCURRED. THE ACTUATION RESULTED IN THE AUTOMATIC CLOSURE OF THE INBOARD RHCU/PRIMARY CONTAINMENT SYSTEM ISOLATION VALVE (MO-1201-2) AND THEREBY CAUSED A TEMPORARY INTERRUPTION IN THE OPERATION OF THE RWCU SYSTEM. THE ACTUATION OCCURRED DURING THE INSTALLATION OF A CALIBRATED TEMPERATURE SWITCH IN THE INBOARD PCIS LOGIC CIRCUIT THAT CONTROLS THE ISOLATION VALVE. DURING THE INSTALLATION, THE TEMPERATURE SWITCH BECAME INADVERTENTLY GROUNDED AND CAUSED THE LOGIC CIRCUIT'S FUSE TO BLOW. THE CAUSE FOR THE ACTUATION WAS NON-LICENSED UTILITY PERSONNEL ERROR. THE FUSE WAS REPLACED AND THE RHCU SYSTEM RETURNED TO SERVICE APPROXIMATELY 20 MINUTES AFTER THE ACTUATION. LONG TERM CORRECTIVE ACTIONS HAVE BEEN IDENTIFIED AND WILL BE TRACKED. THE CORRECTIVE ACTIONS ARE EXPECTED TO REDUCE THE LIKELIHOOD OF A SIMILAR EVENT IN THE FUTURE. THE ACTUATION OCCURRED DURING AN EXTENDED OUTAGE WHILE IN THE COLD SHUTDOWN CONDITION WITH THE MODE SWITCH IN THE SHUTDOWN POSITION AND WITH NEGLIGIBLE CORE DECAY HEAT. THE REACTOR VESSEL HEAD WAS INSTALLED AND THE CONTROL RODS WERE INSERTED IN THE CORE AT THE TIME OF THE ACTUATION. THE INTERRUPTION IN THE OPERATION OF THE RWCU SYSTEM POSED NO THREAT TO THE PUBLIC HEALTH AND SAFETY OR TO PLANT OPERATION.

[223] PILGRIM 1 DOCKET 50-293 LER 87-020
FAILURE OF FIRE DAMPER TO CLOSE DUE TO ORIENTATION OF CLOVER HOOKS.
EVENT DATE: 121787 REPORT DATE: 011488 NSSS: GE TYPE: BWR
VENDOR: AIR BALANCE, INC.

(NSIC 207887) ON DECEMBER 17, 1987, AT 1015 HOURS, THREE FIRE DAMPERS CPR-2, 4, AND 5 WERE ACTUATED WHEN THEIR FUSIBLE LINKS WERE INADVERTENTLY ENERGIZED DURING

PERFORMANCE OF PROCEDURE 8.B.4, SECTION I, "PHOTOELECTRIC SMOKE DETECTOR FUNCTIONAL TESTS". FOLLOWING THESE DAMPER ACTUATIONS IT WAS IDENTIFIED THAT FIRE DAMPER CPR-2 FAILED TO FULLY CLOSE DUE TO THE ORIENTATION OF THE CLOVER HOOKS USED TO ATTACH THE FUSIBLE LINK TO THE DAMPER. DAMPERS CPR 4 AND 5 DID FULLY CLOSE. FIRE DAMPER CPR-2 DID NOT CLOSE DUE TO THE CLOVER HOOK CATCHING ON THE EDGE OF THE DAMPER BLADE PREVENTING IT FROM CLOSING. FIELD REVISION NOTICE (FRN) 86-31-129 WAS ISSUED TO CHANGE THE PHYSICAL ORIENTATION OF THE CLOVER HOOKS TO FACE OUTWARD (AWAY) FROM THE DAMPER. IN ADDITION, THE ORIENTATION OF THE CLOVER HOOKS ON THE OTHER FIRE DAMPERS WAS INSPECTED. AT THE TIME OF THIS EVENT, THE PLANT WAS IN AN EXTENDED OUTAGE WITH THE MODE SWITCH IN SHUTDOWN AND CONTROL RODS FULLY INSERTED INTO THE CORE. THE PUBLIC HEALTH AND SAFETY WERE NOT AFFECTED BY THESE EVENTS. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73 (A)(2)(I)(B).

[224] POINT BEACH 2 DOCKET 50-301 LER 87-006
POTENTIAL LOSS OF CONTAINMENT INTEGRITY DUE TO MISADJUSTED VALVE.
EVENT DATE: 121987 REPORT DATE: 012788 NSSS: WE TYPE: PWR
VENDOR: FISHER CONTROLS CO.

(NSIC 208021) ON DECEMBER 19, 1987, DURING AN INSTRUMENT AIR SYSTEM WALKDOWN, THE HANDWHEEL GAG ON THE INSTRUMENT AIR CONTAINMENT ISOLATION VALVE, IA- 3048, WAS FOUND TO BE PARTIALLY ENGAGED SUCH THAT IF THE VALVE WERE AUTOMATICALLY CLOSED, IT WOULD NOT HAVE FULLY SHUT. THE GAG WAS IMMEDIATELY REMOVED, AND THE VALVE WAS TESTED TO VERIFY OPERABILITY. THE INSTRUMENT AIR CONTAINMENT ISOLATION VALVE IS AN AIR-OPERATED VALVE WHICH CLOSES UPON A CONTAINMENT ISOLATION SIGNAL. IT IS IN SERIES WITH A CHECK VALVE. THE CHECK VALVE DID NOT LEAK DURING THE TESTING OF IA-3048. THE GAGS ON ALL CONTAINMENT ISOLATION VALVES WERE VERIFIED TO BE DISENGAGED, AND RED LOCKS WERE PLACED ON THEM TO CONTROL THEIR OPERATION ADMINISTRATIVELY. THIS EVENT OCCURRED BECAUSE A LOCKNUT ON THE VALVE GAG MECHANISM BECAME LOOSE AND OUT OF POSITION, THUS PREVENTING THE GAG FROM BEING COMPLETELY REMOVED. TOTAL CONTAINMENT LEAKAGE WAS NOT AFFECTED BECAUSE THE IN-SERIES CHECK VALVE DID NOT LEAK; THEREFORE, THERE WERE NO ADVERSE SAFETY CONSEQUENCES.

[225] PRAIRIE ISLAND 1 DOCKET 50-282 LER 85-013 REV 01
UPDATE ON INOPERABILITY OF SEVERAL SAFEGUARDS VALVES CAUSED BY MOTOR OPERATED VALVE FAILURE.
EVENT DATE: 091585 REPORT DATE: 010788 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.
LIMITORQUE CORP.
VELAN VALVE CORP.

(NSIC 207663) ON SEPTEMBER 15, 1985, UNIT 1 WAS RECOVERING FROM A TRIP AND WAS AT 2% POWER. OPERATORS WERE IN THE PROCESS OF ALIGNING THE MAIN FEEDWATER SYSTEM FOR OPERATION. ONE VALVE, MV-32024, FEEDWATER TO 1B STEAM GENERATOR CONTAINMENT ISOLATION VALVE, FAILED TO OPEN. DURING THE THIRD ATTEMPT TO OPEN THE VALVE, THE FEEDER BREAKER TO ITS MOTOR CONTROL CENTER (MCC) TRIPPED, RESULTING IN LOSS OF POWER TO SEVERAL SAFEGUARDS MOTOR-OPERATED VALVES IN VARIOUS SYSTEMS; ONLY ONE TRAIN WAS AFFECTED. INVESTIGATION SHOWED ALSO THAT THE BREAKER FOR MV-32024 WAS TRIPPED. THERMAL OVERLOAD RELAYS FOR MV-32024 WERE HEAT-DAMAGED, AND THE VALVE OPERATOR MOTOR ITSELF WAS DAMAGED BY OVERHEATING. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(VII). HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED SINCE ALL REDUNDANT EQUIPMENT FOR THE AFFECTED SYSTEMS REMAINED OPERABLE. CORRECTIVE ACTIONS TAKEN WERE AS FOLLOWS: STARTUP OPERATIONS WERE DISCONTINUED UNTIL THE MCC WAS RETURNED TO SERVICE AND MV-32024 WAS MADE OPERABLE. AN INVESTIGATION INTO ELECTRICAL PROTECTION COORDINATION BETWEEN MCC FEEDER BREAKERS AND LOAD BREAKERS WAS COMPLETED; MODIFICATIONS ARE UNDERWAY. INSTRUCTIONS FOR RESETTING OF BREAKERS WERE CLARIFIED. SEVERAL PROJECTS ARE UNDERWAY TO IMPROVE FEEDWATER SYSTEM DESIGN AND OPERATION.

[226] PRAIRIE ISLAND 1 DOCKET 50-282 LER 87-002
UNIT SHUTDOWN RESULTING FROM STEAN GENERATOR TUBE LEAKAGE.
EVENT DATE: 022087 REPORT DATE: 012788 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 207957) ON 2/20/87, UNIT 1 WAS AT 97% POWER, COASTING DOWN TO REFUELING. AT 2045, A NORMAL SHUTDOWN TO COLD SHUTDOWN WAS STARTED DUE TO A STEAM GENERATOR PRIMARY-TO-SECONDARY LEAK WHICH HAD INCREASED OVER A PERIOD OF SEVERAL DAYS TO 0.33 GPM AS MEASURED BY XENON ACTIVITY AT THE CONDENSER AIR EJECTOR. RADIOCHEMISTRY ANALYSIS OF STEAM GENERATOR BLOWDOWN SAMPLES ON FEBRUARY 20TH CONFIRMED THAT THE LEAK WAS IN NO. 12 STEAM GENERATOR. ON FEBRUARY 23, R16C41 WAS IDENTIFIED AS THE LEAKING TUBE USING A CAMERA MOUNTED IN THE CHANNELHEAD. LEAKAGE WAS 1 DROP PER 29 SECONDS AT 225 PSIG NITROGEN ON THE SECONDARY SIDE AND 13 DROPS PER MINUTE AT 450 PSIG. A 100% EDDY CURRENT TEST PROGRAM WAS DONE; 19 TUBES WERE PLUGGED. THE PRAIRIE ISLAND WESTINGHOUSE MODEL 51 STEAM GENERATORS WERE FABRICATED WITH A 2 1/4 INCH ROLLED REGION AT THE BOTTOM OF THE TUBESHEET AND AN OPEN CREVICE BETWEEN THE TUBE AND TUBESHEET HOLE FOR THE REMAINDER OF THE 22 INCH THICK TUBESHEET. IMPURITIES IN STEAM GENERATOR BULK WATER CONCENTRATE IN THE CREVICE REGIONS AND CAN CAUSE INTERGRANULAR CORROSION OF THE MILL-ANNEALED ALLOY 600 STEAM GENERATOR TUBING. A CAUSTIC CREVICE ENVIRONMENT IS THE MOST LIKELY CAUSE OF THIS TYPE OF CORROSION AT PRAIRIE ISLAND. TECH SPEC 3.1.C.6 REQUIRES UNIT SHUTDOWN TO COLD SHUTDOWN AND AN INSERVICE STEAM GENERATOR TUBE INSPECTION WHENEVER PRIMARY-TO- SECONDARY LEAKAGE EXCEEDS 1.0 GALLONS PER MINUTE.

[227] QUAD CITIES 1 DOCKET 50-254 LER 87-003 REV 01
UPDATE ON RCIC INOPERABLE DUE TO FLOW CONTROLLER FAILURE CAUSED BY LOOSE SOLDER JOINT.
EVENT DATE: 020587 REPORT DATE: 012988 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 208015) AT 1050 HOURS, ON FEBRUARY 5, 1987, UNIT ONE WAS IN THE RUN MODE AT 100 PERCENT CORE THERMAL POWER. WHILE PERFORMING QOS 1300-S2 (REACTOR CORE ISOLATION COOLING (RCIC) PUMP OPERABILITY TEST), IT WAS DISCOVERED THAT THE FLOW CONTROLLER DID NOT RESPOND TO AUTOMATIC FLOW CONTROL SIGNALS. HOWEVER, IT DID WORK SATISFACTORILY IN MANUAL. RCIC WAS DECLARED INOPERABLE AND APPROPRIATE NOTIFICATION PER 10CFR50.72 AND OPERABILITY TESTING PER TECHNICAL SPECIFICATION 3.5.E.2 WAS COMPLETED. THE EXACT CAUSE FOR THIS FLOW CONTROLLER FAILURE WAS DETERMINED TO BE A LOOSE COLD SOLDER JOINT IN THE SETPOINT TAPE CHASSIS SECTION OF THE ELECTRONIC CONTROLLER. A REPLACEMENT FLOW CONTROLLER, CONTAINING A NEW POWER SUPPLY AND SETPOINT TAPE CONTROLLER WAS INSTALLED BY THE IMD. THE ORIGINAL CONTROLLER WAS REPAIRED, CALIBRATED AND REINSTALLED ON MARCH 14, 1987 WHILE UNIT ONE WAS IN COLD SHUTDOWN. RCIC WAS VERIFIED OPERABLE AT 0310 HOURS ON MARCH 16, 1987, DURING THE UNIT ONE STARTUP. SIMILAR FLOW CONTROLLERS WILL BE INSPECTED TO ENSURE NO OTHER PROBLEMS EXIST. THIS REPORT IS SUBMITTED TO SATISFY THE REQUIREMENTS OF 10CFR50.73(A)(2)(V).

[228] QUAD CITIES 1 DOCKET 50-254 LER 87-018 REV 01
UPDATE ON RECIRCULATION MOTOR GENERATOR FIELD BREAKER FAILURE TO TRIP DUE TO UNKNOWN CAUSE.
EVENT DATE: 091387 REPORT DATE: 011188 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 208016) ON SEPTEMBER 13, 1987 QUAD CITIES UNIT ONE WAS IN THE SHUTDOWN MODE AT 0% THERMAL POWER. AT 0228 HOURS, WHILE ATTEMPTING TO SHUTDOWN THE 1B RECIRCULATION MOTOR GENERATOR SET FOR INSPECTION, IT WAS FOUND THAT THE ASSOCIATED FIELD BREAKER FAILED TO TRIP. THE TRIP COIL WAS FOUND BURNT OUT. THE ARMATURE IN THE CLOSING LINKAGE WAS FOUND TO BE BINDING AND HAD TO BE FREED BY OPERATING PERSONNEL. NRC NOTIFICATION WAS COMPLETED AT 0600 HOURS TO SATISFY THE REQUIREMENTS OF 10 CFR 50.72. THE RECIRCULATION MG SET FIELD BREAKERS ARE

INVOLVED IN THE ANTICIPATED TRANSIENT WITHOUT A SCRAM (ATWS) SYSTEM AND ARE DESIGNED TO TRIP UPON EITHER A LOW LOW REACTOR WATER LEVEL OR HIGH REACTOR PRESSURE. THE CAUSE FOR THE BREAKER'S FAILURE TO TRIP COULD NOT BE DETERMINED BY GENERAL ELECTRIC. VARIOUS COMPONENTS WERE REPLACED AND THE BREAKER HAS BEEN REINSTALLED. A PROCEDURE REVISION WILL BE COMPLETED TO IDENTIFY THE RECOMMENDED LUBRICANT. IN ADDITION, ACTION ITEM RECORD 4-87-13 IS INITIATED TO INVESTIGATE THE SUITABILITY OF APPLICATION FOR THIS FIELD BREAKER. THIS REPORT IS PROVIDED TO COMPLY WITH 10 CFR 50.73(A)(2)(V)(D).

[229] QUAD CITIES 1 DOCKET 50-254 LER 87-025
CONTROL ROOM HABITABILITY STUDY DESIGN BASIS ASSUMPTION ERROR RESULTS IN EXCEEDING ALLOWABLE FILTER EFFICIENCY.
EVENT DATE: 112587 REPORT DATE: 122287 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 207500) ON NOVEMBER 25, 1987, QUAD CITIES UNIT ONE WAS IN THE REFUEL MODE AT 0 PERCENT POWER AND UNIT TWO WAS IN THE RUN MODE AT 98 PERCENT POWER. AT 1505 HOURS, THE STATION WAS NOTIFIED OF A DESIGN BASIS ASSUMPTION USED IN THE CONTROL ROOM HABITABILITY STUDY WHICH CONFLICTS WITH CURRENT TECH SPEC REQUIREMENTS. TECH SPECS REQUIRE 90 PERCENT IODINE REMOVAL EFFICIENCIES FOR BOTH STANDBY GAS TREATMENT SYSTEMS (SBGTS) AND THE CONTROL ROOM VENTILATION AIR FILTRATION UNIT. THE STUDY PERFORMED FOR CONTROL ROOM HABITABILITY ASSUMED 99 PERCENT EFFICIENCIES FOR THESE TRAINS. NRC NOTIFICATION OF THIS CONDITION WAS COMPLETE AT 1528 HOURS. THE CAUSE OF THIS CONDITION IS DUE TO AN ANALYSIS DEFICIENCY IN THAT INADEQUATE DESIGN REVIEW WAS PERFORMED. A REVIEW OF RECORDS INDICATES THAT CURRENTLY THESE TRAINS MEET THE REQUIREMENTS OF THE CONTROL ROOM HABITABILITY STUDY; HOWEVER, TWO OCCASIONS WERE IDENTIFIED WHERE THE 1/2B SBGTS DID NOT MEET THESE REQUIREMENTS. A SUPPLEMENTAL REPORT WILL BE ISSUED TO DESCRIBE THE RESOLUTION OF THE DISCREPANCIES IDENTIFIED. THIS REPORT IS PROVIDED PER 10CFR50.73(A)(2)(II).

[230] QUAD CITIES 1 DOCKET 50-254 LER 87-026 REV 01
UPDATE ON PIPING SUPPORT OUTSIDE COMPLIANCE WITH SAFETY ANALYSIS REPORT DUE TO DESIGN/CONSTRUCTION ERROR.
EVENT DATE: 113087 REPORT DATE: 011388 NSSS: GE TYPE: BWR

(NSIC 207968) ON 12/28/87, QUAD CITIES UNIT ONE WAS IN THE RUN MODE AT 15% REACTOR THERMAL POWER. AT 1335 HOURS, THE STATION WAS NOTIFIED THAT TWO PIPING SUPPORTS LOCATED ON REACTOR CORE ISOLATION COOLING (RCIC) SUCTION LINE DID NOT COMPLY WITH THE FINAL SAFETY ANALYSIS REPORT (FSAR) CRITERIA FOR ALLOWABLE STRESS. HOWEVER, THE SYSTEM WAS OPERABLE. THIS EVENT WAS REPORTED TO NRC REGION III IN ACCORDANCE WITH THE AGREEMENT FOR THE PIPING CONFIGURATION VERIFICATION PROGRAM. PREVIOUSLY, IN LER 254/87-026, REVISION 00, A PIPING SUPPORT ON 1B CORE SPRAY DISCHARGE LINE WAS FOUND TO BE BEYOND FSAR CRITERIA FOR ALLOWABLE STRESS. THE PCVP IS ONGOING AND A SUPPLEMENT WILL BE PROVIDED UPON PROGRAM COMPLETION. THE CAUSE OF THESE CONDITIONS IS CONSTRUCTION/DESIGN ERROR DURING A MODIFICATION IN 1980 BECAUSE THE AS-BUILT CONFIGURATION WAS NOT IN CONFORMANCE WITH AS-DESIGNED/ENGINEERED DRAWINGS USED FOR THE ORIGINAL PIPING STRESS ANALYSIS. CORRECTIVE ACTION WAS TO SHIM THE EXCESS CLEARANCES BETWEEN THE PIPING LUGS AND THE SUPPORT'S WIDE FLANGE, STRUT REMOVAL AND SUPPORT RELOCATION. THE NEW MODIFICATION PROGRAM IN EFFECT SHOULD PREVENT RECURRENCE. THIS REPORT IS PROVIDED PER 10CFR50.73(A)(2)(II).

[231] QUAD CITIES 1 DOCKET 50-254 LER 87-027
CONTROL ROOM VENTILATION ISOLATIONS DUE TO TOXIC GAS ANALYZER TRIPS.
EVENT DATE: 120187 REPORT DATE: 122987 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 207503) ON DECEMBER 1, 1987, QUAD CITIES UNIT ONE WAS IN THE REFUEL MODE AT

0% POWER AND UNIT TWO WAS IN THE RUN MODE AT 95% POWER. AT 2200 HOURS, THE CONTROL ROOM HEATING, VENTILATION, AIR CONDITIONING (HVAC) SYSTEM AUTOMATICALLY ISOLATED WHEN THE TOXIC GAS ANALYZER SAMPLE POINT SELECTOR POSITION WAS CHANGED. THE SWITCH POSITION REQUIRED CHANGING BECAUSE THE APPROPRIATE PROCEDURE WAS NOT USED WHEN THE HVAC SYSTEM WAS RESET FOLLOWING MAINTENANCE ON NOVEMBER 30, 1987. THE ISOLATION PROBABLY OCCURRED DUE TO A SMALL AMOUNT OF MOISTURE PRESENT IN THE SAMPLE LINE. ON DECEMBER 2, 1987, AT 2331 HOURS, DURING A TRAINING SESSION BEING CONDUCTED AS CORRECTIVE ACTION FOR THE PREVIOUS DAY'S EVENT, THE HVAC ISOLATED AGAIN WHEN IT WAS NOTED THAT THE SULFUR DIOXIDE MONITOR RANGE SWITCH WAS THOUGHT TO BE MISPOSITIONED. WHEN THE RANGE SWITCH WAS RANGED DOWN, THE ISOLATION OCCURRED. THE CAUSE FOR THIS EVENT WAS LACK OF PROCEDURAL GUIDANCE AND TRAINING. NRC NOTIFICATION OF BOTH THESE EVENTS WAS COMPLETED AS REQUIRED BY 10 CFR 50.72(B)(2)(II). CORRECTIVE ACTIONS INCLUDE INSTALLATION OF PERMANENT SIGNS, UPGRADED TRAINING, PROCEDURE REVISIONS, A POSSIBLE MODIFICATION. THIS REPORT IS PROVIDED PER 10 CFR 50.73(A)(2)(IV).

[232] QUAD CITIES 1 DOCKET 50-254 LER 87-031
FAILURE OF HPCI MINIMUM FLOW VALVE TO OPEN DUE TO AIR IN FLOW SWITCH SENSING LINES BECAUSE OF OUTAGE SCHEDULING DEFICIENCY.
EVENT DATE: 122387 REPORT DATE: 011488 NSSS: GE TYPE: BWR

(NSIC 207969) AT 1310 HOURS, ON DECEMBER 23, 1987, QUAD CITIES UNIT ONE WAS PERFORMING A VALVE OPERABILITY TEST (QOS 2300-3) ON THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM AS PART OF NORMAL UNIT STARTUP TESTING. DURING THIS TEST, HPCI WAS DECLARED INOPERABLE DUE TO THE FAILURE OF THE HPCI PUMP MINIMUM FLOW VALVE (MO-1-2301-14) TO AUTOMATICALLY OPEN WHEN THE HPCI TURBINE INLET VALVE MO-1-2301-3 WAS OPENED. NRC NOTIFICATION OF THIS EVENT PER 10 CFR 50.72 WAS COMPLETED AT 1445 HOURS. THE CAUSE OF THIS PROBLEM WAS DUE TO THE PRESENCE OF AIR IN THE SENSING LINES TO THE FLOW SWITCH THAT CONTROLS THE HPCI MINIMUM FLOW VALVE. THE SENSING LINES WERE BACKFILLED AND HPCI WAS DEEMED OPERABLE AT 1450 HOURS AFTER VERIFYING PROPER VALVE OPERATION. THE ROOT CAUSE FOR THIS EVENT IS MANAGEMENT DEFICIENCY BECAUSE NO METHOD WAS USED TO ENSURE THESE INSTRUMENT SENSING LINES WERE BACKFILLED PRIOR TO UNIT RESTART. THE WORK PLANNING DEPARTMENT IS TO INCORPORATE BACKFILLING OF HPCI INSTRUMENT LINES TO INSTRUMENT MAINTENANCE OUTAGE TASKS. THIS REPORT IS PROVIDED PER 10CFR50.73(A)(2)(V)(D).

[233] QUAD CITIES 1 DOCKET 50-254 LER 87-032
RCIC INOPERABLE DUE TO CHECK VALVE STUCK CLOSED BECAUSE OF WORN PARTS.
EVENT DATE: 122387 REPORT DATE: 011388 NSSS: GE TYPE: BWR
VENDOR: ROCKWELL MANUFACTURING COMPANY

(NSIC 208001) ON DECEMBER 23, 1987 QUAD CITIES UNIT ONE WAS OPERATING IN THE RUN MODE AT APPROXIMATELY 12% THERMAL POWER. AT 1653 HOURS, WHILE PERFORMING REACTOR CORE ISOLATION COOLING (RCIC) MANUAL INITIATION TEST PER TEMPORARY PROCEDURE 5156, IT WAS DISCOVERED THAT THE RCIC PUMP WOULD NOT INJECT WATER INTO THE REACTOR VESSEL. NRC NOTIFICATION PER 10CFR50.72 WAS COMPLETED AT 1840 HOURS. AN INVESTIGATION OF THIS EVENT REVEALED THAT THE RCIC PUMP DISCHARGE CHECK VALVE FAILED TO OPEN. MAINTENANCE PERSONNEL WERE ABLE TO JAR THE VALVE LOOSE AND THEN RCIC INJECTED INTO THE VESSEL AS REQUIRED. THE CHECK VALVE WAS DISASSEMBLED AND THE CAUSE FOR THIS EVENT WAS DUE TO WORN HINGE PINS AND VALVE BUSHING WHICH ALLOWED THE VALVE DISK TO BECOME WEDGED INTO THE DISC SEAT. THE VALVE WAS REPAIRED, RE-ASSEMBLED, AND TESTED. RCIC WAS DECLARED OPERABLE ON DECEMBER 28, 1987. THIS IS THE FIRST OCCURRENCE OF THIS TYPE. THIS REPORT IS PROVIDED PER 10CFR50.73(A)(2)(V)(B).

[234] QUAD CITIES 1 DOCKET 50-254 LER 87-033
INADVERTENT CONTROL ROD SCRAM DURING SCRAM TIMING DUE TO TEST PANEL DESIGN DEFICIENCY AND PERSONNEL ERROR (OPERATOR BUMPED TEST SWITCH).
EVENT DATE: 122687 REPORT DATE: 011488 NSSS: GE TYPE: BWR

(NSIC 208002) ON DECEMBER 26, 1987, QUAD CITIES UNIT ONE WAS IN THE RUN MODE AT 24% THERMAL POWER. AT 1048 HOURS, WHILE PERFORMING HOT SCRAM TIMING UTILIZING THE CONTROL ROD SCRAM TEST PANEL, AN INADVERTENT CONTROL ROD SCRAM OCCURRED WHEN CONTROL ROD H-9 (30-35) WAS SCRAMMED PER PROCEDURE. A DIAGONALLY ADJACENT CONTROL ROD, G-10 (26-39) WAS OBSERVED TO SCRAM TO ITS FULLY INSERTED POSITION ALONG WITH CONTROL ROD H-9. CONTROL ROD G-10 WAS RESTORED TO ITS IN-SEQUENCE POSITION PER APPROPRIATE PROCEDURES AT 1052 HOURS. NRC NOTIFICATION OF THIS EVENT WAS COMPLETED AT 1120 HOURS PER 10 CFR 50.72. THE CAUSE OF THIS EVENT IS A COMBINATION OF PERSONNEL ERROR AND THE COMPACT DESIGN OF THE CONTROL ROD SCRAM TEST PANEL. THE ADJACENT CONTROL ROD WAS INADVERTENTLY BUMPED DURING SCRAM TIMING BECAUSE THE TEST SWITCHES ON THE PANEL ARE SMALL AND CLOSELY SPACED. THE CORRECTIVE ACTION WAS TO RETURN THE CONTROL ROD TO ITS PRESCRAM POSITION AND SUBSEQUENTLY REVIEWING THIS EVENT WITH EACH ON-COMING SHIFT. FURTHER CORRECTIVE ACTION WILL BE A PROCEDURE REVISION AND INSTALLATION OF A CAUTION SIGN. THIS REPORT IS PROVIDED PER 10 CFR 50.73(A)(2)(IV).

[235] QUAD CITIES 2 DOCKET 50-265 LER 87-005
UNIT TWO GENERATOR/TURBINE TRIP/REACTOR SCRAM DUE TO SPURIOUS ACTUATION OF TRANSFORMER SUDDEN PRESSURE RELAY.
EVENT DATE: 032187 REPORT DATE: 040987 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 208033) ON MARCH 21, 1987, QUAD CITIES UNIT TWO WAS IN THE RUN MODE OPERATING AT APPROXIMATELY 100% POWER. AT 0143 HOURS, THE UNIT TWO MAIN GENERATOR TRIPPED DUE TO AN ACTUATION OF THE MAIN TRANSFORMER SUDDEN PRESSURE RELAY. THIS RESULTED IN A REACTOR SCRAM DUE TO A TURBINE/GENERATOR LOAD MISMATCH. A NORMAL SCRAM RECOVERY FOLLOWED. TESTING BY THE OPERATIONAL ANALYSIS DEPARTMENT (OAD) REVEALED NO PROBLEMS WITH THE MAIN TRANSFORMER, THE SUDDEN PRESSURE RELAY OR ITS ASSOCIATED INHIBIT RELAY. BOTH RELAYS WERE REPLACED LIKE FOR LIKE AND UNIT TWO STARTUP WAS INITIATED AT 2148 HOURS ON MARCH 21, 1987. NO PROBLEMS WERE ENCOUNTERED AND THE UNIT TWO GENERATOR WAS SYNCHRONIZED AT 1150 HOURS ON MARCH 22, 1987. COMMONWEALTH EDISON'S TECHNICAL CENTER IS CONTINUING TO ANALYZE AND TEST THE RELAYS INVOLVED WITH THIS EVENT IN AN EFFORT TO DISCOVER THE ROOT CAUSE FOR THE RELAY ACTUATION. THIS REPORT IS SUBMITTED TO COMPLY WITH 10CFR50.73(A)(2)(IV).

[236] QUAD CITIES 2 DOCKET 50-265 LER 87-019
PIPING SUPPORTS OUTSIDE COMPLIANCE WITH SAFETY ANALYSIS REPORT DUE TO DESIGN ERROR.
EVENT DATE: 111687 REPORT DATE: 120487 NSSS: GE TYPE: BWR

(NSIC 207505) ON NOVEMBER 16, 1987, QUAD CITIES UNIT TWO WAS IN THE RUN MODE AT 92% THERMAL POWER. AT 1645 HOURS, THE STATION WAS NOTIFIED THAT THREE PIPING SUPPORTS (TWO ON RESIDUAL HEAT REMOVAL (RHR) AND ONE ON HIGH PRESSURE COOLANT INJECTION (HPCI)) DID NOT COMPLY WITH THE FINAL SAFETY ANALYSIS REPORT (FSAR) CRITERIA FOR ALLOWABLE STRESS. ALL SYSTEMS WERE STILL OPERABLE. THIS EVENT WAS REPORTED VIA THE EMERGENCY NOTIFICATION SYSTEM AT 1740 HOURS TO COMPLY WITH 10CFR50.72. THE CAUSE FOR THIS SITUATION IS DESIGN ERROR DURING A MODIFICATION IN 1980 BECAUSE AS-BUILT CONFIGURATIONS WERE NOT ACCURATELY DOCUMENTED ON DRAWINGS USED FOR THE ORIGINAL PIPING STRESS ANALYSIS. CORRECTIVE ACTIONS INVOLVE ADJUSTMENTS TO VARIABLE SPRING CANS AND INSTALLATION OF A RIGID STRUT SUPPORT. THE NEW MODIFICATION PROGRAM IN EFFECT SHOULD PREVENT RECURRENCE. THIS REPORT IS PROVIDED PER 10CFR50.73(A)(2)(II).

[237] RANCHO SECO DOCKET 50-312 LER 87-027 REV 01
UPDATE ON B&W RPS SHUTDOWN BYPASS KEYS USED WHILE NOT PERMITTED BY TECHNICAL SPECIFICATIONS.
EVENT DATE: 041987 REPORT DATE: 021788 NSSS: BW TYPE: PWR

(NSIC 208023) TECHNICAL SPECIFICATION 3.5.1.4 SPECIFICALLY PROHIBITS THE USE OF THE "SHUTDOWN BYPASS" KEY(S) DURING REACTOR POWER OPERATION. DURING COLD SHUTDOWN CONDITIONS, THE DISTRICT IDENTIFIED INSTANCES WHERE THAT TECHNICAL SPECIFICATION WAS NOT FOLLOWED DURING A REACTOR PROTECTION SYSTEM (RPS) CHANNEL CALIBRATION. BECAUSE THE SAFETY REVIEW COMMITTEE APPROVED THE SURVEILLANCE PROCEDURES INVOLVED, AND THE PROCEDURES REQUIRE AN ACTION SPECIFICALLY PROHIBITED BY TECHNICAL SPECIFICATIONS FOR ALL FOUR SAFETY TRAINS, THIS EVENT IS REPORTABLE PURSUANT TO 10 CFR PART 50.73(A)(2)(I)(B). SURVEILLANCE PROCEDURES I.108A/B/C/D HAVE BEEN REVISED TO TEST THE "SHUTDOWN BYPASS" FUNCTION MONTHLY ONLY WHILE THE REACTOR IS NOT CRITICAL. IN ADDITION, THE LIMITS AND PRECAUTION SECTIONS OF THESE PROCEDURES HAVE BEEN REVISED TO BE CONSISTENT WITH TECHNICAL SPECIFICATION 3.5.1.4. PROCEDURE B.4, REVISION 40, "PLANT SHUTDOWN AND COOLDOWN" HAS BEEN REVISED TO INCLUDE NOTIFICATION OF THE PLANT INSTRUMENT AND CONTROL GROUP TO TEST THE SHUTDOWN PRESSURE SETPOINT.

[238] RANCHO SECO DOCKET 50-312 LER 87-025 REV 01
UPDATE ON DECAY HEAT THERMAL RELIEF VALVES DID NOT MEET ACCEPTANCE CRITERIA DURING AS FOUND TESTING.
EVENT DATE: 051087 REPORT DATE: 012788 NSSS: BW TYPE: PWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV

(NSIC 207960) DURING COLD SHUTDOWN CONDITIONS, THREE THERMAL RELIEF VALVES ON THE DECAY HEAT REMOVAL SYSTEM WERE FOUND TO BE OUTSIDE OF THE ACCEPTANCE CRITERION FOR PRESSURE RELIEF. TECHNICAL SPECIFICATION 4.5.1.2.A COMMITS THE DISTRICT TO ASME SECTION XI, EXCEPT WHERE SPECIFIC RELIEF WAS GRANTED. BECAUSE THE PLANT WAS SHUTDOWN WITH BOTH STEAM GENERATORS OPERABLE, AND THE DECAY HEAT SYSTEM TRAINS WERE IN A PLANNED OUTAGE DURING THIS EVENT, TECHNICAL SPECIFICATIONS LCOS 3.3.1.A.4 AND 3.1.1.5.A FOR DECAY HEAT SYSTEM OPERABILITY DID NOT APPLY. THE THREE THERMAL RELIEF VALVES HAVE SAFETY SIGNIFICANCE. TWO VALVES DID, IN FACT, RELIEVE BELOW THE RATED HYDROSTATIC TEST PRESSURE OF THE DECAY HEAT SYSTEM. ONE VALVE DID NOT LIFT DURING THE AS-FOUND TEST. THESE RELIEF VALVE PROBLEMS WERE UNCOVERED DURING THE NORMAL COURSE OF ASME SECTION XI TESTING PERFORMED ON THE REQUIRED PLANT SYSTEMS. AS A DIRECT RESULT OF THIS EVENT, TWO OTHER VALVES IN THE DESIGNATED GROUP WERE TESTED RE: WR #127426 AND 133367 AND FOUND TO BE UNACCEPTABLE. THE GROUPING OF VALVES IS PLANT SYSTEM BASED. THE VALVE THAT WAS FUNCTIONING AS PSV-26101 DURING THIS EVENT WILL BE REPAIRED PRIOR TO RETURNING THE "B" TRAIN OF DECAY HEAT TO OPERABLE STATUS.

[239] RANCHO SECO DOCKET 50-312 LER 87-029 REV 01
UPDATE ON FIRE PROTECTION PROGRAM DEFICIENCIES WITH RESPECT TO TECHNICAL SPECIFICATIONS AND COMMITMENTS.
EVENT DATE: 051687 REPORT DATE: 011488 NSSS: BW TYPE: PWR

(NSIC 207929) THE SURVEILLANCE TESTING OF UNSUPERVISED CIRCUITS BETWEEN THE PYROTRONICS OR NOTIFIER PANELS AND THE AUXILIARY RELAYS USED TO TRANSMIT SIGNALS TO THE CONTROL ROOM WAS NOT ADEQUATELY PERFORMED IN CONFORMANCE WITH TECHNICAL SPECIFICATION 4.18.1.3. THIS SYSTEM WAS CORRECTED BY ESTABLISHING A CONTINUOUS FIRE WATCH POST AT THE LOCAL FIRE PANELS TO MONITOR FIRE ALARM SYSTEM ANNUNCIATION, THEREBY ELIMINATING RELIANCE ON THE UNSUPERVISED CIRCUITS. THE DISTRICT COMMITTED TO COMPLY WITH VARIOUS REQUIREMENTS OF THE NATIONAL FIRE PROTECTION ASSOCIATION STANDARDS VIA THE DISTRICT RESPONSE DATED AUGUST 31, 1976 TO THE NRC BRANCH TECHNICAL POSITION (BTP) APCSB 9.5-1, WHICH WAS ISSUED BY THE NRC ON MAY 1, 1976. THIS COMMITMENT WAS ESSENTIALLY RESTATED BY THE NRC IN THE SAFETY EVALUATION REPORT ISSUED AS AMENDMENT 19 TO THE FACILITY OPERATING LICENSE

ON FEBRUARY 28, 1978. THIS REPORT PROVIDES THE NRC WITH INFORMATION CONCERNING DEVIATIONS FROM SEVERAL OF THOSE COMMITMENTS WHICH MAY OR MAY NOT AFFECT THE NRC'S SER FOR THE RANCHO SECO FIRE PROTECTION PROGRAM. THE SPECIFIC DEVIATIONS ARE CONTAINED IN DISTRICT INTERNAL REPORT NUMBERS ERPT E-033, -034, -035, -037, -039 -047, AND -048.

[240] RANCHO SECO DOCKET 50-312 LER 87-046
SURVEILLANCE PROCEDURES NOT PERFORMED BY TECHNICAL SPECIFICATION DATE DUE TO PERSONNEL ERROR.
EVENT DATE: 121587 REPORT DATE: 011388 NSSS: BW TYPE: PWR

(NSIC 207888) ON NOVEMBER 26, 1987, DECAY HEAT SYSTEM VALVES HV-20001 AND HV-20002 WERE RETURNED TO OPERABLE STATUS. THE VALVES WERE ENERGIZED, STROKED, AND LEFT IN THE OPEN POSITION. SURVEILLANCE PROCEDURE (SP) 203.06C/D REQUIRES VERIFICATION OF VALVE STROKE TIME AND POSITION INDICATION FOLLOWING A COLD SHUTDOWN OF 72 HOURS OR GREATER, PROVIDED IT HAS BEEN THREE MONTHS SINCE THE VALVES WERE LAST EXERCISED. THE VALVES HAD BEEN SCHEDULED FOR THE QUARTERLY PERFORMANCE OF SP-203-06C/D ON SEPTEMBER 18, 1987, BUT THE PROCEDURE WAS NOT PERFORMED BECAUSE THE VALVES WERE NOT IN SERVICE AT THAT TIME. SP- 203-06C/D SHOULD HAVE BEEN PERFORMED ON NOVEMBER 26, 1987 WHEN THE VALVES WERE RETURNED TO SERVICE. THE PROCEDURE WAS NOT PERFORMED AT THAT TIME. ON DECEMBER 15, 1987, THE ROUTINE QUARTERLY SURVEILLANCE SP- 31B (SP-31B SUPERSEDED SP-203-06C/D) WAS SUCCESSFULLY PERFORMED. BETWEEN NOVEMBER 26, 1987 AND DECEMBER 15, 1987 THE VALVES DID NOT MEET THE OPERABILITY REQUIREMENTS OF THE TECHNICAL SPECIFICATIONS. THE VALVES WERE IN THE OPEN POSITION AND WERE FUNCTIONAL; THEREFORE, THE DECAY HEAT SYSTEM WAS FUNCTIONAL DURING THE PERIOD THAT THE VALVES WERE TECHNICALLY INOPERABLE. A SEARCH OF PREVIOUSLY SUBMITTED LERS DISCLOSED FOUR SIMILAR OCCURRENCES. LERS 85-08, 86-06, 87-12, AND 87-45 RELATED TO TECHNICAL SPECIFICATION REQUIRED SURVEILLANCES MISSED DUE TO SCHEDULING PROBLEMS.

[241] RIVERBEND 1 DOCKET 50-458 LER 86-024 REV 06
UPDATE ON ESF ACTUATIONS DUE TO AN EPA BREAKER FAILURE.
EVENT DATE: 032586 REPORT DATE: 011588 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 207950) AT 1626 ON 3/25/86 WITH THE UNIT SUBCRITICAL, POWER TO REACTOR PROTECTION SYSTEM (RPS) BUS B WAS LOST DUE TO A TRIP OF AN ELECTRICAL PROTECTION ASSEMBLY (EPA) BREAKER. THE CAUSE WAS UNKNOWN AND UNDER INVESTIGATION WHEN ANOTHER TRIP OF THE SAME BREAKER OCCURRED AT 2211 THE SAME DAY. THE LOSS OF THE RPS BUS CAUSED RESIDUAL HEAT REMOVAL (RHR) TRAIN A TO BE ISOLATED WHICH RESULTED IN A LOSS OF SHUTDOWN COOLING. AT 1636, RHR TRAIN A WAS RESTARTED IN THE SHUTDOWN COOLING MODE AND THE ISOLATIONS WERE RESET. THE CAUSE OF THE EPA BREAKER TRIP WAS TRACED TO A POTENTIAL PROBLEM IN THE TRIP CONDITION CIRCUITRY OF THE EPA BREAKERS WHICH WERE EXERCISED DURING THE PERFORMANCE OF A SURVEILLANCE TEST PROCEDURE. THE BREAKER WAS REWORKED PER DESIGN, TESTED AND RETURNED TO SERVICE ON 4/2/86. AT 1911 ON 12/24/86 WITH THE UNIT AT FULL POWER, POWER TO THE REACTOR PROTECTION SYSTEM (RPS) BUS B WAS AGAIN LOST RESULTING IN A HALF-SCRAM AND THE DIVISION 2 ISOLATIONS OF REACTOR WATER CLEANUP AND OTHER DIVISION 2 CONTAINMENT ISOLATION VALVES. IMMEDIATE ACTION WAS TAKEN TO PLACE RPS BUS B ON ITS ALTERNATE POWER SUPPLY AND POWER WAS RESTORED IN APPROXIMATELY ONE MINUTE. THE ISOLATIONS WERE RESET AND THE SYSTEM LINEUPS WERE RETURNED TO NORMAL. AN IMMEDIATE FUNCTIONAL TEST OF THE EPA WAS PERFORMED SHOWING NO APPARENT MALFUNCTION.

[242] RIVERBEND 1 DOCKET 50-458 LER 87-023 REV 01
UPDATE ON RADIATION MONITOR HEAT EXCHANGERS PLUGGED WITH CORROSION DUE TO SERVICE WATER CHEMISTRY.
EVENT DATE: 101987 REPORT DATE: 012988 NSSS: GE TYPE: BWR

VENDOR: GENERAL ATOMIC CO.

(NSIC 208045) ON 10/19/87 DURING SHUTDOWN (MODE 5), THE COOLING WATER LINES TO HEAT EXCHANGER 1RMS*HX11A SERVING ONE OF THE TWO REDUNDANT ANNULUS EXHAUST: RADIATION MONITORS (1RMS*RE11A) WERE FOUND TO BE PLUGGED DUE TO A BUILDUP OF CORROSION. THIS DISCOVERY WAS MADE DURING ROUTINE INSPECTION OF THE HEAT EXCHANGERS. ON 11/18/87, THE COOLING WATER LINES TO THE HEAT EXCHANGER IN THE REDUNDANT TRAIN WERE ALSO FOUND TO BE PLUGGED DUE TO CORROSION BUILDUP. THE ROOT CAUSE THAT LED TO THE PLUGGING OF THESE LINES HAS BEEN ATTRIBUTED TO THE PAST CONDITION OF SERVICE WATER CHEMISTRY. SUBSEQUENT EVALUATION HAS CONCLUDED THAT THE ANNULUS EXHAUST RADIATION MONITORS WILL PERFORM THEIR REQUIRED SAFETY FUNCTION OF INITIATING THE STANDBY GAS TREATMENT SYSTEM IN THE EVENT OF A HIGH RADIATION CONDITION IN THE REACTOR BUILDING ANNULUS WITHOUT COOLING WATER BEING SUPPLIED TO 1RMS*HX11A AND 11B. THEREFORE, THIS CONDITION NO LONGER SATISFIES THE REPORTING REQUIREMENTS OF 10CFR50.73 AND HENCE, THIS REPORT IS BEING PROVIDED FOR INFORMATION ONLY. THE CORRECTIVE ACTION WHICH WAS TAKEN TO ELIMINATE THE POSSIBILITY OF CORROSION IN THE SERVICE WATER SUPPLY AND RETURN LINES TO HEAT EXCHANGERS 1RMS*HX11A AND 11B WAS TO REPLACE THE CARBON STEEL PIPING WITH STAINLESS STEEL PIPING. THIS CONDITION DID NOT RESULT IN ANY INCREASE IN RISK TO THE PUBLIC.

[243] RIVERBEND 1 DOCKET 50-458 LER 87-027
TECHNICAL SPECIFICATION VIOLATION DUE TO INCORRECTLY POSITIONED INSTRUMENT ROOT VALVE.
EVENT DATE: 111787 REPORT DATE: 121187 NSSS: GE TYPE: BWR

(NSIC 207491) ON 11/17/87 WITH THE UNIT IN MODE 4 (COLD SHUTDOWN), SYSTEM VALVE LINE UPS WERE BEING PERFORMED WHEN VALVE LRCS-V122 WAS DISCOVERED CLOSED. THIS VALVE SERVES AS THE PROCESS ISOLATION ROOT VALVE FOR PRESSURE TRANSMITTERS SERVING THE HIGH DRYWELL PRESSURE SCRAM, VARIOUS INBOARD CONTAINMENT ISOLATION VALVES, AUTO START OF DIVISION II STANDBY GAS TREATMENT (SBGT), ANNULUS MIXING (HVR), FUEL BUILDING VENTILATION (FBV), CONTAINMENT ATMOSPHERIC MONITORING (CMS), DIVISION II LOAD SHED BREAKERS AND HIGH PRESSURE CORE SPRAY (HPCS) INITIATION. INVESTIGATION REVEALED THIS ROOT VALVE WAS NOT SIGNED OFF AS OPENED DURING THE INITIAL PERFORMANCE OF THE SYSTEM LINE-UP IN AUGUST 1985. THIS INCORRECT VALVE POSITION WAS RESTORED TO THE CORRECT VALVE POSITION (OPEN). SAFETY RELATED VALVE LINE-UPS WILL NOW RECEIVE INDEPENDENT COMPLETION REVIEW IN ADDITION TO NORMAL REVIEW AND APPROVAL PRIOR TO STARTUP FROM THE CURRENT REFUELING OUTAGE. THESE REVIEWS WILL ENSURE THE VALVE POSITIONING AND INDEPENDENT VERIFICATION IS COMPLETED ON SAFETY RELATED VALVE LINE- UPS. GSU HAS DETERMINED THAT NO ADVERSE EFFECTS ON THE HEALTH AND SAFETY OF THE PUBLIC EXISTED AS A RESULT OF THIS EVENT. THERE WERE NO HIGH DRYWELL PRESSURE EVENTS DURING THIS PERIOD.

[244] RIVERBEND 1 DOCKET 50-458 LER 87-033
LOSS OF SHUTDOWN COOLING DUE TO SPURIOUS EPA BREAKER TRIP.
EVENT DATE: 121187 REPORT DATE: 011188 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 207701) AT APPROXIMATELY 1030 ON 12/11/87 WITH THE UNIT IN COLD SHUTDOWN (MODE 4), A LOSS OF THE DIVISION II REACTOR PROTECTION SYSTEM (RPS) BUS OCCURRED DUE TO THE TRIP OF NORMAL POWER SUPPLY ELECTRICAL PROTECTION ASSEMBLY (EPA) BREAKER C71-S003D. THE LOSS OF THE RPS BUS RESULTED IN ALL REQUIRED DIVISION II ISOLATIONS WHICH CAUSED A LOSS OF RHR "A" SHUTDOWN COOLING. THE BUS WAS RE-ENERGIZED FROM THE ALTERNATE POWER SOURCE, ALL ISOLATIONS RESET AND SHUTDOWN COOLING RESTARTED. THE LOGIC CARD IN THE EPA BREAKER WAS FOUND TO BE MALFUNCTIONING AND TWO INTEGRATED CIRCUIT (IC) CHIPS WERE REPLACED. THE SURVEILLANCE TEST PROCEDURE (STP) WAS SUCCESSFULLY PERFORMED AND THE BREAKER WAS RETURNED TO SERVICE. AT APPROXIMATELY 1857 ON 12/19/87 WITH THE UNIT AGAIN IN MODE 4 FOLLOWING A MANUAL UNIT SHUTDOWN, A SECOND TRIP OF EPA BREAKER C71-S003D

OCCURRED. AGAIN, THE BREAKER LOGIC CARD WAS FOUND TO BE DEFECTIVE. THREE INTEGRATED CIRCUIT (IC) CHIPS WERE REPLACED, THE STP SUCCESSFULLY PERFORMED AND THE BREAKER WAS RETURNED TO SERVICE. THE OUTPUT VOLTAGE OF THE MOTOR-GENERATOR SET WAS SUBSEQUENTLY INCREASED SLIGHTLY TO HELP PREVENT RECURRENCE. IN EACH CASE, THE UNIT WAS ALREADY IN COLD SHUTDOWN AND ALL SAFETY SYSTEMS FUNCTIONED AS DESIGNED. FOLLOWING THE FIRST EVENT SHUTDOWN COOLING WAS RESTORED WITHIN TEN MINUTES BY UTILIZING THE ALTERNATE POWER SUPPLY.

[245] RIVERBEND 1 DOCKET 50-458 LER 87-034
FULL REACTOR SCRAM DUE TO MANUAL HALF-SCRAM AND AN INTERMEDIATE RANGE MONITOR SPIKE.
EVENT DATE: 122187 REPORT DATE: 011988 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 207996) AT 0510 ON 12/21/87 WITH THE REACTOR AT LESS THAN 1% POWER IN MODE 2 (HOT SHUTDOWN), A REACTOR SCRAM OCCURRED. THE SCRAM RESULTED 1 FROM A SPURIOUS TRIP OF INTERMEDIATE RANGE MONITOR (IRM) CHANNEL "A" (DIVISION 1) OF THE NEUTRON MONITORING SYSTEM CAUSING A HALF-SCRAM 1 WITH AN EXISTING MANUAL HALF-SCRAM BEING PRESENT ON CHANNEL "B" (DIVISION 2) OF THE REACTOR PROTECTION SYSTEM (RPS). THE HALF-SCRAM ON RPS HAD BEEN INITIATED TO COMPLY WITH THE TECH SPEC AS A RESULT OF A REACTOR WATER LEVEL INSTRUMENT READING VALUES IN EXCESS OF THE CHANNEL CHECK LIMITS. AN EVALUATION OF THE REACTOR WATER LEVEL INDICATION DISCREPANCIES DETERMINED THE CAUSE TO BE AIR TRAPPED IN THE INSTRUMENT REFERENCE LINE. SINCE THE REACTOR VESSEL WAS AT A VERY LOW PRESSURE DURING STARTUP (APPROXIMATELY 100 PSI), THE CONDENSING CHAMBER DID NOT FUNCTION AT A RATE ADEQUATE TO MAKEUP THE LOSS IN THE REFERENCE LINE WATER COLUMN LEVEL. THE REACTOR VESSEL WAS PRESSURIZED TO A RANGE THAT WOULD PERMIT PROPER OPERATION OF THE REFERENCE LINE CONDENSING CHAMBER AND AS EXPECTED, ACCEPTABLE WATER LEVEL CHANNEL CHECKS WERE OBTAINED PRIOR TO EXCEEDING 150 PSI REACTOR PRESSURE. INVESTIGATION INTO THE IRM SPIKE HAS BEEN INCONCLUSIVE. THE UNIT WAS SUCCESSFULLY RETURNED TO OPERATION ON 12/23/87. THERE WAS NO IMPACT ON THE SAFE OPERATION OF THE PLANT OR TO THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT.

[246] ROBINSON 2 DOCKET 50-261 LER 87-029
SERVICE WATER FLANGE LEAK IN CONTAINMENT DUE TO MINIMUM COMPONENT REDUNDANCY VIOLATION.
EVENT DATE: 111887 REPORT DATE: 121787 NSSS: WE TYPE: PWR
VENDOR: ALLIS CHALMERS
FAIRBANKS CO, THE
WESTINGHOUSE ELECTRIC CORP.

(NSIC 207495) AT 1230 HOURS, NOVEMBER 18, 1987, WHILE OPERATING AT 100% POWER A SMALL LEAK WAS IDENTIFIED IN A SERVICE WATER LINE FROM THE MOTOR COOLER OF CONTAINMENT AIR RECIRCULATION COOLING UNIT HVH-2 INSIDE THE CONTAINMENT VESSEL (CV). UTILITY LICENSED OPERATORS ISOLATED HVH-2, THE FLANGE LEAK WAS REPAIRED, AND THE UNIT RETURNED TO SERVICE. POST-MAINTENANCE TESTING REVEALED THAT THE OUTLET ISOLATION VALVE, V6-34B, HAD REMAINED PARTIALLY OPEN (ALTHOUGH CONTROL ROOM INDICATION SHOWED IT CLOSED) WHILE THE FLANGE SPOOL PIECE INSIDE THE CV WAS REMOVED. THIS CONSTITUTED A BREACH OF CONTAINMENT INTEGRITY. AT 0300 HOURS, NOVEMBER 19, THE LICENSEE NOTIFIED THE NRC OF A ONE-HOUR NONEMERGENCY EVENT, PURSUANT TO 10CFR50.72(B)(1)(II). AT 0902, AFTER CORRECTING THE VALVE POSITION PROBLEM, THE FLOW PATH WAS DECLARED OPERABLE. LATER, THE LICENSEE DETERMINED THAT ISOLATING HVH-2 HAD JEOPARDIZED COMPONENT REDUNDANCY REQUIREMENTS IN THAT TWO OF FOUR CONTAINMENT COOLING UNITS HAD NO EMERGENCY POWER SUPPLY AT THE TIME HVH-2 WAS REMOVED FROM SERVICE. THE LICENSEE PROVIDED THE NRC FOLLOW-UP NOTIFICATION AT 1451 HOURS, NOVEMBER 25 TO REPORT THE CONDITION. AT 0949 HOURS, DECEMBER 3, THE LICENSEE PROVIDED AN ADDITIONAL NRC FOLLOW-UP NOTIFICATION.

[247] ROBINSON 2 DOCKET 50-261 LER 87-030
NON-REDUNDANT POWER SUPPLY TO REDUNDANT VITAL EQUIPMENT DUE TO ORIGINAL SYSTEM DESIGN.
EVENT DATE: 120287 REPORT DATE: 121787 NSSS: WE TYPE: PWR
VENDOR: ALOYCO, INC.
INGERSOLL-RAND CO.

(NSIC 207421) DECEMBER 2, 1987, AT 1230 HOURS, A POTENTIAL FOR A COMMON FAILURE WAS IDENTIFIED THEREBY THE SAFETY INJECTION AND RESIDUAL HEAT REMOVAL (SI/RHR) SYSTEMS MAY NOT BE ABLE TO BE SHIFTED FROM THE INJECTION PHASE TO THE RECIRCULATION PHASE WITHIN THE THREE MINUTES REQUIRED BY EMERGENCY OPERATING PROCEDURES (EOP) TO RESTORE POST-ACCIDENT FLOW TO THE REACTOR COOLANT SYSTEM. TWO REDUNDANT VALVES IN THE SI/RHR SYSTEM (SIS-863A AND B), OF WHICH ONE IS REQUIRED TO BE OPEN FOR RECIRCULATION MODE, CONTAIN INTERLOCKS WHICH ARE SUPPLIED POWER BY A SINGLE VITAL POWER SUPPLY, CREATING A POTENTIAL SINGLE FAILURE THAT COULD PREVENT THE VALVES FROM REMOTELY OPENING FROM THE CONTROL ROOM. IT WAS DETERMINED THAT THERE WAS NOT REASONABLE ASSURANCE THAT ONCE THIS CONDITION WAS RECOGNIZED, MANUAL OPERATION OF ONE OF THE VALVES WOULD BE COMPLETED WITHIN THE THREE MINUTE TIME FRAME REQUIRED BY EOPS. THE CAUSE OF THIS CONDITION, WHICH EXISTED SINCE ORIGINAL CONSTRUCTION, IS ATTRIBUTED TO ADEQUATE DESIGN OF THE SYSTEMS' INTERLOCK LOGIC POWER ARRANGEMENT. AN AUXILIARY OPERATOR WAS IMMEDIATELY ASSIGNED THE SPECIFIC RESPONSIBILITY TO MANUALLY OPEN THE VALVES IF NECESSARY. A TEMPORARY MODIFICATION WAS INSTALLED TO JUMPER THE INTERLOCK, WHICH IS BEING PROCEDURALLY CONTROLLED, UNTIL MORE PERMANENT CORRECTIVE ACTION BE CAN BE IMPLEMENTED.

[248] SALEM 1 DOCKET 50-272 LER 87-017 REV 02
UPDATE ON DISC. LEAK PATHS FROM 13(23) AFW PUMP COMPARTMENTS - CONTROL OF DESIGN REQUIREMENTS.
EVENT DATE: 111387 REPORT DATE: 012688 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 207971) ON 11/13/87 AN NRC INSPECTOR IDENTIFIED LEAKAGE PATHS FROM THE NO. 13 AUXILIARY FEEDWATER (AFW) TURBINE DRIVEN PUMP COMPARTMENT (BA). SUBSEQUENT INVESTIGATION REVEALED SIX (6) UNSEALED OPENINGS FROM THE UNIT 1 PUMP AND ONE UNSEALED OPENING FROM THE UNIT 2 PUMP. THE COMPARTMENT ENCLOSES THE STEAM FEED PIPING TO THE AFW TURBINE DRIVEN PUMP SUCH THAT A POSTULATED PIPE BREAK WOULD NOT DAMAGE ADJACENT VITAL ELECTRICAL EQUIPMENT LOCATED OUTSIDE THE COMPARTMENT. THE ROOT CAUSE OF THE STEAM DRIVEN AFW PUMP COMPARTMENT DEFICIENCIES HAS BEEN ATTRIBUTED TO CONTROL OF DESIGN REQUIREMENTS. THE LEAKAGE PATHS IDENTIFIED ON THE SALEM UNITS 1 & 2 STEAM DRIVEN AFW PUMP ENCLOSURES HAVE BEEN SEALED. A REVIEW OF OTHER PROTECTIVE PIPE RUPTURE ENCLOSURES TO VERIFY THEIR INTEGRITY HAS BEEN MAINTAINED, IS CONTINUING. TO ENSURE THAT ADEQUATE ATTENTION IS GIVEN TO THE MAINTENANCE OF THESE STRUCTURES IN THE FUTURE, THE PROGRAM ANALYSIS GROUP (PAG) WILL ISSUE A FIELD DIRECTIVE IDENTIFYING THE AREAS THAT ARE DESIGNED TO ACCOMMODATE THE EFFECTS OF A PIPE RUPTURE. BASED ON THE FIELD DIRECTIVE, ADMINISTRATIVE CONTROLS TO MAINTAIN PROTECTIVE STRUCTURES IN A CONDITION THAT MEETS THEIR DESIGN REQUIREMENTS WILL BE REVIEWED AND CHANGES MADE AS NECESSARY.

[249] SALEM 1 DOCKET 50-272 LER 87-018
LEAD/LAG AND DERIVATIVE AMPLIFIERS IMPROPERLY CALIBRATED DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 120987 REPORT DATE: 010888 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 207709) ON 12/08/87, IT WAS DETERMINED BY TECHNICAL DEPARTMENT SYSTEM ENGINEERING PERSONNEL THAT THE LEAD/LUG AND DERIVATIVE AMPLIFIERS IN THE PROCESS AND PROTECTION CONTROL SYSTEM HAVE BEEN INCORRECTLY CALIBRATED. THE PROCESS AND PROTECTION EQUIPMENT AFFECTED BY THE IMPROPER CALIBRATION INCLUDE: LOW STEAMLINE

PRESSURE TRIP, OVERTEMPERATURE DELTA T TRIP, OVERPOWER DELTA T TRIP, LOW PRESSURIZER PRESSURE TRIP; MAIN TURBINE IMPULSE CONTROL; COOLANT AVERAGE TEMPERATURE CONTROL (PROGRAM); POWER MISMATCH CHANNEL IMPULSE CONTROL AND STEAM DUMP CONTROL. THE ROOT CAUSE OF THIS EVENT WAS PROCEDURAL AND STEAM DUMP CONTROL. THE ROOT CAUSE OF THIS EVENT WAS PROCEDURAL INADEQUACY. AN INVESTIGATION IS ON-GOING TO DETERMINE WHY THE PROCEDURES WERE INCORRECT. INVESTIGATION OF THE POTENTIAL EFFECTS OF THE INCORRECT CALIBRATION ON THE PROCESS AND PROTECTION SYSTEM EQUIPMENT NOTED ABOVE IS CONTINUING.

[250] SALEM 1 DOCKET 50-272 LER 87-019
WASTE GAS OXYGEN CONCENTRATION GREATER THAN 2 PERCENT.
EVENT DATE: 122787 REPORT DATE: 012288 NSSS: WE TYPE: PWR

(NSIC 208032) OXYGEN CONCENTRATION WITHIN THE WASTE GAS HOLDUP SYSTEM WAS GREATER THAN 2% FOR MORE THAN 48 HOURS CONTRARY TO THE REQUIREMENTS OF TECH SPEC ACTION 3.11.2.5.A. ON DECEMBER 25, 1987 AT 0940 HOURS, CHEMISTRY SAMPLING OF NO. 14 WASTE GAS DECAY TANK (WGDT) INDICATED AN OXYGEN CONCENTRATION OF 4.4%. ADDITIONALLY, NO. 13 WGDT INDICATED AN OXYGEN CONCENTRATION OF 4.3% AT 2110 HOURS THE SAME DAY. OXYGEN CONCENTRATION WITHIN NO. 14 WGDT AND NO. 13 WGDT WERE NOT REDUCED TO LESS THAN 2% UNTIL 2125 HOURS ON DECEMBER 27, 1987 AND 0855 HOURS ON DECEMBER 28, 1987, RESPECTIVELY. OXYGEN CONCENTRATION OF GREATER THAN 2% WITHIN THE WASTE GAS HOLDUP SYSTEM IS AN ANTICIPATED CONCERN AT THE CONCLUSION OF A REFUELING OUTAGE, AS WAS THE CASE HERE. NORMALLY, THE OXYGEN CAN BE DILUTED/PURGED FROM THE SYSTEM WITHIN 48 HOURS, HOWEVER, DUE TO THE APPARENTLY LARGER QUANTITY OF OXYGEN FROM THE OUTAGE ACTIVITIES, IT COULD NOT BE REDUCED WITHIN 48 HOURS. PSE&G SYSTEM ENGINEERING IS INVESTIGATING THIS EVENT IN ORDER TO IDENTIFY ITS ROOT CAUSE AND TO DEVELOP METHODS/PROCEDURES TO PREVENT RECURRENCE.

[251] SALEM 2 DOCKET 50-311 LER 87-009 REV 06
UPDATE ON APPENDIX R CRITERIA NON-CONFORMANCE DUE TO INADEQUATE DESIGN REVIEW.
EVENT DATE: 112587 REPORT DATE: 122487 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 1 (PWR)

(NSIC 207525) THE FOLLOWING SYSTEM/COMPONENT CONDITIONS WERE IDENTIFIED BY A PSE&G TASK FORCE REVIEWING/EVALUATING SALEM STATION'S COMPLIANCE WITH THE REQUIREMENTS OF 10CFR 50 APPENDIX R. WHERE APPROPRIATE, FIRE WATCHES WERE ESTABLISHED. LONG TERM CORRECTIVE ACTION INCLUDES MAKING DESIGN CHANGE MODIFICATIONS, AS APPLICABLE. LER 87-009-00 ADDRESSED A SW SYSTEM CABLING APPENDIX R SEPARATION CRITERIA INADEQUACY. THE ROOT CAUSE WAS INADEQUATE DESIGN REVIEW. LER 87-009-01 ADDRESSED NON-SEISMICALLY QUALIFIED MARINITE WALLS LOCATED IN SALEM UNITS 1 & 2 460V SWITCHGEAR ROOM. THE WALLS HAVE BEEN REINFORCED TO SEISMIC CRITERIA. A SAMPLE OF DESIGN CHANGES INSTALLED BEFORE IMPLEMENTATION OF CURRENT DESIGN CONTROL PROCEDURES IS BEING CONDUCTED. LER 87-009-02 ADDRESSES RHR ROOM COOLERS (VF) CABLING APPENDIX R INADEQUACIES AND CONTROL CABLING APPENDIX R INADEQUACIES IS INADEQUATE DESIGN REVIEW. LER 87-009-03 ADDRESSED A D/G POWER CABLING APPENDIX R SEPARATION CRITERIA DEFICIENCY. THE ROOT LOCATED IN BOTH UNITS CO(2) EQUIPMENT ROOMS, IDENTIFIED ON 9/10/87. THE ROOT CAUSE WAS INADEQUATE DESIGN REVIEW. LER 87-009-05 ADDRESSED TWO ISSUES: UHF COMMUNICATION CONCERNS (BOTH UNITS) DURING A POSTULATED FIRE AND THE DECLARED INOPERABILITY OF ALL THREE D/G'S (EK) (BOTH UNITS) UPON POSTULATED ACUTATION OF THE LOW PRESSURE CO(2) FLOODING SYSTEM.

[252] SALEM 2 DOCKET 50-311 LER 87-015
POTENTIAL FOR CERTAIN SW MCC CONTROL CIRCUITS TO PICK-UP STARTER COIL DUE TO INADEQUATE DESIGN REVIEW.
EVENT DATE: 112787 REPORT DATE: 122487 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 1 (PWR)

(NSIC 207553) ON NOVEMBER 27, 1987 AN ENGINEERING REVIEW OF SALEM ELECTRICAL SYSTEMS REVEALED THAT WITH A DEGRADED GRID CONDITION, A LOCA AND ASSUMING THE 13.8 KV TO 4 KV STATION POWER TAP CHANGER FAILS TO FUNCTION, CERTAIN MCC CONTROL CIRCUITS (ED) MAY NOT HAVE ADEQUATE VOLTAGE TO PICK-UP THEIR RESPECTIVE MCC STARTER COIL. TO ASSURE PICK-UP BY THE STARTER COIL, 89% OF THE 115 VAC IS REQUIRED AS PER MANUFACTURER SPECIFICATIONS. THE ROOT CAUSE OF THIS EVENT IS INADEQUATE DESIGN REVIEW OF CONTROL CIRCUITS FOR DEGRADED VOLTAGE CONDITIONS. AS APPLICABLE, MODIFICATIONS TO CORRECT THE ELECTRICAL CONCERNS INCLUDED THE ADDITION OF INTERPOSING RELAYS TO LESSEN THE CONTROL CIRCUIT VOLTAGE DROP AND ENHANCEMENT OF THE POWER CIRCUIT CONDUCTOR SIZE WHICH INCREASED THE BUS VOLTAGE AVAILABLE AT THE MCC. THE UNIT 2 ANALYSIS AND MODIFICATIONS ARE BEING MADE AS APPLICABLE. UNIT 1 IS CURRENTLY IN A REFUELING OUTAGE AND WILL NOT BE RETURNED TO SERVICE UNTIL COMPLETION OF THE ENGINEERING ANALYSIS AND APPLICABLE DESIGN MODIFICATIONS.

[253] SALEM 2 DOCKET 50-311 LER 87-016
2A DIESEL GENERATOR SURVEILLANCE MISSED DUE TO PERSONNEL ERROR.
EVENT DATE: 120787 REPORT DATE: 010688 NSSS: WE TYPE: PWR

(NSIC 207636) ON DECEMBER 07, 1987 TECHNICAL SPECIFICATION SURVEILLANCE 4.8.1.1.2. FOR 2A DIESEL GENERATOR (D/G) WAS NOT PERFORMED WITHIN THE FOURTEEN DAY REQUIRED TIME FRAME. TECHNICAL SPECIFICATION SURVEILLANCE 4.8.1.1.2 REQUIRES A ONE HOUR LOADED RUN OF THE D/G. THE SURVEILLANCE INCLUDES STARTING AND LOADING THE D/G WITHIN A SPECIFIED TIME PERIOD. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR BY A SHIFT SUPERVISOR. CORRECTIVE ACTION INCLUDES APPROPRIATE CORRECTIVE DISCIPLINARY ACTION TO THE INDIVIDUAL INVOLVED. THE SURVEILLANCE WAS SUBSEQUENTLY SUCCESSFULLY COMPLETED. THIS EVENT WILL BE REVIEWED BY THE NUCLEAR TRAINING DEPARTMENT FOR INCORPORATION INTO APPLICABLE TRAINING PROGRAMS. ADDITIONALLY, STATION MANAGEMENT HAS INITIATED A REVIEW/AUDIT OF 1987 INCIDENT REPORTS AND REPORTABLE EVENTS INVOLVING PERSONNEL ERROR.

[254] SALEM 2 DOCKET 50-311 LER 87-017
TECHNICAL SPECIFICATION NON-COMPLIANCE DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 120887 REPORT DATE: 010688 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 1 (PWR)
VENDOR: C & D BATTERIES, DIV OF ELTRA CORP.

(NSIC 207653) ON DECEMBER 10, 1987, THE UNIT 2 VITAL 28 VOLT AND RB5 VOLT ACTUATORS (EC) WERE DECLARED INOPERABLE WHEN IT WAS FOUND THAT SOME OF THE PRESSURE MEASURED SPECIFIC GRAVITIES DID NOT MEET TECH SPEC SURVEILLANCES 4.8.2.3.2. AND 4.8.2.5.2. ON DECEMBER 8, 1987 IT WAS DISCOVERED THAT THE ELECTROLYTE SPECIFIC GRAVITIES LEVEL CORRECTION FACTOR DID NOT AGREE WITH THE BATTERY MANUFACTURER'S INSTALLATION AND OPERATING INSTRUCTIONS MANUAL. THE MANUFACTURER'S CORRECTION FACTOR EQUATES TO 0.002 FOR EACH 1/16 INCH THE ELECTROLYTE LEVEL IS BELOW THE CELL'S HIGH LEVEL MARKER WHILE THE PSE&G VALVE EACH WAS 0.001. THE ROOT CAUSE OF THIS DISCOVERY HAS BEEN DETERMINED TO BE PROCEDURAL INADEQUACY. THE BATTERY MAINTENANCE PROCEDURES WERE NOT IN CONFORMANCE WITH THE MANUFACTURER'S ELECTROLYTE LEVEL CORRECTION FACTOR VALUE. CORRECTIVE ACTION INCLUDED REVISING THE BATTERY TEST PROCEDURE TO INCORPORATE THE 0.002 LEVEL CORRECTION FACTOR. ADDITIONALLY, THE UNIT 1 AND 2 VITAL BATTERIES WERE FULL CHARGED AND ALL CELL ELECTROLYTE LEVELS WERE TOPPED-UP TO THEIR HIGH LEVEL MARKER WITH 1.210 SPECIFIC GRAVITY ELECTROLYTE IN ACCORDANCE WITH MANUFACTURER'S RECOMMENDATIONS. A PROGRAM TO TREND BATTERY TESTING RESULTS HAS BEEN INITIATED.

[255] SALEM 2 DOCKET 50-311 LER 87-018
FUNCTIONAL TEST FOR BOTH UNIT 1 AND 2 WASTE GAS OXYGEN MONITORS PERFORMED LATE DUE TO INADVERTENT ADMINISTRATIVE CONTROLS.
EVENT DATE: 122387 REPORT DATE: 012288 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 1 (PWR)

(NSIC 207989) ON DECEMBER 23, 1987 AT 0825 HOURS, IT WAS DISCOVERED THAT THE REQUIRED FUNCTIONAL TEST FOR BOTH UNIT 1 AND UNIT 2 WASTE GAS OXYGEN MONITORS WERE NOT PERFORMED WITHIN THEIR REQUIRED TIME FRAME. TECH SPEC SURVEILLANCE 4.3.3.9.1.B. REQUIRES THE PERFORMANCE OF THIS FUNCTIONAL TEST EVERY 31 DAYS. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INADEQUATE ADMINISTRATIVE CONTROLS ASSOCIATED WITH THE IMPLEMENTATION OF THE NEW COMPUTER BASED WORK ACTIVITY SYSTEM, MMIS. TECH SPEC SURVEILLANCE 4.3.3.9.1.B WAS NOT SCHEDULED AND PERFORMED AS REQUIRED DUE TO A MISINTERPRETATION OF THE SCHEDULING RESPONSIBILITIES. THE REQUIRED FUNCTIONAL SURVEILLANCES WERE SUCCESSFULLY COMPLETED ON 12/23/87. A REVIEW OF CHEMISTRY DEPARTMENT TECH SPEC REQUIRED SURVEILLANCES, DURING THE PERIOD IN QUESTION, WAS CONDUCTED. NO ADDITIONAL MISSED SURVEILLANCES WERE DISCOVERED. THIS EVENT WILL BE REVIEWED WITH ALL CHEMISTRY DEPARTMENT SUPERVISION BY CHEMISTRY DEPARTMENT MANAGEMENT. ADMINISTRATIVE CONCERNS ASSOCIATED WITH THE IMPLEMENTATION OF THE NEW MMIS SYSTEM ARE BEING REVIEWED AND ADDRESSED. THE CORRECTIVE ACTIONS ASSOCIATED WITH THESE ADMINISTRATIVE CONCERNS WILL PREVENT RECURRENCE OF THIS TYPE EVENT.

[256] SAN ONOFRE 1 DOCKET 50-206 LER 86-011 REV 01
UPDATE ON MAIN FEEDWATER PUMP FAILURE.
EVENT DATE: 090486 REPORT DATE: 123087 NSSS: WE TYPE: PWR
VENDOR: BYRON JACKSON PUMPS, INC.

(NSIC 207665) ON SEPTEMBER 4, 1986 AT 2143, WITH THE REACTOR AT 52% POWER AND BOTH MAIN FEEDWATER PUMPS (MFP) IN OPERATION, THE WEST MFP LOW LUBE OIL PRESSURE ALARM WAS RECEIVED AND THE AUXILIARY LUBRICATING OIL PUMP AUTO-STARTED. IMMEDIATE INSPECTION REVEALED THAT THE WEST MFP INBOARD BEARING TEMPERATURE WAS INCREASING AND THE PUMP WAS STOPPED AT 2145. THE PUMP INBOARD BEARING SEAL WATER DEFLECTOR RING WAS OBSERVED TO BE DAMAGED AND THE PUMP WAS DECLARED INOPERABLE. BECAUSE THE MFP ALSO SERVES AS THE SAFETY INJECTION PUMP AT SAN ONOFRE UNIT 1, UNIT SHUTDOWN WAS INITIATED AS REQUIRED BY TECHNICAL SPECIFICATIONS 3.3.1 AND 3.0.3, AT 2235. AN UNUSUAL EVENT WAS ALSO DECLARED AS REQUIRED BY THE SAN ONOFRE SITE EMERGENCY PLAN. THE UNUSUAL EVENT WAS CLOSED AT 0120 ON SEPTEMBER 5, 1986. SUBSEQUENT DISASSEMBLY AND INSPECTION REVEALED THAT THE PUMP SHAFT HAD FAILED IN A THREADED AREA WHERE THE THRUST NUT IS ENGAGED TO PRE-LOAD THE THRUST DISK ASSEMBLY. THE FAILURE ANALYSES CONCLUDED THAT THE CAUSE OF THIS FAILURE WAS LOSS OF THRUST DISK ASSEMBLY PRE-LOAD. NEW ASSEMBLY INSTRUCTIONS WERE DEVELOPED IN CONJUNCTION WITH THE PUMP MANUFACTURER TO AVOID LOSS OF PRE-LOAD. IN ADDITION, VIBRATION AMPLITUDE WILL BE MONITORED THROUGHOUT THE REMAINDER OF THE FUEL CYCLE TO DETECT INCIPIENT FAILURE.

[257] SAN ONOFRE 1 DOCKET 50-206 LER 86-014 REV 01
UPDATE ON VOLUNTARY ENTRY INTO TECHNICAL SPECIFICATION 3.0.3 DURING DC GROUND TROUBLESHOOTING.
EVENT DATE: 123086 REPORT DATE: 122487 NSSS: WE TYPE: PWR

(NSIC 207664) ON 12/30/86, WITH THE UNIT 1 REACTOR AT 92% POWER, A GROUND INDICATION WAS RECEIVED ON DC BUS 2. LOCATING THE GROUND REQUIRES OPENING THE DC BUS FEEDER BREAKERS AND/OR LIFTING LEADS WHICH DE-ENERGIZE CONTROL POWER TO TRAIN A FEEDWATER VALVES(S) REQUIRED FOR SAFETY INJECTION (SI) SUBJECT TO TECHNICAL SPECIFICATION (TS) 3.3.1. TS 3.3.1 DOES NOT EXPLICITLY PERMIT INOPERABILITY OF THESE VALVES FOR ANY PERIOD OF TIME AND THEREFORE, OPENING THE DC BUS 2 FEEDER BREAKER OR LIFTING LEADS CONSTITUTES BRIEF ENTRY INTO TS 3.0.3. THE DURATION OF BREAKER OPENING AND/OR LEAD LIFTING WAS LIMITED, IN EACH CASE, TO THAT NECESSARY

TO DETERMINE IF THE GROUND EXISTED ON THE CIRCUIT. THE TOTAL TIME TRAIN A SI WAS SO IMPAIRED WAS LESS THAN THE ONE HOUR PERMITTED BY TS 3.0.3. THERE WERE NO SAFETY CONSEQUENCES TO THESE EVENTS SINCE TRAIN B SI WAS OPERABLE AT ALL TIMES. THE GROUND DID NOT AFFECT TRAIN A SI OPERABILITY EXCEPT FOR THE TROUBLESHOOTING PERIODS. THE GROUND WAS FOUND TO BE CAUSED BY MOISTURE IN A CONTROL CIRCUITRY JUNCTION BOX FOR ONE OF TWO SOLENOID VALVES WHICH CONTROL THE RATE OF FEEDWATER ISOLATION VALVE HV-854A CLOSURE. OVER A PERIOD OF TIME, MOISTURE FROM VALVE PACKING LEAKS AND THE SURROUNDING ENVIRONMENT HAD PENETRATED THE JUNCTION BOX INSIDE THE VALVE ACTUATOR.

[258] SAN ONOFRE 1 DOCKET 50-206 LER 87-003 REV 01
UPDATE ON LOSS OF GENERATOR FIELD TRIP.
EVENT DATE: 031087 REPORT DATE: 122487 NSSS: WE TYPE: PWR
VENDOR: CRANE VALVE CO.
WESTINGHOUSE ELEC CORP.-NUCLEAR ENERGY SYS

(NSIC 207656) AT 1024 ON MARCH 10, 1987, WITH THE UNIT AT 92% POWER, A TRIP SIGNAL INITIATED IN RESPONSE TO A LOSS OF GENERATOR FIELD, RESULTED IN A TURBINE TRIP AND SUBSEQUENT REACTOR TRIP. THE AUXILIARY FEEDWATER SYSTEM AUTOMATICALLY ACTUATED ON LOW STEAM GENERATOR LEVEL AS EXPECTED FROM A TRIP AT THIS POWER LEVEL. DURING THE TRIP RECOVERY, EQUIPMENT DEFICIENCIES WERE NOTED WHICH INCLUDED: FAILURE OF TWO ROD BOTTOM INDICATING LIGHTS TO ILLUMINATE; ONE OF TWO INTERMEDIATE RANGE NUCLEAR INSTRUMENTATION CHANNELS FAILED OFF-SCALE LOW; EIGHT MOISTURE SEPARATOR REHEATER STEAM DUMP VALVES AND THEIR ASSOCIATED MOTOR OPERATED ISOLATION VALVES FAILED TO CLOSE, WHICH IN TURN CAUSED PERFORATIONS IN ONE OF THE LOW PRESSURE TURBINE CASING RUPTURE DISCS; AND, THE MAIN FEEDWATER BLOCK VALVE TO "A" STEAM GENERATOR FAILED TO FULLY CLOSE UPON DEMAND. THE LOSS OF GENERATOR FIELD WAS DUE TO AN ELECTRICAL SHORT RESULTING FROM A BRUSH BEING INADVERTENTLY DROPPED INTO THE MAIN EXCITER COMMUTATOR ASSEMBLY DURING PERFORMANCE OF ROUTINE BRUSH REPLACEMENT ON THE COMMUTATOR AND BRUSH ASSEMBLIES. INADEQUATE LIGHTING WITHIN THE COMMUTATOR ENCLOSURE WAS A CONTRIBUTING FACTOR SINCE THE ELECTRICIAN HELD A FLASHLIGHT IN ONE HAND WHILE REPLACING THE BRUSH WITH THE OTHER, THEREBY LIMITING HIS DEXTERITY.

[259] SAN ONOFRE 1 DOCKET 50-206 LER 87-017
ENTRIES INTO TECHNICAL SPECIFICATION 3.0.3 TO PERFORM VENTS OF SAFETY INJECTION HEADER.
EVENT DATE: 120187 REPORT DATE: 123187 NSSS: WE TYPE: PWR
VENDOR: MAROTTA SCIENTIFIC CONTROLS, INC.

(NSIC 207657) PLANT PROCEDURES REQUIRE THAT EACH LOOP OF THE SAFETY INJECTION SYSTEM HEADER BE VENTED ON A QUARTERLY BASIS IN ORDER TO AVOID A WATER HAMMER IN THE EVENT OF SYSTEM ACTUATION. UPON COMPLETION OF VENTING OF THE LOOP B HEADER ON 9/10/87, IT WAS NOTED THAT THE REMOTELY OPERATED, NORMALLY CLOSED, CONTAINMENT ISOLATION VALVES IN THE VENT LINE FOR THAT HEADER INITIALLY DID NOT FULLY CLOSE UPON DEMAND. THE 3/4 INCH VENT LINE WAS SUCCESSFULLY ISOLATED, AS REQUIRED. ACCORDINGLY, ON 12/1/87 WHEN THE VENTING OF THE LOOP B HEADER WAS AGAIN REQUIRED, OPENING OF THE CONTAINMENT ISOLATION VALVES CONSTITUTED A VOLUNTARY ENTRY INTO TECHNICAL SPECIFICATION (TS) 3.0.3, SINCE ASSURANCE COULD NOT BE PROVIDED THAT THE ISOLATION VALVES WOULD RELIABLY CLOSE FROM THE OPEN POSITION, IF REQUIRED. THIS IS CONTRARY TO THE REQUIREMENTS OF TS 3.6.2. IT WAS RECOGNIZED AT THIS TIME THAT THE INITIAL FAILURE OF THE VALVES TO CLOSE ON 9/10/87 ALSO REPRESENTED ENTRY INTO TS 3.0.3. THIS ENTRY HAD NOT PREVIOUSLY BEEN RECOGNIZED. THE CONDITIONS OF TS 3.0.3 WERE MET ON BOTH 12/1/87 AND 9/10/87, AS THE CONTAINMENT PENETRATION WAS OPEN FOR ONLY APPROXIMATELY 30 MINUTES ON EACH OCCASION. THE VALVES WILL BE INSPECTED AND REPAIRED OR REPLACED AS NECESSARY DURING THE NEXT UNIT OUTAGE, SCHEDULED TO COMMENCE IN FEBRUARY 1988.

[260] SAN ONOFRE 1 DOCKET 50-206 LER 87-018
VOLUNTARY ENTRY INTO TECHNICAL SPECIFICATION 3.0.3 DURING DC GROUND TROUBLESHOOTING.
EVENT DATE: 121687 REPORT DATE: 011588 NSSS: WE TYPE: PWR

(NSIC 207729) AT 1745 ON 12/16/87, WITH REACTOR POWER AT 92%, A GROUND INDICATION WAS RECEIVED ON DC BUS 2. LOCATING THE GROUND REQUIRES OPENING THE DC BUS FEEDER BREAKERS WHICH DENERGIZE CONTROL POWER TO SAFETY INJECTION (SI) VALVES SUBJECT TO TECHNICAL SPECIFICATION (TS) 3.3.1. TS 3.3.1 DOES NOT EXPLICITLY PERMIT INOPERABILITY OF THESE VALVES FOR ANY PERIOD OF TIME AND THEREFORE, OPENING OF SUCH BREAKERS CONSTITUTES ENTRY INTO TS 3.0.3. DC FEEDER BREAKERS TO SI VALVES WERE OPENED ON THREE OCCASIONS (ONCE AT 200 AND TWICE AT 2234). ON EACH OCCASION, THE BREAKER WAS OPEN ONLY MOMENTARILY IN ACCORDANCE WITH ADMINISTRATIVE CONTROLS. CONSEQUENTLY, THE TOTAL TIME SI COMPONENTS WERE INOPERABLE WAS LESS THAN THE ONE HOUR PERMITTED BY TS 3.0.3. THERE WERE NO SAFETY CONSEQUENCES TO THIS EVENT SINCE AT LEAST ONE TRAIN OF SI WAS OPERABLE AT ALL TIMES. THE GROUND WAS DUE TO THE PRESENCE OF WATER INSIDE THE HOUSING OF A SOLENOID (LOCATED OUTDOORS) THAT CONTROLS CLOSURE OF THE NORTH TURBINE PLANT COOLING WATER PUMP DISCHARGE VALVE. THE COVER TO THE SOLENOID WAS FOUND LOOSE WITH TWO OF ITS SCREWS STRIPPED. ALTHOUGH THE CAUSE OF THE STRIPPED SCREWS COULD NOT BE DETERMINED, THE MOST PROBABLE CAUSE IS EXCESSIVE TIGHTENING DURING THE PERFORMANCE OF A MAINTENANCE ACTIVITY. THE GROUND WAS ELIMINATED BY DRYING THE ELECTRICAL CONNECTIONS.

[261] SAN ONOFRE 2 DOCKET 50-361 LER 85-061
MAIN STEAM SAFETY VALVES SETPOINTS OUTSIDE TECHNICAL SPECIFICATION LIMITS.
EVENT DATE: 121885 REPORT DATE: 021788 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)
VENDOR: CROSBY VALVE

(NSIC 208010) ON 12/18/85, UNIT 3 ENTERED MODE 3 AND CONTINUED TO OPERATE WITH THREE MAIN STEAM SAFETY VALVES (MSSVS) SET OUTSIDE THE ALLOWED BAND REQUIRED BY TECH SPEC 3.7.1.1 A. SINCE THIS CONDITION WAS NOT RECOGNIZED AT THE TIME, THE LCO ACTION STATEMENT, WHICH REQUIRES THAT THE POWER LEVEL-HIGH TRIP SETPOINT BE REDUCED, WAS NOT MET. ON 6/2/86, UNIT 2 ENTERED MODE 3 AND CONTINUED TO OPERATE WITH THREE MSSVS SIMILARLY SET OUTSIDE THE ALLOWABLE BAND. IN BOTH CASES, THE DEVIATIONS EXCEEDED THE MAXIMUM ALLOWED VALUE BY LESS THAN 10 PSI. THERE WAS NO SAFETY SIGNIFICANCE TO THESE OCCURRENCES SINCE UNDER THESE CONDITIONS, PEAK PRIMARY PRESSURE DURING THE MOST SEVERE TRANSIENT (LOSS OF CONDENSER VACUUM) WOULD NOT BE EXPECTED TO INCREASE ABOVE THAT PREDICTED BY THE SAFETY ANALYSIS. THE CAUSE OF THE MSSVS SETPOINTS DEVIATING FROM THE TS SETPOINTS WAS THE ERRONEOUS USE NUMERICAL DATA IN PERFORMING THE MSSV TESTING. A TABLE USED IN THE TEST PROCEDURE WAS DEVELOPED BASED UPON INFORMATION OBTAINED FROM THE TESTING DEVICE MANUAL AND MSSV INFORMATION STATED IN THE TECH SPECS. THE DATA USED TO DEVELOP THE TABLE WERE INCORRECTLY USED, AND THERE WAS NO VERIFICATION TO ESTABLISH WHETHER THE TABLE WAS CORRECT. NORMAL OPERATION OF UNITS 2 AND 3 CONTINUED UNTIL THE ERROR WAS DISCOVERED IN FEBRUARY 1987, AT WHICH TIME UNIT 2 WAS IN MODE 3 AND UNIT 3 WAS IN A REFUELING OUTAGE.

[262] SAN ONOFRE 2 DOCKET 50-361 LER 86-029 REV 02
UPDATE ON TRIP DURING TRANSFER OF NON-1E POWER SUPPLY.
EVENT DATE: 121086 REPORT DATE: 123187 NSSS: CE TYPE: PWR
VENDOR: RCA ELECTRONIC COMPONENTS

(NSIC 207666) ON 12/10/86 AT 1037, WITH UNIT 2 AT 93% POWER, THE TURBINE TRIPPED DURING A POWER INTERRUPTION TO THE TURBINE GOVERNOR CONTROL SYSTEM (TGCS), CAUSING A REACTOR TRIP. THE STEAM BYPASS CONTROL SYSTEM (SBCS) DID NOT INITIALLY ACTUATE AND A MAIN STEAM SAFETY VALVE BRIEFLY ACTUATED. THE TRIP RECOVERY PROCEEDED NORMALLY, ALTHOUGH START-UP CHANNEL 'B' FAILED, AND PLANT PROTECTION

SYSTEM (PPS) CHANNEL 'A' DID NOT TRIP. ALL OTHER REQUIRED SAFETY RELATED EQUIPMENT FUNCTIONED AS DESIGNED, AND THERE WERE NO SAFETY CONSEQUENCES. THE NON-IE 120 VAC LOAD WAS BEING TRANSFERRED FROM THE NON-IE UNINTERRUPTIBLE POWER SUPPLY (UPS) INVERTER TO THE ALTERNATE SOURCE. A PROCEDURAL STEP TO DEFEAT THE AUTOMATIC RETRANSFER CIRCUIT WAS NOT PERFORMED, CAUSING THE LOAD TO TRANSFER BACK TO THE PRIMARY SOURCE. WHEN THE UPS INVERTER WAS DISCONNECTED UNDER LOAD, THE AUTOMATIC TRANSFER TO THE ALTERNATE SOURCE DID NOT OCCUR IN TIME TO PREVENT THE TRIP. THE UPS IS EQUIPPED WITH AN AUTOMATIC TRANSFER SWITCH WHICH AUTOMATICALLY TRANSFERS THE LOAD TO THE ALTERNATE SOURCE ON LOSS OF INVERTER OUTPUT VOLTAGE. THE TRANSFER SWITCH WAS FOUND TO OPERATE CORRECTLY; HOWEVER, THE ENSUING TRANSIENT IS BELIEVED TO HAVE CAUSED THE TRIP. THE EVENT RESULTED FROM THE FAILURE TO FOLLOW THE PROCEDURE; ADDITIONALLY, THE JOB DID NOT RECEIVE THE CORRECT LEVEL OF ATTENTION BY OPERATIONS PERSONNEL.

[263] SAN ONOFRE 2 DOCKET 50-361 LER 87-004 REV 01
UPDATE ON UNIT 2 TRIP FROM LOW STEAM GENERATOR LEVEL.
EVENT DATE: 032887 REPORT DATE: 123187 NSSS: CE TYPE: PWR
VENDOR: COPES-VULCAN, INC.

(NSIC 207661) AT 0010 ON 3/29/87, WHILE REDUCING UNIT 2 REACTOR POWER FROM 100% TO 85% TO PERMIT CLEANING OF A CONDENSER WATER BOX, THE REACTOR TRIPPED ON A LOW LEVEL IN STEAM GENERATOR E-089 AFTER MAIN FEEDWATER CONTROL VALVE 2FV-1111 UNCOUPLED FROM THE ACTUATOR, CAUSING A REDUCTION IN FLOW. THE EMERGENCY FEEDWATER ACTUATION SYSTEM (EFAS) ACTUATED AS DESIGNED TO PROVIDE AUXILIARY FEEDWATER AND RESTORE STEAM GENERATOR LEVELS TO NORMAL. INSPECTION OF 2FV-1111 FOUND THE LOCKNUT THAT SECURES THE VALVE STEM TO THE ACTUATOR HAD LOOSENED. INSPECTION OF THE REMAINING FEEDWATER CONTROL VALVES IN BOTH UNITS 2 AND 3 WAS PERFORMED AND ALL VALVE STEMS WERE FOUND TIGHTLY SECURED TO THE COUPLING BLOCK. SUBSEQUENT INVESTIGATION DETERMINED THAT THE LOCKNUT MAY NOT HAVE BEEN ADEQUATELY TORQUED DURING CALIBRATION OF THE VALVE ON 3/17/87 SINCE TORQUING REQUIREMENTS FOR THE LOCKNUT WERE NOT SPECIFIED IN THE CALIBRATION PROCEDURE. APPROPRIATE PROCEDURES HAVE BEEN REVISED. ANTI-ROTATIONAL LOCKING WASHERS HAVE BEEN INSTALLED ON THE UNIT 2 MAIN FEEDWATER CONTROL VALVES. RELATIVE POSITION INDICATION OF MAIN FEEDWATER FLOW CONTROL VALVES WILL BE REVIEWED DAILY TO IDENTIFY CHANGES INDICATIVE OF IMPENDING UNCOUPLING UNTIL THE ANTI-ROTATIONAL LOCKING WASHERS ARE INSTALLED ON THE UNIT 3 MAIN FEEDWATER FLOW CONTROL VALVES AT THE NEXT REFUELING OUTAGE.

[264] SAN ONOFRE 2 DOCKET 50-361 LER 87-007 REV 01
UPDATE ON CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) SPURIOUS ACTUATIONS WERE INITIATED BY ELECTRICAL NOISE SPIKES.
EVENT DATE: 060787 REPORT DATE: 012788 NSSS: CE TYPE: PWR

(NSIC 208050) WITH UNIT 2 IN MODE 1 AT 100% POWER, FOUR SPURIOUS ACTUATIONS OF THE SAN ONOFRE UNIT 2 RAIN "A" CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) OCCURRED BETWEEN JUNE 7, 1987 AND JUNE 9, 1987. THESE ACTUATIONS WERE INITIATED BY ELECTRICAL NOISE SPIKES WHICH AFFECTED CONTAINMENT AREA RADIATION MONITOR 2RT-7856. THESE NOISE SPIKES ARE MOST FREQUENTLY GENERATED BY OPERATION OF TRAIN "A" HIGH PRESSURE SAFETY INJECTION (HPSI) SYSTEM COMPONENTS. THE MINI-PURGE ISOLATION VALVES WHICH WERE IN THE OPEN POSITION, CLOSED IN ACCORDANCE WITH DESIGN REQUIREMENTS. IN ALL CASES, CONTAINMENT RADIATION LEVELS WERE BELOW THE PIS ACTUATION SET POINT. CABLING FOR THE TRAIN "A" CPIS MONITOR 2RT-7856 IS ROUTED NEAR CABLE TRAYS CONTAINING CONTROL AND POWER CABLES ASSOCIATED WITH THE TRAIN "A" HPSI PUMP P-017, THE TRAIN "A" CABLING FOR HPSI SWING PUMP P-018 AND THE TRAIN "A" HPSI VALVES. CORRECTIVE MEASURES PREVIOUSLY TAKEN, INCLUDED REPLACING THE DETECTOR AND ENSURING PROPER GROUNDING OF THE DETECTOR AND ASSOCIATED CABLING. THE RESULTS OF INVESTIGATIONS INTO THE CAUSE OF THESE AND OTHER SIMILAR SPURIOUS ACTUATIONS, AND CORRECTIVE ACTIONS, ARE DISCUSSED IN DETAIL N LER 2-87-15 (DOCKET NO. 50-361). IMPLEMENTATION OF THESE CORRECTIVE

ACTIONS ON UNIT HAS, THUS FAR, EFFECTIVELY PREVENTED THE TYPE OF SPURIOUS ACTUATIONS REPORTED HERE.

[265] SAN ONOFRE 2 DOCKET 50-361 LER 87-030
DELINQUENT COLLECTION AND ANALYSIS OF CONTAINMENT PURGE EFFLUENT SAMPLES.
EVENT DATE: 112787 REPORT DATE: 123187 NSSS: CE TYPE: PWR

(NSIC 207644) TECHNICAL SPECIFICATION (TS) SURVEILLANCE REQUIREMENT 4.11.2.1.2, TABLE 4.11-2 REQUIRES THAT CONTINUOUS IODINE AND PARTICULATE SAMPLES FROM THE CONTAINMENT PURGE STACK BE TAKEN, COLLECTED EVERY SEVEN DAYS AND ANALYZED WITHIN 48 HOURS OF BEING COLLECTED. FROM 11/18/87 AT 1855 UNTIL 11/21/87 AT 1120, WITH THE NORMAL CONTAINMENT PURGE MONITOR 2RT-7828 INOPERABLE FOR SURVEILLANCE TESTING, THE CONTAINMENT PURGE RELEASE WAS BEING MONITORED BY EFFLUENT MONITOR 2RT- 7865. ON 12/1/87 AT 1440, DURING A ROUTINE REVIEW OF EFFLUENT DATA, IT WAS DISCOVERED THAT THE 11/18 TO 11/21 SAMPLES FROM 2RT-7865 HAD NOT BEEN COLLECTED AND ANALYZED. THE TS INTERVAL, INCLUDING THE 25% EXTENSION, FOR THE COLLECTION OF THESE SAMPLES WAS EXCEEDED ON 11/27. SUBSEQUENT ANALYSES OF THE SAMPLES INDICATED THE DISCHARGE CONSTITUENTS TO BE CONSISTENT WITH INDEPENDENT PURGE SAMPLES TAKEN FROM 11/21 TO 11/24. BASED ON THIS, EFFLUENT RELEASES WERE DETERMINED TO BE WELL WITHIN TS LIMITS. THERE WAS, THEREFORE, NO SAFETY SIGNIFICANCE TO THIS EVENT. THE CAUSE OF THIS EVENT WAS INADEQUATE ADMINISTRATIVE CONTROLS. AFTER 2RT-7828 WAS RETURNED TO SERVICE ON 11/21, THE DAY SHIFT CHEMISTRY TECHNICIAN VERIFIED THAT THE RADIATION MONITORS AND ASSOCIATED SAMPLE LINES HAD BEEN PROPERLY REALIGNED, BUT COLLECTION OF THE SAMPLES FROM 2RT-7865 WAS DEFERRED TO THE SWING SHIFT.

[266] SAN ONOFRE 2 DOCKET 50-361 LER 87-027
TECHNICAL SPECIFICATION FIRE DOOR SURVEILLANCE DISCREPANCIES.
EVENT DATE: 113087 REPORT DATE: 123087 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 207641) ON 11/30/87, WITH UNIT 2 IN MODE 5 AND UNIT 3 AT 100% POWER, A QUALITY ASSURANCE AUDIT DETERMINED THAT THE TECHNICAL SPECIFICATION (TS) 4.7.9.1 SURVEILLANCE REQUIREMENTS HAD NOT BEEN FULFILLED FOR 11 FIRE DOORS. SPECIFICALLY, THE 6-MONTH VISUAL INSPECTION OF THE CLOSING MECHANISM AND LATCHES ON 7 OF THE 11 DOORS HAD NOT BEEN ADEQUATELY PERFORMED AND THE 18-MONTH FUNCTIONAL TESTING HAD NOT BEEN PERFORMED ON ALL 11 DOORS. IN ADDITION, IT WAS IDENTIFIED THAT THE 6-MONTH AND 18-MONTH SURVEILLANCES ON FIRE DOORS CONTAINING TWO LEAVES (DOUBLE DOORS) MAY NOT HAVE BEEN PERFORMED CORRECTLY. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE SUBSEQUENT SURVEILLANCES OF THE FIRE DOORS INVOLVED DEMONSTRATED THEM TO BE OPERABLE. THE CAUSES OF THESE SURVEILLANCE DISCREPANCIES ARE PROCEDURAL DEFICIENCY AND INADEQUATE TRAINING. THE FIRE DOOR SURVEILLANCE PROCEDURE DOES NOT REQUIRE VISUAL INSPECTION AND FUNCTIONAL TESTING OF VARIOUS DOORS DUE TO ALARA CONSIDERATIONS. THE PROCEDURE ALSO DOES NOT REQUIRE CYCLING OF WATER-TIGHT DOORS TO SATISFY FUNCTIONAL TESTING REQUIREMENTS. ADDITIONALLY, THE TRAINING PROGRAM FOR FIRE DOOR INSPECTORS IS NOT SUFFICIENTLY PRESCRIPTIVE TO ENSURE COMPLETE SURVEILLANCE OF BOTH LEAVES ON DOUBLE DOORS.

[267] SAN ONOFRE 2 DOCKET 50-361 LER 87-028
CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) ACTUATION DUE TO HIGH IMPEDANCE.
EVENT DATE: 120687 REPORT DATE: 010588 NSSS: CE TYPE: PWR

(NSIC 207642) AT 0019 ON 12/06/87, WITH UNIT 2 IN MODE 3, A SPURIOUS ACTUATION OF TRAIN "B" CPIS WAS INITIATED FROM CONTAINMENT AREA RADIATION MONITOR 2RT-7857. THE CONTAINMENT PURGE WAS NOT IN OPERATION AT THE TIME OF THE ACTUATION. AT 0055 ON 12/06/87, FOLLOWING VERIFICATION THAT CONTAINMENT RADIATION LEVELS WERE BELOW THE ACTUATION SETPOINT, CPIS WAS RESET. INVESTIGATION INTO THE CAUSE OF THE

ACTUATION DETERMINED THAT A POOR ELECTRICAL CONNECTION EXISTED BETWEEN THE 2RT-7857 MONITOR MODULE AND ITS CABLE CONNECTOR WHICH RESULTED IN A HIGH IMPEDANCE CONDITION. THE POOR CONNECTION WAS CAUSED BY A BUILDUP OF DEPOSITS ON THE CONNECTOR PINS. SUCH CONNECTOR PINS HAVE NOT BEEN CLEANED ON A PERIODIC BASIS. THE 18-MONTH CALIBRATION FOR THOSE RADIATION MONITORS WHICH ACTUATE EQUIPMENT AND HAVE SIMILAR MODULE-TO-CABLE CONNECTORS WILL BE REVISED TO REQUIRE THE CLEANING OF CONNECTOR PINS. THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE ALL CPIS COMPONENTS OPERATED IN ACCORDANCE WITH DESIGN.

[268] SAN ONOFRE 2 DOCKET 50-361 LER 87-029
FUEL HANDLING ISOLATION SYSTEM (FHIS) TRAIN "B" ACTUATION.
EVENT DATE: 121287 REPORT DATE: 011188 NSSS: CE TYPE: PWR

(NSIC 207643) ON DECEMBER 12, 1987, AT 0043, WITH UNIT 2 IN MODE 2 AND THE REACTOR AT 1% POWER, A SPURIOUS ACTUATION OF TRAIN "B" OF THE FUEL HANDLING ISOLATION SYSTEM (FHIS) OCCURRED. THERE WAS NO INDICATION OF INCREASED RADIATION LEVELS IN THE FUEL HANDLING BUILDING (FHB). AFTER THE FHB AIRBORNE ACTIVITY LEVELS WERE CONFIRMED TO BE NORMAL, THE FHIS WAS RESET AND FHB VENTILATION WAS RETURNED TO NORMAL. ALL FHIS TRAIN "B" COMPONENTS FUNCTIONED AS DESIGNED. AT THE TIME OF THE FHIS ACTUATION, THE CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) TRAIN "B" MONITOR WAS ACTUATED AS A PART OF THE MONTHLY CHANNEL FUNCTIONAL SURVEILLANCE TEST. THE FHIS ACTUATION HAS BEEN ATTRIBUTED TO AN INDUCED VOLTAGE SPIKE WHICH CAN OCCUR WHEN THE TRAIN "B" CPIS MONITOR ACTUATES. AS PREVIOUSLY REPORTED, THE CPIS AND FHIS WIRING IS ROUTED IN COMMON WIRE BUNDLES. IT IS BELIEVED THAT ACTUATION OF THE CPIS CIRCUIT, WHICH CAUSES DE-ENERGIZATION OF RELAY OPERATING COILS, IS SUFFICIENT TO INDUCE A VOLTAGE SPIKE IN THE OTHER CIRCUIT'S WIRING. THE CABINET HOUSING THE CPIS AND FHIS WILL BE MODIFIED TO SEPARATE THE WIRING CURRENTLY ROUTED IN COMMON BUNDLES. THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT AS ALL FHIS TRAIN "B" COMPONENTS OPERATED IN ACCORDANCE WITH DESIGN.

[269] SAN ONOFRE 2 DOCKET 50-361 LER 87-031
MANUAL REACTOR TRIP DUE TO FEEDWATER ISOLATION VALVE FAILING CLOSED.
EVENT DATE: 121787 REPORT DATE: 011888 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 1 (PWR)
VENDOR: CONTROL COMPONENTS
COOPER INDUSTRIES
MAROTTA SCIENTIFIC CONTROLS, INC.

(NSIC 207931) AT 0831 ON 12/17/87, WITH UNIT 2 AT 75% POWER, ONE OF TWO MAIN FEEDWATER ISOLATION VALVES (MFIV) FAILED CLOSED, CAUSING BOTH MAIN FEEDWATER PUMPS TO TRIP ON HIGH DISCHARGE PRESSURE. UNIT 2 WAS MANUALLY TRIPPED IN ACCORDANCE WITH OPERATING PRACTICE TO MINIMIZE THE EFFECTS OF THE LOSS OF FEEDWATER. CONCURRENT WITH THE AUTOMATIC INITIATION OF THE EMERGENCY FEEDWATER ACTUATION SYSTEM FOR STEAM GENERATOR #2 (EFAS 2) AS A RESULT OF LOW LEVEL IN STEAM GENERATOR #2, EFAS 1 (FOR SG #1) AND EFAS 2 WERE MANUALLY ACTUATED. PLANT CONDITIONS WERE STABILIZED, AND RECOVERY PROCEEDED NORMALLY. THIS EVENT HAD NO EFFECT ON THE HEALTH AND SAFETY OF PLANT PERSONNEL OR THE PUBLIC SINCE ALL SAFETY SYSTEMS OPERATED AS DESIGNED. THE THREADED CONDUIT CONNECTION TO THE AFFECTED MFIV SOLENOID WAS FOUND LOOSE, AND THE CABLE PENETRATION AREA INTO THE CONDUIT CONNECTOR WAS NOT SEALED. BY ONE OR BOTH OF THESE PATHS, WATER AND FOREIGN MATERIAL ENTERED THE SOLENOID HOUSING AND CAUSED CORROSION OF THE POWER LEADS AND TERMINAL BLOCK. THIS RESULTED IN FAILURE OF THE POWER LEAD TO THE MFIV SOLENOID AND CLOSURE OF THE MFIV. THE MAINTENANCE PROCEDURE FOR REASSEMBLY OF THE SOLENOID DID NOT PROVIDE SUFFICIENT GUIDANCE REGARDING THE INSTALLATION OF THE CONDUIT AND SEALING OF THE CABLE PENETRATION TO ENSURE THEIR WATER TIGHTNESS.

[270] SAN ONOFRE 3 DOCKET 50-362 LER 87-004 REV 01
UPDATE ON STEAM GENERATOR BLOWDOWN EFFLUENT SAMPLE NOT TAKEN.
EVENT DATE: 031787 REPORT DATE: 122987 NSSS: CE TYPE: PWR

(NSIC 207688) ON 3/17/87, AT 1249, WITH UNIT 3 AT 100% POWER, THE STEAM GENERATOR BLOWDOWN (EIIS SYSTEM CODE WI), NORMALLY ALIGNED TO THE CONDENSER, WAS DIRECTED TO THE OUTFALL DISCHARGE. TECHNICAL SPECIFICATION (TS) TABLE 4.11-1 REQUIRES THAT A SAMPLE BE OBTAINED DURING PERIODS OF CONTINUOUS DISCHARGE FOR USE IN THE WEEKLY COMPOSITE SAMPLE ANALYSIS. HOWEVER, THIS SAMPLE WAS NOT OBTAINED WHILE BLOWDOWN WAS ALIGNED TO THE OUTFALL. BLOWDOWN WAS REALIGNED TO THE CONDENSER AT 1453. A CHEMISTRY TECHNICIAN DISCOVERED THE MISSED SAMPLE ON 3/19/87 DURING A ROUTINE REVIEW OF BLOWDOWN RECORDS. THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT, AS TWO EFFLUENT MONITORS WERE IN SERVICE FOR THE RELEASE PATH, AND RELEASE LIMITS WERE NOT EXCEEDED. THE CONTROL OPERATOR (CO) HAD DIRECTED THE NUCLEAR PLANT EQUIPMENT OPERATOR (NPEO) TO REALIGN BLOWDOWN TO THE OUTFALL TO TEST FOR CONDENSER AIR IN-LEAKAGE. THE PROCEDURE WHICH WAS BEING USED BY THE NPEO IN THE FIELD REFERENCES THE TS IN THE PRECAUTIONS SECTION, AND ALSO INSTRUCTS THAT CHEMISTRY BE NOTIFIED TO INITIATE SAMPLING. HOWEVER, DUE TO A MISCOMMUNICATION BETWEEN THE CO AND NPEO, CHEMISTRY WAS NOT NOTIFIED TO INITIATE SAMPLING. THE OPERATORS INVOLVED NEITHER REVIEWED THE PRECAUTIONS REGARDING THE SAMPLING REQUIREMENTS. PERIODIC OPERATOR TRAINING WILL CONTINUE TO EMPHASIZE THE IMPORTANCE OF PROCEDURAL PRECAUTIONS.

[271] SEABROOK 1 DOCKET 50-443 LER 87-019 REV 01
UPDATE ON FAILURE OF GOULD/TELEMECANIQUE J-10 RELAYS IN NON-SAFETY-RELATED CIRCULATING WATER SYSTEM APPLICATION.
EVENT DATE: 101387 REPORT DATE: 012988 NSSS: WE TYPE: PWR
VENDOR: GOULD INC.

(NSIC 207986) THREE ITE GOULD/TELEMECANIQUE CLASS J CONTROL RELAYS FAILED IN NON-SAFETY RELATED CIRCULATING WATER (CW) SYSTEM CIRCUIT APPLICATIONS DURING A FOUR MONTH PERIOD BETWEEN APRIL 1987 AND AUGUST 1987. THE IMMEDIATE CAUSE OF THE RELAY FAILURES WAS THE BREAK UP OF EMBRITTLED PLASTIC AT THE END OF THE MAGNET YOKE ASSEMBLY. THE BROKEN PIECES CAUSED THE MOVEABLE PART OF THE RELAY TO JAM. THE BINDING COULD OCCUR IN EITHER THE CLOSED OR OPEN CONDITION. THE EMBRITTLEMENT RESULTED WHEN THE PLASTIC WAS SUBJECTED TO HIGH TEMPERATURES FOR A LONGER DURATION THAN THE MATERIAL COULD WITHSTAND. THE SOURCE OF THE HIGH TEMPERATURE WAS DETERMINED TO BE EXCESSIVE HEAT GENERATED BY THE MAGNETIC COIL WHEN THE RELAY IS ENERGIZED. ON OCTOBER 9, 1987, TELEMECANIQUE NOTIFIED THE NRC PURSUANT TO 10 CFR 21 REGARDING EMBRITTLEMENT OF THE ARMATURE CARRIER IN CLASS J INDUSTRIAL CONTROL RELAYS THAT HAVE NUMBER 816 A.C. COILS. ON OCTOBER 13, 1987, NEW HAMPSHIRE YANKEE NOTIFIED THE NRC OF A POTENTIAL UNANALYZED CONDITION IN ACCORDANCE WITH 10 CFR 50.72. J-10 RELAYS WILL BE REPLACED WITH NEW TELEMECANIQUE RELAYS CONTAINING COILS AND PLASTIC QUALIFIED FOR THE SPECIFIED RANGE OF VOLTAGES AND ENVIRONMENTAL CONDITIONS. CORRECTIVE MEASURES WILL BE COMPLETED FOR SAFETY-RELATED J-10 RELAYS PRIOR TO ENTRY INTO MODE 4.

[272] SEABROOK 1 DOCKET 50-443 LER 87-025
INCOMPLETE SURVEILLANCE TESTING DATA.
EVENT DATE: 120787 REPORT DATE: 010688 NSSS: WE TYPE: PWR

(NSIC 207700) ON DECEMBER 7, 1987, IT WAS DISCOVERED THAT PUMP VIBRATION READINGS FOR THE PRIMARY COMPONENT COOLING WATER (PCCW), RESIDUAL HEAT REMOVAL (RHR), AND SERVICE WATER (SW) SYSTEM PUMPS HAD NOT BEEN TAKEN IN ACCORDANCE WITH INSERVICE TESTING REQUIREMENTS. THE LOWER END OF THE REQUIRED FREQUENCY RANGE FOR VIBRATION DISPLACEMENT VALUES WAS NOT EFFECTIVELY MONITORED BECAUSE IT WAS FILTERED OUT BY THE TEST EQUIPMENT. AS A RESULT THE OPERATING RHR PUMP WAS DECLARED INOPERABLE DUE TO INCOMPLETE SURVEILLANCE DATA. THE ROOT CAUSE OF THIS PROBLEM WAS THAT INCORRECT INFORMATION WAS PROVIDED FOR THE OPERATION OF THE TEST

EQUIPMENT. THIS PROBLEM WAS RECOGNIZED DURING THE PERFORMANCE OF SURVEILLANCE TESTS ON THE PCCW PUMPS. FURTHER INVESTIGATION DETERMINED THAT NONE OF THE PCCW, RHR OR SW SYSTEM PUMPS WERE BEING MONITORED OVER THE FULL FREQUENCY RANGE REQUIRED BY THE ASME CODE SECTION XI. UPON IDENTIFICATION OF THIS PROBLEM, THE OPERABLE RHR PUMP WAS DECLARED INOPERABLE AND SURVEILLANCE TESTING WAS PERFORMED TO ESTABLISH NEW BASELINE REFERENCE VALUES AND ACCEPTANCE CRITERIA. ADDITIONALLY, A REVISION WAS MADE TO THE RHR SURVEILLANCE PROCEDURE ADDING A PRECAUTION TO ALERT OPERATORS TO THE NEED FOR MONITORING VIBRATION DISPLACEMENT ON THE LOWER FREQUENCY RANGE. OPERATORS WERE MADE AWARE OF THE PROBLEM AND THE REQUIREMENTS FOR PROPER MEASUREMENT OF VIBRATION DISPLACEMENT.

[273] SEQUOYAH 1 DOCKET 50-327 LER 87-050 REV 01
UPDATE ON THE CONTAINMENT SPRAY PUMPS WILL NOT DELIVER THE DESIGN BASIS FLOW RATE DUE TO A DESIGN DEFICIENCY.
EVENT DATE: 072387 REPORT DATE: 010788 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207715) THIS LER IS BEING REVISED TO CHANGE A PART OF THE CORRECTIVE ACTION. IT WAS ORIGINALLY DETERMINED THAT VERIFYING THAT THE CONTAINMENT SPRAY SYSTEM (CSS) PUMPS WOULD DEVELOP A HEAD PRESSURE OF 157 PSID AT 4,750 GALLONS PER MINUTE (GPM) WOULD ENSURE COMPLIANCE WITH SURVEILLANCE REQUIREMENT (SR) 4.6.2.1.B. HOWEVER, A MORE PRECISE CALCULATION HAS CONCLUDED ONLY 143 PSID IS REQUIRED TO MEET THE SR. ON JULY 23, 1987, WITH UNITS 1 AND 2 IN MODE 5 (0% POWER, 4 PSIG, 133 DEGREES F AND 0% POWER, ATMOSPHERIC PRESSURE, 134 DEGREES F, RESPECTIVELY), IT WAS DETERMINED THAT THE CSS PUMPS COULD NOT DELIVER THE 4,750 GPM (FSAR) AND ACCIDENT ANALYSIS. DUE TO AN UNDERSIZED FLOW ORIFICE AND A PUMP DISCHARGE PRESSURE ACCEPTANCE CRITERIA WHICH WAS TOO LOW, THE PUMPS MAY ONLY DELIVER A FLOW RATE OF 4,400 GPM AT THE CSS HEADERS. THIS CONDITION WAS DETERMINED TO BE THE RESULT OF A DESIGN DEFICIENCY. DNE COULD NOT LOCATE THE ORIGINAL DESIGN CALCULATION, AND WHEN NEW CALCULATIONS WERE GENERATED, THE EXISTING DESIGN COULD NOT BE JUSTIFIED. TO CORRECT THIS DEFICIENCY, THE FLOW ORIFICE HAS BEEN BORED OUT SUCH THAT ESSENTIALLY, NO ORIFICE WILL REMAIN IN THE CSS LINE. ALSO, A TECH SPEC CHANGE HAS BEEN SUBMITTED TO INCREASE THE ACCEPTANCE CRITERIA FOR THE PUMPS' TOTAL DISCHARGE HEAD PRESSURE.

[274] SEQUOYAH 1 DOCKET 50-327 LER 87-063 REV 01
UPDATE ON CONTROL ROOM ISOLATION CAUSED BY SIMULTANEOUS CONTACT OF TWO TERMINALS WITH AN OPEN ALLIGATOR CLIP DEFEATING BLOCK FUNCTION DURING RADIATION MONITOR TESTING.
EVENT DATE: 092487 REPORT DATE: 012188 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
VENDOR: GENERAL ATOMIC CO.

(NSIC 207990) THIS REPORT IS BEING REVISED TO PROVIDE UPDATED INFORMATION IN THE AREA OF CORRECTIVE ACTION. ON SEPTEMBER 24, 1987, AT 1057 EST WITH UNITS 1 AND 2 IN MODE 5 (0 PERCENT POWER, 3 PSIG, 120 DEGREES F AND 0 PERCENT POWER, 65 PSIG, 119 DEGREES F, RESPECTIVELY), AN INADVERTENT CONTROL ROOM VENTILATION ISOLATION WAS INITIATED. THIS WAS CAUSED BY AN ALLIGATOR CLIP ACCIDENTALLY CONTACTING TWO TERMINAL POINTS SIMULTANEOUSLY DURING PERFORMANCE OF A RADIATION MONITOR SURVEILLANCE INSTRUCTION (SI)-82, "FUNCTIONAL TESTS FOR THE RADIATION MONITORING SYSTEM." THIS DEFEATED THE BLOCK FUNCTION ON THE MAIN CONTROL ROOM INTAKE MONITOR O-RM-90-126 AND INITIATED A HIGH RADIATION SIGNAL. THE HIGH RADIATION SIGNAL THEN GENERATED A MAIN CONTROL ROOM ISOLATION. UPON DETERMINATION THAT THE SIGNAL WAS INVALID, THE VENTILATION SYSTEM WAS RESET AND AN INVESTIGATION TO DETERMINE THE CAUSE WAS PERFORMED. DURING TROUBLESHOOTING TO DETERMINE THE CAUSE OF THE ISOLATION, A PREPLANNED ISOLATION OCCURRED INDICATING THE HIGH RADIATION SIGNAL WAS GENERATED BY AN OPEN ALLIGATOR CLIP MAKING CONTACT WITH TWO TERMINALS. SI-82 HAS BEEN PERFORMED THREE ADDITIONAL TIMES AFTER THIS OCCURRENCE BEFORE THE

PROCEDURE WAS REVISED BUT NO ADDITIONAL INADVERTENT CONTROL ROOM VENTILATION ISOLATIONS OCCURRED.

[275] SEQUOYAH 1 DOCKET 50-327 LER 87-064 REV 01
UPDATE ON IMPROPER FIT OF EMERGENCY RAW COOLING WATER FLOOD MODE SPOOL PIECES DUE TO MINOR PIPING ORIENTATION CHANGES.
EVENT DATE: 101387 REPORT DATE: 011588 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207893) THIS REPORT IS REVISED TO PROVIDE ADDITIONAL INFORMATION REGARDING COMPLETED CORRECTIVE ACTIONS. THIS REPORT WAS SUBMITTED AS A "VOLUNTARY REPORT" TO IDENTIFY A POTENTIAL PROBLEM WITH FLOOD MODE SPOOL PIECES AND TO KEEP NRC INFORMED OF ONGOING ACTIVITIES AT SEQUOYAH NUCLEAR PLANT. ON OCTOBER 13, 1987, WITH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN), A CONDITION WHERE 13 EMERGENCY RAW COOLING WATER (ERCW) FLOOD MODE SPOOL PIECES DID NOT FIT PROPERLY WAS DISCOVERED DURING THE INTEGRATED DESIGN INSPECTION (IDI) FOR THE ERCW SYSTEM. A CONDITION ADVERSE TO QUALITY REPORT WAS INITIATED TO INVESTIGATE THE POTENTIAL PROBLEM. DURING THE STAGE II FLOOD PREPARATION PLAN, THESE SPOOL PIECES ARE REQUIRED TO BE INSTALLED TO REPLACE COMPONENT COOLING SYSTEM WITH ERCW AS COOLING WATER FOR SPENT FUEL POOL HEAT EXCHANGERS, REACTOR COOLANT PUMP THERMAL BARRIERS, SAMPLE SYSTEM HEAT EXCHANGERS, AND TO REPLACE RAW COOLING WATER WITH ERCW FOR THE ICE CONDENSER REFRIGERATION SYSTEM. THE ROOT CAUSE OF THE CONDITION COULD NOT BE SPECIFICALLY IDENTIFIED. THE MOST PROBABLE CAUSE IS THAT GRADUAL CHANGES IN INTERNAL PIPING STRESSES OVER THE YEARS ALONG WITH SYSTEM MODIFICATIONS HAVE RESULTED IN SLIGHT CHANGES IN THE ORIENTATION OF THE PIPING. EACH SPOOL PIECE WAS INSPECTED, AND IT WAS DETERMINED THAT ALL COULD BE INSTALLED WITH HANGER OR GASKET ADJUSTMENTS, OR ALTERATIONS.

[276] SEQUOYAH 1 DOCKET 50-327 LER 87-065 REV 02
UPDATE ON ERCW SCREEN WASH PUMPS WERE OMITTED FROM THE ASME SECTION XI TEST PROGRAM RESULTING IN POTENTIAL DEGRADATION OF ERCW FLOW TO BOTH UNITS.
EVENT DATE: 102387 REPORT DATE: 011588 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207941) THIS LER IS BEING REVISED TO UPDATE THE CORRECTIVE ACTION SECTION OF THIS REPORT. THE LER PREVIOUSLY INDICATED A SYSTEM OPERATING INSTRUCTION (SOI) WOULD DETAIL NORMAL AND ACCIDENT CONDITIONS FOR THE ESSENTIAL RAW COOLING WATER (ERCW) SCREEN WASH PUMPS. HOWEVER, THE SOI ONLY ADDRESSES NORMAL CONDITIONS, AND THE ABNORMAL OPERATING INSTRUCTION (AOI) ADDRESSES ACCIDENT CONDITIONS. ON OCTOBER 23, 1987, WITH UNITS 1 AND 2 IN MODE 5 (0 PERCENT POWER, 3 PSIG, 125 DEGREES F AND 0 PERCENT POWER, 100 PSIG, 127 DEGREES F, RESPECTIVELY), IT WAS DETERMINED THAT THE ERCW SCREEN WASH PUMPS WERE NOT INCLUDED IN THE ASME SECTION XI TEST PROGRAM. PERIODIC TESTING OF THE ERCW SCREEN WASH PUMPS IN ACCORDANCE WITH COMPLETE FAILURE OF THE ERCW SCREEN WASH SYSTEM COULD RESULT IN THE GRADUAL CLOGGING OF THE ERCW TRAVELING SCREENS AND SUBSEQUENT DEGRADATION OF ERCW FLOW TO BOTH UNITS. HOWEVER, THERE IS A HIGH DEGREE OF CONFIDENCE THAT THESE PUMPS WOULD HAVE FUNCTIONED PROPERLY DURING ACCIDENT CONDITIONS. THE PUMPS ARE TVA CLASS C SAFETY-RELATED PUMPS, WHICH HAVE BEEN IN USE SINCE INITIAL PLANT STARTUP AND THERE HAS BEEN NO HISTORY OF CLOGGING PROBLEMS WITH THE TRAVELING SCREENS DUE TO INSUFFICIENT FLOW FROM THE SCREEN WASH PUMPS.

[277] SEQUOYAH 1 DOCKET 50-327 LER 87-072 REV 01
UPDATE ON ENGINEERED SAFETY FEATURE EQUIPMENT ROOM COOLERS ARE INADEQUATE FOR HEAT LOADS DUE TO INADEQUATE DESIGN INPUTS INTO DESIGN CALCULATIONS.
EVENT DATE: 111087 REPORT DATE: 011588 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207528) THIS LER IS BEING REVISED TO UPDATE THE CORRECTIVE ACTION SECTION OF THIS REPORT. THIS LER PREVIOUSLY INDICATED A TEMPORARY FILTRATION SYSTEM WOULD BE ADDED BEFORE THE PLANT ENTERED MODE 4. HOWEVER, ADDITIONAL ANALYSIS DETERMINED THE TEMPORARY FILTRATION SYSTEM IS NOT NECESSARY TO ENSURE ENGINEERED SAFETY FEATURE (ESF) COOLER OPERABILITY. A PERMANENT FILTRATION SYSTEM WILL BE ADDED AS DISCUSSED IN THE ORIGINAL REPORT. ON NOVEMBER 10, 1987, UNIT 1 AND UNIT 2 WERE IN MODE 5 (0% POWER, 3 PSIG, 125 DEGREES F AND 0% POWER, 120 PSIG, 100 DEGREES F, RESPECTIVELY) WHEN A POTENTIAL REPORTABLE OCCURRENCE WAS INITIATED WHICH CONCLUDED THAT SEVERAL OF THE ENGINEERED SAFETY FEATURE (ESF) COOLERS HAD INSUFFICIENT AIR FLOW TO MAINTAIN TEMPERATURES WITHIN THE ENVIRONMENTAL QUALIFICATION TEMPERATURE PROFILES. THIS DEFICIENCY WAS DISCOVERED THROUGH A CALCULATION REVIEW PROGRAM WHERE IT WAS FOUND THAT IMPROPER DESIGN INPUTS WERE USED IN THE CALCULATIONS, RESULTING IN INADEQUATE COOLER DESIGN PARAMETERS. THIS DEFICIENCY IS NOT CONSIDERED TO HAVE A SIGNIFICANT SAFETY IMPACT BASED ON CURRENT PLANT CONDITIONS (MODE 5) AND BECAUSE LOWER COOLING WATER TEMPERATURES EXIST THAN THOSE USED IN THE DESIGN CALCULATIONS. THE ROOT CAUSE WAS ATTRIBUTED TO THE LACK OF AN ADEQUATE DESIGN CONTROL PROGRAM AT THE TIME THE CALCULATIONS WERE PERFORMED.

[278] SEQUOYAH 1 DOCKET 50-327 LER 87-073 REV 01
UPDATE OF INADEQUATE DESIGN OF CENTRIFUGAL CHARGING PUMP'S AUXILIARY LUBE OIL SYSTEM COULD RESULT IN THE FAILURE OF HIGH HEAD SAFETY INJECTION TO START ON A MANUAL SIGNAL.
EVENT DATE: 111087 REPORT DATE: 011588 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207529) THIS LER IS BEING REVISED TO UPDATE THE CORRECTIVE ACTION TAKEN FOR THIS CONDITION. THE REPORT ORIGINALLY GAVE TWO OPTIONS AS CORRECTIVE ACTION FOR THIS CONDITION. TVA HAS DECIDED MODIFYING THE CENTRIFUGAL CHARGING PUMP (CCP) START CIRCUITRY IS THE MORE PRUDENT DESIGN. ON NOVEMBER 10, 1987, WITH UNITS 1 AND 2 IN MODE 5 (0% PERCENT POWER, PSIG 3, 125 DEGREES F AND 0% POWER, 120 PSIG, 100 DEGREES F, RESPECTIVELY), IT WAS DETERMINED THAT, UNDER CERTAIN CONDITIONS, THE CCPS MAY NOT START USING THE MANUAL HANDSWITCH IN THE MAIN CONTROL ROOM. THE CONTROL LOGIC FOR MANUALLY STARTING THE CCPS CONTAINS AN INTERLOCK THAT REQUIRES THE AUXILIARY LUBE OIL PUMP (AOP) TO START FIRST AND INCREASE THE CCP OIL PRESSURE BEFORE ALLOWING THE CCP TO START. HOWEVER, SINCE THE AOPS AND ASSOCIATED INTERLOCKS HAVE NOT BEEN QUALIFIED IN ACCORDANCE WITH CLASS 1E STANDARDS, THEY MAY NOT BE AVAILABLE FOLLOWING AN ACCIDENT. WITHOUT AN HANDSWITCH IN THE CONTROL ROOM AND MAY NOT BE AVAILABLE TO MITIGATE THE CONSEQUENCES OF A POSTULATED ACCIDENT. THE EVENT WAS CAUSED BY AN INADEQUATE DESIGN REVIEW OF INFORMATION RECEIVED FROM THE NUCLEAR STEAM SUPPLY SYSTEM (NSSS) STARTING THE CCPS ARE AVAILABLE AND THAT THEY HAVE BEEN CAPABLE OF PERFORMING THEIR DESIGNED SAFETY FUNCTION.

[279] SEQUOYAH 1 DOCKET 50-327 LER 87-075
INADEQUATE PROCEDURE AND MISINTERPRETATION OF AUXILIARY FEEDWATER BYPASS LEVEL CONTROL VALVES' OPERATIONAL MODES RESULTS IN AN SR NOT BEING MET AND THE CONDITION NOT BEING REPORTED.
EVENT DATE: 120387 REPORT DATE: 123087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207716) ON DECEMBER 3, 1987, AT 0830 EST WITH UNITS 1 AND 2 IN MODE 5 (0 PERCENT POWER, 4 PSIG, 124 DEGREES F AND 0 PERCENT POWER, 120 PSIG, 113 DEGREES F, RESPECTIVELY), IT WAS DETERMINED THAT THE REVISION 14 OF SURVEILLANCE INSTRUCTION (SI)-118, "MOTOR-DRIVEN AUXILIARY FEEDWATER PUMP AND VALVE AUTOMATIC ACTUATION," DID NOT COMPLETELY TEST THE AUXILIARY FEEDWATER (AFW) BYPASS LEVEL CONTROL VALVES (BPLCVS) AS REQUIRED BY TECH SPEC 4.7.1.2.B.1. THE PROCEDURAL DEFICIENCIES WERE DISCOVERED DURING THE INITIAL IMPLEMENTATION OF AN UPGRADED SI REVIEW PROGRAM AND DOCUMENTED ON POTENTIALLY REPORTABLE OCCURRENCE (PRO) 1-87-129, INITIATED ON MARCH 25, 1987. THE INITIAL EVALUATION OF THE PRO

RESULTED IN A DETERMINATION OF NOT REPORTABLE BASED ON THE EVALUATION THAT THE INTENT OF THE TECH SPEC WAS MET. HOWEVER, AN AUDIT CONDUCTED BY AN NRC INSPECTOR REVEALED THAT THE EVALUATION WAS INCORRECT. THE CAUSE OF NOT MEETING THE TECH SPEC SURVEILLANCE REQUIREMENT WAS DETERMINED TO BE AN INCOMPLETE TEST PROCEDURE. THE CAUSE OF THE INCOMPLETE TEST PROCEDURE WAS AN INADEQUATE SI REVIEW PROGRAM IN PLACE AT THE TIME SI-118 WAS WRITTEN. THE CAUSE OF THE INCORRECT REPORTABILITY EVALUATION HAS BEEN ATTRIBUTED TO A MISUNDERSTANDING OF ALL OPERATIONAL MODES OF THE AFW BPLCV BY THE REVIEWER EVALUATING IT FOR REPORTABILITY.

[280] SEQUOYAH 1 DOCKET 50-327 LER 87-076
A CONTAINMENT VENTILATION ISOLATION OCCURRED AS THE RESULT OF TEST PERSONNEL CONNECTING TEST EQUIPMENT TO INCORRECTLY SPECIFIED TERMINALS.
EVENT DATE: 120687 REPORT DATE: 010488 NSSS: WE TYPE: PWR

(NSIC 207654) ON DECEMBER 6, 1987, WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN) AT APPROXIMATELY 0950 EST, AN "A" TRAIN CONTAINMENT VENTILATION ISOLATION (CVI) OCCURRED. THE CVI OCCURRED WHILE INSTRUMENT MAINTENANCE (IM) PERSONNEL WERE INVESTIGATING THE CAUSE OF CVIS WHICH OCCURRED ON DECEMBER 5 (LER 328/87009) AND NOVEMBER 27, 1987 (LER 328/87008). AN OBJECTIVE OF THIS INVESTIGATION INCLUDED VERIFYING THE TIME REQUIRED FOR RADIATION MONITOR (RM) 2-RM-90-106 TRIP RELAYS TO ACTUATE. PURSUANT TO THIS VERIFICATION, A WORK REQUEST (WR) WAS WRITTEN. THE WORK REQUEST INSTRUCTED IM PERSONNEL TO CONNECT RECORDER TEST LEADS TO UNIT 1 CVI TRIP RELAYS. THE INTENT OF THE INVESTIGATION WAS TO CONNECT THE RECORDER LEADS TO UNIT 2 TRIP RELAYS. OPERATIONS WAS THEN INSTRUCTED TO BLOCK THE UNIT 2 RM (2-RM-90-106) RELAY TRIP SIGNAL USING A HANDSWITCH IN THE CONTROL ROOM TO ENSURE THAT A CVI WOULD NOT BE INITIATED AS A RESULT OF THE RECORDER CONNECTION. HOWEVER, OPERATIONS WAS NOT INSTRUCTED TO BLOCK THE UNIT 1 RM. BLOCKING THE UNIT 1 RM WAS NOT REQUIRED TO VERIFY THE ACTUATION TIME OF THE UNIT 2 CVI TRIP RELAYS. THE NONBLOCKED STATUS OF THE UNIT 1 RM IN COMBINATION WITH INCORRECT RECORDER CONNECTIONS TO UNIT 1 TRIP RELAYS PROVIDED A SIGNAL FLOW PATH SUCH THAT AN "A" TRAIN CVI WAS INITIATED AS IM PERSONNEL ADJUSTED THE RECORDER.

[281] SEQUOYAH 1 DOCKET 50-327 LER 87-077
INADEQUATE DESIGN OF THE CONTAINMENT ISOLATION SYSTEM FOR THE HYDROGEN ANALYZERS COULD RESULT IN BYPASS LEAKAGE FOLLOWING A LOCA.
EVENT DATE: 120787 REPORT DATE: 123087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207717) ON 12/7/87, WITH UNITS 1 AND 2 IN MODE 5 (0% POWER, 4 PSIG, 123F AND 0% POWER, 115 PSIG, 120F, RESPECTIVELY), A CONDITION WAS IDENTIFIED WHICH COULD HAVE RESULTED IN BYPASS LEAKAGE FOLLOWING A DESIGN BASIS LOCA. THE HYDROGEN ANALYZER SYSTEM INSTALLED IN EACH UNIT OF THE SEQUOYAH NUCLEAR PLANT (SQN) UTILIZES A SMALL AMOUNT OF CONTROL AIR TO ACT AS A REAGENT GAS. A SAMPLE FROM THE CONTAINMENT ATMOSPHERE IS MIXED WITH THE REAGENT GAS IN THE HYDROGEN ANALYZER MODULE, ANALYZED FOR HYDROGEN CONTENT, AND RETURNED TO CONTAINMENT. DURING THE REVIEW OF AN HAS RELATED ENGINEERING CHANGE NOTICE (ECN), TVA DISCOVERED THAT THE CURRENT HAS DESIGN REPRESENTED A POTENTIAL PATHWAY FOR RADIONUCLIDES TO ESCAPE TO THE ENVIRONMENT. THE HAS ISOLATION VALVES LOCATED INSIDE CONTAINMENT ARE AIR OPERATED VALVES WHICH FAIL IN THE OPEN POSITION. IF A LOCA OCCURRED CONCURRENT WITH A SINGLE FAILURE OF ONE TRAIN OF CONTROL AIR

[282] SEQUOYAH 1 DOCKET 50-327 LER 87-078
AN INADEQUATE PROCEDURE FOR REACTOR COOLANT SYSTEM CHEMICAL ADDITION RESULTED IN NON-COMPLIANCE WITH A TECHNICAL SPECIFICATION ACTION STATEMENT.
EVENT DATE: 122187 REPORT DATE: 011588 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207942) ON DECEMBER 21, 1987, WITH UNIT 1 IN MODE 5 (0 PERCENT POWER, 4 PSIG, 124 DEGREES F), IT WAS DETERMINED THAT A CHEMICAL ADDITION MADE TO THE UNIT 1 REACTOR COOLANT SYSTEM (RCS) ON NOVEMBER 25, 1987, CAUSED A DILUTION OF THE RCS BORON CONCENTRATION AND SUBSEQUENT POSITIVE REACTIVITY CHANGE. SINCE BOTH TRAINS OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) WERE INOPERABLE AT THE TIME OF THE CHEMICAL ADDITION, ACTION STATEMENT (B) TO TECH SPEC 3.7.7, WHICH REQUIRES ALL OPERATIONS INVOLVING A POSITIVE REACTIVITY CHANGE TO BE SUSPENDED, WAS NOT COMPLIED WITH. FURTHER INVESTIGATION INTO THIS EVENT IDENTIFIED OTHER OCCASIONS WHERE BORON DILUTION OR RCS COOLDOWN HAVE RESULTED IN SMALL POSITIVE REACTIVITY CHANGES WHICH MAY NOT HAVE COMPLIED WITH ACTION STATEMENT (B) TO TECH SPEC 3.7.7. TVA IS CURRENTLY INVESTIGATING THESE EVENTS AND WILL PROVIDE A SUPPLEMENT TO THIS REPORT BY MARCH 31, 1988. THE EVENT WAS CAUSED BY THE LACK OF AN ADEQUATE PROCEDURE FOR ENSURING THAT RCS CHEMICAL ADDITIONS, WHICH WERE MADE DURING THE PERIOD OF TIME WHEN BOTH TRAINS OF CREVS WERE INOPERABLE, DID NOT RESULT IN POSITIVE REACTIVITY CHANGES. AS IMMEDIATE CORRECTIVE ACTION, TVA HAS REVISED THE CHEMICAL ADDITION PROCEDURES SUCH THAT RCS CHEMICAL ADDITIONS WILL NOT RESULT IN DILUTION OF THE RCS BORON CONCENTRATION.

[283] SEQUOYAH 1 DOCKET 50-327 LER 87-074
EMERGENCY PROCEDURES DO NOT ADEQUATELY ADDRESS OPENING CERTAIN HIGH HEAD SAFETY INJECTION VALVES FOLLOWING AN ACCIDENT IN HOT SHUTDOWN.
EVENT DATE: 122287 REPORT DATE: 012188 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 208006) ON DECEMBER 22, 1987, WITH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN), IT WAS IDENTIFIED THAT THE SEQUOYAH NUCLEAR PLANT (SQN) EMERGENCY PROCEDURES MAY NOT SUFFICIENTLY ADDRESS THE REQUIRED OPERATOR ACTIONS FOLLOWING A POSTULATED LOSS OF COOLANT ACCIDENT (LOCA) DURING MODE 4 OPERATION. SINCE THE SQN TECH SPECS REQUIRE THE BORON INJECTION (BIT) TO BE OPERABLE IN MODES 1 THROUGH 3 ONLY, GENERAL OPERATING INSTRUCTION (GOI)-3, "PLANT SHUTDOWN FROM MINIMUM LOAD TO COLD SHUTDOWN," INSTRUCTS PLANT OPERATORS TO DEENERGIZE THE BIT ISOLATION VALVES FOLLOWING ENTRY INTO MODE 4 IN ORDER TO PRECLUDE THE POSSIBILITY OF A SPURIOUS SAFETY INJECTION SIGNAL CAUSING AN OVERPRESSURIZATION OF THE REACTOR COOLANT SYSTEM (RCS). ISOLATION OF THE BIT ALSO RESULTS IN THE ISOLATION OF THE HIGH HEAD SAFETY INJECTION SYSTEM FLOW PATH REQUIRED BY LIMITING CONDITION FOR OPERATION (LCO) 3.5.3. HOWEVER, LCO 3.5.3 ALLOWS REALIGNMENT OF THE EMERGENCY CORE COOLING SYSTEM (ECCS) DURING MODE 4. FURTHER INVESTIGATION HAS SHOWN THAT EMERGENCY PROCEDURE E-0, "REACTOR TRIP OR SAFETY INJECTION," REQUIRES THE PLANT OPERATORS TO VERIFY AT LEAST ONE TRAIN OF HIGH HEAD SAFETY INJECTION IS PROVIDING FLOW THROUGH THE BIT; HOWEVER, IT DOES NOT PROVIDE SPECIFIC INSTRUCTIONS FOR REENERGIZING AND OPENING THE BIT ISOLATION VALVES IN THE EVENT OF A LOCA.

[284] SEQUOYAH 1 DOCKET 50-327 LER 88-001
AN INACCURATE COMPUTER DATABASE CAUSES A TECH SPEC SURVEILLANCE INTERVAL TO BE EXCEEDED RESULTING IN INOPERABLE DIESEL GENERATORS.
EVENT DATE: 011188 REPORT DATE: 012388 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207997) ON JANUARY 11, 1988, WITH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN) ALL FOUR EMERGENCY DIESEL GENERATORS (D/GS) WERE DECLARED INOPERABLE BECAUSE THE INTERVAL FOR TECH SPEC SURVEILLANCE REQUIREMENT (SR) 4.8.1.2 (CHEMICAL ANALYSIS OF D/G FUEL OIL) WAS EXCEEDED. IN ADDITION, BECAUSE THE HIGH PRESSURE FIRE PUMPS USE THE D/G FUELS AS AN EMERGENCY POWER SOURCE, THE PLANT FIRE SUPPRESSION SYSTEM

WAS ALSO DECLARED INOPERABLE, AND A BACKUP FIRE SUPPRESSION SYSTEM WAS ESTABLISHED. THIS REPORT PROVIDES DETAILS ON THE ABOVE DESCRIBED EVENT AND FULFILLS THE SPECIAL REPORTING REQUIREMENT FOR INOPERABLE FIRE PUMPS. BECAUSE OF AN OVERSIGHT DURING THE RECENT CONVERSION TO A NEW COMPUTER PROGRAM USED TO SCHEDULE SURVEILLANCE INSTRUCTIONS (SIS), SI-116, "QUARTERLY CHEMISTRY REQUIREMENTS ON DIESEL GENERATOR FUEL OIL," WAS NOT PERFORMED WITHIN THE TIME INTERVAL REQUIRED BY THE TECH SPECS. IMMEDIATELY UPON DISCOVERY OF THE EVENT, A SPECIAL SI-116 PACKAGE WAS ISSUED AND PERFORMED, AND ALL D/GS AND FIRE SUPPRESSION PUMPS WERE RETURNED TO OPERABLE STATUS ON JANUARY 12, 1988. IN ADDITION, AN IMMEDIATE REVIEW OF ALL SIS WAS INITIATED TO ENSURE THAT THE MODE REQUIREMENTS IN THE SCHEDULING PROGRAM WERE CONSISTENT WITH THE APPLICABILITY SECTION IN THE SIS. ALL SIS COMMON TO UNITS 1 AND 2 OR APPLICABLE TO UNIT 2 ONLY HAVE BEEN REVIEWED.

[285] SEQUOYAH 2 DOCKET 50-328 LER 87-008 REV 01
UPDATE ON ELECTROMAGNETIC INTERFERENCE SPIKE INITIATING A CONTAINMENT VENTILATION ISOLATION AS A RESULT OF THE DETECTOR CABLE GROUND.
EVENT DATE: 112787 REPORT DATE: 123087 NSSS: WE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 207718) THIS LER IS BEING REVISED TO PROVIDE ADDITIONAL INFORMATION IN THE CORRECTIVE ACTION. ON NOVEMBER 27, 1987, WITH UNIT 2 IN MODE 5 (0 PERCENT POWER, 75 PSIG, 114 DEGREES F), A TRAIN "A" CONTAINMENT VENTILATION ISOLATION (CVI) OCCURRED ON UNIT 2. AT APPROXIMATELY 0320 EST, WITH AN INSTRUMENT MALFUNCTION ALARM PRESENT ON THE CONTAINMENT LOWER COMPARTMENT AIR RADIATION MONITOR (RM), OPERATIONS PERSONNEL DETERMINED THAT A SPURIOUS HIGH RADIATION SPIKE HAD OCCURRED AT APPROXIMATELY 0230 EST. SINCE THE SPURIOUS RADIATION SPIKE EXCEEDED THE RM TRIP CRITERIA, A CVI AND INSTRUMENT MALFUNCTION ALARM WERE GENERATED. INSTRUMENT MAINTENANCE (IM) PERSONNEL DETERMINED THAT THE MOST PROBABLE CAUSE OF THE SPURIOUS HIGH RADIATION SPIKE WAS AN ELECTROMAGNETIC PULSE GENERATED BY THE ACTUATION OF THE LOW SAMPLE FLOW SWITCH THAT WAS TRANSFERRED TO THE RM DETECTOR CABLE. INVESTIGATION INTO THIS AND THREE SUBSEQUENT CVIS REVEALED THAT THE DETECTOR CABLE DID NOT HAVE A GOOD GROUND. AS THE RESULT OF A RECENT TREND OF CVI ACTUATIONS (SIX WITHIN THE INTERVAL OF NOVEMBER 27 TO DECEMBER 21), TVA HAS ESTABLISHED A SPECIAL TASK GROUP TO INVESTIGATE THE ROOT CAUSE OF THE SUBJECT CVIS. TVA BELIEVES THIS INVESTIGATION WILL PROVIDE AN OVERVIEW OF THE EVENTS AND ALLOW FOR A DETERMINATION AS TO THE POSSIBILITIES OF A COMMON OR EVENT SPECIFIC ROOT CAUSE.

[286] SEQUOYAH 2 DOCKET 50-328 LER 87-009
TWO SIMILAR CONTAINMENT VENTILATION ISOLATIONS OCCURRED AS A RESULT OF INDUCED ELECTROMAGNETIC INTERFERENCE.
EVENT DATE: 120587 REPORT DATE: 123087 NSSS: WE TYPE: PWR

(NSIC 207655) THIS REPORT DESCRIBES TWO CONTAINMENT VENTILATION ISOLATIONS (CVIS) THAT OCCURRED ON DECEMBER 5, 1987. AS ALLOWED BY NUREG 1022, SUPPLEMENT 1, PARAGRAPH 6.12, THIS REPORT COMBINES THESE EVENTS AS THE TWO EVENTS ARE SIMILAR. ON DECEMBER 5, 1987, WITH UNIT 2 IN MODE 5 (COLD SHUTDOWN) (0 PERCENT POWER, 75 PSIG, 114 DEGREES F), TWO CVIS OCCURRED AT APPROXIMATELY 0911 EST AND 1553 EST. INVESTIGATION INTO THE EVENTS REVEALED THAT WELDING AND GRINDING WORK WAS BEING PERFORMED IN UNIT 2 CONTAINMENT. AS A RESULT OF THIS WORK, THE AIR CONDITIONS CLOGGED RADIATION MONITOR (RM) SAMPLE GAS PREFILTERS, CREATING A LOW FLOW CONDITION. THE LOW FLOW CONDITION ACTIVATED RM 2-RM-90-106 SAMPLE LOW FLOW SWITCH. TVA BELIEVES THIS SWITCH ACTUATION INDUCED A ELECTROMAGNETIC INTERFERENCE (EMI) SPIKE TO THE RM DETECTOR CABLE. SINCE THIS SPURIOUS RADIATION SPIKE EXCEEDED THE RM TRIP CRITERIA, THE SUBJECT CVIS WERE GENERATED. TVA HAS CONCLUDED THAT THE MOST PROBABLE CAUSE OF THE CIVS WAS AN ELECTROMAGNETIC PULSE GENERATED BY THE ACTUATION OF THE LOW SAMPLE FLOW SWITCH THAT WAS INDUCED TO THE RM DETECTOR CABLE.

[287] SEQUOYAH 2 DOCKET 50-328 LER 87-010
TWO CONTAINMENT VENTILATION ISOLATIONS OCCURRED AS THE RESULT OF A SPURIOUS HIGH RADIATION SPIKE.
EVENT DATE: 122187 REPORT DATE: 011588 NSSS: WE TYPE: PWR
VENDOR: LUNDELL CONTROLS - TECHNOLOGY INC.
MALLORY CAPACITOR CO

(NSIC 207943) THIS REPORT DESCRIBES TWO CONTAINMENT VENTILATION ISOLATIONS (CVIS) THAT OCCURRED ON DECEMBER 21, 1987. AS ALLOWED BY NUREG 1022, SUPPLEMENT 1, THIS REPORT COMBINES THESE EVENTS SINCE THE TWO EVENTS ARE SIMILAR. ON DECEMBER 21, 1987, WITH UNIT 2 IN MODE 5 (COLD SHUTDOWN) (0 PERCENT POWER, 225 PSIG, 125 DEGREES F), TWO CVI ACTUATIONS OCCURRED. THE FIRST CVI OCCURRED AT APPROXIMATELY 0750 EST AND WAS CAUSED BY A SPURIOUS HIGH RADIATION SPIKE ON CONTAINMENT RADIATION MONITORS (RMS) 2-RM-90-106 AND/OR 2-RM-90-112. SINCE BOTH RM CHART RECORDERS SHOWED A HIGH RADIATION SPIKE AT APPROXIMATELY THE SAME TIME AND NEITHER HIGH RADIATION ALARMS ACTUATED, IT WAS NOT POSSIBLE TO DETERMINE WHICH RM GENERATED THE CVI. AT APPROXIMATELY 0810 EST, THE CVI WAS RESET, AND NORMAL MODE 5 OPERATION WAS RESUMED. THE SECOND CVI OCCURRED AT APPROXIMATELY 0900 EST AND WAS CAUSED BY A SPURIOUS HIGH RADIATION SPIKE ON RM 2-RM-90-112. AT APPROXIMATELY 0906 EST, THE CVI WAS RESET, AND NORMAL MODE 5 OPERATION WAS RESUMED. AS A RESULT OF A RECENT TREND IN CVI ACTUATIONS (SIX WITHIN THE INTERVAL FROM NOVEMBER 27 TO DECEMBER 21), TVA HAS ESTABLISHED A SPECIAL TASK FORCE TO INVESTIGATE THE ROOT CAUSE OF THE SUBJECT CVIS. ONCE A ROOT CAUSE IS ESTABLISHED, APPROPRIATE CORRECTIVE ACTIONS WILL BE DETERMINED AND IMPLEMENTED.

[288] SHEARON HARRIS 1 DOCKET 50-400 LER 87-065 REV 01
UPDATE ON FIRST STAGE TURBINE PRESSURE SETPOINTS FOR P-13 PERMISSIVE WERE INCORRECTLY SET DUE TO PERSONNEL ERROR.
EVENT DATE: 112487 REPORT DATE: 011488 NSSS: WE TYPE: PWR

(NSIC 207907) THE PLANT WAS OPERATING IN MODE 1, POWER OPERATION, AT 100 PERCENT REACTOR POWER ON NOVEMBER 24, 1987. AT APPROXIMATELY 1500 HOURS, IT WAS DISCOVERED THAT THE FIRST STAGE TURBINE PRESSURE SETPOINTS P-13, WHICH UNBLOCK REACTOR TRIP BLOCK PERMISSIVE P-7 ABOVE 10 PERCENT POWER, WERE INCORRECTLY SET. THE INSTALLED SETPOINTS WERE BASED ON PRESSURE EQUIVALENT TO 10 PERCENT GENERATOR LOAD IN LIEU OF 10 PERCENT RATED THERMAL POWER WHICH IS REQUIRED BY TECHNICAL SPECIFICATIONS TABLE 2.2-1, ITEM 19. THE P-13 PORTION OF THE CHANNEL WAS THEN DECLARED INOPERABLE AT 1635 HOURS AND TECHNICAL SPECIFICATION ACTION STATEMENTS WERE PUT INTO EFFECT. PERSONNEL ERROR WAS RESPONSIBLE FOR ESTABLISHING THE SETPOINTS IN PERCENT LOAD IN LIEU OF PERCENT RATED THERMAL POWER. PERSONNEL HAD USED A WESTINGHOUSE DOCUMENT INSTEAD OF THE SHNPP TECHNICAL SPECIFICATIONS TO ESTABLISH THE CONTROLLING SETPOINTS. THE IMMEDIATE CORRECTIVE ACTION WAS TO DECLARE THE P-13 CHANNEL INOPERABLE AND ENTER APPLICABLE COMPENSATORY ACTIONS. A PLANT CHANGE REQUEST HAS BEEN INITIATED TO CHANGE THE SETPOINTS TO REFLECT FIRST STAGE TURBINE PRESSURE VERSUS RATED THERMAL POWER. THIS EVENT IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV) AS A VIOLATION OF TECHNICAL SPECIFICATIONS.

[289] SHEARON HARRIS 1 DOCKET 50-400 LER 87-066
FAILURE TO PERFORM SATISFACTORY ACTUATION LOGIC TEST FOR CONTAINMENT VENTILATION ISOLATION DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 120487 REPORT DATE: 011088 NSSS: WE TYPE: PWR

(NSIC 207646) THE PLANT WAS OPERATING IN MODE 1, POWER OPERATION, AT 100 PERCENT REACTOR POWER ON DECEMBER 4, 1987. AT 0830 HOURS IT WAS DISCOVERED THAT PART OF THE CONTAINMENT VENTILATION ISOLATION CIRCUITRY ACTUATION LOGIC FOR CONTAINMENT AREA RADIATION MONITORS WAS NOT BEING TESTED ON A STAGGERED MONTHLY BASIS AS REQUIRED BY TECHNICAL SPECIFICATION 4.3.2.1 TABLE 4.3-2 ITEM 3.C.2. THE PROCEDURE THAT WAS DESIGNATED TO PERFORM THE SURVEILLANCE TESTED ONLY THE PORTION

OF THE CIRCUITRY WITHIN THE SOLID STATE PLANT PROTECTION SYSTEM (SSPPS). A SEPARATE PROCEDURE ADEQUATELY ADDRESSED THE REQUIREMENT BUT WAS ONLY BEING DONE ON AN 18 MONTH FREQUENCY RATHER THAN THE REQUIRED FREQUENCY IN TECHNICAL SPECIFICATIONS. THIS ERROR WAS DISCOVERED WHILE DOING A REVIEW OF SCHEDULING TASK SHEETS FOR THE LATTER PROCEDURE. UPON NOTIFICATION OF THIS EVENT OPERATIONS PERSONNEL IMMEDIATELY DECLARED THE FOUR CONTAINMENT AREA RADIATION MONITOR CHANNELS INOPERABLE AND ENTERED TECHNICAL SPECIFICATIONS ACTION STATEMENT WHICH REQUIRED THE CLOSING OF THE CONTAINMENT PURGE MAKEUP AND EXHAUST ISOLATION VALVES. THE SURVEILLANCE WAS RUN SATISFACTORILY AND THE CIRCUITS DECLARED OPERABLE. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR THROUGH INADEQUATE PREPARATION AND REVIEW OF THE APPLICABLE PROCEDURE.

[290] SHEARON HARRIS 1 DOCKET 50-400 LER 87-067
CONTAINMENT VENTILATION ISOLATION DUE TO THE SPURIOUS HIGH RADIATION ALARM WHILE SAMPLING THE MONITOR.
EVENT DATE: 121487 REPORT DATE: 011488 NSSS: WE TYPE: PWR

(NSIC 207908) THE PLANT WAS OPERATING IN MODE 1, POWER OPERATION, AT 99 PERCENT REACTOR POWER ON DECEMBER 14, 1987. AT 0340 HOURS WHILE A CHEMISTRY TECHNICIAN WAS OBTAINING A GRAB SAMPLE FROM CONTAINMENT LEAK DETECTION RADIATION MONITOR, REM-1LT-3502A-SA, A HIGH RADIATION ALARM WAS GENERATED. THE HIGH RADIATION ALARM GENERATED A CONTAINMENT VENTILATION ISOLATION SIGNAL WHICH RESULTED IN THE CLOSURE OF ALL CONTAINMENT VENTILATION VALVES IN ACCORDANCE WITH PLANT DESIGN. THE CAUSE OF THE ALARM WAS A SPURIOUS ELECTRONIC SIGNAL SPIKE. THE ALARM WAS CLEARED, THE MONITOR WAS VERIFIED TO BE OPERATING CORRECTLY, AND THE CONTAINMENT VENTILATION NORMAL VALVE LINE-UP WAS RESTORED. THE GRAB SAMPLE TAKEN WHEN THE MONITOR ALARMED WAS ANALYZED AND FOUND TO BE WITHIN ACCEPTABLE LIMITS. ACTIONS TO PREVENT RECURRENCE ARE: GRAB SAMPLE LINES HAVE BEEN CLEARLY LABELED TO ENSURE CORRECT HOOK-UP, SAMPLING FREQUENCIES HAVE BEEN REDUCED, AND A PLANT CHANGE REQUEST WILL BE INSTALLED WHICH PROVIDES AN IMPROVED SIGNAL SMOOTHING ALGORITHM. THIS EVENT IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV) AS AN AUTOMATIC ACTUATION OF AN ENGINEERED SAFETY FEATURE.

[291] SHEARON HARRIS 1 DOCKET 50-400 LER 87-064
SPECIAL REPORT CONCERNING INOPERATIVE RADIATION MONITOR WAS NOT SUBMITTED WITHIN THE REQUIRED TIME FRAME.
EVENT DATE: 121887 REPORT DATE: 011888 NSSS: WE TYPE: PWR

(NSIC 207935) ON NOVEMBER 2, 1987, AT 1400, WHILE THE PLANT WAS IN MODE 5, THE WASTE PROCESSING BUILDING VENT STACK 5A WIDE RANGE GAS MONITOR (WRGM), #1WV-3547-1, (EIIS:IL) WAS DECLARED INOPERABLE UNDER TECHNICAL SPECIFICATION 3.3.3.11, RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION. THIS WAS DUE TO A POTENTIAL LOSS OF SAMPLE FLOW CAUSED BY CALIBRATION OF THE STACK FLOW TRANSMITTER AND ISO-KINETIC SAMPLING SKID. AT THAT TIME, IT WAS NOT RECOGNIZED THAT THE CALIBRATION WOULD ALSO MAKE THE MONITOR INOPERABLE UNDER TECH SPEC 3.3.3.6, ACCIDENT MONITORING INSTRUMENTATION. THIS DETERMINATION WAS NOT MADE BY OPERATIONS PERSONNEL UNTIL DECEMBER 18, 1987. EFFORTS TO RETURN THE MONITOR TO SERVICE WITHIN 7 DAYS WERE UNSUCCESSFUL, AND A SPECIAL REPORT SHOULD HAVE BEEN WRITTEN WITHIN 14 DAYS AS REQUIRED BY TECH SPECS 3.3.3.6, ACTION C. BECAUSE OF THE LATE DECLARATION OF INOPERABILITY UNDER TECH SPEC 3.3.3.6, THIS SPECIAL REPORT WAS NOT SUBMITTED WITHIN THE SPECIFIED TIME PERIOD. THE DELAY IN SUBMITTING THE SPECIAL REPORT WAS CAUSED BY PERSONNEL ERROR. CORRECTIVE ACTION INCLUDED A REVISION TO PROCEDURE OMM-003, EQUIPMENT INOPERABLE RECORD (EIR), WHICH WILL PROVIDE BETTER GUIDANCE FOR TRACKING EIRS AND FOR MAINTAINING COGNIZANCE OF TIME REQUIREMENTS FOR LCOS.

[292] SHOREHAM DOCKET 50-322 LER 87-033
WEEKLY LIQUID EFFLUENT SAMPLES REQUIRED BY TECHNICAL SPECIFICATIONS WERE DISCARDED PRIOR TO BEING UTILIZED FOR MONTHLY AND QUARTERLY ANALYSIS DUE TO PERSONNEL ERROR.
EVENT DATE: 113087 REPORT DATE: 122387 NSSS: GE TYPE: BWR

(NSIC 207545) ON NOVEMBER 30, 1987 AT APPROXIMATELY 1130, IT WAS DISCOVERED BY A RADIOCHEMISTRY TECHNICIAN THAT A WEEKLY LIQUID EFFLUENT SAMPLE WAS DISCARDED PRIOR TO BEING UTILIZED FOR MONTHLY AND QUARTERLY ANALYSES REQUIRED BY TECH SPEC 4.11.1.1.1. THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) WITH THE MODE SWITCH IN SHUTDOWN AND ALL RODS INSERTED IN THE CORE. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR. THE TECHNICIAN WHO HAD PREPARED THE WEEKLY SAMPLE FOR THE NOVEMBER 16-22, COMPLETED REQUIRED ANALYSIS OF THE SAMPLE, BUT NEGLECTED TO LABEL AND STORE THE SAMPLE AS REQUIRED BY PROCEDURE WHICH WAS SUBSEQUENTLY DISCARDED DURING NORMAL LAB CLEAN-UP. PLANT MANAGEMENT WAS NOTIFIED OF THE EVENT AND THE NRC WAS NOTIFIED PER LICENSE CONDITION NPF-36, 2.F ON DECEMBER 1 AT 1027. THE INDIVIDUAL RESPONSIBLE RECEIVED A WRITTEN REPRIMAND. VALUES FOR TRITIUM AND ALPHA IN THE MISSING WEEKLY COMPOSITE WILL BE INTERPOLATED BASED ON ANALYSIS OF THE PRECEEDING AND FOLLOWING WEEKS COMPOSITE SAMPLES. IN ADDITION, ALTHOUGH THE PROCEDURE UTILIZED TO PERFORM THE SAMPLING AND ANALYSIS (SP 74.020.10) WAS ADEQUATE, AN ADDITIONAL SIGN-OFF SPACE WILL BE INCLUDED TO ENSURE THAT THE INDIVIDUAL PERFORMING THE PROCEDURE LABELS AND STORES THE SAMPLE PROPERLY.

[293] SHOREHAM DOCKET 50-322 LER 87-034
CALIBRATION OF MAIN CONTROL ROOM VENTILATION RADIATION MONITORS CONTRARY TO TECHNICAL SPECIFICATION REQUIREMENTS.
EVENT DATE: 120187 REPORT DATE: 122387 NSSS: GE TYPE: BWR

(NSIC 207554) ON DECEMBER 1, 1987 AT 0930 HOURS, IT WAS DETERMINED THAT THE MAIN CONTROL ROOM VENTILATION RADIATION MONITORS HAD BEEN CALIBRATED CONTRARY TO TECH SPEC REQUIREMENTS. STATION TECH SPEC TABLE 3.3.7.1-1 REQUIRES THAT THE MAIN CONTROL ROOM VENTILATION RADIATION MONITOR HAVE AN ALARM/TRIP SETPOINT OF LESS THAN OR EQUAL TO TWO TIMES BACKGROUND. THE NUCLEAR ENGINEERING DEPARTMENT, RADIATION PROTECTION DIVISION AT SNPS HAD PREVIOUSLY INTERPRETED THIS REQUIREMENT TO MEAN TWO TIMES BACKGROUND OVER BACKGROUND. HENCE, THE MONITORS HAVE BEEN CALIBRATED TO ALARM AT THIS HIGHER SETPOINT SINCE JUNE 1986. HOWEVER, RECENT DISCUSSIONS AMONG MEMBERS OF THE PLANT STAFF HAVE RESULTED IN A DIFFERENT INTERPRETATION OF THIS SETPOINT REQUIREMENT. THE PLANT STAFF HAS CONCLUDED THAT THE PREVIOUS INTERPRETATION ACTUALLY PROVIDED A SETPOINT OF THREE TIMES BACKGROUND AND THAT AS A CONSEQUENCE, TECH SPEC 3.3.7.1 HAD BEEN VIOLATED. PLANT MANAGEMENT WAS NOTIFIED OF THE EVENT AND THE NRC WAS NOTIFIED AT 0825 ON DECEMBER 2, 1987 PER LICENSE NPF-36, 2.F. AT THE TIME OF NOTIFICATION, THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) WITH ALL RODS INSERTED INTO THE CORE. THE MONITORS HAVE SINCE BEEN DECLARED INOPERABLE WITH THE PLANT IN THE ASSOCIATED LIMITING CONDITION FOR OPERATION UNTIL THE SETPOINTS ARE CHANGED IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS.

[294] SHOREHAM DOCKET 50-322 LER 87-035
ELECTRICAL NOISE BETWEEN GROUNDS FOR TEMPERATURE MONITORING UNITS RESULTS IN HIGH ENERGY LINE BREAK LOGIC INITIATION.
EVENT DATE: 122187 REPORT DATE: 012088 NSSS: GE TYPE: BWR

(NSIC 207949) ON 12/21/87 AND 1/6/88 AT 1633 AND 0825 RESPECTIVELY, HIGH ENERGY LINE BREAK LOGIC ISOLATIONS OF THE REACTOR WATER CLEAN-UP (RWCU) AND MAIN STEAM LINE (MSL) DRAINS VALVE (1G33*MOV-034 AND 1B21*MOV-032) OCCURRED MOST LIKELY DUE TO ELECTRICAL NOISE POTENTIAL BETWEEN THE TWO GROUNDS UTILIZED WITHIN THE PANEL FOR TEMPERATURE MONITORING UNITS (TMU) 1G11*TMU-500A AND 1G11*TMU-500B. THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) WITH ALL RODS INSERTED INTO THE CORE. OPERATORS VERIFIED THE SIGNAL AS FALSE AND RETURNED THE SYSTEMS TO

THEIR NORMAL CONFIGURATION PRIOR TO THE EVENT. PLANT MANAGEMENT WAS INFORMED OF THE EVENTS AND THE NRC WAS NOTIFIED AT 1753 HRS ON DECEMBER 21, 1987 AND 1058 HRS ON JANUARY 6, 1988. THE INVESTIGATION INTO THE CAUSE OF EVENTS HAS LED TO THE DISCOVERY OF A GROUNDING PROBLEM WITHIN THE PRIMARY CONTAINMENT MONITORING PANEL (PCM) WHERE THE UNITS ARE LOCATED. LILCO BELIEVES THAT THE PROBLEM IS DUE TO THE 24 VDC POWER SUPPLIES FOR THE TMUS BEING GROUNDED BY THE ISOLATED INSTRUMENT GROUND. VOLTAGE MEASUREMENTS BETWEEN THE TWO GROUNDS HAS REVEALED THAT AN ELECTRICAL NOISE POTENTIAL EXISTS, WHICH COULD LEAD TO FALSE SIGNALS BEING GENERATED WITHIN THE TMUS AND IN TURN INITIATING THE ISOLATIONS. AN ENGINEERING CHANGE HAS BEEN GENERATED TO ALLOW THE 24 VDC POWER SUPPLIES FOR THE TMUS TO BE GROUNDED TO THE ISOLATED INSTRUMENT GROUND.

[295] SOUTH TEXAS 1 DOCKET 50-498 LER 87-016
HYDRAULIC TRANSIENTS IN THE AUXILIARY FEEDWATER SYSTEM DUE TO A DESIGN ERROR.
EVENT DATE: 110587 REPORT DATE: 010588 NSSS: WE TYPE: PWR
VENDOR: BINGHAM PUMP CO.
VALTEK INC.
WKM VALVE DIVISION

(NSIC 207726) ON NOVEMBER 5, 1987, WHILE THE PLANT WAS IN MODE 4, PRIOR TO INITIAL CRITICALITY, A ONE INCH DOUBLE VALVE VENT LINE IN THE PUMP DISCHARGE PIPING OF AUXILIARY FEEDWATER (AFW) TRAIN A BROKE OFF. TRAIN A WAS DECLARED INOPERABLE AND THE VENT LINE WAS REPAIRED. ON NOVEMBER 8, 1987 A SECOND FAILURE OCCURRED IN A DOUBLE VALVE INSTRUMENT TAP FOR THE TRAIN D FLOW ELEMENT. AFW IS NOT REQUIRED TO BE OPERABLE PRIOR TO MODE 3. AN INITIAL ASSESSMENT DETERMINED THE CAUSE OF THESE EVENTS TO BE WATER HAMMER ATTRIBUTABLE TO IMPROPER VENTING OF THE SYSTEM. DESIGN AND PROCEDURAL CHANGES WERE MADE AND THE UNIT WENT TO MODE 3 ON NOVEMBER 22, 1987. SUBSEQUENTLY, SUSTAINED PIPING VIBRATION IN TRAINS A AND C WAS OBSERVED. ADDITIONAL PIPING DAMAGE OCCURRED AND THE UNIT WAS RETURNED TO MODE 4. BASED ON CONCERNS ABOUT THE DESIGN OF THE AFW SYSTEM EVALUATIONS AND TESTING HAVE DETERMINED THAT PRESSURE PULSATIONS WERE BEING SET UP WHEN THE FLOW CONTROL VALVES WERE IN A HIGHLY THROTTLED POSITION. A COMBINATION OF BOTH HYDRAULIC AND STRUCTURAL RESONANCES SET UP AS A RESULT OF CHANGES TO ELIMINATE THIS PROBLEM WILL BE IMPLEMENTED AND PROOF TESTED PRIOR TO ENTRY INTO MODE 3. THERE WERE NO ADVERSE SAFETY OR RADIOLOGICAL CONSEQUENCES FROM THIS EVENT. THIS EVENT IS BEING REPORTED PURSUANT TO 10CFR50.73(A)(V).

[296] SOUTH TEXAS 1 DOCKET 50-498 LER 87-014 REV 01
UPDATE ON CONTROL ROOM VENTILATION ACTUATION TO RECIRCULATION MODE DUE TO A TOXIC GAS MONITOR DETECTING PAINT FUMES.
EVENT DATE: 111387 REPORT DATE: 121887 NSSS: WE TYPE: PWR

(NSIC 207550) AT APPROXIMATELY 2326 HOURS ON NOVEMBER 12, 1987 WITH UNIT 1 IN MODE 4 (INITIAL STARTUP TESTING), AN AUTO-ACTUATION OF THE CONTROL ROOM VENTILATION SYSTEM TO THE RECIRCULATION MODE OCCURRED AS A RESULT OF A TOXIC GAS MONITOR DETECTING HIGH LEVELS OF TOXIC GAS. THE CONTROL ROOM VENTILATION ACTUATION TO RECIRCULATION MODE IS AN ENGINEERED SAFETY FEATURE (ESF). THE CONTROL ROOM OPERATORS VERIFIED THE RECIRCULATION MODE DAMPER LINEUP AND INITIATED AN INVESTIGATION OF THE EVENT. THE INVESTIGATION DETERMINED THAT THE EVENT WAS CAUSED BY THE TOXIC GAS MONITOR DETECTING PAINT FUMES CONTAINING HYDROCARBONS WHICH HAVE INFRARED ABSORPTION CHARACTERISTICS SIMILAR TO ANHYDROUS AMMONIA AND AMMONIA HYDROXIDE. THOSE ARE TOXIC GASES OF CONCERN AT THE STPEGS. PAINTING OF A NEWLY INSTALLED DOOR PROVED TO BE THE SOURCE OF THE FUMES. PAINTING IN THE VICINITY HAS BEEN SUSPENDED UNTIL CORRECTIVE ACTIONS ARE IMPLEMENTED. THE PAINTING CONTROL PROCEDURE WILL BE REVISED TO INCLUDE A TOXIC GAS EVALUATION BEFORE PAINTING IS ALLOWED WITHIN THE CONTROL ROOM AIR INLET CHASE. THIS EVENT REPRESENTS A BREAKDOWN IN THE MANAGEMENT CONTROL PROCESS.

[297] SOUTH TEXAS 1 DOCKET 50-498 LER 87-015
INITIATION OF COOLDOWN DUE TO INOPERABILITY OF TWO ESSENTIAL CHILLER UNITS.
EVENT DATE: 112187 REPORT DATE: 122187 NSSS: WE TYPE: PWR
VENDOR: YORK, O. H. COMPANY

(NSIC 207538) ON NOVEMBER 21, 1987 AT APPROXIMATELY 1005 HOURS WITH THE UNIT IN MODE 4 AND PRIOR TO INITIAL CRITICALITY, AN OPERATOR DISCOVERED OIL HEATER SWITCHES DEENERGIZED FOR TWO OF THE SIX ESSENTIAL CHILLED WATER (CH) SYSTEM CHILLER UNITS ON THE LOCAL CONTROL PANELS. THE CHILLERS WERE IN THE STANDBY MODE AND WERE NOT RUNNING. THE HEATERS ARE REQUIRED TO BE ENERGIZED AT LEAST 12 HOURS PRIOR TO OPERATION OF THE UNITS. THIS CONDITION RESULTED IN TWO OF THE THREE INDEPENDENT TRAINS OF THE CH SYSTEM BEING DECLARED INOPERABLE. THE PLANT ENTERED TECHNICAL SPECIFICATION 3.0.3 AND INITIATED A COOLDOWN TO MODE 5. THE CHILLER OIL HEATERS WERE SUBSEQUENTLY REENERGIZED AND THE CHILLERS WERE INSPECTED FOR PROPER OPERATION AND DECLARED OPERABLE AT 2254 HOURS THE SAME DAY. ALTHOUGH IT WAS NOT POSSIBLE TO CONFIRM HOW OR WHY THE SWITCHES WERE MOVED THE ROOT CAUSE HAS BEEN ATTRIBUTED TO INADEQUATE PROTECTION OF EQUIPMENT CONTROLS. ADDITION OF CAUTION TAGS, REVISION TO OPERATOR LOGS AND INSPECTIONS, AND EVALUATION OF MODIFICATIONS TO THE LOCAL PANELS HAVE BEEN INITIATED. NO SAFETY CONSEQUENCES RESULTED FROM THE EVENT.

[298] SOUTH TEXAS 1 DOCKET 50-498 LER 87-017
PRESSURIZER LOW PRESSURE SAFETY INJECTION SETPOINT TOO LOW DUE TO PROCEDURAL ERROR.
EVENT DATE: 112487 REPORT DATE: 122187 NSSS: WE TYPE: PWR

(NSIC 207551) ON NOVEMBER 24, 1987 A DISCREPANCY BETWEEN THE TECHNICAL SPECIFICATION PRESSURIZER LOW PRESSURE SAFETY INJECTION SETPOINT AND THE ACTUAL SETPOINT OF THE EQUIPMENT IN THE PLANT WAS IDENTIFIED BY HL&P. THE SETPOINT IS EXISTED IN THE TECHNICAL SPECIFICATIONS AS GREATER THAN OR EQUAL TO 1850 PSIG WITH NOTATION REFERRING TO 1869 PSIG ON A SEPARATE PAGE. TECHNICAL SPECIFICATIONS REQUIRE THE SETPOINT TO BE GREATER THAN OR EQUAL TO 1869 PSIG AS AN INTERIM RESOLUTION OF THE VERITRAK TRANSMITTER UNCERTAINTY ISSUE FOR SOUTH TEXAS. THE SURVEILLANCE PROCEDURES WERE WRITTEN LISTING 1850 PSIG AS THE SET POINT. THE UNIT HAD NOT OPERATED IN A CONDITION WHERE THE AFFECTED INSTRUMENTATION WAS REQUIRED OR IN A CONDITION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. THE AFFECTED SURVEILLANCE PROCEDURES AND INSTRUMENTATION SETPOINTS HAVE BEEN CORRECTED. OTHER PLANT SURVEILLANCE PROCEDURES HAVE BEEN REVIEWED FOR SIMILAR DISCREPANCIES.

[299] SOUTH TEXAS 1 DOCKET 50-498 LER 87-018
INITIATION OF COOLDOWN DUE TO INOPERABILITY OF TWO TRAINS OF CONTAINMENT SPRAY.
EVENT DATE: 112487 REPORT DATE: 122387 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SOUTH TEXAS 2 (PWR)

(NSIC 207552) ON NOVEMBER 24, 1987 AT APPROXIMATELY 1249 HOURS WITH THE UNIT IN MODE 4 AND PRIOR TO INITIAL CRITICALITY, OPERATIONS PERSONNEL DECLARED TRAIN A OF THE ESSENTIAL COOLING WATER (ECW) SYSTEM TO BE INOPERABLE DUE TO A CRACK FOUND IN THE THREADED CONNECTION OF A ONE HALF INCH PIPE ATTACHED TO A SYSTEM ANNUBAR FLOW ELEMENT. ECW IS A SUPPORT SYSTEM FOR CONTAINMENT SPRAY. AS A RESULT, TRAIN A OF THE CONTAINMENT SPRAY SYSTEM WAS ALSO DECLARED INOPERABLE. AT THIS TIME TRAIN C OF CONTAINMENT SPRAY WAS INOPERABLE DUE TO THE PLANT BEING IN A 72 HOUR TECHNICAL SPECIFICATION OUTAGE FOR RESTORATION OF SPRAY ADDITIVE TANK LEVEL. THEREFORE, TWO OF THE THREE TRAINS OF CONTAINMENT SPRAY WERE INOPERABLE AND, AS A RESULT, THE PLANT ENTERED TECHNICAL SPECIFICATION 3.0.3 AND INITIATED A COOLDOWN TO MODE 5. AT 0700 ON NOVEMBER 25 TRAIN C OF CONTAINMENT SPRAY WAS RETURNED TO OPERABLE STATUS AND THE COOLDOWN WAS STOPPED PRIOR TO ENTRY INTO MODE 5. THE ROOT CAUSE OF THE CRACKED CONNECTION HAS BEEN ATTRIBUTED TO DESIGN ERROR ASSOCIATED WITH THE ANNUBAR AND SUPPORT. MODIFICATION OF THE ANNUBAR OPPOSITE SIDE AND SUPPORTS FOR

ALL THREE TRAINS OF ECW WILL BE INITIATED. NO SAFETY CONSEQUENCES RESULTED FROM THE EVENT.

[300] SOUTH TEXAS 1 DOCKET 50-498 LER 87-019 REV 01
UPDATE ON SLAVE RELAY SURVILLANCE DEFICIENCY DUE TO PERSONNEL ERROR.
EVENT DATE: 112487 REPORT DATE: 011888 NSSS: WE TYPE: PWR

(NSIC 207923) ON NOVEMBER 24, 1987 AT APPROXIMATELY 1330 HOURS DURING THE REVIEW OF A SLAVE RELAY SURVEILLANCE PROCEDURE, IT WAS DETERMINED THAT THE PROCEDURE HAD NOT PROPERLY TESTED THE CONTINUITY OF A SLAVE RELAY CONTACT IN TRAIN A OF THE CONTAINMENT SPRAY (CS) SYSTEM WHICH WAS NECESSARY TO INITIATE A CONTAINMENT SPRAY ACTUATION. A FIELD CHANGE TO THE PROCEDURE HAD DELETED A STEP WHICH WOULD HAVE PROPERLY TESTED THE SLAVE RELAY CONTACT. THE UNIT HAD ENTERED MODE 4 ON OCTOBER 31, 1987 AND THE FAILURE TO ADEQUATELY TEST THE SLAVE RELAY CONTACT PRIOR TO ENTERING MODE 4 WAS A VIOLATION OF THE TECHNICAL SPECIFICATIONS. TRAIN A OF THE CS SYSTEM WAS IMMEDIATELY DECLARED INOPERABLE AND THE SLAVE RELAY CONTACT WAS SATISFACTORILY TESTED ON NOVEMBER 25, 1987. THE CAUSE OF THE EVENT WAS DETERMINED TO BE PERSONNEL ERROR, IN THAT THE TECHNICAL REVIEWERS OF THE PROCEDURE MISREAD A DRAWING DURING A SUPPLEMENTARY PROCEDURE REVIEW JUST PRIOR TO THE PERFORMANCE OF THE PROCEDURE AND AN INADEQUATE TECHNICAL VERIFICATION OF THE ENSUING FIELD CHANGE TO THE PROCEDURE WAS CONDUCTED. TO PREVENT RECURRENCE OF THE EVENT, THE SLAVE RELAY SURVEILLANCE PROCEDURES FOR TRAINS B AND C HAVE BEEN REVIEWED TO ENSURE THAT IDENTICAL ERRORS DID NOT EXIST AND THE INSTRUMENTATION AND CONTROLS GROUP TECHNICAL SUPERVISORS HAVE RECEIVED TRAINING CONCERNING THE NECESSITY OF INDEPENDENT REVIEW OF FIELD CHANGES.

[301] SOUTH TEXAS 1 DOCKET 50-498 LER 87-020
CONTROL ROOM VENTILATION ACTUATION TO RECIRCULATION MODE DUE TO FAILURE OF A TOXIC GAS MONITOR COMPUTER CHIP.
EVENT DATE: 112887 REPORT DATE: 122287 NSSS: WE TYPE: PWR
VENDOR: FOXBORO CO., THE

(NSIC 207540) AT APPROXIMATELY 1556 HOURS ON NOVEMBER 28, 1987 WITH UNIT 1 IN MODE 4, AN AUTO-ACTUATION OF THE CONTROL ROOM VENTILATION TO RECIRCULATION MODE OCCURRED AS A RESULT OF A MALFUNCTION OF A TOXIC GAS MONITOR. THE CONTROL ROOM VENTILATION ACTUATION TO RECIRCULATION MODE IS AN ENGINEERED SAFETY FEATURE (ESF). THE CONTROL ROOM OPERATORS VERIFIED THE RECIRCULATION MODE DAMPER LINEUP AND INITIATED AN INVESTIGATION OF THE EVENT. THE INVESTIGATION DETERMINED THAT THE EVENT WAS CAUSED BY A FAILURE OF A TOXIC GAS MONITOR COMPUTER CHIP. ON NOVEMBER 29, 1987 THE FAILED COMPUTER CHIP WAS REPLACED AND THE SYSTEM WAS RESTORED TO NORMAL OPERATION ON NOVEMBER 30, 1987. A DISCUSSION WITH THE MANUFACTURER INDICATED THAT THE FAILURE HISTORY OF MONITORS AT OTHER PLANTS DID NOT INDICATE GENERIC PROBLEM WITH THE COMPUTER CHIP. A TASK FORCE IS CONTINUING TO EVALUATE THE RELIABILITY OF THE TOXIC GAS MONITORING SYSTEM AND AS A RESULT A DESIGN MODIFICATION OF THE ESF ACTUATION LOGIC HAS BEEN INITIATED WHICH WILL REQUIRE BOTH TOXIC GAS MONITORS TO BE INOPERABLE (MONITOR MALFUNCTION OR LOSS OF POWER) TO INITIATE AN ESF ACTUATION. THE ESF ACTUATION LOGIC IN RESPONSE TO AN INDICATION OF THE PRESENCE OF TOXIC GAS WILL NOT BE MODIFIED.

[302] SOUTH TEXAS 1 DOCKET 50-498 LER 87-021
ACTUATION OF ESF LOAD SEQUENCER AND STANDBY DIESEL GENERATOR DUE TO PERSONNEL ERROR.
EVENT DATE: 113087 REPORT DATE: 010888 NSSS: WE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 207702) AT APPROXIMATELY 2145 HOURS ON NOVEMBER 30, 1987 WITH THE UNIT IN MODE 4 PRIOR TO INITIAL CRITICALITY, WHILE TROUBLESHOOTING THE TRAIN B ENGINEERED SAFETY FEATURE (ESF) LOAD SEQUENCER, AN UNANTICIPATED BUS STRIP AND DIESEL

GENERATOR START OCCURRED WHEN A MAINTENANCE TECHNICIAN DEENERGIZED THE SEQUENCER. THE SEQUENCER WAS REENERGIZED TO CLEAR THE BUS STRIP AND THE SEQUENCER CORRECTLY BEGAN A LOSS OF OFFSITE POWER (LOOP) LOADING SEQUENCE. AFTER CONFIRMING THAT NO LOOP ACTUALLY EXISTED, THE SEQUENCER WAS DEENERGIZED, THE DIESEL GENERATOR WAS STOPPED, AND AFFECTED EQUIPMENT WAS RETURNED TO NORMAL ALIGNMENT. THE CAUSE OF THE OCCURRENCE HAS NOT BEEN CONCLUSIVELY IDENTIFIED. THE MOST LIKELY CAUSE APPEARS TO BE THE CREATION OF UNUSUAL LOGIC STATES IN THE SEQUENCER MAIN PROCESSOR WHEN IT WAS PARTIALLY DEENERGIZED. THE METHOD FOR DEENERGIZATION OF THE SEQUENCERS HAS BEEN REVISED TO PRECLUDE THE POSSIBILITY OF THIS TYPE OF FAILURE.

[303] SOUTH TEXAS 1 DOCKET 50-498 LER 87-022
INOPERABILITY OF TWO TOXIC GAS MONITORS.
EVENT DATE: 120687 REPORT DATE: 010788 NSSS: WE TYPE: PWR

(NSIC 207703) AT APPROXIMATELY 0530 HOURS ON DECEMBER 6, 1987 WITH UNIT 1 IN MODE 4, BOTH CONTROL ROOM TOXIC GAS MONITORS WERE FOUND INOPERABLE. THE CONTROL ROOM VENTILATION WAS IMMEDIATELY PLACED INTO THE RECIRCULATION MODE. CHANNEL CHECKS ON BOTH TOXIC GAS MONITORS HAD BEEN PERFORMED BY A REACTOR PLANT OPERATOR IN TRAINING FOR CONTROL ROOM OPERATOR, ON DECEMBER 4, 1987. THE METHOD HE USED IN PERFORMING THE CHANNEL CHECKS REQUIRED PLACING THE MONITORS IN THE SUMMARY MODE. WITH THE TOXIC GAS COMPUTERS IN THIS MODE THE ACTUATION FUNCTION OF THE MONITORS WAS DISABLED. UPON COMPLETING THE CHANNEL CHECKS THE OPERATOR FAILED TO RETURN THE MONITORING SYSTEMS TO NORMAL OPERATION. THE CAUSE OF THE OCCURRENCE WAS FAILURE TO PROVIDE ADEQUATE TRAINING. CORRECTIVE ACTIONS WHICH ARE BEING TAKEN INCLUDE PROVIDING TRAINING FOR THE OPERATORS ON THE PROPER OPERATION AND USE OF THE TOXIC GAS MONITORS, EVALUATING THE FEASIBILITY OF A MORE POSITIVE METHOD OF DETERMINING THE OPERABILITY OF THE TOXIC GAS SYSTEM, AND REVISING THE LOG KEEPING PROCEDURE TO PROVIDE ADEQUATE INSTRUCTION IN PERFORMING CHANNEL CHECKS ON THE TOXIC GAS MONITORS. PLANT BRIEFINGS WILL BE CONDUCTED AND THE PLANT CONDUCT OF OPERATIONS PROCEDURE WILL BE REVISED TO SPECIFICALLY DESIGNATE HOW STUDENTS CAN BE USED.

[304] SOUTH TEXAS 1 DOCKET 50-498 LER 87-023
LOOSE VALVE-SHAFT-TO ACTUATOR-DRIVE KEYS IN MOTOR OPERATED VALVES SUPPLIED BY ROCKWELL INTERNATIONAL.
EVENT DATE: 120887 REPORT DATE: 010788 NSSS: WE TYPE: PWR
VENDOR: HILLS-MCCANNA COMPANY

(NSIC 207704) PRIOR TO THE UNIT'S INITIAL ENTRY INTO MODE 4 ON 10/10/87, HL&P IDENTIFIED A VALVE WITH A DISPLACED VALVE-TO-ACTUATOR-DRIVE KEY. ON 10/12/87 WITH THE UNIT STILL IN MODE 5, A SECOND VALVE WAS IDENTIFIED WITH THE SAME CONDITION. INSPECTIONS IDENTIFIED 2 OTHER VALVES WITH DISPLACED KEYS AND 10 OTHERS WITH AT LEAST 1 LOOSE KEY. LOOSE KEYS WERE REPLACED WITH KEYS OF THE PROPER SIZE OR SHIMMED TO ENSURE A SNUG-TIGHTFIT WITH THEIR KEYWAYS PRIOR TO ENTRY INTO MODE 4. ON 12/8/87, AT APPROXIMATELY 1755 HOURS WITH UNIT 1 IN MODE 4, HL&P COMPLETED THE ENGINEERING EVALUATION THAT DETERMINED THAT SAFETY RELATED EQUIPMENT COULD HAVE BEEN PREVENTED FROM PERFORMING ITS SAFETY RELATED FUNCTION DUE TO THE LOOSE OR DISPLACED KEYS IN FOUR OF THE AFFECTED MOTOR OPERATED VALVES (MOVS). EACH OF THREE TRAINS OF THE STPEGS REACTOR CONTAINMENT FAN COOLERS (RCFCS) WERE AFFECTED. THE RCFCS ARE REQUIRED TO MITIGATE THE CONSEQUENCES OF A LOSS OF COOLANT ACCIDENT OR A HIGH ENERGY LINE BREAK INSIDE CONTAINMENT. THE KEYS IN QUESTION WERE SUPPLIED BY ROCKWELL WITH A LOOSE FIT AND MAY HAVE BEEN DISPLACED DURING SHIPMENT OR INSTALLATION. THE LACK OF ADEQUATE GUIDANCE FROM THE VENDOR RESULTED IN THE DEFICIENT CONDITION AT STP. VENDOR MANUALS HAVE BEEN AMENDED TO REQUIRE A SNUG-TIGHTFIT ENGAGEMENT AND AN INSPECTION ATTRIBUTE WILL BE ADDED TO IDENTIFY LOOSE OR DISPLACED KEYS.

[305] SOUTH TEXAS 1 DOCKET 50-498 LER 87-024
CONTROL ROOM VENTILATION ACTUATION TO RECIRCULATION MODE DUE TO INADVERTENT OPERATION OF PUSHBUTTON BY TECHNICIAN.
EVENT DATE: 120887 REPORT DATE: 010788 NSSS: WE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 207705) AT APPROXIMATELY 2153 HOURS ON DECEMBER 8, 1987, PRIOR TO INITIAL CRITICALITY AND WHILE IN MODE 4, AN AUTO-ACTUATION OF THE CONTROL ROOM VENTILATION SYSTEM TO RECIRCULATION MODE OCCURRED AS A RESULT OF A PERSONNEL ERROR BY A TECHNICIAN PERFORMING MAINTENANCE ON THE CONTROL ROOM RADIATION MONITORING SYSTEM REMOTE CONTROL CONSOLE (ZCP-023). TECHNICIANS WERE PERFORMING WORK ON A FUEL HANDLING BUILDING RADIATION MONITOR AND INADVERTENTLY PRESSED THE "FLOW" PUSHBUTTON FOR THE ADJACENT CONTROL ROOM VENTILATION RADIATION MONITOR SAMPLE PUMP CONTROL MODULE. THIS CAUSED THE SAMPLE PUMP TO STOP WHICH RESULTED IN AN ESF ACTUATION OF THE CONTROL ROOM VENTILATION SYSTEM TO RECIRCULATION MODE AS DESIGNED. FOLLOWING THE EVENT, THE OPERATORS VERIFIED THE STATUS OF THE AFFECTED EQUIPMENT AND THE VENTILATION SYSTEM WAS LEFT IN THE RECIRCULATION MODE TO ALLOW COMPLETION OF THE TECHNICIANS' WORK. THIS EVENT WAS CAUSED BY A PERSONNEL ERROR. THE "FLOW" PUSHBUTTONS WHICH COULD CAUSE INADVERTENT ESF ACTUATIONS HAVE SINCE BEEN DISABLED ON THE CONTROL ROOM REMOTE PANEL. ADDITIONALLY, THE IMPORTANCE OF ATTENTION TO DETAIL IS BEING STRESSED TO TECHNICIANS.

[306] SOUTH TEXAS 1 DOCKET 50-498 LER 87-025
STANDBY DIESEL GENERATOR ACTUATION DUE TO PROCEDURAL ERRORS.
EVENT DATE: 120987 REPORT DATE: 010888 NSSS: WE TYPE: PWR
VENDOR: AGASTAT RELAY CO.

(NSIC 207706) AT 0424 HOURS ON DECEMBER 9, 1987 WITH UNIT 1 IN MODE 4, PRIOR TO INITIAL CRITICALITY, A TRAIN A LOSS OF OFFSITE POWER (LOOP) ACTUATION OCCURRED. THIS CAUSED AN ENGINEERED SAFETY FEATURE (ESF) TRAIN A BUS STRIP, STANDBY DIESEL GENERATOR START AND SUBSEQUENT SEQUENCING OF THE TRAIN A ESF COMPONENTS. NORMAL POWER WAS RESTORED TO THE BUS AND THE DIESEL GENERATOR WAS REMOVED FROM SERVICE AT 0455 HOURS. THE ROOT CAUSE OF THIS ACTUATION COULD NOT BE CONCLUSIVELY ESTABLISHED; HOWEVER, THE MOST LIKELY CAUSES ARE PROCEDURAL ERRORS COUPLED WITH A HARDWARE FAILURE. THE ESF TRAIN COMPONENT ACTUATIONS OCCURRED AS DESIGNED. A NUMBER OF CORRECTIVE ACTIONS HAVE BEEN OR WILL BE IMPLEMENTED TO ADDRESS THE MOST LIKELY CAUSES OF THE EVENT. THERE WERE NO ADVERSE SAFETY OR RADIOLOGICAL CONSEQUENCES AS A RESULT OF THIS EVENT.

[307] SOUTH TEXAS 1 DOCKET 50-498 LER 87-026
DEGRADED UNDERVOLTAGE COINCIDENT WITH A SAFETY INJECTION CIRCUITRY SURVEILLANCE DEFICIENCY DUE TO A DEFICIENT PROCEDURE.
EVENT DATE: 121287 REPORT DATE: 011188 NSSS: WE TYPE: PWR

(NSIC 207924) ON DECEMBER 12, 1987, AT APPROXIMATELY 1857 HOURS WITH UNIT 1 IN MODE 4, PRIOR TO INITIAL CRITICALITY, DURING REVIEW OF WORK INSTRUCTIONS FOR THE REPLACEMENT OF A TIME DELAY RELAY IN THE DEGRADED UNDERVOLTAGE CIRCUIT, IT WAS DETERMINED THAT THE TRIP ACTUATION DEVICE OPERATIONAL TEST (TADOT) ON DEGRADED UNDERVOLTAGE COINCIDENT WITH SAFETY INJECTION HAD NOT BEEN TESTED AS REQUIRED. ALL THREE ENGINEERED SAFETY FEATURES (ESF) BUSSES WERE DECLARED INOPERABLE. THE PLANT ENTERED TECHNICAL SPECIFICATION 3.0.3 AND A PLANT COOLDOWN TO MODE 5 WAS INITIATED. THE CAUSE OF THE EVENT WAS DETERMINED TO BE A DEFICIENT SURVEILLANCE PROCEDURE RESULTING FROM A PERSONNEL ERROR IN INTERPRETING THE REQUIREMENTS OF THE MONTHLY TADOT. TO PREVENT RECURRENCE, A NEW PROCEDURE WAS WRITTEN AND SATISFACTORILY PERFORMED ON EACH ESF BUS. TESTING WAS COMPLETED AT APPROXIMATELY 1300 ON DECEMBER 13, 1987 PRIOR TO COMPLETING THE COOLDOWN TO MODE 5. ADDITIONALLY, COMPREHENSIVE REVIEWS OF INSTRUMENTATION & CONTROLS AND ELECTRICAL SURVEILLANCE PROCEDURES WERE CONDUCTED TO ENSURE OTHER TESTING REQUIREMENTS WERE

COVERED IN SURVEILLANCE PROCEDURES. THERE WERE NO ADVERSE SAFETY RADIOLOGICAL CONSEQUENCES AS A RESULT OF THIS EVENT.

[308] ST. LUCIE 1 DOCKET 50-335 LER 87-017
REACTOR TRIP DUE TO REACTOR PROTECTIVE SYSTEM HI START-UP RATE B CHANNEL IN TRIP AND THE LOSS OF 1MD 120V AC INSTRUMENT BUS DUE TO PERSONNEL ERROR.
EVENT DATE: 122187 REPORT DATE: 012088 NSSS: CE TYPE: PWR

(NSIC 207963) ON 12/21/87, WHILE OPERATING IN MODE 1 AT 100% POWER, ST. LUCIE UNIT #1 TRIPPED DUE TO THE LOSS OF THE 1MD 120 VOLT AC BUS. WITH THE REACTOR PROTECTIVE SYSTEM (RPS) HI START-UP RATE B CHANNEL BISTABLE IN TRIP, THE RPS LOGIC FOR HI START-UP RATE WAS 1 OF 3. THE LOSS OF THE 1D INSTRUMENT INVERTER CAUSED THE SUBSEQUENT LOSS OF THE 1MD 120 VOLT AC BUS, WHICH RESULTED IN THE ACTUATION OF THE DEENERGIZE TO ACTUATE FUNCTION OF THE RPS D CHANNEL TRIP BISTABLES THUS SATISFYING THE RPS TRIP LOGIC. THE ROOT CAUSE OF THE EVENT WAS A COGNITIVE PERSONNEL ERROR BY A UTILITY NON-LICENSED OPERATOR NOT ADEQUATELY FOLLOWING A PLANT APPROVED PROCEDURE FOR OPERATION OF THE 120 VOLT INSTRUMENT AC CLASS 1E SYSTEM. THE NON-LICENSED OPERATOR WAS COUNSELED BY HIS SUPERVISOR ON THE IMPORTANCE OF ADEQUATELY FOLLOWING APPROVED PROCEDURES AND THE NEED FOR GREATER ATTENTION TO DETAIL WHILE PERFORMING CRITICAL JOB RESPONSIBILITIES. A PROCEDURE FOR INFREQUENT OPERATIONS OR MANIPULATIONS IS BEING DRAFTED. THIS IS TO ASSURE A DETAILED REVIEW AND BRIEFING BY THE SHIFT SUPERVISOR WITH APPROPRIATE PERSONNEL FOR SAFE AND SATISFACTORY PERFORMANCE. THE PLANT TRAINING DEPARTMENT WILL EVALUATE THIS ITEM TO DETERMINE APPROPRIATE TRAINING METHODS AND REQUIREMENTS. A HUMAN PERFORMANCE EVALUATION IS BEING CONDUCTED TO IDENTIFY ANY AREAS THAT MAY BE OF CONCERN.

[309] ST. LUCIE 2 DOCKET 50-389 LER 87-007 REV 01
UPDATE ON REACTOR TRIP ON LOSS OF LOAD CAUSED BY MAIN GENERATOR EXCITER BEARING FAILURE DUE TO PERSONNEL ERROR.
EVENT DATE: 112587 REPORT DATE: 012988 NSSS: CE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 207936) ON 25 NOVEMBER 1987, ST. LUCIE UNIT TWO WAS OPERATING AT 50 PERCENT POWER STEADY STATE IN MODE 1. AT 2331 HOURS, THE REACTOR TRIPPED ON LOSS OF LOAD DUE TO A TURBINE TRIP. THE TURBINE TRIPPED ON A MAIN GENERATOR LOCKOUT. THE FAILURE OF THE MAIN GENERATOR EXCITER BEARING CAUSED THE ARMATURE OF THE PERMANENT MAGNET GENERATOR (PMG) TO GRIND INTO ITS STATOR. WHEN THIS OCCURRED, THE PMG DISCONTINUED SUPPLYING VOLTAGE TO THE EXCITER OF THE MAIN GENERATOR. THE TRIP WAS UNCOMPLICATED AND THE UNIT WAS QUICKLY STABILIZED IN MODE 3, HOT STANDBY. THE ROOT CAUSE OF THE EVENT WAS A PERSONNEL ERROR DURING THE PERFORMANCE OF THE WEEKLY GENERATOR EXCITER GROUND CHECK ON WHICH A PROLONGED GROUND RESULTED IN THE FAILURE OF THE GENERATOR EXCITER BEARING. THE FOLLOWING CORRECTIVE ACTIONS HAVE BEEN IMPLEMENTED: REPLACED THE PMG AND THE EXCITER BEARING, REMOVED THE EXCITER BEARING THERMOCOUPLE TO PREVENT FUTURE GROUNDING OCCURRENCE OF THIS TYPE, PROCEDURES HAVE BEEN REVISED TO ENSURE ANY FUTURE GROUNDS ARE RECOGNIZED, AND TRAINING WAS GIVEN TO ALL ELECTRICAL MAINTENANCE PERSONNEL FOR BETTER UNDERSTANDING OF THE EXCITER GROUND CHECK.

[310] SUMMER 1 DOCKET 50-395 LER 87-029
ENGINEERED SAFETY FEATURE (ESF) ACTUATION DURING TESTING DUE TO PERSONNEL ERROR.
EVENT DATE: 121387 REPORT DATE: 010788 NSSS: WE TYPE: PWR

(NSIC 208036) ON DECEMBER 13 AND 14, 1987, THE STAND-BY COMPONENT COOLING WATER (CCW) PUMP, WHICH HAD PREVIOUSLY BEEN DECLARED INOPERABLE, AUTO STARTED AS THE RESULT OF A VALID SIGNAL DURING TESTING. THE OPERATORS, ALTHOUGH AWARE OF THE AUTO-START FEATURE, DID NOT TAKE INTO CONSIDERATION THE "LOW PRESSURE AUTO START" DURING AN UNCOUPLED RUN OF THE "A" CCW PUMP FOR POST MAINTENANCE TESTING ON

DECEMBER 13. ON DECEMBER 14, THE DISCHARGE FLOW OF THE "A" PUMP WAS BEING ADJUSTED PER AN APPROVED TEST PROCEDURE, WITH THE "C" PUMP IN STAND-BY, WHEN A LOW PRESSURE SPIKE CAUSED AN AUTO-START OF THE STAND-BY PUMP. THE CAUSE OF THE FIRST EVENT WAS A RESULT OF NOT PERFORMING AN IN-DEPTH REVIEW OF THE POST MAINTENANCE TESTS. THE CAUSE OF THE SECOND EVENT WAS THAT THE SHIFT SUPERVISOR HAD NOT PREVIOUSLY EXPERIENCED "AUTO-STARTING" OF THE STAND-BY PUMP WHILE PERFORMING THESE ADJUSTMENTS AND CHOSE NOT TO ALIGN THE SYSTEM TO PRECLUDE ACTUATION. THE OPERATORS IMMEDIATELY SECURED THE PUMP FOLLOWING EACH AUTO-START. CORRECTIVE ACTION TO PRECLUDE RECURRENCE WILL BE TO PLACE A COPY OF THIS REPORT INTO THE OPERATORS REQUIRED READING. IN ADDITION, EACH SHIFT SUPERVISOR WILL REVIEW THIS EVENT WITH HIS SHIFT TO ENSURE ANY FUTURE POST MAINTENANCE TESTING OF ESF EQUIPMENT TAKES INADVERTENT AUTO-START INTO CONSIDERATION.

[311] SUMMER 1 DOCKET 50-395 LER 87-030
FAILURE TO ESTABLISH A CONTINUOUS FIRE WATCH DUE TO PERSONNEL ERROR.
EVENT DATE: 122987 REPORT DATE: 012188 NSSS: WE TYPE: PWR

(NSIC 207966) AT APPROXIMATELY 2240 HOURS, DECEMBER 29, 1987, DURING THE REVIEW OF STATION LOGS, THE CONTROL ROOM SUPERVISOR NOTED THAT AN AUXILIARY OPERATOR (AO) HAD MADE A LOG ENTRY NOTING THAT A FIRE DOOR HAD NOT BEEN FULLY SHUT WHEN EXITING THE PENETRATION AREA DUE TO WIRES RUNNING UNDER THE DOOR. BECAUSE THE INTEGRATED FIRE COMPUTER SYSTEM (IFCS) HAD PREVIOUSLY BEEN REMOVED FROM SERVICE, NOT FULLY CLOSING THE FIRE DOOR REQUIRED A CONTINUOUS FIRE WATCH. THE CONTROL ROOM SUPERVISOR INITIATED IMMEDIATE ACTION TO FULLY SHUT THE FIRE DOOR. THE CAUSE OF THIS EVENT WAS DUE TO PERSONNEL ERROR. THE AO WAS AWARE THAT THE IFCS WAS REMOVED FROM SERVICE. HOWEVER, THE AO HAD THE MISCONCEPTION THAT A CONTINUOUS FIRE WATCH AND A CONTINUOUS ROVING FIRE WATCH WERE THE SAME. ON JANUARY 6, 1988, A MANAGEMENT REVIEW BOARD MEETING, CHAIRED BY THE VICE PRESIDENT, NUCLEAR OPERATIONS CONVENED TO REVIEW THIS EVENT. THE FOLLOWING CORRECTIVE ACTION WILL BE TAKEN AS A RESULT OF THIS EVENT: 1. THE AO INVOLVED IN THE EVENT ATTENDED THE MRB AND WILL BRIEF EACH SHIFT OF THIS EVENT AND LESSONS LEARNED. 2. A COPY OF THIS REPORT WILL BE PLACED IN OPERATIONS REQUIRED READING. 3. AO TRAINING WILL BE REVIEWED AND UPGRADED IF REQUIRED.

[312] SUMMER 1 DOCKET 50-395 LER 87-031
MISSED FIRE WATCH DUE TO INADEQUATE PROCEDURE.
EVENT DATE: 122987 REPORT DATE: 012888 NSSS: WE TYPE: PWR

(NSIC 207967) AT APPROXIMATELY 0810 HOURS, 12/29/87, THE INTEGRATED FIRE COMPUTER SYSTEM (IFCS) WAS REMOVED FROM SERVICE. A FIRE WATCH PATROL WAS ESTABLISHED AND THE FIRE WATCH LOG IN THE FIRE SERVICE SYSTEM OPERATING PROCEDURE (SOP-509) WAS UTILIZED TO DOCUMENT THE FIRE WATCH PATROL. AT APPROXIMATELY 2110 HOURS, A DETAILED REVIEW OF THE FIRE WATCH PATROL LOG BY THE FIRE PROTECTION COORDINATOR REVEALED THAT FOUR ROOMS WERE NOT LISTED. IMMEDIATE ACTION WAS TAKEN TO ESTABLISH ROVING PATROLS IN THESE ROOMS AND TO CORRECT THE FIRE WATCH LOG. THE EVENT WAS CAUSED BY AN INADEQUATE FIRE SERVICE SYSTEM OPERATING PROCEDURE, SOP-509. ON JANUARY 6, 1988, A MANAGEMENT REVIEW BOARD MEETING CHAIRED BY THE VICE PRESIDENT, NUCLEAR OPERATIONS, CONVENED TO REVIEW THIS EVENT. THE FOLLOWING CORRECTIVE ACTIONS WERE DETERMINED TO ADDRESS IMMEDIATE CONCERNS: 1. THE SYSTEM OPERATING PROCEDURE FOR FIRE SERVICE (SOP-509) WAS REVIEWED AGAINST THE TECH SPECS IN ORDER TO PROVIDE ASSURANCE THAT THERE WERE NO FURTHER DISCREPANCIES. 2. THE PROVISION OF FIRE PROTECTION SHIFT LEADERS (OFFICERS) AS REFERENCED IN LER 87018 IS ON SCHEDULE TO MEET THE 3/31/88 COMMITMENT DATE. THE ADDITION OF THESE FIRE PROTECTION OFFICERS WILL PROVIDE ADDED ATTENTION TO THE DAILY FIRE PROTECTION ACTIVITIES AND ALLOW SINGLE POINT ACCOUNTABILITY REGARDING FIRE PROTECTION.

[313] SURRY 1 DOCKET 50-280 LER 87-035
REQUIRED SURVEILLANCE FOR EDGS NOT PERFORMED DUE TO HUMAN ERROR.
EVENT DATE: 071787 REPORT DATE: 123187 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 207633) ON JULY 17, 1987, WITH UNITS 1 AND 2 AT 100% POWER, IT WAS DETERMINED THAT THE EMERGENCY DIESEL GENERATOR (EDG) (EIIS-DG) DEGRADED/UNDERVOLTAGE PROTECTION (EIIS-EB) SURVEILLANCE TEST WAS NOT BEING PERFORMED. THIS TEST DEMONSTRATES THAT THE DEGRADED/UNDERVOLTAGE PROTECTION IS DEFEATED WHEN THE EDG IS THE SOLE SOURCE OF POWER TO AN EMERGENCY BUS, AND THAT THE PROTECTION IS REINSTATED WHEN THE DIESEL OUTPUT BREAKER IS OPEN. TECHNICAL SPECIFICATION 4.6.1.B REQUIRES THAT THIS TEST BE PERFORMED DURING EACH REFUELING SHUTDOWN. FAILURE TO PERFORM THE DIESEL DEGRADED/UNDERVOLTAGE SURVEILLANCE AS REQUIRED BY TECHNICAL SPECIFICATIONS WAS CAUSED BY A FAILURE TO INCORPORATE THE REQUIREMENTS OF A TECHNICAL SPECIFICATION AMENDMENT INTO THE EXISTING TEST PROGRAM. THIS WAS DETERMINED TO BE A REPORTABLE OCCURRENCE BY THE STATION NUCLEAR SAFETY AND OPERATING COMMITTEE ON DECEMBER 4, 1987. SPECIAL TESTS FOR FUNCTIONAL TESTING OF THE ENABLE/DISABLE INTERLOCKS WERE PERFORMED AND COMPLETED ON THE EMERGENCY BUSES IN SEPTEMBER, 1987. THIS TESTING VERIFIED THAT THE RELAYS ARE FULLY OPERABLE AND THAT THE PROTECTION MATRIX FUNCTIONS AS DESIGNED. THE REFUELING PERIODIC TESTS WILL BE REVISED TO INCLUDE TESTING OF THE ENABLE/DISABLE INTERLOCK. THE REVISIONS WILL BE COMPLETED PRIOR TO THE NEXT REFUELING OUTAGE.

[314] SURRY 1 DOCKET 50-280 LER 87-034
IODINE SPIKE DUE TO DEFECTIVE FUEL ELEMENT.
EVENT DATE: 112687 REPORT DATE: 122287 NSSS: WE TYPE: PWR

(NSIC 207431) ON NOVEMBE 26, 1987, AT 0415 HOURS, FOLLOWING A UNIT 1 SHUTDOWN FOR REPAIRS TO STEAM GENERATOP 'C' SECONDARY MANWAY, THE SPECIFIC ACTIVITY SAMPLE OF THE REACTOR COOLANT SHOWED A DOSE EQUIVALENT I-131 OF 1.19 MICROCURIES/CC. THIS EXCEEDS THE DOSE EQUIVALENT I-131 TECHNICAL SPECIFICATION OF LESS THAN 1.0 MICROCURIES/CC SPECIFIED IN SECTION 3.1.D.2 AND IS BEING REPORTED IN ACCORDANCE WITH THE SPECIAL REPORTING REQUIREMENTS OUTLINED IN TECHNICAL SPECIFICATION 3.1.D.4. THE IODINE SPIKE WAS CAUSED BY A KNOWN, BUT NOT SPECIFICALLY LOCATED, FUEL ELEMENT DEFECT IN THE REACTOR CORE. POST SHUTDOWN CONDITIONS ENHANCED THE RELEASE OF FISSION PRODUCTS, SPECIFICALLY I-131. THIS CAUSED AN INCREASE IN REACTOR COOLANT SPECIFIC ACTIVITY. THE IMMEDIATE CORRECTIVE ACTION WAS TO IMPLEMENT THE ACTIONS REQUIRED BY TECHNICAL SPECIFICATION TABLE 4.1.2B. SPECIFICALLY, THE LEVEL OF DOSE-EQUIVALENT I-131 WAS MONITORED AT LEAST EVERY FOUR HOURS UNTIL THE LEVEL RETURNED TO LESS THAN 1 MICROCURIES/CC.

[315] SURRY 1 DOCKET 50-280 LER 87-036
CONTAINMENT AIR LOCK LEAKAGE DUE TO INADEQUATE SEAL.
EVENT DATE: 120287 REPORT DATE: 123187 NSSS: WE TYPE: PWR
VENDOR: WOOLLEY, W. J. COMPANY

(NSIC 207676) ON DECEMBER 2, 1987 AT 0415 HOURS, WITH UNIT 1 AT HOT SHUTDOWN, THE INNER DOOR OF THE CONTAINMENT PERSONNEL AIR LOCK (EIIS-AL) EXCEEDED THE LEAKAGE ACCEPTANCE CRITERIA FOR PERIODIC TEST 16.7, "PERSONNEL AIR LOCK LEAKAGE TEST". CONTAINMENT INTEGRITY, AS DEFINED IN TECHNICAL SPECIFICATION 1.0.H.4, WAS BRIEFLY VIOLATED WHEN OPERATIONS PERSONNEL OPENED THE OUTER AIR LOCK DOOR IN ORDER TO EXIT THE CONTAINMENT FOLLOWING THE COMPLETION OF THE TEST. AT 0655 HOURS, A SATISFACTORY LEAKAGE TEST WAS PERFORMED ON THE OUTER AIR LOCK DOOR. CONTAINMENT INTEGRITY WAS AGAIN BRIEFLY VIOLATED WHEN MECHANICS OPENED THE OUTER DOOR TO GAIN ACCESS TO THE PERSONNEL AIR LOCK IN ORDER TO MAKE REPAIRS TO THE INNER DOOR. AT 0925 HOURS, UPON COMPLETION OF THE REPAIRS, A SATISFACTORY LEAKAGE TEST WAS PERFORMED ON BOTH THE INNER AND OUTER AIR LOCK DOORS. THIS EVENT WAS DUE TO THE FAILURE OF THE INNER DOOR OF THE PERSONNEL AIR LOCK TO SEAL TIGHTLY. THE LEAKAGE WAS ATTRIBUTED TO SPACING IN THE ACTUATOR MECHANISM WHICH DID NOT ALLOW FOR

PROPER SEATING OF THE ACTUATOR PISTON. A 1/2 INCH SHIM WAS WELDED TO THE ACTUATOR MECHANISM TO CORRECT THE SPACING BETWEEN THE PISTON AND THE SEATING SURFACE TO ENSURE PROPER SEALING OF THE DOOR.

[316] SURRY 1 DOCKET 50-280 LER 87-038
INCREASED OFF-SITE THYROID DOSE CALCULATIONS FROM STEAM GENERATOR TUBE RUPTURE DUE TO POST-TRIP STEAM GENERATOR TUBE UNCOVERY.
EVENT DATE: 120887 REPORT DATE: 010788 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 207634) ON DECEMBER 8, 1987, WITH UNIT 1 AT 100% POWER AND UNIT 2 IN COLD SHUTDOWN, IT WAS DETERMINED THAT THE UPPER PORTION OF THE STEAM GENERATOR TUBE BUNDLE MAY UNCOVER FOLLOWING A REACTOR TRIP. THE PREVIOUS ANALYSIS HAD ASSUMED THAT THE TUBE BUNDLE WOULD REMAIN COVERED. THIS CONDITION WAS DISCOVERED WHEN A REVIEW OF THE EXPECTED SURRY STATION STEAM GENERATOR POST-TRIP RESPONSE WAS PERFORMED AS A FOLLOW-UP ITEM TO THE JULY 15, 1987 NORTH ANNA STEAM GENERATOR TUBE RUPTURE EVENT. UNCOVERING THE BREAK LOCATION HAS THE EFFECT OF REDUCED IODINE PARTITIONING AND RESULTS IN INCREASED CALCULATED OFFSITE THYROID DOSE FOR THE POSTULATED STEAM GENERATOR TUBE RUPTURE EVENT. A RE-ANALYSIS OF THIS EVENT WAS MADE AND INDICATES THAT WHILE THE CALCULATED OFFSITE THYROID DOSES ARE INCREASED, THEY REMAIN WELL BELOW THE 10CFR100 LIMITS.

[317] SURRY 1 DOCKET 50-280 LER 87-039
PROTECTION SYSTEM CHANNEL INOPERABLE DUE TO FAILED SUMMATOR IN SIGNAL CONDITIONING CIRCUIT.
EVENT DATE: 121587 REPORT DATE: 011388 NSSS: WE TYPE: PWR
VENDOR: HAGAN CORPORATION

(NSIC 207884) ON DECEMBER 15, 1987, AT 1623 HOURS, WITH UNIT 1 AT 100% STEADY STATE POWER, THE UNIT 1 REACTOR COOLANT SYSTEM (RCS) (EIIS AB) 'C' LOOP AVERAGE TEMPERATURE (TAVG) PROTECTION CHANNEL SUMMATOR FAILED LOW. THIS FAILURE CAUSED THE CHANNEL 3 OVER-TEMPERATURE AND OVER-POWER DELTA TEMPERATURE (OT OP DELTA T) REACTOR TRIPS AND TURBINE RUNBACK SETPOINTS TO FAIL HIGH. THIS EVENT IS CONTRARY TO TECHNICAL SPECIFICATION TABLE 3.7-1 WHICH REQUIRES A MINIMUM DEGREE OF REDUNDANCY OF ONE (1) FOR THE OT OP DELTA T REACTOR TRIPS. IN RESPONSE TO THE INSTRUMENT FAILURE, OPERATORS VERIFIED THAT THE CONTROL CHANNELS FOR TAVG AND DELTA T AND 'A' AND 'B' LOOP PROTECTION CHANNELS TAVG AND DELTA T WERE NORMAL. IN ADDITION, THE AFFECTED PROTECTION CHANNELS WERE PLACED IN THE TRIP MODE AT 1640 HOURS. THE 'C' LOOP TAVG PROTECTION CHANNEL FAILED DUE TO A FAULTY SUMMATOR IN THE SIGNAL CONDITIONING CIRCUITRY. THE SUMMATOR FAILED DUE TO THE FAILURE OF ITS POWER SUPPLY. THE SUMMATOR WAS REPLACED AND THE AFFECTED PROTECTION CHANNELS WERE RETURNED TO NORMAL AT 1720 HOURS.

[318] SURRY 1 DOCKET 50-280 LER 87-040
REQUIRED STRONTIUM 89 AND 90 ANALYSES NOT PERFORMED DUE TO PERSONNEL ERROR.
EVENT DATE: 123187 REPORT DATE: 012988 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 208019) ON DECEMBER 31, 1987, WITH UNITS 1 AND 2 AT 100% POWER, IT WAS DISCOVERED DURING ROUTINE ACCOUNTABILITY RECORDS REVIEW THAT THE CONTAINMENT HOG PARTICULATE FILTERS FOR UNIT 1 (DATED DECEMBER 11, 1987) AND UNIT 2 (DATED DECEMBER 19, 1987) WERE UNINTENTIONALLY DISCARDED. THESE FILTERS WERE TO BE RETAINED FOR STRONTIUM (SR) 89 AND 90 ANALYSES IN ACCORDANCE WITH HEALTH PHYSICS (HP) PROCEDURE 3.2.1. BOTH THE HP PROCEDURE AND TECHNICAL SPECIFICATION 4.9, TABLE 4.9-2, REQUIRE QUARTERLY ANALYSIS OF CONTAINMENT HOGGING EFFFICIENTS FOR SR-89 AND SR-90 ACTIVITY. THESE ANALYSES ENSURE THAT GASEOUS RADIOACTIVE RELEASES ARE MAINTAINED AS LOW AS PRACTICABLE AND WITHIN LIMITS SET FORTH IN 10CFR20 AND 10CFR50, APPENDIX I. THE CONTAINMENT HOG ACCOUNTABILITY SR-89 AND

SR-90 RELEASES HAVE BEEN ESTIMATED FROM PREVIOUS DATA BY HP PERSONNEL; BASED ON THESE ESTIMATES, RELEASES FOR THIS QUARTER WERE WITHIN ESTABLISHED LIMITS. HP SUPERVISORY AND MANAGEMENT PERSONNEL WERE NOTIFIED OF THE MISSED ANALYSES AND WILL REVIEW THE REQUIREMENTS OF HP PROCEDURE 3.2.1 AND TECHNICAL SPECIFICATION TABLE 4.9-2 WITH HP PERSONNEL. IN ADDITION, THE PROCEDURE WILL BE REVISED TO PROVIDE ADDITIONAL CONTROL OVER FILTER RETENTION.

[319] SUSQUEHANNA 1 DOCKET 50-387 LER 87-036
ASSIGNMENT OF SRO WITH INACTIVE LICENSE STATUS RESULTS IN OPERATION PROHIBITED BY TECHNICAL SPECIFICATIONS.
EVENT DATE: 091887 REPORT DATE: 122387 NSSS: GE TYPE: BWR

(NSIC 207497) DURING THE PERIOD OF 0323 TO 1520 HOURS ON 9/18/87, ADMINISTRATIVE TECHNICAL SPECIFICATION 6.2.2.D WAS VIOLATED ON UNIT 1 RESULTING IN AN OPERATION PROHIBITED BY TECHNICAL SPECIFICATIONS. UNIT ONE WAS IN REFUELING. THE INDIVIDUALS ASSIGNED REFUELING FLOOR SENIOR REACTOR OPERATOR DUTIES WERE ASSISTANT UNIT SUPERVISORS (AUS) WHO DID NOT POSSESS ACTIVE LICENSES AS DESCRIBED IN 10CFR55.53. THE INDIVIDUALS WERE QUALIFIED AND LICENSED AS SROS BUT HAD NOT PERFORMED SRO DUTIES DURING THE PREVIOUS CALENDAR QUARTER; THEREFORE THE INDIVIDUALS' LICENSES WERE CONSIDERED AS INACTIVE. THUS THE INTENT OF TECHNICAL SPECIFICATION 6.2.2.D WAS NOT MET. CAUSE OF THE EVENT WAS INSUFFICIENT TRAINING AND ADMINISTRATIVE CONTROLS CONCERNING THE REQUIREMENTS FOR UPGRADING AN SRO LICENSE FROM AN INACTIVE TO AN ACTIVE STATUS. THE ASSIGNMENT OF TWO AUSS FROM THEIR NORMAL SHIFT DUTIES TO REFUELING DUTIES WAS CONSISTENT WITH PRACTICES ESTABLISHED DURING PRIOR REFUELING OUTAGES. THIS PRACTICE WAS ACCEPTABLE PRIOR TO THE REVISION TO 10CFR55 ON MARCH 31, 1987. THE AUSS INVOLVED WERE NOT AWARE OF THE REVISED REQUIREMENTS NOR WERE PROCEDURES ADEQUATE FOR IMPLEMENTING THE NEW REQUIREMENTS. THIS EVENT WAS REVIEWED WITH ALL LICENSED OPERATORS.

[320] SUSQUEHANNA 1 DOCKET 50-387 LER 87-031 REV 01
UPDATE ON UNANTICIPATED ESF ACTUATION DUE TO INSTALLATION OF JUMPER IN WRONG PANEL.
EVENT DATE: 110587 REPORT DATE: 123087 NSSS: GE TYPE: BWR

(NSIC 207724) AT 0338 ON 11-5-87, AN UNANTICIPATED ENGINEERED SAFEGUARDS FEATURE ACTUATION OCCURRED ON UNIT ONE WHICH WAS IN REFUELING. TESTING AND OPERATING PERSONNEL WERE PERFORMING SURVEILLANCE TEST SE-159-200, "18 MONTH LOGIC SYSTEM FUNCTIONAL TEST OF THE PRIMARY AND SECONDARY CONTAINMENT ISOLATION SYSTEM", WHEN THE "B" TRAINS OF STANDBY GAS TREATMENT SYSTEM (SGTS), CONTROL ROOM EMERGENCY OUTSIDE AIR SUPPLY SYSTEM (CREOASS), AND REACTOR BUILDING RECIRCULATION FANS UNEXPECTEDLY ACTUATED. THIS ACTUATION WAS CAUSED BY THE INSTALLATION OF A JUMPER IN THE WRONG PANEL. THE JUMPER WAS REMOVED AND AFFECTED SYSTEMS RESTORED. CAUSES OF THE EVENT WERE COGNITIVE PERSONNEL ERROR AND INCOMPLETE VERBAL COMMUNICATIONS. FOLLOWING NORMAL PRACTICE, THE TEST DIRECTOR VERBALLY INSTRUCTED THE ELECTRICIAN TO INSTALL THE JUMPER. THE TEST DIRECTOR SPECIFIED THE CORRECT TERMINAL POINTS TO THE ELECTRICIAN BUT FAILED TO SPECIFY THE PANEL. FURTHERMORE, THE TEST DIRECTOR FAILED TO RECOGNIZE THAT THE JUMPER WAS TO BE PLACED IN A PANEL DIFFERENT FROM THE PANEL USED IN THE PRECEDING TWENTY NINE STEPS OF THE TEST PROCEDURE. THE TEST DIRECTOR INVOLVED IN THE EVENT WAS COUNSELLED. THIS EVENT WILL BE REVIEWED WITH TEST DIRECTORS ON THE PLANT STAFF TECHNICAL SECTION STAFF ALONG WITH GUIDANCE ON PROPER VERBAL COMMUNICATIONS TECHNIQUES.

[321] SUSQUEHANNA 1 DOCKET 50-387 LER 87-035
PROCEDURAL INADEQUACY RESULTS IN AN INADVERTENT ENGINEERED SAFETY FEATURE (ESF) ACTUATION.
EVENT DATE: 111887 REPORT DATE: 121887 NSSS: GE TYPE: BWR

(NSIC 207496) AT 1511 ON NOVEMBER 18, 1987, AN UNPLANNED ENGINEERED SAFETY

FEATURE (ESF) ACTUATION OCCURRED ON UNIT ONE. UTILITY INSTRUMENTATION AND CONTROL (I&C) TECHNICIANS WERE OPERATING SURVEILLANCE TEST "18 MONTH TIME RESPONSE TEST OF RPS AND EOC/RPT TRIPS FOR TURBINE STOP VALVE AND TURBINE CONTROL VALVE FAST CLOSURE" (SI-183-413) WHEN A MAIN STEAM ISOLATION VALVE (MSIV) CLOSURE SIGNAL WAS GENERATED. THE MSIVS WERE CLOSED PRIOR TO AND AFTER THE CLOSURE SIGNAL WAS GENERATED. THE ACTUATION WAS THE RESULT OF LEADS LIFTED FROM THE PANEL SIDE RATHER THAN THE FIELD SIDE OF A TERMINAL BLOCK DURING THE SURVEILLANCE TEST. THE LEADS WERE RELANDED AND THE ACTUATION SIGNAL WAS RESET. THE SURVEILLANCE WAS REPERFORMED BY LIFTING THE FIELD SIDE LEADS AND WAS COMPLETED WITHOUT INCIDENT. THE CAUSE OF THE EVENT WAS A DEFICIENCY IN AN APPROVED PROCEDURE. SI-183-413 DID NOT SPECIFY LIFTING THE FIELD SIDE OR THE PANEL SIDE LEADS. THE EVENT WILL BE REVIEWED AT THE NEXT I&C MONTHLY SHOP MEETING. IN ADDITION, SI-183/283-413 WILL BE REVISED. AN EXISTING NOTE PRECEEDING THE LIST OF LEADS TO BE LIFTED WILL BE EXPANDED TO SPECIFY LIFTING LEADS ON THE FIELD SIDE ONLY. THE PREREQUISITES/LIMITATIONS WILL BE REVISED SO THAT THE SURVEILLANCE CAN ONLY BE PERFORMED IN PLANT CONDITIONS 4 AND 5 WITH THE MSIVS CLOSED.

[322] SUSQUEHANNA 1 DOCKET 50-387 LER 87-034
TECHNICIAN INADVERTENTLY CONNECTS TEST EQUIPMENT TO THE INCORRECT SYSTEM MODULE RESULTING IN AN ENGINEERED SAFETY FEATURE ACTUATION.
EVENT DATE: 112987 REPORT DATE: 122887 NSSS: GE TYPE: BWR

(NSIC 207560) ON NOVEMBER 29, 1987, AT APPROXIMATELY 1900 HOURS, THE HIGH PRESSURE COOLANT INJECTION (HPCI) INBOARD STEAM ISOLATION VALVE INADVERTENTLY CLOSED DURING SURVEILLANCE TESTING. AT THE TIME, I&C TECHNICIANS WERE PERFORMING SI-183-328 "QUARTERLY CALIBRATION OF MAIN STEAM LINE TUNNEL (TURBINE BLDG) TEMPERATURE CHANNELS TSH-B21-10100A, B, C, D." UNIT ONE WAS OPERATING NEAR 80% RATED POWER DURING THE SURVEILLANCE TESTING. THE TECHNICIAN IDENTIFIED THE CORRECT MODULE, B21-10100A, BY REFERRING TO A DRAWING OF THE MODULE'S LOCATION WHICH WAS ATTACHED TO THE CABINET DOOR AND BY LOCATING THE MODULE'S IDENTIFICATON TAG INSIDE THE CABINET. AFTER THE TECHNICIAN CONFIRMED THAT HE HAD CHOSE THE CORRECT MODULE, HE BENT DOWN TO PICK UP A SCREW DRIVER. THE TECHNICIAN THEN ATTEMPTED TO RELOCATE THE MODULE BY LOCATING THE TAG. HE SAW THE TAG AND PROCEEDED TO WORK ON THE MODULE LOCATED BELOW THE TAG. WHEN THE TECHNICIAN ADJUSTED THE THERMOCOUPLE CALIBRATOR, TO DETERMINE THE TRIP SETPOINT, THE HPCI STEAM SUPPLY INBOARD ISOLATION VALVE CLOSED. THE INVOLVED TECHNICIANS REVIEWED THE ERROR WITH THEIR SUPERVISOR. THE EVENT WILL ALSO BE REVIEWED BY ALL OTHER APPLICABLE I&C PERSONNEL ON AN UPCOMING SHOP MEETING.

[323] SUSQUEHANNA 1 DOCKET 50-387 LER 88-001
BOTH TRAINS OF CONTROL ROOM EMERGENCY AIR SUPPLY SYSTEM INOPERABLE DUE TO FAILURE OF LATCHING MECHANISM ON DOOR.
EVENT DATE: 010188 REPORT DATE: 012688 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)
VENDOR: CHICAGO BULLIT PROOF EQUIP CO.

(NSIC 207948) ON JANUARY 1, 1988 AT 2030 HOURS WITH UNIT 1 OPERATING AT 60% POWER AND UNIT 2 OPERATING AT 100% POWER, LIMITING CONDITION FOR OPERATION (LCO) 3.0.3 WAS ENTERED FOR BOTH UNITS WHEN THE LATCHING MECHANISM ON CONTROL STRUCTURE DOOR #455 FAILED. THIS DOOR IS PART OF THE PRESSURE BOUNDARY FOR THE CONTROL ROOM EMERGENCY OUTSIDE AIR SUPPLY SYSTEM (CREOASS: EIIS CODE: BH). AS A RESULT OF THE DOOR'S INOPERABILITY, BOTH CREOASS TRAINS WERE CONSIDERED INOPERABLE. AT 2100 HOURS ON JANUARY 1, 1988, THE LCO WAS CLEARED WHEN THE DOOR WAS SECURED CLOSED AND SUBSEQUENTLY REPAIRED. THE PLANT STAFF IS CONDUCTING A REVIEW OF THE CREOASS DESIGN REQUIREMENTS VERSUS EXISTING TECH SPEC REQUIREMENTS TO DETERMINE IF AN INOPERABLE BOUNDARY DOOR NECESSITATES ENTRY INTO LCO 3.0.3 WHEN OTHER POSITIVE CONTROLS FOR MAINTAINING THE CREOASS BOUNDARY ARE EMPLOYED.

[324] THREE MILE ISLAND 2 DOCKET 50-320 LER 87-012
FAILURE OF THE LOCKNUT AND STEMNUT FOR ISOLATION VALVE DH-V-4A.
EVENT DATE: 100987 REPORT DATE: 020388 NSSS: BW TYPE: PWR
VENDOR: LIMITORQUE CORP.
VELAN VALVE CORP.

(NSIC 208034) ON OCTOBER 9, 1987, A FLUSH OF THE DECAY HEAT REMOVAL SYSTEM PIPING WAS COMPLETED WHICH REQUIRED CONTAINMENT ISOLATION VALVE DH-V-4A TO BE CLOSED. FOLLOWING THE EVOLUTION AT 0230 HOURS ON OCTOBER 9, 1987, THE BREAKER FOR DH-V-4A WAS ENERGIZED. CONTROL ROOM OPERATORS NOTICED THAT THE VALVE POSITION INDICATOR WAS CYCLING OPEN AND CLOSED. THE BREAKER FOR DH-V-4A WAS DE-ENERGIZED AND AN UNIT WORK INSTRUCTION (UWI) WAS ISSUED TO REPAIR THE VALVE. ON OCTOBER 24, 1987, MAINTENANCE PERSONNEL ENERGIZED THE BREAKER FOR DH-V-4A FROM THE MOTOR CONTROL CENTER. SUBSEQUENT OPERATION OF THE VALVE INDICATED A PROBLEM WITH THE VALVE ASSEMBLY. ON OCTOBER 25, 1987, MAINTENANCE INSPECTED DH-V-4A AND OBSERVED THAT THE LOCKNUT WAS UNCOUPLED FROM THE DRIVE SLEEVE ASSEMBLY, WHICH WAS CRACKED. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE THE FAILURE TO STAKE THE LOCKNUT AND DRIVE SLEEVE ASSEMBLY RESULTING IN THE GEARED LIMIT SWITCH ALTERNATIVELY INDICATING OPEN AND CLOSED. THE LOCKNUT AND DRIVE SLEEVE ASSEMBLY WERE REPLACED AND THE LOCKNUT WAS STAKED TO THE DRIVE SLEEVE ASSEMBLY. DH-V--4A WAS RESTORED TO OPERABLE STATUS AT 1818 HOURS ON OCTOBER 30, 1987. ISOLATION VALVE DH-V-4B WAS INSPECTED AND OBSERVED TO BE IN A SATISFACTORY CONDITION.

[325] THREE MILE ISLAND 2 DOCKET 50-320 LER 87-011
FAILURE TO COMPLY WITH TECHNICAL SPECIFICATIONS 3.7.10.2 FOR HOURLY FIREWATCH DUE TO OPERATOR ERROR.
EVENT DATE: 112387 REPORT DATE: 122887 NSSS: BW TYPE: PWR

(NSIC 207527) AT 1040 HOURS ON 11/23/87, THE TECH SPECS 3.7.10.3 REQUIRED CABLE ROOM AND TRANSFORMER ROOM HALON SYSTEM WAS REMOVED FROM SERVICE TO PERFORM SURVEILLANCE 4224-SUR-3812.03, "FIRE SYSTEM HALON SYSTEM CHECK." SWITCHING ORDER NO. 13662 AUTHORIZED THE HALON SYSTEM MAIN/RESERVE BANK KEY SWITCH TO BE POSITIONED TO THE MAIN BANK. UPON TAKING THE HALON SYSTEM OUT-OF-SERVICE, THE UNIT ENTERED INTO THE ACTION STATEMENT OF TECH SPEC 3.7.10.3.A WHICH REQUIRED A ROVING HOURLY FIREWATCH. UPON COMPLETION OF THE SURVEILLANCE, THE UTILITY MAINTENANCE FOREMAN REPORTED TO THE SHIFT FOREMAN AND THE OPERATIONS MANAGER THAT THE SURVEILLANCE WAS COMPLETED AND THAT THE SYSTEM COULD BE RETURNED ON 11/23/87. NOTED THAT THE SURVEILLANCE WAS COMPLETED SATISFACTORILY, THE SYSTEM WAS RETURNED-TO-SERVICE, AND THE HOURLY FIREWATCH WAS SECURED. HOWEVER, SWITCHING ORDER NO. 13662 WAS NOT CLEARED AND THE MAIN/RESERVE BANK KEY SWITCH WAS NEVER REPOSITIONED. THIS CONDITION WAS DISCOVERED ON 11/28/87, DURING A WEEKLY AUDIT OF THE SWITCHING AND TAGGING LOG BY THE OPERATIONS DEPARTMENT. THE HALON SYSTEM WAS RETURNED-TO-SERVICE AT 1610 HOURS ON 11/28/87. THUS, FROM 1625 HOURS ON 11/23/87, TO 1610 HOURS ON 11/28/87, THE CABLE ROOM AND TRANSFORMER ROOM HALON SYSTEM WAS INOPERABLE WITHOUT THE TECH SPEC ACTION STATEMENT REQUIRED HOURLY FIREWATCH.

[326] TROJAN DOCKET 50-344 LER 87-032 REV 01
UPDATE ON RCS WIDE RANGE PRESSURE TRANSMITTER OUT-OF-CALIBRATION DUE TO INSTRUMENT DRIFT.
EVENT DATE: 041787 REPORT DATE: 010888 NSSS: WE TYPE: PWR
VENDOR: ITT-BARTON

(NSIC 207639) ON APRIL 17, 1987, REACTOR COOLANT SYSTEM (RCS) WIDE RANGE PRESSURE TRANSMITTERS PT-403 AND 405 WERE FOUND OUT-OF-CALIBRATION. THIS WOULD HAVE RESULTED IN AUTOMATIC ISOLATION OF THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM FROM THE RCS AT A PRESSURE HIGHER THAN THE 600 PSIG LIMIT IN THE TECH SPECS. IN ADDITION, THE RCS POWER-OPERATED RELIEF VALVES WOULD HAVE LIFTED AT PRESSURES HIGHER THAN THE VALUES ALLOWED IN THE TECH SPECS (440 PSIG AND 490 PSIG) DURING A

POSTULATED COLD OVERPRESSURE TRANSIENT. ON DECEMBER 10, 1987, PT-405 WAS AGAIN FOUND TO BE OUT-OF-CALIBRATION. THE CAUSE OF THIS EVENT WAS PRESSURE TRANSMITTER DRIFT. THESE TRANSMITTERS WERE INSTALLED IN 1986 AND THE VENDOR HAS INDICATED THAT NEW TRANSMITTERS CAN EXHIBIT GREATER DRIFT. THE IMMEDIATE CORRECTIVE ACTION WAS TO RECALIBRATE THE TRANSMITTERS. THE PRE-INSTALLATION ACCEPTANCE TEST ON NEW TRANSMITTERS OF THIS TYPE IS BEING EVALUATED FOR ADEQUACY. BECAUSE OF THE SIMILARITIES BETWEEN THIS EVENT AND THE OUT-OF-CALIBRATION DISCOVERED IN THE MAIN STEAM PRESSURE TRANSMITTERS MANUFACTURED BY THE SAME VENDOR (LER 87-33, 11-25-87), FURTHER EVALUATION OF BOTH OF THESE EVENTS IS CONTINUING. PT-403 AND 405 WILL BE CALIBRATION CHECKED THE NEXT TIME THE RCS PRESSURE IS REDUCED TO LESS THAN 700 PSIG.

[327] TROJAN DOCKET 50-344 LER 87-021 REV 01
UPDATE ON OPENING OF INCORRECT FUSE DRAWER CAUSED 12.47 KV BUS UNDERVOLTAGE CONDITION AND EDG STARTED.
EVENT DATE: 082187 REPORT DATE: 010888 NSSS: WE TYPE: PWR

(NSIC 207720) ON AUGUST 21, 1987, ACTIONS WERE IN PROGRESS TO MAKE 12.47 KV BUSES H1 AND H2 OPERATIONAL FROM THE UNIT AUXILIARY TRANSFORMER. AN OPERATOR HAD JUST COMPLETED INSTALLING POTENTIAL TRANSFORMER FUSES FOR BUSES H1 AND H2 WHEN HE OPENED THE STARTUP TRANSFORMER FUSE DRAWER BY MISTAKE. THIS CAUSED A SENSED UNDERVOLTAGE CONDITION, AND THE "A" EMERGENCY DIESEL GENERATOR STARTED AUTOMATICALLY. THE CAUSE WAS PERSONNEL ERROR. THE OPERATOR OPENED THE POTENTIAL TRANSFORMER FUSE DRAWER FOR THE STARTUP TRANSFORMER THINKING THAT THIS WAS THE DRAWER FOR THE MAIN GENERATOR POTENTIAL TRANSFORMER FUSES. A CONTRIBUTING CAUSE WAS A LACK OF A PROCEDURE FOR REPLACING THE POTENTIAL TRANSFORMER FUSES. THE OPERATOR WAS COUNSELED ON THE CORRECT METHOD FOR POTENTIAL TRANSFORMER FUSE REPLACEMENT. A WRITTEN PROCEDURE WILL BE PREPARED PROVIDING INSTRUCTIONS ON FUSE REPLACEMENT. THE ADMINISTRATION OF LOAD DISPATCHER SWITCHING ORDERS AND THE PRACTICES ON FUSE REPLACEMENT WILL BE REVIEWED FOR IMPROVEMENT. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[328] TROJAN DOCKET 50-344 LER 87-030
LOW CCW FLOW TO RHR HEAT EXCHANGER DUE TO PROCEDURE INADEQUACY.
EVENT DATE: 102187 REPORT DATE: 112087 NSSS: WE TYPE: PWR

(NSIC 207234) ON OCTOBER 31, 1987 DURING A TEMPORARY PLANT TEST, IT WAS DISCOVERED THAT THE COMPONENT COOLING WATER (CCW) FLOW TO THE "A" RESIDUAL HEAT REMOVAL (RHR) HEAT EXCHANGER, WAS ONLY 4460 GALLONS PER MINUTE (GPM). THE FINAL SAFETY ANALYSIS REPORT (FSAR) SPECIFIES A MINIMUM CCW FLOW TO EACH RHR HEAT EXCHANGER OF 5000 GPM. THE "B" RHR HEAT EXCHANGER CCW FLOW RATE WAS CONFIRMED TO BE GREATER THAN 5000 GPM. THE CAUSE OF THIS EVENT WAS PROCEDURE INADEQUACY. OPERATING INSTRUCTION (OI) 4-1, "RESIDUAL HEAT REMOVAL", PROVIDES INSTRUCTIONS FOR ADJUSTING THE RHR HEAT EXCHANGER CCW OUTLET VALVES. HOWEVER, THE VALVE ADJUSTMENTS, PERFORMED PER OI 4-1 WERE NOT DONE WITH THE CCW SYSTEM IN THE ACCIDENT CONFIGURATION. THE IMMEDIATE CORRECTIVE ACTION WAS TO ADJUST THE "A" RHR HEAT EXCHANGER CCW OUTLET VALVE TO PROVIDE 5100 GPM. OI 4-1 WILL BE REVISED TO ENSURE THAT THROTTLING OF THE RHR HEAT EXCHANGER CCW OUTLET VALVES PROVIDES ADEQUATE FLOW TO THE RHR HEAT EXCHANGERS WHILE IN THE ACCIDENT CONFIGURATION. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[329] TROJAN DOCKET 50-344 LER 87-036
SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE MISSED DUE TO PERSONNEL ERROR.
EVENT DATE: 120287 REPORT DATE: 123187 NSSS: WE TYPE: PWR

(NSIC 207530) ON DECEMBER 2, 1987, DURING A REVIEW OF SURVEILLANCE RECORDS, IT WAS DETERMINED THAT THE MONTHLY CHANNEL CHECK OF THE SEISMIC MONITORING INSTRUMENTATION REQUIRED BY TECH SPEC 4.3.3.3.1 WAS NOT PERFORMED IN NOVEMBER

1987. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THE SURVEILLANCE WAS DEFERRED DUE TO EXCAVATION OCCURRING IN THE VICINITY OF THE SEISMIC SENSOR. FOLLOWING DEFERRAL OF THE SURVEILLANCE, PERSONNEL FAILED TO ENSURE THAT THE SURVEILLANCE WAS PERFORMED WITHIN THE TECH SPEC ALLOWED EXTENSION OF 1.25 TIMES THE NORMAL SURVEILLANCE INTERVAL. THIS WAS ATTRIBUTED TO AN INADEQUATE TRACKING SYSTEM FOR SURVEILLANCES WHICH ARE INTENTIONALLY DEFERRED BEYOND THEIR NORMAL SCHEDULED COMPLETION DATE. THE REQUIRED SURVEILLANCE WAS PERFORMED AND THE SEISMIC MONITORING INSTRUMENTATION WAS VERIFIED AS OPERABLE. AN ALTERNATE MEANS FOR TRACKING SURVEILLANCES WHICH HAVE BEEN DEFERRED WILL BE DEVELOPED. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[330] TROJAN DOCKET 50-344 LER 87-037
EHC SWITCH FAILURE CAUSED LOAD REJECTION AND REACTOR TRIPPED MANUALLY.
EVENT DATE: 120687 REPORT DATE: 010588 NSSS: WE TYPE: PWR
VENDOR: COPES-VULCAN, INC.
MASTER SPECIALTIES

(NSIC 207640) ON DECEMBER 6, 1987, THE REACTOR WAS MANUALLY TRIPPED FOLLOWING A LOAD REJECTION TRANSIENT. ABOUT TEN MINUTES AFTER THE REACTOR TRIPPED, IT WAS REPORTED TO THE CONTROL ROOM THAT THE TURBINE-DRIVEN AUXILIARY FEEDWATER (AFW) PUMP HAD NOT AUTOMATICALLY STARTED. THE PUMP WAS MANUALLY STARTED. STEAM GENERATOR BLOWDOWN ISOLATION VALVE MO-2808 FAILED TO INDICATE COMPLETELY CLOSED FOLLOWING THE MAIN TURBINE TRIP. THE UPSTREAM ISOLATION VALVE CLOSED AS DESIGNED. THE CAUSE OF THE LOAD REJECTION TRANSIENT WAS A FAILURE OF THE LOAD DECREASE PUSH BUTTON IN THE ELECTRO HYDRAULIC CONTROL (EHC) SYSTEM. THE FAILURE OF THE TURBINE-DRIVEN AFW PUMP TO AUTO-START WAS DUE TO A LOOSE ELECTRICAL CONNECTION IN THE AUTO-START CIRCUITRY. THE FAILURE OF VALVE MO-2808 TO INDICATE FULLY CLOSED WAS DUE TO THE THREADED VALVE SEATING BACKING OUT AND PREVENTING COMPLETE CLOSURE INDICATION FOR THE VALVE. THE PLANT WAS PLACED IN MODE 3 (HOT STANDBY). THE EHC LOAD DECREASE PUSH BUTTON WAS REPLACED. THE LOOSE CONNECTION IN THE TURBINE-DRIVEN AFW PUMP AUTO-START CIRCUITRY WAS TIGHTENED. OTHER ACCESSIBLE TERMINAL CONNECTIONS WERE CHECKED FOR TIGHTNESS, AND NO DISCREPANCIES WERE FOUND. BLOWDOWN VALVE MO-2808 WAS REPAIRED AND TESTED SATISFACTORILY. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[331] TURKEY POINT 3 DOCKET 50-250 LER 87-031
CONTROL ROOM VENTILATION ISOLATION DUE TO LOSS OF POWER TO CONTAINMENT PARTICULATE MONITOR CAUSED BY GROUND IN BLOWDOWN EFFLUENT RADIATION MONITOR.
EVENT DATE: 120987 REPORT DATE: 010888 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)

(NSIC 207632) ON DECEMBER 9, 1987, WITH UNIT 3 IN MODE 5 AND UNIT 4 IN MODE 1 AT 100% POWER, THE BLOWDOWN EFFLUENT RADIATION MONITOR (R-19) WAS OUT OF SERVICE. MAINTENANCE TO RETURN R-19 TO SERVICE WAS INITIATED. UPON REMOVAL OF POWER FROM R-19, AS THE TECHNICIAN PERFORMING THE MAINTENANCE BEGAN TO REMOVE R-19'S RATEMETER COVER, BREAKER 3P08 (WHICH SUPPLIES POWER TO THE PROCESS RADIATION MONITOR SYSTEM (PRMS) RACK 66) TRIPPED. BECAUSE THE CONTAINMENT RADIOACTIVE GAS MONITOR (R-12) IS ALSO POWERED FROM BREAKER 3P08, IT ALSO LOST POWER. R-12 IS DESIGNED TO ACTUATE CONTAINMENT AND CONTROL ROOM VENTILATION ISOLATION UPON A LOSS OF POWER. AT 0822, THE CONTROL ROOM VENTILATION AND THE CONTAINMENT VENTILATION SYSTEM COMPONENTS ISOLATED. FOLLOWING THE ALARMS, THE AUTOMATIC ACTIONS DUE TO R-12 ACTUATING WERE VERIFIED TO HAVE OCCURRED AND WORK ON R-19 WAS HALTED. BREAKER 3P08 WAS THEN RECLOSED, HOWEVER IT TRIPPED AGAIN. PRMS RACK 66 WAS CLEANED AND INSPECTED. THE INSPECTION REVEALED A SMALL STRAND OF BARE WIRE ON AN INTERNAL RIBBON CONNECTOR IN R-19 WHICH PROBABLY CAUSED THE POWER SUPPLY TO SHORT TO GROUND AS THE DRAWER WAS BEING WORKED ON. THE PROPER ACTUATION OF THE COMPONENTS REQUIRED TO ISOLATE THE CONTROL ROOM AND THE CONTAINMENT WAS VERIFIED.

[332] TURKEY POINT 3 DOCKET 50-250 LER 87-032
CONTROL ROOM VENTILATION ISOLATION UPON CONTAINMENT RADIOACTIVE PARTICULATE MONITOR ACTUATION DUE TO FAULTY REMOTE READOUT ON BLOWDOWN EFFLUENT RADIATION MONITOR.
EVENT DATE: 121787 REPORT DATE: 011988 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)

(NSIC 207953) ON DECEMBER 17, 1987, WITH UNIT 3 IN MODE 5 AND UNIT 4 IN MODE 1 AT 94% POWER, THE UNIT 3 BLOWDOWN EFFLUENT RADIATION MONITOR (R-19) WAS UNDERGOING PERIODIC SURVEILLANCE. R-19'S HIGH ALARM WAS INITIATED. COINCIDENT WITH THIS, BREAKER 3P08-19 (WHICH SUPPLIES POWER TO PROCESS RADIATION MONITOR SYSTEM (PRMS) RACK 66) TRIPPED. BECAUSE THE CONTAINMENT RADIOACTIVE PARTICULATE MONITOR (R-11) IS ALSO POWERED FROM BREAKER 3P08-19, IT ALSO LOST POWER. R-11 IS DESIGNED TO ACTUATE CONTAINMENT AND CONTROL CONTROL ROOM VENTILATION ISOLATION UPON A LOSS OF POWER. AT 0421, THE CONTROL ROOM VENTILATION AND THE CONTAINMENT VENTILATION SYSTEM COMPONENTS ISOLATED FOLLOWING THE TRIP OF BREAKER 3P08-19. TROUBLESHOOTING DISCOVERED THAT BREAKER 3P08-19 WAS TRIPPING DUE TO A GROUND IN THE FIELD WIRING, WHICH OCCURRED WHEN ONE OF THE RELAYS IN R-19 WAS DEENERGIZED. THE CAUSE OF BREAKER 3P08-19 TRIPPING WAS DETERMINED TO BE A GROUND CAUSED BY CORROSION FOUND ON A REMOTE BUZZER COIL AND LIGHT SOCKET. R-19 WAS RECENTLY REPLACED WITH A NEW DRAWER OF DIFFERENT DESIGN. AFTER EVALUATING THE PURPOSE OF THE REMOTE METER, BUZZER, AND LIGHT, IT WAS DETERMINED THAT THEY WERE NOT REQUIRED. A CHANGE REQUEST LIFTING THE POWER LEADS TO THE REMOTE METER AND ELIMINATING THE GROUND WAS GENERATED.

[333] TURKEY POINT 3 DOCKET 50-250 LER 87-033
REACTOR TRIP OCCURS DURING CONTROLLED SHUTDOWN WHEN SOURCE RANGE HIGH NEUTRON FLUX TRIP UNBLOCKED AND A DETECTOR OUT OF SERVICE WITHOUT BYPASSING THE TRIP SIGNAL.
EVENT DATE: 122587 REPORT DATE: 012288 NSSS: WE TYPE: PWR
VENDOR: COPES-VULCAN, INC.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 207954) ON DECEMBER 25, 1987, UNIT 3 EXPERIENCED A SUBCRITICAL REACTOR TRIP. UNIT 3 WAS IN THE PROCESS OF A CONTROLLED SHUTDOWN DUE TO PRESSURIZER PRESSURE CONTROL PROBLEMS. THE UNIT WAS TAKEN OFF THE LINE AND THE OPERATORS PROCEEDED TO FULLY INSERT THE CONTROL RODS. WHEN THE INDICATION ON BOTH INTERMEDIATE RANGE DETECTORS DECREASED TO BELOW 10 E-10 AMPS, THE SOURCE RANGE DETECTORS (N-31 AND N-32) ENERGIZED AND BEGAN INDICATING IN COUNTS PER SECOND (CPS). N-31 HAD BEEN TAKEN OUT OF SERVICE FOR A PREVIOUS FAILURE AND THE INSTRUMENT FUSES IN THE DRAWER HAD BEEN REMOVED, BUT THE LEVEL TRIP BYPASS SWITCH HAD BEEN LEFT IN NORMAL POSITION. THIS PLACES THE CHANNEL IN THE TRIPPED CONDITION AND LOCKS IN THE INPUT TO THE REACTOR TRIP LOGIC. THEREFORE, WHEN THE SOURCE RANGE DETECTORS REENERGIZED, THE REACTOR TRIP LOGIC FOR A SOURCE RANGE HIGH FLUX REACTOR TRIP AT SHUTDOWN WAS FULFILLED. THE REACTOR TRIP OCCURRED AS DESIGNED AND THE UNIT WAS STABILIZED IN MODE 3 (HOT STANDBY). A CONTRIBUTING FACTOR WAS THAT THE OFF NORMAL OPERATING PROCEDURE DID NOT ADDRESS THE FAILURE OF A SOURCE RANGE DETECTOR IN MODE 1 (POWER OPERATION). APPROPRIATE PROCEDURE CHANGES WILL BE MADE AND A POST TRIP REVIEW WAS PERFORMED TO ASSURE PROPER OPERATION OF SAFETY RELATED EQUIPMENT.

[334] TURKEY POINT 4 DOCKET 50-251 LER 87-027
STEAM GENERATOR BLOWDOWN FLOWRATE NOT ESTIMATED DUE TO MISUNDERSTANDING OF TECHNICAL SPECIFICATION REQUIREMENTS.
EVENT DATE: 121487 REPORT DATE: 011288 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 3 (PWR)
VENDOR: FOXBORO CO., THE

(NSIC 207876) ON DECEMBER 14 AT 1930 IT WAS DETERMINED THAT STEAM GENERATOR (SG)

4B BLOWDOWN EFFLUENT FLOWRATE WAS NOT BEING ESTIMATED, CONTRARY TO THE REQUIREMENTS OF TECHNICAL SPECIFICATION (TS) TABLE 3.9-2 ITEM 2.B. AT THE TIME OF THE EVENT, FLOW TRANSMITTER FT-6277B WAS OUT OF SERVICE. TS TABLE 3.9-2 ITEM 2.B REQUIRES 1 FLOW RATE MEASUREMENT DEVICE TO BE OPERABLE DURING BLOWDOWN OPERATIONS. THE ACTION STATEMENT STATES THAT EFFLUENT RELEASES MAY CONTINUE PROVIDED THE FLOW RATE IS ESTIMATED AT LEAST ONCE PER 4 HOURS. AFTER CONCLUDING THAT TS REQUIREMENTS WERE NOT BEING MET, SG BLOWDOWN WAS IMMEDIATELY SECURED. FOLLOWING REPAIR, FT- 6277B WAS RETURNED TO SERVICE AT 0210 ON DECEMBER 17, 1987. THE CAUSE OF THE EVENT WAS INADEQUATE IDENTIFICATION OF THE SPECIFIC HARDWARE WHICH WAS SUBJECT TO TS TABLE 3.9-2, ITEM 2.B. IT WAS BELIEVED THAT THIS TS APPLIED TO THE FLOW RATE MONITORS ASSOCIATED WITH R-19 AS THIS FLOWPATH DISCHARGES DIRECTLY TO THE DISCHARGE CANAL. A LETTER DEFINING THE METHODS OF COMPLIANCE WITH TS TABLE 3.9-2 ITEM 2.B WAS ISSUED. PROPER BLOWDOWN FLOW ESTIMATION PROCEDURES WILL BE DEVELOPED AND IMPLEMENTED.

[335] TURKEY POINT 4 DOCKET 50-251 LER 87-028 REV 01
UPDATE ON UNIT SHUTDOWN COMMENCED WHEN TWO INTAKE COOLING WATER PUMPS WERE DECLARED INOPERABLE DUE TO A BROKEN SHAFT COUPLING AND A PACKING BOX BEARING FAILURE.
EVENT DATE: 121887 REPORT DATE: 021088 NSSS: WE TYPE: PWR
VENDOR: JOHNSTON PUMP CO.

(NSIC 208014) ON 12/18/87, WHILE UNIT 4 WAS AT 100% POWER, TWO INTAKE COOLING WATER (ICW) PUMPS WERE DECLARED OUT OF SERVICE WHICH EXCEEDED THE LIMITS OF TECH SPEC 3.4.5. ON 12/18/87 THE ANNUNCIATOR FOR ICW LOW PRESSURE ALARMED. THE OPERATORS NOTICED THAT THE 4C ICW PUMP WAS INDICATING LOW AMPS. AN IMMEDIATE ATTEMPT WAS MADE TO START THE 4B ICW PUMP BUT THE PUMP WOULD NOT OPERATE. ANOTHER ATTEMPT WAS MADE TO START THE 4B ICW PUMP WITH THE 4C ICW PUMP ISOLATED BUT THIS WAS UNSUCCESSFUL. BOTH PUMPS WERE DECLARED OUT OF SERVICE. TS 3.4.5 REQUIRES THREE (3) ICW PUMPS TO BE OPERABLE IN MODE 1 AND THE ACTION STATEMENT ONLY ALLOWS ONE (1) ICW PUMP TO BE INOPERABLE FOR UP TO 24 HOURS. THEREFORE, 2 ICW PUMPS OUT OF SERVICE EXCEEDS THE REQUIREMENTS OF TS 3.4.5 REQUIRING THE UNIT TO BE SHUTDOWN AND PLACED IN HOT STANDBY WITHIN 7 HOURS. A SHUTDOWN OF UNIT 4 WAS COMMENCED AND AN UNUSUAL EVENT WAS DECLARED IN ACCORDANCE WITH THE APPLICABLE TURKEY POINT EMERGENCY PROCEDURES. AN EVENT RESPONSE TEAM WAS FORMED TO DETERMINE ROOT CAUSE OF THE FAILURES AND PROVIDE CORRECTIVE ACTIONS. AT THIS TIME DISCRETIONARY ENFORCEMENT WAS REQUESTED AND GRANTED BY NRC REGION II TO ALLOW 24 HOURS OF OPERATION TO REPAIR BOTH PUMPS. BOTH PUMPS WERE REPAIRED AND RETURNED TO SERVICE BEFORE THE END OF THE 24 HOUR PERIOD.

[336] VERMONT YANKEE DOCKET 50-271 LER 87-019
MISSED SURVEILLANCE OF LPCI REACTOR VESSEL SHROUD LEVEL PERMISSIVE DUE TO PERSONNEL ERROR.
EVENT DATE: 121587 REPORT DATE: 011488 NSSS: GE TYPE: BWR

(NSIC 207939) ON 12/15/87, DURING NORMAL OPERATION AT 99% POWER, IT WAS DISCOVERED THAT SURVEILLANCE TESTING OF THE LOW PRESSURE COOLANT INJECTION (LPCI) REACTOR VESSEL SHROUD LEVEL PERMISSIVE FOR CONTAINMENT SPRAY (EIIS IDENTIFIER = 69) HAD NOT BEEN COMPLETED AS SPECIFIED BY TECH SPEC TABLE 4.2.1. THE MISSED SURVEILLANCE TESTS ARE ATTRIBUTED TO ERRORS MADE IN UPDATING THE MONTHLY SURVEILLANCE TESTING SCHEDULES. SUBSEQUENT INSPECTION AND TESTING WAS COMPLETED ON 12-15-87 AND DETERMINED THAT THE INSTRUMENTATION WAS FUNCTIONING PROPERLY, AND HAD REMAINED OPERABLE AT ALL TIMES. THE SURVEILLANCE TEST PROGRAM WAS REVIEWED, AND ALL ASSOCIATED PERSONNEL HAVE BEEN INSTRUCTED ON THE IMPORTANCE OF ACCURATE REVISION PROCESSES.

[337] VOGTLE 1 DOCKET 50-424 LER 87-025 REV 01
UPDATE ON REACTOR TRIP DUE TO STARTUP TEST PROCEDURE INADEQUACY.
EVENT DATE: 050987 REPORT DATE: 021088 NSSS: SS TYPE: PWR

(NSIC 207982) ON MAY 9, 1987 AT 0332 CDT WHILE IN MODE 1 AT 80% REACTOR POWER, AN AUTOMATIC REACTOR TRIP OCCURRED WHEN STEAM GENERATOR (SG) #4 WATER LEVEL REACHED THE LOW-LOW LEVEL TRIP SETPOINT. THE PLANT WAS CONDUCTING STARTUP TEST PROCEDURE 1-6SC-02 "LOAD SWING TEST" AND WAS PERFORMING THE 10% STEP LOAD INCREASE PART OF THE TEST WHEN THE TRIP OCCURRED. A MAIN TURBINE LOAD LIMITER POTENTIOMETER WAS INCORRECTLY SET AND ALLOWED TURBINE LOADING TO OVERSHOOT TO 82% REACTOR POWER DURING THE LOAD INCREASE. WITH A SINGLE MAIN FEEDWATER (MFW) PUMP OPERATING IN AUTOMATIC, THE OPERATOR COULD NOT MAINTAIN STEAM GENERATOR WATER LEVEL IN THE OPERATING BAND. THIS RESULTED IN A REACTOR TRIP ON STEAM GENERATOR LOW-LOW WATER LEVEL. THE TRIP WAS CAUSED BY PROCEDURAL INADEQUACY. THE TEST PROCEDURE DID NOT CONTAIN THE NECESSARY INSTRUCTIONS TO CORRECTLY ADJUST THE TURBINE LOAD LIMITER POTENTIOMETER AND DID NOT CONTAIN INSTRUCTIONS TO THE OPERATORS CONCERNING CORRECTIVE ACTIONS FOR AN OVERSHOOT CONDITION. CORRECTIVE ACTIONS WERE TAKEN TO CLARIFY THE TEST PROCEDURE TO ENSURE THAT THE MAIN TURBINE LOAD LIMITER IS CORRECTLY ADJUSTED AND APPROPRIATE OPERATOR ACTIONS ARE TAKEN.

[338] VOGTLE 1 DOCKET 50-424 LER 87-058 REV 01
UPDATE ON FALSE SIGNAL FROM A RADIATION MONITOR LEADS TO CONTROL ROOM ISOLATION.
EVENT DATE: 092187 REPORT DATE: 020488 NSSS: SS TYPE: PWR
VENDOR: AMPEREX ELECTRONIC CORP.

(NSIC 207983) ON SEPTEMBER 21, 1987, UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT 100% RATED THERMAL POWER. AT 0204 CDT, A CONTROL ROOM ISOLATION (EMERGENCY MODE) (CRI) OCCURRED AS A RESULT OF RADIATION MONITOR 1RE-12116 SPURIOUSLY OR FALSELY GENERATING A HIGH RADIATION SIGNAL. ANOTHER CONTROL ROOM AIR INTAKE RADIATION MONITOR, 1RE-12117, WAS CHECKED, AND DISPLAYED NORMAL READINGS. AIR SAMPLES WERE TAKEN WHICH VERIFIED THAT HIGH AIRBORNE ACTIVITY DID NOT EXIST. BY 0328 CDT, CONTROL ROOM ISOLATION WAS DISCONTINUED AND NORMAL HVAC OPERATION RESUMED. THE CAUSE OF THIS EVENT WAS A FALSE HIGH RADIATION SIGNAL RECEIVED FROM RADIATION MONITOR 1RE-12116. SUBSEQUENT EVENTS (SEE LER 50-424/1987- 073) HAVE LED TO THE CONCLUSION THAT THIS EVENT WAS CAUSED BY A COMBINATION OF A FAULTY SENSING TUBE FOR THE RADIATION MONITOR AND THE COMPUTER SOFTWARE DESIGN, I.E., THE STATISTICAL COUNTING ALGORITHM WITHIN THE DATA PROCESSING MODULE (DPM) FOR THE MONITOR. CORRECTIVE ACTION HAS INCLUDED REPLACEMENT OF THE TUBE AND A PROPOSED SOFTWARE CHANGE TO PROVIDE A MORE STATISTICALLY ACCURATE HIGH ALARM.

[339] VOGTLE 1 DOCKET 50-424 LER 87-074
TECHNICAL SPECIFICATION VIOLATION WHEN CORE EXIT THERMOCOUPLES NOT DECLARED INOPERABLE.
EVENT DATE: 100687 REPORT DATE: 011788 NSSS: SS TYPE: PWR

(NSIC 207919) ON APRIL 13, 1987, A MAINTENANCE WORK ORDER (MWO) WAS WRITTEN WHICH IDENTIFIED THAT DOCUMENTATION COULD NOT BE LOCATED ADDRESSING THE INSTALLATION OF THE "O" RINGS FOR THE CORE EXIT THERMOCOUPLES JUNCTION BOXES (TRAIN "A" AND TRAIN "B"). ON OCTOBER 6, 1987, IT WAS IDENTIFIED THAT THE MWO HAD NOT BEEN CLOSED, AND A DEFICIENCY CARD (DC) WAS WRITTEN WHICH SPECIFIED THE "O" RINGS WERE NOT INSTALLED AND THE COVERS NOT PROPERLY SECURED. THE DC WAS PROCESSED WITHOUT IDENTIFYING THE THERMOCOUPLES AS BEING INOPERABLE BY THE SHIFT SUPERVISOR (SS). DURING THE SUBSEQUENT OCTOBER 1987 OUTAGE, THE BOXES WERE PROPERLY SEALED AND RETURNED TO OPERABLE STATUS ON OCTOBER 18, 1987. THIS EVENT OCCURRED BECAUSE THE REFERENCE JUNCTION BOXES WERE NOT PROPERLY SEALED PRIOR TO REACTOR OPERATION, AND THEY WERE NEVER IDENTIFIED AS AN OPEN ITEM PRIOR TO TURNOVER TO OPERATIONS. SUBSEQUENTLY, THE JUNCTION BOXES WERE IDENTIFIED AS A DEFICIENCY, BUT AN APPROPRIATE OPERABILITY REVIEW WAS NOT PERFORMED IN A TIMELY MANNER. CORRECTIVE ACTIONS TAKEN INCLUDE SCHEDULED TRAINING, "COMMITMENT TO SAFETY", WHICH INCLUDES

A DISCUSSION RELATED TO OPERABILITY DETERMINATION, FOR APPROPRIATE NUCLEAR OPERATIONS PERSONNEL.

[340] VOGTLE 1 DOCKET 50-424 LER 87-071
MISCOMMUNICATION CAUSES INADEQUATE ANALYSIS OF UNIT 1 DIESEL FUEL OIL.
EVENT DATE: 110987 REPORT DATE: 010488 NSSS: SS TYPE: PWR

(NSIC 207699) ON NOVEMBER 9, 1987, DIESEL FUEL OIL WAS RECEIVED THAT WAS INTENDED FOR PLANT VOGTLE - UNIT 2. SATISFACTORY RESULTS WERE OBTAINED FROM A SAMPLE THAT WAS TAKEN AND TESTED IN THE CHEMISTRY LAB. A UNIT 2 SHIFT SUPERVISOR (SS) TRANSFERRED APPROXIMATELY 6200 GALLONS OF THE SHIPMENT INTO THE UNIT 1 STORAGE TANKS. ON DECEMBER 4, 1987, A CHEMISTRY SUPERVISOR, WHILE REVIEWING DIESEL FUEL OIL DOCUMENTATION, REALIZED THAT FUEL OIL WAS ADDED TO THE UNIT 1 STORAGE TANKS WITHOUT MEETING ALL UNIT 1 TECHNICAL SPECIFICATION (TS) ANALYSIS REQUIREMENTS (I.E., ONE TEST REMAINED TO BE PERFORMED). A SAMPLE OF THIS SHIPMENT WAS TESTED AND RESULTS INDICATED THAT ALL UNIT 1 TS REQUIREMENTS HAD BEEN SATISFIED. THIS EVENT WAS CAUSED BY A MISCOMMUNICATION BETWEEN THE UNIT 1 CHEMISTRY FOREMAN AND THE UNIT 2 SS. THE UNIT 2 SS ASSUMED THE FUEL OIL WAS SAMPLED PER UNIT 1 TS REQUIREMENTS. CORRECTIVE ACTIONS TAKEN INCLUDE REVISING THE ANALYSIS PROCEDURE FOR UNIT 2, COUNSELING OF THE SS INVOLVED AND ISSUING A MEMO TO ALL SS'S TO ENSURE TS ANALYSIS REQUIREMENTS ARE SATISFIED PRIOR TO ADDITION OF NEW FUEL TO THE UNIT 1 STORAGE TANKS.

[341] VOGTLE 1 DOCKET 50-424 LER 87-068
CONTROL ROOM ISOLATION DUE TO FAULTY SENSING TUBE AND SOFTWARE DESIGN.
EVENT DATE: 111787 REPORT DATE: 121787 NSSS: SS TYPE: PWR
VENDOR: AMPEREX ELECTRONIC CORP.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 207485) AT 0731 CST ON NOVEMBER 17, 1987 UNIT 1 WAS OPERATING AT 89 PERCENT RATED THERMAL POWER WHEN A HIGH LEVEL ALARM WAS RECEIVED FROM CONTROL ROOM AIR INTAKE RADIOGAS MONITOR, 1RE-12116, INITIATING A CONTROL ROOM VENTILATION ISOLATION (CRI) ACTUATION. THIS EVENT WAS CAUSED BY A COMBINATION OF A FAULTY MONITOR SENSING TUBE AND THE COMPUTER SOFTWARE (I.E., STATISTICAL COUNTING ALGORITHM) DESIGN WHICH INTERPRETS THE INPUT DATA TO PROVIDE AN ANTICIPATORY HIGH LEVEL ALARM. CORRECTIVE ACTION HAS INCLUDED REPLACEMENT OF THE TUBE AND A PROPOSED SOFTWARE CHANGE TO PROVIDE A MORE REALISTIC ANTICIPATORY HIGH LEVEL ALARM.

[342] VOGTLE 1 DOCKET 50-424 LER 87-069
OPERATING ABOVE THE MAXIMUM POWER LEVEL SPECIFIED IN OPERATING LICENSE.
EVENT DATE: 112087 REPORT DATE: 010588 NSSS: SS TYPE: PWR

(NSIC 207698) ON SEPTEMBER 9, 1987, DURING A REVIEW OF THE HAND CALCULATION AND THE PROTEUS COMPUTER CALORIMETRIC METHODS FOR DETERMINING REACTOR THERMAL POWER, A POSSIBLE VIOLATION OF LICENSE CONDITION 2.C.(1) OF THE PLANT VOGTLE UNIT 1 FACILITY OPERATING LICENSE NO. NPF-68 WAS DISCOVERED. A REVIEW OF PLANT OPERATING DATA WAS UNDERTAKEN. BY NOVEMBER 20, 1987, INVESTIGATION DETERMINED THAT THE MAXIMUM LICENSED POWER LEVEL HAD BEEN EXCEEDED ON NUMEROUS OCCASIONS. SUBSEQUENT REVIEW DETECTED ONE INSTANTANEOUS DATA POINT WHERE THE RATED THERMAL POWER MAY HAVE REACHED A MAXIMUM VALUE OF 3484 MWT (APPROXIMATELY 102.1% RATED THERMAL POWER). CORRECTIVE ACTIONS WERE TAKEN TO REVISE PROCEDURES SUCH THAT CALORIMETRICS FOR NI CALIBRATION ARE PERFORMED UNDER MORE STABILIZED PLANT CONDITIONS AND AT INCREASED FREQUENCIES. THIS INVESTIGATION IS ONGOING. A SUPPLEMENTAL REPORT IS SCHEDULED TO BE SUBMITTED ON OR ABOUT FEBRUARY 12, 1988.

[343] VOGTLE 1 DOCKET 50-424 LER 87-072
INADEQUATE TRAINING CAUSES A SURVEILLANCE TO BE IMPROPERLY PERFORMED.
EVENT DATE: 112187 REPORT DATE: 010888 NSSS: SS TYPE: PWR

(NSIC 207917) ON DECEMBER 9, 1987, A TECHNICAL SPECIFICATION (TS) SURVEILLANCE FOR UNIT 1 WAS IDENTIFIED AS BEING INADEQUATELY PERFORMED ON NOVEMBER 21, 1987. THE AUXILIARY PLANT OPERATOR (APO) TAKING THE READINGS FROM THE PLANT SAFETY MONITORING SYSTEM (PSMS) DATA DISPLAYS FAILED TO OBTAIN THE CORRECT READINGS FOR THE REACTOR VESSEL LEVEL INDICATION SYSTEM (RVLIS) AND THE REACTOR COOLANT SYSTEM (RCS) SUBCOOLING CHANNELS. A REVIEW OF THE SURVEILLANCE DATA BY THE UNIT 1 SHIFT SUPERVISOR (SS) DID NOT IDENTIFY THESE READINGS AS BEING INCORRECT. THIS EVENT WAS CAUSED BY AN APO WHO FAILED TO OBTAIN THE CORRECT READINGS FROM THE DISPLAY SCREEN. THE APO HAD NOT RECEIVED ADEQUATE "HANDS-ON" TRAINING FOR THE PSMS CONSOLE AND WAS UNAWARE OF THE PROPER OPERATION OF THE CONSOLE AND THE EXPECTED VALUES OF THESE INSTRUMENT CHANNELS. ALSO, THERE ARE NO PSMS CONSOLE AND DISPLAY SCREENS ON THE PLANT SIMULATOR. CORRECTIVE ACTIONS INCLUDE A TRAINING DISCUSSION AND COUNSELING OF THE INDIVIDUALS INVOLVED. A PSMS CONSOLE AND DISPLAY SCREEN WILL BE PROCURED FOR THE PLANT SIMULATOR.

[344] VOGTLE 1 DOCKET 50-424 LER 87-076
PERSONNEL ERROR CAUSES LOSS OF MONITOR OPERABILITY RESULTING IN TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 112787 REPORT DATE: 012688 NSSS: SS TYPE: PWR

(NSIC 207984) ON NOVEMBER 26, 1987, WHILE IN MODE 1 (POWER OPERATIONS) AT 98 PERCENT RATED THERMAL POWER, PLANT PERSONNEL WERE PERFORMING TECHNICAL SPECIFICATION (TS) SURVEILLANCE TESTING ON THE TRAIN A CONTAINMENT HYDROGEN MONITOR. INSTRUMENT PANEL 1-1513-P5-HMA HAD BEEN UNBOLTED IN ORDER TO REACH TEST POINTS BEHIND THE PANEL. UPON COMPLETION OF THE TESTING THE PANEL WAS SHUT AND ONLY 1 OF THE 4 PANEL BOLTS WAS REINSTALLED. IT REMAINED INOPERABLE FOR LONGER THAN THE 7 DAY PERIOD ALLOWED BY THE TS LIMITING CONDITION FOR OPERATION (LCO). ON DECEMBER 27, 1987, PLANT PERSONNEL WERE AGAIN PERFORMING SURVEILLANCE TESTING WHEN THEY DISCOVERED THAT THE 3 PANEL BOLTS WERE NOT INSTALLED. UPON COMPLETION OF THE TESTING, THEY REPLACED THE MISSING BOLTS AND INFORMED THE CONTROL ROOM OF THE SITUATION. THE CAUSE OF THIS EVENT IS PERSONNEL ERROR. CORRECTIVE ACTION INCLUDES BRIEFING APPROPRIATE PERSONNEL ON CONTROL AND TEMPORARY STORAGE OF MATERIALS DURING IN-PROCESS WORK.

[345] VOGTLE 1 DOCKET 50-424 LER 87-073
CONTAINMENT VENTILATION ISOLATION DUE TO SENSING TUBE FAILURE AND SOFTWARE DESIGN.
EVENT DATE: 122187 REPORT DATE: 011588 NSSS: SS TYPE: PWR
VENDOR: AMPEREX ELECTRONIC CORP.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 207918) ON DECEMBER 21, 1987, AT 2223 CST, THE PLANT WAS OPERATING AT 99 PERCENT RATED THERMAL POWER. A HIGH LEVEL ALARM WAS RECEIVED FROM THE CONTROL ROOM AIR INTAKE RADIOGAS MONITOR, 1RE-12116, INITIATING A CONTROL ROOM VENTILATION ISOLATION (CRI) ACTUATION. THIS EVENT WAS CAUSED BY A COMBINATION OF A FAULTY SENSING TUBE FOR THE RADIATION MONITOR AND THE COMPUTER SOFTWARE DESIGN, I.E., THE STATISTICAL COUNTING ALGORITHM WITHIN THE DATA PROCESSING MODULE (DPM) FOR THE MONITOR, WHICH INTERPRETS THE INPUT DATA TO PROVIDE THE HIGH ALARM. CORRECTIVE ACTION HAS INCLUDED REPLACEMENT OF THE TUBE AND A PROPOSED SOFTWARE CHANGE TO PROVIDE A MORE STATISTICALLY ACCURATE HIGH ALARM.

[346] VOGTLE 1 DOCKET 50-424 LER 87-075
MISSING SCREWS IN THE NUCLEAR INSTRUMENTATION DRAWERS LEADS TO TECH SPEC 3.0.3. ENTRY.
EVENT DATE: 122287 REPORT DATE: 012188 NSSS: SS TYPE: PWR

(NSIC 207934) ON DECEMBER 22, 1987, AT 1406, WITH UNIT 1 AT APPROXIMATELY 99 PERCENT OF RATED THERMAL POWER (RTP), AN INSTRUMENTATION AND CONTROL (I&C) TECHNICIAN, WHILE TAKING TEST READINGS IN A POWER RANGE NUCLEAR INSTRUMENTATION (NI) DRAWER, IDENTIFIED THAT THE SCREWS WERE MISSING FROM A HOLD DOWN (COVER) PLATE ON A PRINTED CIRCUIT (PC) CARD RACK. THE PLATE FUNCTIONS AS A HOLD DOWN PLATE FOR THE CARD ASSEMBLIES TO AID IN THE RESTRICTION OF CARD MOVEMENT. ALL OTHER NI DRAWERS WERE CHECKED AND ALL CHANNELS OF SOURCE, INTERMEDIATE, AND POWER (EXCEPT CHANNEL I) RANGES WERE AFFECTED. THE NIS WERE DECLARED INOPERABLE AND THE PLANT ENTERED SECTION 3.0.3 OF TECHNICAL SPECIFICATIONS (TS). THE EXACT CIRCUMSTANCES BY WHICH THE SCREWS CAME TO BE MISSING IS NOT KNOWN. IT IS THOUGHT THAT THIS EVENT OCCURRED BECAUSE A FAILURE TO INITIALLY INSTALL THESE SCREWS WAS NOT DISCOVERED DURING THE CONSTRUCTION ACCEPTANCE TEST PROGRAM NOR DURING THE SUBSEQUENT CALIBRATION OF THE NI'S. CORRECTIVE ACTIONS INCLUDE INSTALLING THE MISSING SCREWS. SEVERAL OTHER PANELS WERE INSPECTED AND, SINCE A SIMILAR PROBLEM WAS NOT IDENTIFIED, THIS IS NOT CONSIDERED TO BE A GENERIC PROBLEM.

[347] WATERFORD 3 DOCKET 50-382 LER 87-028
REACTOR TRIP DUE TO A FAILED SOLENOID VALVE DURING MAIN STEAM ISOLATION VALVE TESTING.
EVENT DATE: 121187 REPORT DATE: 011188 NSSS: CE TYPE: PWR
VENDOR: FLUID CONTROLS CORP.

(NSIC 207904) ON DECEMBER 11, 1987, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 90% POWER WHEN A REACTOR TRIP OCCURRED DUE TO THE INADVERTENT SHUTTING OF A MAIN STEAM ISOLATION VALVE (MSIV). AT 0754 HOURS, WITH THE AUXILIARY FEEDWATER PUMP TAGGED OUT AND THE HIGH PRESSURE STEAM SUPPLY TO MAIN FEEDWATER PUMP (MFWP) TURBINE A ISOLATED, MSIV TESTING WAS IN PROGRESS WHEN MSIV #2 WENT PARTIALLY SHUT DUE TO A FAILED SOLENOID VALVE. THIS RESULTED IN A CORE PROTECTION CALCULATOR TRIP DUE TO A LARGE DIFFERENCE IN REACTOR COLD LEG TEMPERATURES. THE MSIV CLOSURE ALSO CAUSED A STEAM GENERATOR (SG) SECONDARY SIDE PRESSURE RELIEF VALVE TO LIFT. BOTH MFWP'S TRIPPED AFTER THE REACTOR TRIP, BUT THE EMERGENCY FEEDWATER SYSTEM FUNCTIONED PROPERLY THROUGHOUT THE EVENT TO MAINTAIN SG LEVELS. THE MSIV WENT PARTIALLY SHUT BECAUSE A SOLENOID VALVE (SV) STUCK OPEN DURING TESTING ALLOWING BOTH HYDRAULIC SYSTEM DUMP VALVES TO BE OPEN SIMULTANEOUSLY AND DRAIN THE HYDRAULIC FLUID FROM THE MSIV ACTUATOR. THE SV WAS REPLACED AND THE MSIV TESTED SUCCESSFULLY PRIOR TO RESTARTING THE PLANT. THE SV HAS BEEN RETURNED TO THE MANUFACTURER FOR ANALYSIS AND RECOMMENDATIONS. SINCE ALL PROTECTIVE FEATURES FUNCTIONED AS DESIGNED, THERE WAS NO THREAT TO THE HEALTH AND SAFETY OF THE PUBLIC OR PLANT PERSONNEL.

[348] WOLF CREEK 1 DOCKET 50-482 LER 87-052
INSTRUMENT TERMINATION SPLICES INSTALLED WHICH FAIL TO MEET ENVIRONMENTAL QUALIFICATION REQUIREMENTS.
EVENT DATE: 111887 REPORT DATE: 121887 NSSS: WE TYPE: PWR
VENDOR: RAYCHEM CORP.

(NSIC 207562) ON 11/18/87, AT APPROXIMATELY 1520 CST, IT WAS DETERMINED THAT ENVIRONMENTALLY QUALIFIED (EQ) TERMINATIONS ON EQ INSTRUMENTATION CIRCUITS IN MULTIPLE SYSTEMS AND REDUNDANT SAFETY TRAINS WERE NOT INSTALLED IN ACCORDANCE WITH THE DESIGN, VENDOR RECOMMENDATIONS, OR THE EQ PROGRAM REQUIREMENTS. THE MAJORITY OF THESE SPLICES WERE INSTALLED DURING PLANT CONSTRUCTION. SEVERAL CAUSES LED TO THIS EVENT. WORK INSTRUCTIONS DID NOT PROVIDE NECESSARY DETAIL TO ASSURE THE SIZE HEAT SHRINK TUBING USED AGREED WITH DESIGN AND MANUFACTURER'S REQUIREMENTS. TRAINING PROVIDED TO PERSONNEL INSTALLING THE SPLICES DID NOT EMPHASIZE THAT THE SIZE OF HEAT SHRINK TUBING SELECTED FOR USE WAS ESSENTIAL TO MEET DESIGN AND MANUFACTURER'S REQUIREMENTS. FURTHER, WHILE REFERENCED BY THE DESIGN DOCUMENTS, THE MANUFACTURER'S INSTRUCTIONS WERE NOT PROVIDED WITH THE WORK PACKAGES. FOLLOWING PROGRAMMATIC ENHANCEMENTS, A REWORK OF EQ INSTRUMENTATION TERMINATION SPLICES WAS PERFORMED TO BRING THEM INTO COMPLIANCE WITH THE DESIGN

REQUIREMENTS. IN ORDER TO PREVENT RECURRENCE, DURING FUTURE INSTALLATION OF EQ TERMINATION SPLICES WRITTEN INSTALLATION INSTRUCTIONS UTILIZED IN THE FIELD WILL CONTAIN DETAILED INFORMATION AND CRITERIA FOR THE PROPER SELECTION OF HEAT SHRINK TUBING. PERSONNEL INSTALLING THE SPLICES WILL BE TRAINED ON THE INSTALLATION INSTRUCTIONS .

[349] WOLF CREEK 1 DOCKET 50-482 LER 87-053
ENGINEERED SAFETY FEATURES ACTUATION - CONTROL ROOM VENTILATION ISOLATION SIGNAL CAUSED BY PAPER TAPE BUNCHING UP ON CHLORINE MONITOR.
EVENT DATE: 112387 REPORT DATE: 122287 NSSS: WE TYPE: PWR
VENDOR: M D A SCIENTIFIC, INC.

(NSIC 207535) ON NOVEMBER 23, 1987, AT 1107 CST, A CONTROL ROOM VENTILATION ISOLATION SIGNAL (CRVIS) OCCURRED DUE TO CHLORINE MONITOR GK-AITS-3 INDICATING HIGH CHLORINE LEVEL IN THE OUTSIDE AIR MAKEUP TO THE CONTROL BUILDING HEATING, VENTILATING AND AIR CONDITIONING SYSTEM. UPON RECEIPT OF THE CRVIS, ALL ENGINEERED SAFETY FEATURES EQUIPMENT REQUIRED TO OPERATE RESPONDED PROPERLY. DURING THIS EVENT, THE PLANT WAS IN MODE 6, REFUELING, WITH THE REACTOR COOLANT SYSTEM DEPRESSURIZED AND A REACTOR COOLANT AVERAGE TEMPERATURE OF 102 DEGREES FAHRENHEIT. NO CHLORINE WAS PRESENT AS EVIDENCED BY NORMAL READINGS ON THE REDUNDANT CHLORINE MONITOR. THE CONTROL BUILDING HEATING, VENTILATING AND AIR CONDITIONING SYSTEM WAS RETURNED TO A NORMAL CONFIGURATION AT 1127 CST, NOVEMBER 23, 1987, AND THE AFFECTED MONITOR WAS PLACED IN BYPASS FOR TROUBLESHOOTING. EXAMINATION OF THE MONITOR AFTER THE EVENT REVEALED THAT THE CHEMICALLY SENSITIVE PAPER TAPE USED TO DETECT CHLORINE HAD STOPPED WINDING ONTO THE TAKE-UP SPOOL AND WAS, INSTEAD, BUNCHING UP. THIS EVENTUALLY CAUSED LESS LIGHT TO BE ABLE TO PASS THROUGH THE PAPER, INITIATING THE CRVIS. THE PAPER TAPE WAS REPLACED. NO FURTHER PROBLEMS WITH THE MONITOR WERE FOUND AND IT WAS RETURNED TO OPERATION AT 1518 CST ON NOVEMBER 23, 1987.

[350] WOLF CREEK 1 DOCKET 50-482 LER 87-057
FAILURE TO FULLY UNDERSTAND REQUIREMENTS CAUSES TECHNICAL SPECIFICATION VIOLATIONS DUE TO HOURLY RATHER THAN CONTINUOUS FIRE WATCHES ESTABLISHED.
EVENT DATE: 122187 REPORT DATE: 011588 NSSS: WE TYPE: PWR

(NSIC 207933) ON DECEMBER 21, 1987, AT APPROXIMATELY 1500 CST, IT WAS DISCOVERED BY THE FIRE PROTECTION COORDINATOR THAT A CONTINUOUS RATHER THAN AN HOURLY FIRE WATCH SHOULD HAVE BEEN ESTABLISHED FOR ROOM 1403, LOAD CENTER AND MOTOR GENERATOR SETS ROOM. ON JANUARY 8, 1988, AT APPROXIMATELY 1700 CST, IT WAS DETERMINED BY THE FIRE PROTECTION COORDINATOR THROUGH REVIEW OF FIRE IMPAIRMENT CONTROL PERMITS THAT A CONTINUOUS RATHER THAN AN HOURLY FIRE WATCH SHOULD HAVE BEEN ESTABLISHED IN AREA 1301/1320 OF THE AUXILIARY BUILDING WHERE SAFE SHUTDOWN CIRCUITS AND REDUNDANT EQUIPMENT IS LOCATED. THE ROOT CAUSE OF THESE EVENTS WAS DETERMINED TO BE COGNITIVE PERSONNEL ERROR IN DETERMINING THE TYPE OF FIRE WATCH REQUIRED. SHORT TERM ACTIONS TAKEN TO PREVENT RECURRENCE WERE RETRAINING OF THE FIRE PROTECTION EMPLOYEE, DISSEMINATION OF PROCEDURE REVISIONS, AND DISTRIBUTION OF LETTERS FROM THE FIRE PROTECTION COORDINATOR EMPHASIZING THE DISTINCTION BETWEEN TECHNICAL SPECIFICATIONS 3.7.10.2, 3.7.10.3 AND 3.7.11 IN ESTABLISHING THE APPROPRIATE FIRE WATCH. LONG TERM ACTIONS ARE THE ADDITION OF THIS LICENSEE EVENT REPORT TO REQUIRED READING AND THE IMPROVEMENT OF CONTINUING TRAINING FOR FIRE PROTECTION PERSONNEL AND PLANT OPERATORS IN THE RECOGNITION OF CONDITIONS THAT LED TO THIS EVENT.

[351] WOLF CREEK 1 DOCKET 50-482 LER 87-060
PROCEDURAL INADEQUACY RESULTING IN TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 122287 REPORT DATE: 012188 NSSS: WE TYPE: PWR

(NSIC 207872) ON DECEMBER 22, 1987, DURING A REVIEW OF THE SURVEILLANCE TESTING

PROCEDURE FOR THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) P-4 REACTOR TRIP SYSTEM INTERLOCK, IT WAS DISCOVERED THAT THE P-4 TURBINE TRIP ON REACTOR TRIP FUNCTION HAD NOT BEEN INCORPORATED INTO THE SURVEILLANCE TESTING PROCEDURE, AND THEREFORE HAD NOT BEEN TESTED IN ACCORDANCE WITH TECH SPEC REQUIREMENTS. THE CAUSE OF THIS PROCEDURAL INADEQUACY WAS A COGNITIVE PERSONNEL ERROR, BY INSTRUMENTATION AND CONTROL (I&C) PERSONNEL, WHO FAILED TO INCORPORATE THE P-4 TURBINE TRIP ON REACTOR TRIP FUNCTION INTO THE SURVEILLANCE TESTING PROCEDURE. ALTHOUGH THE BASES IN THE TECH SPEC DISCUSSES THE P-4 TURBINE TRIP ON REACTOR TRIP FUNCTION, BECAUSE I&C PERSONNEL VIEWED THIS P-4 FUNCTION AS A TRIP FUNCTION OUTSIDE THE SCOPE OF ESFAS, AND THEREFORE OUTSIDE THE SCOPE OF THE TECH SPEC SURVEILLANCE REQUIREMENT, THIS P-4 FUNCTION WAS NOT INCLUDED IN THE PROCEDURE. THIS AMBIGUITY IN THE TECH SPEC REQUIREMENT CONTRIBUTED TO THE PROCEDURAL INADEQUACY. THE PROCEDURE WAS REVISED TO INCLUDE TESTING OF THE P-4 TURBINE TRIP ON REACTOR TRIP FUNCTION, AND THE SURVEILLANCE WAS PERFORMED SATISFACTORILY. A REVIEW OF THE TECH SPEC WILL BE CONDUCTED TO ENSURE THAT OTHER FUNCTIONS DISCUSSED IN THE BASES ARE INCLUDED IN THE SURVEILLANCE TESTING PROCEDURES.

[352] WOLF CREEK 1 DOCKET 50-482 LER 87-058
CONTAINMENT HIGH RANGE RADIATION MONITORS NOT INSTALLED PROPERLY DUE TO ERROR IN DESIGN DOCUMENT.
EVENT DATE: 122387 REPORT DATE: 012288 NSSS: WE TYPE: PWR
VENDOR: CONAX CORP.
GENERAL ATOMIC CO.
RAYCHEM CORP.

(NSIC 208008) AT 0729 CST, DECEMBER 23, 1987, IT WAS DETERMINED THAT A VIOLATION OF THE PLANT'S TECH SPECS HAD OCCURRED BECAUSE THE DETECTOR CABLES FOR THE CONTAINMENT HIGH RANGE RADIATION MONITORS WERE NOT INSTALLED IN ACCORDANCE WITH EQUIPMENT QUALIFICATION REQUIREMENTS. THIS CONDITION HAS EXISTED SINCE THE RECEIPT OF THE OPERATING LICENSE ON MARCH 11, 1985. THE EVENT IS BELIEVED TO HAVE OCCURRED BECAUSE THE CONTRACT DESIGN ENGINEER MISINTERPRETED A VENDOR MANUAL WHILE COMPILING THE ELECTRICAL PENETRATION LIST. WHEN DESIGN ENGINEERING DETERMINED THAT THIS DESIGN DOCUMENT WAS IN ERROR ON DECEMBER 23, 1987, THE CONTAINMENT HIGH RANGE RADIATION MONITORS WERE DECLARED INOPERABLE. HEAT SHRINK MOISTURE SEALS WERE INSTALLED AT 0140 CST ON DECEMBER 29, 1987, PURSUANT TO A DESIGN CHANGE THAT ALSO REVISED THE ELECTRICAL TERMINATION LIST AND VENDOR MANUAL. A REVIEW OF DESIGN DOCUMENTS AND EQUIPMENT QUALIFICATION RECORDS DETERMINED THAT THESE CONNECTORS ARE THE ONLY COAXIAL CABLES AFFECTED BY THE DESIGN ERROR.

[353] WOLF CREEK 1 DOCKET 50-482 LER 87-059
WIRED GLASS INSERT DISCOVERED IN FIRE DOOR CAUSES LOSS OF THREE-HOUR FIRE RATING.
EVENT DATE: 122387 REPORT DATE: 012288 NSSS: WE TYPE: PWR
VENDOR: CECO CORP

(NSIC 208009) ON DECEMBER 23, 1987, IT WAS DETERMINED THAT A VIOLATION OF TECH SPECS HAD OCCURRED BECAUSE DOOR 14051, THE AUXILIARY BUILDING FILTRATION UNIT ROOM NUMBER B DOOR, WAS NOT AN ACCEPTABLE THREE-HOUR FIRE RATED BARRIER. THE CAUSE OF THIS EVENT IS THAT DOOR 14051 IS IMPROPERLY UNDERWRITER LABORATORY LABELLED AS A THREE-HOUR FIRE PROTECTION RATED DOOR EVEN THOUGH A WIRED GLASS INSERT IS INTALLED IN THE DOOR. INVESTIGATIONS CONCERNING THE DESIGN, PROCUREMENT, INSTALLATION, AND WORK HISTORY OF DOOR 14051 WERE UNSUCCESSFUL IN DETERMINING THE ROOT CAUSE OF HOW OR WHEN THE WIRED GLASS INSERT WAS INSTALLED. THE UNACCEPTABLE FIRE PROTECTION RATING OF DOOR 14051 WAS NOT PREVIOUSLY IDENTIFIED BECAUSE WHETHER THE DOOR CONTAINED GLASS OR NOT WAS NOT AN INSPECTED ATTRIBUTE. A FIRE IMPAIRMENT CONTROL PERMIT WAS ISSUED AGAINST DOOR 14051 ON DECEMBER 23, 1987. FIRE WATCHES HAVE BEEN ESTABLISHED AND ARE BEING MAINTAINED. DOOR 14051 WILL BE REPLACED WITH A DOOR WHICH CONFORMS TO DESIGN REQUIREMENTS.

AN INSPECTION OF REQUIRED FIRE DOORS WILL BE CONDUCTED TO ENSURE THAT THE DOOR TYPE CONFORMS WITH THE DESIGN DRAWING.

[354] WOLF CREEK 1 DOCKET 50-482 LER 87-051
PROCEDURAL DEFICIENCY CAUSES TWO MAIN FEEDWATER ISOLATIONS AND AN AUXILIARY FEEDWATER ACTUATION.
EVENT DATE: 122687 REPORT DATE: 012288 NSSS: WE TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 208007) ON DECEMBER 26, 1987, THREE ENGINEERED SAFETY FEATURES ACTUATIONS OCCURRED. AT APPROXIMATELY 0510 CST, A MAIN FEEDWATER ISOLATION SIGNAL (FWIS) OCCURRED WHEN STEAM GENERATOR 'A' (S/G) LEVEL SWELLED ABOVE THE HI-HI SETPOINT. AT APPROXIMATELY 0517 CST, A REACTOR TRIP SIGNAL AND AUXILIARY FEEDWATER ACTUATION SIGNAL (AFAS) OCCURRED WHEN S/G 'A' LEVEL SHRANK BELOW THE LO-LO LEVEL SETPOINT. AT APPROXIMATELY 0830 CST, A SECOND FWIS OCCURRED WHEN S/G 'A' LEVEL AGAIN SWELLED ABOVE THE HI-HI LEVEL SETPOINT. THE ROOT CAUSE OF THE FIRST FWIS (0510 CST) AND THE AFAS (0517 CST) WERE PROCEDURAL DEFICIENCY. THE PROCEDURE FOR TESTING THE MSIVS HAS BEEN ENHANCED TO REDUCE THE EFFECTS OF SWELL BY REQUIRING A LOW PRESSURE DIFFERENTIAL ACROSS THE MSIV AND LESS THAN 50 PERCENT S/G LEVEL PRIOR TO OPENING THE MSIV. THE ROOT CAUSE OF THE SECOND FWIS (0830 CST) IS A COMBINATION OF PERSONNEL ERROR AND PROCEDURAL DEFICIENCY. THE LICENSED OPERATOR CONTROLLING THE S/G LEVEL ALLOWED THE S/G LEVEL TO REACH APPROXIMATELY 66% PRIOR TO OPENING THE MSIV. THE PROCEDURE DID NOT ADDRESS A LEVEL REQUIREMENT PRIOR TO THIS EVENT. THE PROCEDURE ENHANCEMENTS SHOULD PREVENT BOTH OF THESE EVENTS FROM RECURRING.

[355] WPPSS 2 DOCKET 50-397 LER 87-032
NONCOMPLIANCE WITH TECH SPECS DUE TO RCIC ISOLATION TRIP SYSTEM INOPERABILITY CAUSED BY INCORRECT INSTRUMENT SETPOINT CALCULATION.
EVENT DATE: 121687 REPORT DATE: 011588 NSSS: GE TYPE: BWR

(NSIC 207905) DURING A TECHNICAL REVIEW ON 12/16/87, AN INCORRECT SURVEILLANCE PROCEDURE WAS DISCOVERED WHICH ERRONEOUSLY DIRECTED THE CALIBRATION OF TWO OUT OF FOUR REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) TURBINE HIGH EXHAUST DIAPHRAGM PRESSURE SWITCHES. THE SET POINTS FOR THE SWITCHES WERE INCORRECT DUE TO A PLANT STARTUP ENGINEER ERRONEOUSLY ADDING A WATER COLUMN HEAD CORRECTION FACTOR TO THE SETPOINT CALCULATIONS. THIS ERROR RESULTED IN INOPERABILITY, FOR LENGTHY PERIODS SINCE INITIAL FUEL LOAD, OF ONE OF TWO ISOLATION TRIP SYSTEMS FOR THE RCIC SYSTEM ON TURBINE HIGH EXHAUST PRESSURE. THE ROOT CAUSE OF THE EVENT WAS DETERMINED TO BE PERSONNEL ERROR ON THE PART OF A PLANT STARTUP ENGINEER IN THAT THE ORIGINAL PRESSURE SWITCH SETPOINT CALCULATION WAS PERFORMED INCORRECTLY. THE SURVEILLANCE PROCEDURE WAS CORRECTED AND A CONFIRMATORY CHANNEL CALIBRATION WAS PERFORMED USING THE CORRECTED SETPOINTS. THE ERRONEOUS SETPOINT CALCULATION AFFECTED THE TWO DIVISION I CHANNELS OF THE FOUR CHANNELS OF INSTRUMENTATION WHICH PROVIDE ISOLATION FOR THE RCIC SYSTEM ON HIGH PRESSURE AT THE TURBINE EXHAUST DIAPHRAGMS. THE PROCEDURE FOR THE REMAINING TWO CHANNELS (COMPRISING THE DIVISION II TRIP SYSTEM) WAS WRITTEN CORRECTLY. THESE 4 PRESSURE SWITCHES MONITOR THE SAME SECTION OF PIPING AND CAUSE ISOLATION OF REDUNDANT ISOLATION VALVES.

[356] WPPSS 2 DOCKET 50-397 LER 87-033
PLANT TECHNICAL SPECIFICATION VIOLATION CAUSED BY MISSED ASME VALVE OPERABILITY SURVEILLANCES DUE TO PERSONNEL ERROR.
EVENT DATE: 121887 REPORT DATE: 011588 NSSS: GE TYPE: BWR

(NSIC 207906) ON 12/16/87, WHILE PERFORMING AN OPERATIONAL QUALITY ASSURANCE (QA) SURVEILLANCE, A PLANT QA ENGINEER DISCOVERED A CONDITION THAT POTENTIALLY VIOLATED WNP-2 PLANT TECH SPECS. THE SURVEILLANCE INFORMATION WAS GIVEN TO A PLANT ENGINEER WHO DETERMINED, ON 12/18/87 THAT WNP-2 PLANT TECH SPEC SECTIONS

0.5, ASME VALVE IN-SERVICE TESTING REQUIREMENTS, AND 3.6.3, PRIMARY CONTAINMENT VALVE OPERABILITY REQUIREMENTS, FOR THE POST ACCIDENT SAMPLING SYSTEM (PASS) HAD ON TWO OCCASIONS NOT BEEN MET. DURING THE 1986 AND 1987 SPRING REFUELING AND MAINTENANCE OUTAGES, THE PASS PSR PRIMARY CONTAINMENT ISOLATION VALVE ASME IN-SERVICE TESTING SURVEILLANCE WAS NOT PERFORMED PRIOR TO PLANT STARTUP. THE SURVEILLANCE INTERVAL ALLOWABLE BY WNP-2 PLANT TECH SPEC 4.0.2 WAS EXCEEDED AND, THEREFORE, CHANGING TO OPERATIONAL MODES 1 , 2, OR 3 FROM OPERATIONAL MODE 4 WAS NOT ALLOWED BY TECH SPEC 3.6.3 SINCE THE PSR VALVES WERE BY TECH SPEC REQUIREMENTS INOPERABLE. THE TWO TIME PERIODS WHEN THE PLANT WAS NOT IN COMPLIANCE WITH TECH SPEC WERE FROM JUNE 4, 1986 TO JULY 12, 1986 AND FROM AUGUST 1, 1987 TO AUGUST 10, 1987. THE CAUSE OF THIS EVENT IS PERSONNEL ERROR IN THAT PLANT CHEMISTRY PERSONNEL ASSUMED RESPONSIBILITY FOR THE "PSR VALVE OPERABILITY" SURVEILLANCE WITHOUT ADEQUATELY UNDERSTANDING THE SURVEILLANCE REQUIREMENTS.

[357] ZION 1 DOCKET 50-295 LER 87-016
INOPERABLE CONTAINMENT ISOLATION VALVE.
EVENT DATE: 113087 REPORT DATE: 123087 NSSS: WE TYPE: PWR
VENDOR: LIMITORQUE CORP.
POWELL, WILLIAM COMPANY

(NSIC 207710) ON 11/30/87 WITH UNIT 1 IN MODE 1, POWER OPERATION, ZION STATION MECHANICAL MAINTENANCE DEPARTMENT BLOCKED SHUT THE 1C REACTOR CONTAINMENT FAN COOLER (RCFC) OUTLET ISOLATION VALVE, 1MOV-SW0009, AND REMOVED THE VALVE MOTOR OPERATOR FOR WORK. AT 1845, WHILE MAKING HIS ROUNDS, AN EQUIPMENT OPERATOR NOTICED LOW SERVICE WATER (SW, EEIS CODE (BK) PRESSURE AND HIGH FLOW ON 1C RCFC, INDICATIVE OF AN OPEN OUTLET VALVE. THE VALVE MECHANICAL BLOCK HAD FAILED ALLOWING THE VALVE TO PARTIALLY OPEN. THE OPERATOR NOTIFIED THE CONTROL ROOM AND AT 1900 SHIFT MANAGEMENT VERIFIED THE PRESSURE AND FLOW AND VISUALLY EXAMINED 1MOV-SW0009. SHIFT MANAGEMENT CONCLUDED THE VALVE WAS FULLY CLOSED AND THERE WAS A PROBLEM WITH THE PRESSURE AND FLOW INSTRUMENTS. THE PROBLEM WAS TURNED OVER TO THE NEXT SHIFT WHO TURNED IT OVER TO THE DAY SHIFT. DAY SHIFT REQUESTED TECHNICAL STAFF ASSISTANCE. AT 0900 ON 12/1/87 TECH STAFF DETERMINED THAT 1MOV-SW0009 WAS FAILED PARTIALLY OPEN AND INOPERABLE. SINCE 1MOV-SW0009 IS A CONTAINMENT ISOLATION VALVE, TECH SPEC 3.9.3 REQUIRES THAT THE AFFECTED PENETRATION BE ISOLATED WITHIN FOUR HOURS. THIS WAS ACCOMPLISHED AT 1015 ON 12/1/87 BY SHUTTING THE 1C RCFC AND 1C RCFC MOTOR MANUAL ISOLATION VALVES. THIS WAS ACCOMPLISHED WITHIN FOUR HOURS OF THE TIME THAT THE VALVE WAS DETERMINED TO BE INOPERABLE.

[358] ZION 1 DOCKET 50-295 LER 87-017
DEGRADED FIRE RETARDANT MATERIAL IN PENETRATION FIRE BARRIER DUE TO THERMAL EXPANSION.
EVENT DATE: 121087 REPORT DATE: 010888 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 2 (PWR)

(NSIC 207635) DURING THE PERFORMANCE OF THE 18 MONTH PERIODIC TEST PT-207 ON PENETRATION FIRE BARRIERS, THE STATION FIRE MARSHALL DISCOVERED THAT LIGHT WAS VISIBLE THROUGH A ONE INCH EXPANSION GAP WHERE THE VERTICAL PIPE CHASE WALL ABUTS THE CONTAINMENT WALL. THIS GAP HAD BEEN FILLED WITH STYROFOAM AND COVERED WITH FLAMASTIC, WHICH IS AN ACCEPTABLE FIRE BARRIER ONLY IF NO LIGHT IS VISIBLE THROUGH THE PENETRATION. AN ENGINEERING ANALYSIS CONCLUDED THAT THE BARRIER WAS INOPERABLE PER THE PLANT'S TECH SPECS, BUT THAT THERE WAS NO ADVERSE IMPACT ON SAFE SHUTDOWN CAPABILITY. THIS CONCLUSION WAS BASED ON THE LACK OF SAFETY RELATED EQUIPMENT ON EITHER SIDE OF THE AFFECTED WALLS, AND ON THE LOW COMBUSTIBLE LOADING IN THE AREA. AN ENGINEERING INSPECTION HAD PREVIOUSLY IDENTIFIED THE USE OF STYROFOAM TO FILL THIS PENETRATION FIRE BARRIER, BUT THE BARRIER WAS CONSIDERED OPERABLE DUE TO THE GOOD CONDITION OF THE FLAMASTIC COVERING THE STYROFOAM. COLD TEMPERATURES DURING THE DECEMBER 1987 INSPECTION CAUSED THERMAL CONTRACTION OF THE ADJACENT WALLS, SUFFICIENT TO DEGRADE THE BARRIER SO THAT LIGHT WAS VISIBLE THROUGH THE PENETRATION. CORRECTIVE ACTIONS

INCLUDE AN HOURLY FIRE WATCH AND SCHEDULED REPLACEMENT OF THE BARRIER WITH AN APPROVED FIRE RETARDANT MATERIAL.

[359] ZION 2 DOCKET 50-304 LER 85-029 REV 03
UPDATE ON PURGE ISOLATION DUE TO LOW TEMPERATURE AND HIGH RADIATION SIGNAL.
EVENT DATE: 120185 REPORT DATE: 012888 NSSS: WE TYPE: PWR

(NSIC 207988) ON DECEMBER 1, 1985 WITH UNIT 2 IN COLD SHUTDOWN, DURING A UNIT 2 CONTAINMENT PURGE, THE RUNNING 2A PURGE SUPPLY FAN AND 2A EXHAUST FAN TRIPPED, THE CONTAINMENT PURGE INLET AND OUTLET ISOLATION VALVES 2AOV-RV0001, 2AOV-RV00002, 2AOV-RV0003, 2AOV-RV0004 CLOSED, AND THE "AIR EXHAUST STACK RADIATION HIGH" ANNUNCIATOR ALARMED. THE CAUSE OF THIS ISOLATION WAS A SPURIOUS HIGH RADIATION ALARM FROM CONTAINMENT PURGE EXHAUST STACK AIR PARTICULATE MONITOR 2RT-PRO9C WHICH ISOLATED THE PURGE INLET AND OUTLET VALVES AND TRIPPED THE RUNNING PURGE AND EXHAUST FANS. THE ROOT CAUSE OF THIS EVENT WAS A SPURIOUS SPIKE ON MONITOR 2RT-PRO9C CAUSED BY VOLTAGE SPIKING ON THE AC POWER FEED TO THE MONITOR.

[360] ZION 2 DOCKET 50-304 LER 87-009
MISSED TECHNICAL SPECIFICATION SURVEILLANCE CAUSED BY FAILURE TO PROMPTLY NOTIFY CONTROL ROOM OF PROCEDURE CHANGE.
EVENT DATE: 111887 REPORT DATE: 121887 NSSS: WE TYPE: PWR
VENDOR: BETA CORP.

(NSIC 207524) ON NOVEMBER 18, 1987, BETWEEN THE HOURS OF 00:00 AND 07:00, THE NIS UPPER AND LOWER FLUX DEVIATION AUDIBLE ANNUNCIATORS (IB) WERE DEFEATED ON UNIT 2 BY THE UNIT 2 NUCLEAR STATION OPERATOR (NSO). THE ALARMS WERE FREQUENTLY COMING IN AND THEN RESETTING BECAUSE A NUCLEAR INSTRUMENTATION SYSTEM (NIS) (IG) POWER RANGE CHANNEL WAS STARTING TO DRIFT SLIGHTLY HIGHER THAN THE OTHER THREE POWER RANGES. THE FLASHING LIGHT ON THE MAIN CONTROL BOARD FOR THE ANNUNCIATORS REMAINED OPERABLE. ACCORDING TO TECH SPEC 4.2.2.B.1.B, WHEN BOTH OF THESE ALARMS ARE INOPERABLE, A QUADRANT POWER TILT CALCULATION WILL BE PERFORMED EVERY TWO HOURS. A PROCEDURE CHANGE HAD RECENTLY BEEN MADE TO THE ANNUNCIATOR RESPONSE MANUAL (ARM) THAT DEFINES A DEFEATED AUDIBLE ANNUNCIATOR TO BE AN INOPERABLE ANNUNCIATOR. THIS PROCEDURE CHANGE WAS NOT COMMUNICATED PROMPTLY TO SHIFT PERSONNEL. CONSEQUENTLY, WHEN THE NSO DEFEATED THE ALARMS, HE DID NOT REALIZE THAT HE WAS ALSO MAKING THEM INOPERABLE, AND HE DID NOT CONSULT THE ARM TO SEE IF HIS ACTION REQUIRED ANY ADDITIONAL SURVEILLANCES. THIS CONDITION PERSISTED UNTIL NOVEMBER 20, 1987, WHEN A WESTINGHOUSE REPRESENTATIVE IN CHARGE OF REVISING THE ARM NOTICED THE DEFEATED ANNUNCIATORS AND NOTIFIED SHIFT PERSONNEL OF THE REQUIRED SURVEILLANCE.

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.

SYSTEM INDEX

SYSTEM INDEX

SYSTEM INDEX

SYSTEM INDEX

KEYWORD INDEX

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

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

VENDOR CODE INDEX

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)
NUREG/CR-2000, Vol. 7, No. 3
ORNL/NSIC-200

2. TITLE AND SUBTITLE
Licensee Event Report (LER) Compilation for month of March 1988

3. LEAVE BLANK

4. DATE REPORT COMPLETED
MONTH	YEAR
March	1988

5. AUTHOR(S)

6. DATE REPORT ISSUED
MONTH	YEAR
April	1988

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Oak Ridge National Laboratory
Nuclear Operations Analysis Center
Oak Ridge, TN 37831

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER
FIN A9135

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Office for Analysis and Evaluation of Operational Data
U.S. Nuclear Regulatory Commission
Washington, DC 20555

11a. TYPE OF REPORT
Monthly Report

b. PERIOD COVERED (Inclusive dates)
March 1988

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

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

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS
licensee event report
sequence coding and search system
vendors

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT
Unlimited

16. SECURITY CLASSIFICATION
(This page) Unclassified
(This report) Unclassified

17. NUMBER OF PAGES

18. PRICE

*U.S. GOVERNMENT PRINTING OFFICE:1988-202-292:80040

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

FIRST CLASS MAIL
POSTAGE & FEES PAID
USNRC
PERMIT No. G-67

120555078877 1 1AN1CV1NJ11M1
US NRC-OARM-ADM
DIV OF PUB SVCS
POLICY & PUB MGT BR-PDR NUREG
W-537
WASHINGTON DC 20555