NUREG/CR-2000
ORNL/NSIC-200
Vol. 9, No. 11

Licensee Event Report (LER) Compilation

For month of November 1990

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

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Manuscript Completed: December 1990
Date Published: December 1990

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Nucl. Operations Analysis 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-0161, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports. For those events occurring on and after January 1, 1984, LERs are being submitted in accordance with the revised rule contained in Title 10 Part 50.73 of the Code of Federal Regulations (10 CFR 50.73 - Licensee Event Report System) which was published in the Federal Register (Vol. 48, No. 144) on July 26, 1983. NUREG-1022, Licensee Event Report System - Description of Systems and Guidelines for Reporting, provides supporting guidance and information on the revised LER rule.

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

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[1] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 90-019
FAILURE OF 125 VOLT DC SOLENOID OPERATED VALVE CAUSED INADVERTENT MAIN STEAM ISOLATION VALVE CLOSURE AND SUBSEQUENT REACTOR TRIP.
EVENT DATE: 082190 REPORT DATE: 092090 NSSS: CE TYPE: PWR
VENDOR: ASCO VALVES

(NSIC 219614) ON AUGUST 21, 1990 AT 0216 HOURS, WHILE OPERATING AT FULL POWER THE 'B' STEAM GENERATOR (S/G) MAIN STEAM ISOLATION VALVE (MSIV) CLOSED FULLY RESULTING IN AN AUTOMATIC REACTOR TRIP GENERATED BY THE CORE PROTECTION CALCULATORS (CPCS). THE CPC TRIP WAS INITIATED DUE TO A SENSED TEMPERATURE DIFFERENCE BETWEEN THE REACTOR COOLANT SYSTEM (RCS) COLD LEGS CAUSED BY THE MSIV CLOSURE. THE EMERGENCY FEEDWATER SYSTEM ACTUATED AUTOMATICALLY AND WAS USED TO SUPPLY S/G FEEDWATER FOR DECAY HEAT REMOVAL. THE 'B' S/G MAIN STEAM SAFETY VALVES OPENED TO LIMIT S/G SECONDARY PRESSURE AND RESEATED PROPERLY FOLLOWING THE REACTOR TRIP. OTHER PLANT SYSTEMS RESPONDED PROPERLY TO THE TRANSIENT. SUBSEQUENT INVESTIGATIONS REVEALED THE MSIV CLOSURE WAS DUE TO THE FAILURE OF A NORMALLY ENERGIZED, 125 VOLT DC SOLENOID ON AN AIR SUPPLY VALVE TO THE MSIV. THE ROOT CAUSE OF THE SOLENOID FAILURE COULD NOT BE DETERMINED. THE SOLENOID WAS REPLACED AND THE MSIV WAS STROKE TESTED FOR OPERABILITY. APPROXIMATELY ONE HOUR AFTER THE TRIP THE INDICATION ON THE 'A' S/G BLOWDOWN LINE RADIATION MONITOR WAS NOTED TO BE SLIGHTLY ELEVATED. ANALYSIS OF S/G SECONDARY WATER CONFIRMED THAT A VERY SMALL PRIMARY TO SECONDARY LEAK EXISTED. THE LEAK RATE STABILIZED AT THIS VALUE AND HAS NOT INCREASED. THERE WAS NO ACTUAL SAFETY SIGNIFICANCE AS THE RESULT OF THIS EVENT.

[2] ARNOLD DOCKET 50-331 LER 89-016 REV 01
UPDATE ON HIGH PRESSURE COOLANT INJECTION SYSTEM INOPERABILITY DUE TO FAILURE TO OBTAIN ADEQUATE FLOW IN REQUIRED TIME.
EVENT DATE: 121289 REPORT DATE: 092590 NSSS: GE TYPE: BWR
VENDOR: DRESSER INDUSTRIES, INC.
WOODWARD GOVERNOR COMPANY

(NSIC 219630) ON DECEMBER 12, 1989 AT 1410 HOURS, THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM WAS DECLARED INOPERABLE DURING PERFORMANCE OF SURVEILLANCE TESTING. THE REACTOR WAS AT 100% POWER. DURING THE SURVEILLANCE TESTING, THE HPCI SYSTEM FAILED TO REACH 3000 GALLONS PER MINUTE FLOWRATE WITHIN THE REQUIRED 30 SECONDS DURING A COLD QUICK START. ACTUAL START TIME WAS 30.52 SECONDS. THE REQUIRED EMERGENCY CORE COOLING SYSTEMS WERE PROVEN OPERABLE PER TECHNICAL SPECIFICATIONS LIMITING CONDITIONS FOR OPERATION. THE CAUSE OF THE EVENT WAS DETERMINED TO BE INADEQUATE HPCI TURBINE ELECTRO- HYDRAULIC RESPONSE DURING THE TURBINE STARTUP SEQUENCE. THE ELECTRONIC PORTION OF THE TURBINE GOVERNOR (EG-M) WAS ADJUSTED WITHIN ALLOWABLE TOLERANCES AND A SATISFACTORY START TIME WAS ACHIEVED. SYSTEM IMPROVEMENTS WERE IDENTIFIED AND INCORPORATED IN A DESIGN CHANGE PACKAGE. THE HPCI SYSTEM WAS DECLARED OPERABLE AT 1034 HOURS ON DECEMBER 19, 1989. FOLLOWING COMPLETION OF THE MODIFICATIONS DURING THE REFUEL OUTAGE INITIAL TEST RESULTS FOR THE HPCI SYSTEM SHOW HPCI START TIMES IMPROVED BY APPROXIMATELY TWENTY PERCENT. THIS REPORT IS BEING SUBMITTED AS A SUPPLEMENT TO THE ORIGINAL LICENSEE EVENT REPORT TO DETAIL IMPROVEMENTS MADE TO THE HPCI SYSTEM DURING A RECENT REFUELING OUTAGE.

[3] ARNOLD DOCKET 50-331 LER 90-011
TWO PRIMARY CONTAINMENT ISOLATION SYSTEM ACTUATIONS DURING OUTAGE-RELATED WORK DUE TO A RESTRICTED WORKING ENVIRONMENT AND INADEQUATE HUMAN FACTORS CONSIDERATIONS.
EVENT DATE: 082190 REPORT DATE: 092090 NSSS: GE TYPE: BWR

(NSIC 219612) WITH THE PLANT IN A REFUELING OUTAGE, ON AUGUST 21, 1990, AN AUTOMATIC INITIATION OF ONE SIDE OF THE GROUP III PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) ISOLATION LOGIC OCCURRED DUE TO A BLOWN FUSE. THE ROOT CAUSE OF THE EVENT WAS A RESTRICTED WORKING ENVIRONMENT WITH LIMITED COMPENSATORY MEASURES AVAILABLE. AN INDIVIDUAL RE-TERMINATING WIRES UNDER CRAMPED CONDITIONS INADVERTENTLY TOUCHED A WIRE TO AN INCORRECT TERMINAL POINT IN CLOSE PROXIMITY TO THE DESIRED TERMINATION POINT. AS CORRECTIVE ACTIONS, THE CAUSE OF THE ISOLATION WAS VERIFIED, AND THE LOGIC AND ISOLATION WERE RESET. ON AUGUST 25, 1990, THE

AUTOMATIC CLOSURE OF A GROUP IV PCIS VALVE RESULTED IN A BRIEF LOSS OF RESIDUAL HEAT-REMOVAL SHUTDOWN COOLING. THE VALVE'S MOTOR-OPERATOR (MO) WAS DE-ENERGIZED WITH THE VALVE OPEN DURING A GROUP IV LOGIC TEST. THE MOTOR-OPERATOR CONTROL CIRCUITRY FOR THE VALVE WAS NOT DE-ENERGIZED, HOWEVER, AND A GROUP IV ISOLATION SIGNAL SEALED-IN. UPON RE-ENERGIZING THE VALVE'S MO, THE VALVE CLOSED. THE ROOT CAUSE OF THIS EVENT WAS FOUND TO BE INADEQUATE HUMAN FACTORS CONSIDERATIONS IN THE VALVE MO CONTROL CIRCUITRY DESIGN AND DOCUMENTATION. AS A CORRECTIVE ACTION, MO POWER BREAKERS FOR AFFECTED VALVES WILL BE LABELED TO ALERT PERSONNEL TO THE UNIQUE ASPECTS OF THE CONTROL CIRCUITRY DESIGN WHICH LED TO THE EVENT.

[4] ARNOLD DOCKET 50-331 LER 90-012
REACTOR PROTECTION SYSTEM TRIPS DURING ROUTINE MAINTENANCE WHILE SHUTDOWN.
EVENT DATE: 082390 REPORT DATE: 092190 NSSS: GE TYPE: BWR

(NSIC 219619) DURING THE REFUELING OUTAGE IN AUGUST, 1990, TWO SEPARATE EVENTS OCCURRED THAT RESULTED IN FULL REACTOR PROTECTION SYSTEM (RPS) LOGIC TRIPS. ON AUGUST 23RD, WHILE RESTORING A LEVEL TRANSMITTER TO SERVICE, A PRESSURE SURGE IN THE COMMON REFERENCE LINE OCCURRED WHILE FILLING AND VENTING THE TRANSMITTER. THIS SURGE CAUSED TWO LEVEL INDICATING SWITCHES TO TRIP AND RESULTED IN A FULL RPS LOGIC TRIP. THE ROOT CAUSE WAS DUE TO INSUFFICIENT FORMAL GUIDANCE FOR PERFORMANCE OF THIS OUTAGE SPECIFIC EVOLUTION. ON AUGUST 26TH, WHILE REINSTALLING A LOCAL POWER RANGE MONITOR (LPRM) CABLE GUARD, A WORKER INADVERTENTLY PULLED THE LPRM CABLE OUT OF THE CONNECTOR WHEN ATTEMPTING TO CATCH THE GUARD AFTER IT HAD BEEN DROPPED. THE LPRM SPIKED AND WAS SENSED BY THE 'A' AND 'B' AVERAGE POWER RANGE MONITORS AND RESULTED IN A FULL RPS LOGIC TRIP. THE ROOT CAUSE WAS DUE TO SEVERAL CONTRIBUTING FACTORS EFFECTING WORKING CONDITIONS. IN BOTH EVENTS, CORRECTIVE ACTIONS WERE TAKEN TO RESOLVE THE EVENT AND THE RPS TRIP WAS RESET. THERE WAS NO EFFECT ON THE SAFE SHUTDOWN CONDITION OF THE PLANT IN EITHER CASE.

[5] ARNOLD DOCKET 50-331 LER 90-013
REACTOR WATER CLEANUP ISOLATION DUE TO HIGH AREA DIFFERENTIAL TEMPERATURE.
EVENT DATE: 091090 REPORT DATE: 100290 NSSS: GE TYPE: BWR

(NSIC 219738) ON 9/10/90 AT 0016 HOURS, DURING REACTOR STARTUP, AN AUTOMATIC INITIATION OF THE GROUP V PRIMARY CONTAINMENT ISOLATION SYSTEM ISOLATION LOGIC OCCURRED. AS A RESULT, THE OUTBOARD REACTOR WATER CLEANUP SUCTION AND DISCHARGE VALVES CLOSED AS DESIGNED. THE INITIATING EVENT WAS A HIGH DIFFERENTIAL AREA TEMPERATURE AS SENSED BY TEMPERATURE SWITCH TDS2743F IN THE STEAM LEAK DETECTION LOGIC. THE INTERMEDIATE CAUSE OF THE SYSTEM ISOLATION WAS THE FAILURE TO ESTABLISH A NEW TRIP POINT FOR TEMPERATURE DIFFERENTIAL SWITCH TDS2743F AFTER ONE OF ITS INPUTS WAS RELOCATED BY A DESIGN CHANGE. THE ROOT CAUSE OF THE EVENT WAS INADEQUATE PROCEDURAL CONTROLS IN THE DESIGN CHANGE PROCESS. AS A CORRECTIVE ACTION, AN UPDATED SETPOINT FOR TDS2743F HAS BEEN ESTABLISHED. IN ADDITION, THE DESIGN ENGINEERING DEPARTMENT HAS INITIATED A SELF-EVALUATION OF OVERALL DEPARTMENT PERFORMANCE AND DESIGN VERIFICATION IN PARTICULAR THAT WILL RESULT IN ADDITIONAL VERIFICATION CRITERIA FOR DESIGN CHANGES. THIS EVENT IS BEING REPORTED PURSUANT TO 10CFR50.73(A)(2)(IV) AS AN ENGINEERED SAFETY FEATURE ACTUATION.

[6] ARNOLD DOCKET 50-331 LER 90-014
HIGH PRESSURE REACTOR SCRAM FOLLOWING MOISTURE SEPARATOR REHEATER HIGH LEVEL TURBINE TRIP.
EVENT DATE: 091090 REPORT DATE: 101090 NSSS: GE TYPE: BWR

(NSIC 219739) ON SEPTEMBER 10, 1990 WITH THE REACTOR AT APPROXIMATELY 27% POWER, A TURBINE TRIP OCCURRED AS A RESULT OF A SENSED HIGH LEVEL IN A MOISTURE SEPARATOR REHEATER. REACTOR STEAM PRODUCTION AT THE TIME OF THE TURBINE TRIP WAS SLIGHTLY IN EXCESS OF THE BYPASS VALVE CAPACITY, RESULTING IN A RISING REACTOR PRESSURE, AND A REACTOR SCRAM APPROXIMATELY ONE MINUTE LATER. PLANT RESPONSE TO THE CONDITIONS PRESENT OCCURRED APPROPRIATELY. PRIMARY CONTAINMENT ISOLATION GROUPS 2-5 RESPONDED IN ACCORDANCE WITH DESIGN WHEN REACTOR WATER LEVEL DECREASED AS A RESULT OF VOID REDUCTION IN RESPONSE TO THE REACTOR SCRAM. THE ROOT CAUSE

OF THE EVENT WAS VALVE MISALIGNMENT FOLLOWING MAINTENANCE. THE CORRECTIVE ACTIONS INCLUDED AN IMMEDIATE VALVE LINEUP VERIFICATION AND ENHANCEMENTS TO THE VALVE LINEUP PROCEDURE.

[7] ARNOLD DOCKET 50-331 LER 90-016
REACTOR SCRAM ON THREE MAIN STEAM LINES LESS THAN 90% OPEN DUE TO LOOSE ELECTRICAL CONNECTION COINCIDENT WITH SURVEILLANCE TEST PERFORMANCE.
EVENT DATE: 091890 REPORT DATE: 101290 NSSS: GE TYPE: BWR

(NSIC 219741) ON SEPTEMBER 18, 1990, WITH THE PLANT OPERATING AT APPROXIMATELY 50% POWER, A REACTOR SCRAM OCCURRED WHEN THREE INBOARD MAIN STEAM ISOLATION VALVES (MSIVS) CLOSED UNEXPECTEDLY. JUST PRIOR TO THE MSIVS CLOSING, THE 'A' SIDE OF THE MAIN STEAM LINE RADIATION MONITOR (MSLRM) SURVEILLANCE TEST HAD BEEN SATISFACTORILY COMPLETED WITH ISOLATION SIGNALS RESET. UPON INITIATING THE 'B' SIDE TEST, THE 'B', 'C', AND 'D' INBOARD MSIVS CLOSED RESULTING IN THE SCRAM. THE CAUSE OF THIS EVENT WAS A LOOSE WIRING CONNECTION IN THE INBOARD MSIV CONTROL LOGIC WHICH EFFECTIVELY PUT THE 'B', 'C', AND 'D' INBOARD MSIVS IN A HALF (A SIDE) TRIPPED CONDITION (AC SOLENOIDS DE-ENERGIZED) EVEN THOUGH THE LOGIC WAS RESET. DURING PERFORMANCE OF THE MSLRM SURVEILLANCE, WHEN THE 'B' LOGIC TRIP WAS INSERTED, THE DC SOLENOIDS ON THE INBOARD MSIVS DE-ENERGIZED CAUSING THE 'B', 'C' AND 'D' INBOARD VALVES TO GO CLOSED. IMMEDIATE CORRECTIVE ACTIONS WERE TO REPAIR THE CONNECTION AND PERFORM AN EXTENSIVE INSPECTION OF APPROPRIATE CONTROL ROOM PANELS FOR ADDITIONAL LOOSE CONNECTIONS. LONG TERM CORRECTIVE ACTION WILL INVOLVE PERIODIC INSPECTIONS OF APPROPRIATE PANELS. THIS EVENT HAD NO EFFECT ON THE SAFE OPERATION OF THE PLANT. SAFETY SYSTEMS RESPONDED AS DESIGNED IN RESPONSE TO THE SCRAM SIGNAL AND THE PLANT WAS QUICKLY BROUGHT TO A STABLE CONDITION.

[8] BEAVER VALLEY 1 DOCKET 50-334 LER 90-014
STEAM GENERATOR BLOWDOWN ISOLATION DUE TO AUXILIARY FEEDWATER PRESSURE SWITCH FAILURE.
EVENT DATE: 083190 REPORT DATE: 100190 NSSS: WE TYPE: PWR
VENDOR: BARKSDALE CONTROLS DIV

(NSIC 219642) ON 8/31/90 AT 1744 HOURS, BLOWDOWN FROM ALL STEAM GENERATORS ISOLATED WHEN BLOWDOWN ISOLATION VALVES TV-BD-100A,B,C AND SAMPLING ISOLATION VALVES TV-SS-117A,B,C CLOSED. OPERATORS, AFTER VERIFYING THERE WAS NO CONDITION REQUIRING THE VALVES TO BE SHUT, ATTEMPTED UNSUCCESSFULLY TO REOPEN THEM. FURTHER INVESTIGATION DETERMINED THAT THE VALVES WERE BEING MAINTAINED CLOSED DUE TO PROBLEMS WITH AUXILIARY FEEDWATER PRESSURE SWITCH PS-FW-157-3 (BARKSDALE MODEL B2T-M32SS). THIS SWITCH IS DESIGNED TO ISOLATE BLOWDOWN WHEN THE STEAM DRIVEN AUXILIARY FEEDWATER PUMP IS OPERATING. THE FAILURE OF THE SWITCH WAS CAUSED BY MOISTURE INDUCED CORROSION IN ITS ELECTRICAL CIRCUITRY. THIS SWITCH WAS REPLACED AND ALL ISOLATION VALVES REOPENED BY 1640 HOURS ON 9/1/90. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. THE PRESSURE SWITCH FAILED IN A CONSERVATIVE DIRECTION, CAUSING THE BLOWDOWN AND SAMPLE VALVES TO ISOLATE. REDUNDANT SAFETY-RELATED SIGNALS (CONTAINMENT ISOLATION PHASE A AND HIGH ENERGY LINE BREAK ISOLATION) EXIST AND ARE CAPABLE OF ISOLATING THE BLOWDOWN AND SAMPLING PENETRATIONS IN THE EVENT OF AN ACCIDENT, REGARDLESS OF THE SWITCH FAILURE MODE (UFSAR SECTIONS 5.3, "CONTAINMENT ISOLATION SYSTEM" AND 10.3.8.3, "SECONDARY VENTS AND DRAINS PERFORMANCE ANALYSIS").

[9] BEAVER VALLEY 2 DOCKET 50-412 LER 90-009
LETDOWN ISOLATION ON LOSS OF CONTAINMENT INSTRUMENT AIR PRESSURE.
EVENT DATE: 090290 REPORT DATE: 100190 NSSS: WE TYPE: PWR

(NSIC 219649) ON 9/2/90, THE SAFEGUARDS PROTECTION SYSTEM TRAIN A CIA GO TEST (OST) 2.1.11D) WAS PERFORMED AS SCHEDULED. THIS TEST ACTUATES INDIVIDUAL SOLID STATE PROTECTION SYSTEM RELAYS AND VERIFIES PROPER OPERATION OF AFFECTED COMPONENTS. IN TWO SEPARATE SECTIONS OF THE TEST, THE PROCEDURE ALIGNS A BACKUP AIR SUPPLY TO THE CONTAINMENT INSTRUMENT AIR SYSTEM AND SHUTS DOWN THE CONTAINMENT INSTRUMENT AIR COMPRESSORS WHILE TESTING COMPONENTS REQUIRED FOR COMPRESSOR OPERATION. IN ORDER TO AVOID UNNECESSARY CYCLING OF THE COMPRESSOR, IT WAS DECIDED TO NOT RESTART THE COMPRESSOR AFTER THE FIRST SHUTDOWN, BUT REMAIN

ON THE BACKUP AIR SUPPLY. HOWEVER, DUE TO INADEQUATE REVIEW, WHEN THE STEP TO RESTART THE COMPRESSOR WAS OMITTED, THE STEP TO REOPEN THE CONTAINMENT INSTRUMENT AIR ISOLATION VALVE WAS NOT PERFORMED. THIS RESULTED IN THE GRADUAL LOSS OF CONTAINMENT INSTRUMENT AIR PRESSURE. AFTER 25 MINUTES, THE LETDOWN ISOLATION VALVES FAILED CLOSED DUE TO LOW AIR PRESSURE. OPERATORS RESTORED CONTAINMENT AIR PRESSURE AND REOPENED THE LETDOWN ISOLATION VALVES IN ACCORDANCE WITH PROCEDURES. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. ALL AIR CONTROLLED COMPONENTS ARE DESIGNED TO GO TO A FAIL-SAFE CONDITION ON LOSS OF AIR PRESSURE (REFERENCE: UFSAR SECTION 9.3.1.3.3, "CONTAINMENT INSTRUMENT AIR SYSTEM SAFETY EVALUATION").

[10] BEAVER VALLEY 2 DOCKET 50-412 LER 90-010
LETDOWN ISOLATION DUE TO LOW PRESSURIZER WATER LEVEL.
EVENT DATE: 090490 REPORT DATE: 100390 NSSS: WE TYPE: PWR

(NSIC 219761) ON 9/4/90 AT 0900 HOURS, WITH THE UNIT IN HOT STANDBY (OPERATING MODE 3), A PLANT COOLDOWN WAS COMMENCED TO SUPPORT THE UPCOMING REFUELING OUTAGE. THE REACTOR COOLANT SYSTEM WAS AT 500 DEGREES F AND 2235 PSIG. PRESSURIZER LEVEL WAS NORMAL AT 22 PERCENT. AT 1007 HOURS, A LETDOWN SYSTEM ISOLATION OCCURRED DUE TO PRESSURIZER LEVEL DECREASING TO 14 PERCENT. THE PRESSURIZER LEVEL DECREASE WAS DUE TO THE PLANT RESPONSE AT NO-LOAD CONDITIONS (STEAM DUMPS, BYPASS FEEDWATER REGULATING VALVES AND REACTOR COOLANT SYSTEM PARAMETERS). PRESSURIZER LEVEL WAS RESTORED AND LETDOWN WAS REESTABLISHED. THERE WERE NO SAFETY IMPLICATIONS AS A RESULT OF THIS EVENT. THE LETDOWN ISOLATION FUNCTIONED AS DESIGNED IN RESPONSE TO A LOW PRESSURIZER LEVEL TO CONSERVE THE REACTOR COOLANT SYSTEM INVENTORY FOR DECAY HEAT REMOVAL CONSIDERATIONS DURING ACCIDENT SITUATIONS. A REVIEW OF THIS EVENT SHOWED THAT THE ADMINISTRATIVE COOLDOWN LIMITS WERE NOT EXCEEDED.

[11] BEAVER VALLEY 2 DOCKET 50-412 LER 90-011
CONTAINMENT PURGE ISOLATION DUE TO HIGH RADIATION SIGNAL.
EVENT DATE: 090690 REPORT DATE: 100890 NSSS: WE TYPE: PWR

(NSIC 219762) ON 9/6/90, THE CONTAINMENT BUILDING WAS BEING PURGED IN PREPARATION FOR REFUELING. AT 1719 HOURS, CONTAINMENT RADIATION MONITOR HVR*RQ104B SPIKED HIGH AND INITIATED AN AUTOMATIC PURGE ISOLATION. SUBSEQUENT SAMPLING BY RADIATION CONTROL TECHNICIANS SHOWED NO MEASURABLE AIRBORNE ACTIVITY IN CONTAINMENT. AFTER THE INITIAL SPIKE, THE MONITOR'S INDICATED ACTIVITY RETURNED TO ITS PRE-EVENT LEVEL. INVESTIGATION DETERMINED THAT THE MONITOR HAD SPIKED HIGH DUE TO AN ELECTRICAL DISTURBANCE RESULTING FROM A LIGHTNING STRIKE. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. NO ACTUAL RADIATION INCREASE OCCURRED. THE SYSTEM ACTUATED IN A CONSERVATIVE DIRECTION. ISOLATION OF CONTAINMENT PURGE IN RESPONSE TO A HIGH RADIATION SIGNAL IS DESCRIBED IN UNIT 2 UFSAR SECTION 9.4.7.3, "CONTAINMENT PURGE AIR SYSTEM".

[12] BEAVER VALLEY 2 DOCKET 50-412 LER 90-012
CONTAINMENT PURGE ISOLATION DUE TO IMPROPER RADIATION MONITOR RESTORATION.
EVENT DATE: 091390 REPORT DATE: 101290 NSSS: WE TYPE: PWR

(NSIC 219763) ON 9/13/90 AT 1000 HOURS, INSTRUMENT AND CONTROL (I&C) PERSONNEL AND RADIATION CONTROL (RADCON) PERSONNEL BEGAN A SOURCE CALIBRATION ON THE "B" TRAIN CONTAINMENT VENTILATION RADIATION MONITOR, 2HVR*RQI104B. THIS MONITOR HAD PREVIOUSLY BEEN TAKEN OUT OF SERVICE ON 9/11/90 DUE TO INCREASING BACKGROUND READINGS WITH NO CORRESPONDING INCREASE IN RADIOACTIVE CONCENTRATION. AT 1243 HOURS, THE CONTROL ROOM OPERATORS OBSERVED AN AUTOMATIC CLOSURE OF THE "B" TRAIN CONTAINMENT PURGE EXHAUST DAMPERS, 2HVR*MOD23B AND 25B, DUE TO A HIGH RADIATION SIGNAL FROM 2HVR*RQI104B. THE CLOSURE RESULTED FROM A COMMUNICATIONS ERROR BY RADCON PERSONNEL DURING THE RESTORATION OF THE MONITOR. THE CLOSURE OF THESE DAMPERS IS AN ENGINEERED SAFETY FEATURES SYSTEM ACTUATION, AS THESE DAMPERS PROVIDE A CONTAINMENT PURGE ISOLATION FUNCTION IN THE EVENT OF HIGH RADIATION DURING REFUELING. THE RADIATION MONITOR WAS SUBSEQUENTLY RESTORED TO SERVICE IN THE CORRECT MANNER AND CONTAINMENT VENTILATION WAS RETURNED TO NORMAL STATUS. THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC AS A RESULT OF THIS EVENT. THE

HIGH RADIATION SIGNAL RESULTED FROM A PERSONNEL ERROR AND NOT FROM ACTUAL RADIATION LEVELS. THE CONTAINMENT VENTILATION SYSTEM ISOLATED AS DESIGNED TO CONTAIN ANY RADIOACTIVITY INSIDE CONTAINMENT.

[13] BEAVER VALLEY 2 DOCKET 50-412 LER 90-015
CONTAINMENT LINER TEST CHANNEL VENTS FOUND UNPLUGGED.
EVENT DATE: 091390 REPORT DATE: 101590 NSSS: WE TYPE: PWR

(NSIC 219768) ON 9/13/90, THE CONTAINMENT STRUCTURAL INTEGRITY INSPECTION WAS PERFORMED. THIS INSPECTION DISCOVERED TWENTY-FIVE CONTAINMENT LINER WELD TEST CHANNEL VENT PLUGS MISSING. TWELVE OF THE MISSING PLUGS WERE FOR CHANNELS ON THE CONTAINMENT WALL AND THIRTEEN OF THE PLUGS WERE FOR CHANNELS UNDER THE CONTAINMENT'S CONCRETE FLOOR. A PREVIOUSLY PERFORMED ENGINEERING ANALYSIS OF THE TEST CHANNELS WAS REVIEWED TO VERIFY THAT THE MISSING PLUGS WOULD NOT AFFECT CONTAINMENT STRUCTURAL INTEGRITY. BORESCOPE INSPECTIONS OF THE AFFECTED CHANNELS WERE PERFORMED. SOME CHANNELS WERE PARTIALLY INSPECTED OR INACCESSIBLE DUE TO VENT TUBING CONFIGURATION AND/OR BLOCKAGE. SAMPLES WERE TAKEN OF LIQUID FOUND IN TWO OF THE CHANNELS. THE LIQUID WAS FOUND TO BE A SODIUM HYDROXIDE SOLUTION, LEFT FROM CLEANING DURING CONSTRUCTION. AN ANALYSIS OF THE POTENTIAL CORROSION RATE THAT WOULD OCCUR WITH THIS SOLUTION PRESENT WAS PERFORMED AND VERIFIED TO BE BOUNDED BY THE TEST CHANNEL ANALYSIS. BASED ON THIS ANALYSIS, THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. TECHNICAL SPECIFICATION CHANGES HAVE BEEN INITIATED TO ALLOW CONTINUED OPERATION IN THIS CONFIGURATION.

[14] BEAVER VALLEY 2 DOCKET 50-412 LER 90-014
CONTAINMENT PURGE ISOLATION DURING UPPER INTERNALS LIFT.
EVENT DATE: 091690 REPORT DATE: 101690 NSSS: WE TYPE: PWR

(NSIC 219764) ON 9/16/90, PREPARATIONS WERE UNDERWAY TO LIFT THE UPPER INTERNALS TO ALLOW SUBSEQUENT FUEL MOVEMENT. AT 1204 HOURS, THE UPPER INTERNALS LIFT WAS STARTED. THE LIFT WAS BEING PERFORMED BY MAINTENANCE WITH THE SUPPORT OF RADIATION CONTROL (RADCON) PERSONNEL. RADCON PERSONNEL WERE STATIONED AT THE LOCAL INDICATION FOR THE CONTAINMENT VENTILATION RADIATION MONITORS 2HVR*RQI104A & B. THEY WERE INSTRUCTED TO OBSERVE THE MONITORS RESPONSE DURING THE EVOLUTION, AS AN INCREASE IN THE INDICATIONS WAS EXPECTED. AT 1220 HOURS, AN INCREASE ON 2HVR*RQI104A OCCURRED. BEFORE RADCON COULD ALERT MAINTENANCE PERSONNEL, THE MONITOR ALARMED AND CAUSED AN AUTOMATIC CLOSURE OF THE CONTAINMENT PURGE SUPPLY AND DISCHARGE ISOLATION DAMPERS. THE CAUSE FOR THE ACTUATION WAS AN INCREASE IN THE BACKGROUND RADIATION, CAUSING THE MONITOR TO ALARM HIGH. THE INCREASED BACKGROUND RADIATION WAS THE RESULT OF DIRECT "SHINE" FROM THE UPPER INTERNALS, AS THE MONITOR IS IN A DIRECT LINE WITH THE INTERNALS. RADCON PERSONNEL VERIFIED NO ABNORMAL AIRBORNE RADIOACTIVITY. THE HIGH ALARM WAS RESET AND CONTAINMENT VENTILATION WAS RESTORED AT 1237 HOURS. THERE WERE NO SAFETY IMPLICATIONS AS A RESULT OF THIS EVENT. THE HIGH RADIATION SIGNAL RESULTED FROM INCREASED BACKGROUND READINGS FROM DIRECT "SHINE" AND NOT FROM AN INCREASE IN ACTUAL AIRBORNE RADIATION LEVELS.

[15] BRAIDWOOD 1 DOCKET 50-456 LER 90-015
MISSED AUXILIARY BUILDING VENT STACK GRAB SAMPLE DUE TO PERSONNEL ERROR.
EVENT DATE: 082790 REPORT DATE: 092490 NSSS: WE TYPE: PWR

(NSIC 219627) THE UNIT 1 AUXILIARY BUILDING VENT STACK EFFLUENT (ABVSE) RADIATION MONITOR HAD BEEN DECLARED INOPERABLE ON 8/9/90. AS PART OF THE ACTION STATEMENT, A GRAB SAMPLE WAS REQUIRED EVERY 12 HOURS. RADIATION PROTECTION DEPARTMENT (RP) HAD INITIATED A FREQUENCY OF 1 GRAB SAMPLE AT 0400, 1200, AND 2000 HOURS ON A DAILY BASIS. AT 0400 ON 8/27/90 THE ABVSE GRAB SAMPLE WAS TAKEN IN ACCORDANCE WITH THE SCHEDULE. AT 0700 DURING THE RP SHIFT SUPERVISOR (RPSS) REVIEW OF THE DUTY ASSIGNMENT SHEET, THE RPSS LOST COGNIZANCE OF THE ABVSE GRAB SAMPLE REQUIREMENT AND FAILED TO ENTER IT ON THE SHEET. AT 1430 THE DAY SHIFT RPSS WAS RELIEVED BY THE AFTERNOON SHIFT RPSS. THE NEED TO PERFORM THE 2000 ABVSE GRAB SAMPLE WAS DISCUSSED DURING TURNOVER. THE DAY SHIFT RPSS ASSUMED THAT THE 1200 SAMPLE HAD BEEN TAKEN. AT 1940 THE 2000 ABVSE GRAB SAMPLE WAS TAKEN. AT 2100 THE AFTERNOON SHIFT RPSS IDENTIFIED THAT THERE WAS NO RECORD OF THE 1200 ABVSE

GRAB SAMPLE BEING SUBMITTED FOR ANALYSIS. AT 2245 AFTER DISCUSSIONS WITH THE DAY SHIFT RPSS AND RP TECHNICIAN, IT WAS CONCLUDED THAT A 1200 ABVSE GRAB SAMPLE HAD NOT BEEN TAKEN. THE CAUSES OF THE EVENT WERE PERSONNEL ERROR AND DEFICIENT WORK PRACTICES. TRAINING HAS BEEN PROVIDED, PROGRAMS WILL BE MODIFIED, AND THE STATION IS REVIEWING THE NON-ROUTINE SURVEILLANCE PROCESS. PREVIOUS CORRECTIVE ACTIONS ARE NOT APPLICABLE.

[16] BRAIDWOOD 1 DOCKET 50-456 LER 90-016
SPURIOUS AUTO START OF THE AUXILIARY BUILDING INACCESSIBLF FILTER PLENUM "A" CHARCOAL BOOSTER FAN DUE TO UNKNOWN CAUSE.
EVENT DATE: 082890 REPORT DATE: 092490 NSSS: WE TYPE: PWR

(NSIC 219662) AT 0537 ON 8/28/90, THE AUXILIARY BUILDING INACCESSIBLE FILTER PLENUM A CHARCOAL BOOSTER FAN OB, OVA03CB, SPURIOUSLY AUTO STARTED. THE NUCLEAR STATION OPERATOR (NSO) EXAMINED THE CONTROL SWITCH POSITION AND IDENTIFIED THAT THE SWITCH WAS IN THE AFTER TRIP POSITION. THIS INDICATED THAT THE FAN HAD NOT BEEN STARTED FROM THE MAIN CONTROL ROOM. AN EQUIPMENT ATTENDANT (EA) WAS DISPATCHED TO A LOCAL PANEL THAT CONTAINED A CONTROL SWITCH AND A TRANSFER SWITCH WHICH SELECTED THE CONTROL POINT FOR THE OVA03CB AS EITHER THE MAIN CONTROL ROOM OR THAT PANEL. THE FAN WAS SELECTED FOR CONTROL FROM THE MAIN CONTROL ROOM. THE NSO INSPECTED THE AUTOMATIC ACTUATION RELAYS ON BOTH UNIT 1 AND UNIT 2. THE RELAYS, WHICH ARE LATCHING STYLE RELAYS THAT ACTUATE AS A FUNCTION OF A SAFETY INJECTION, WERE FOUND IN THE "UNLATCHED" STATE. ADDITIONALLY, NO OTHER COMPONENTS ASSOCIATED WITH THE RELAYS HAD OPERATED ON EITHER UNIT. THE NSO SECURED THE OVA03CB AND THE FAN DID NOT RESTART WHEN THE SWITCH WAS RELEASED TO THE AFTER TRIP POSITION. THE CAUSE OF THIS EVENT IS UNDETERMINED. THE FAN CIRCUITRY AND BREAKER WERE CHECKED AND FOUND TO BE FUNCTIONING PROPERLY. RELAY TESTS WERE PERFORMED ON BOTH UNITS AND ALL COMPONENTS FUNCTIONED AS DESIGNED. PREVIOUS CORRECTIVE ACTIONS ARE NOT APPLICABLE TO THIS EVENT.

[17] BRAIDWOOD 1 DOCKET 50-456 LER 90-017
INADEQUATE THRUST CAPABILITY OF AUXILIARY FEEDWATER VALVE OPERATORS DUE TO PRE-SERVICE DESIGN DEFICIENCY.
EVENT DATE: 090690 REPORT DATE: 100590 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BRAIDWOOD 2 (PWR)

(NSIC 219767) AT 1510 ON 9/6/90 BRAIDWOOD STATION WAS NOTIFIED BY NUCLEAR ENGINEERING DEPARTMENT THAT A DISCREPANCY HAD BEEN IDENTIFIED REGARDING THE AUXILIARY FEEDWATER (AF) DISCHARGE HEADER ISOLATION VALVES (AF013) BASED ON THE RESULTS OF CALCULATIONS THAT HAD BEEN PERFORMED IN RESPONSE TO NRC GENERIC LETTER 89-10, IT HAD BEEN CONCLUDED THAT THE VALVE OPERATORS POTENTIALLY COULD NOT DEVELOP ADEQUATE THRUST. THIS COULD POTENTIALLY AFFECT THE ASSUMPTIONS OF THE UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) FOR A MAIN STEAMLINE BREAK INSIDE CONTAINMENT. THE UFSAR ASSUMED THAT THE AF FLOW TO A FAULTED STEAM GENERATOR WOULD BE ISOLATED WITHIN TEN MINUTES. THE METHODOLOGY SPECIFIED IN THE EMERGENCY PROCEDURES FOR THIS FUNCTION DEPENDED EXCLUSIVELY ON THE USE OF THE ASSOCIATED AF013 VALVES FOR THE AFFECTED SG. THE EVENT WAS EVALUATED FOR TECH SPEC APPLICABILITY AND REPORTABILITY. IT WAS CONCLUDED THAT THE DISCREPANCY DID NOT HAVE ANY TECH SPEC APPLICABILITY BUT THAT IT COULD CONSERVATIVELY BE CLASSIFIED AS A CONDITION THAT ALONE COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF STRUCTURES OR SYSTEMS THAT ARE NEEDED TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT. THE CAUSES OF THE EVENT WERE PRESERVICE DESIGN DEFICIENCY AND PROCEDURAL DEFICIENCY. THE VALVE OPERATORS WILL BE MODIFIED. THE AFFECTED PROCEDURES HAVE BEEN REVISED. NO PREVIOUS OCCURRENCES.

[18] BRUNSWICK 1 DOCKET 50-325 LER 90-006 REV 02
UPDATE ON HYDRAULIC PERTURBATION OF REACTOR VESSEL LEVEL INSTRUMENTATION.
EVENT DATE: 042690 REPORT DATE: 100490 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)

(NSIC 219734) UNIT 1 AND UNIT 2 WERE OPERATING AT 100% POWER. AT 1603 ON 4-26-90 WHILE PERFORMING 1MST-RSDP21Q, A HYDRAULIC PERTURBATION OCCURRED ON THE VARIABLE LEG OF INSTRUMENTATION USED TO SENSE REACTOR WATER LEVEL. THIS CAUSED THE

REACTOR WATER CLEANUP SYSTEM (G31-F004) TO ISOLATE, THE STANDBY GAS TREATMENT SYSTEM TO INITIATE, AND THE REACTOR BUILDING VENTILATION SYSTEM TO ISOLATE. THE EVENT WAS CAUSED BY THE INTRODUCTION OF AIR INTO THE TRANSMITTER DURING THE INITIAL FLUSHING OF THE INSTRUMENT. PROCEDURE REVISIONS ARE BEING INITIATED TO CHANGE THE VALVING PROCESS TO ALLEVIATE CONCERNS WITH POTENTIAL AIR ENTRAPMENT DURING THE VALVING EVOLUTION. THE REVISIONS ARE EXPECTED TO BE COMPLETED BY 10/30/90. OTHER PERTURBATIONS HAVE BEEN REPORTED UNDER LERS 2-89-017, 1-87-017, AND 2-86-020.

[19] BRUNSWICK 1 DOCKET 50-325 LER 90-014
RPS BUS A TRIP WHEN EPA-2 TRIPPED ON UNDER FREQUENCY DUE TO A CIRCUIT BOARD FAILURE.
EVENT DATE: 090790 REPORT DATE: 100490 NSSS: GE TYPE: BWR

(NSIC 219735) ON 9/7/90, AT 1407, WITH UNIT 1 AT 87% POWER, THE REACTOR PROTECTION SYSTEM (RPS) BUS A TRIPPED WHEN THE ELECTRICAL PROTECTION ASSEMBLY (EPA) BREAKER NUMBER 2 (EPA-2) OPENED ON UNDERFREQUENCY (UF) AS INDICATED BY A LIGHT ON THE BREAKER. ISOLATIONS AND ACTUATIONS EXPECTED ON THE LOSS OF RPS BUS A OCCURRED PER DESIGN. AN INSPECTION OF THE RPS MOTOR GENERATOR SET AND OF EPA-1 INDICATED THAT THEY WERE OPERATING PROPERLY. AT 1416, RPS BUS A WAS TRANSFERRED TO THE ALTERNATE FEED AND THE ISOLATIONS AND ACTUATIONS WERE RESET. INSTRUMENTATION AND CONTROL PERSONNEL DETERMINED THAT THE EPA-2 TRIP WAS CAUSED BY A FAILURE OF THE CIRCUIT BOARD WITHIN THE BREAKER HOWEVER, EXACTLY WHAT FAILED IN THE CIRCUIT BOARD IS NOT KNOWN. THE CIRCUIT BOARD HAD BEEN INSTALLED WITHIN THE PREVIOUS MONTH. THE CIRCUIT BOARD HAS BEEN SENT TO GENERAL ELECTRIC FOR A FAILURE ANALYSIS. ON 9/19/90, AT 2100, THE RPS A BUS WAS RETURNED TO ITS NORMAL FEED AFTER REPLACING THE DEFECTIVE CIRCUIT BOARD ON 9/14/90, AND LEAVING RPS EPA-2 ENERGIZED BUT NOT TIED TO THE A BUS FOR FIVE DAYS WITHOUT ANY PROBLEMS. A SUPPLEMENT TO THIS REPORT WILL BE SUBMITTED BY 11/25/90. THIS EVENT HAD MINIMAL SAFETY SIGNIFICANCE IN THAT THE SYSTEMS FUNCTIONED PER DESIGN, ISOLATIONS AND ACTUATIONS WERE IN THE CONSERVATIVE DIRECTION AND NO UNEXPECTED RESPONSES WERE INCURRED. SIMILAR EVENTS: LERS 2-89-012, 20 & 21; 2-88-13; 1-87-09 & 21.

[20] BRUNSWICK 2 DOCKET 50-324 LER 90-004 REV 02
UPDATE ON MANUAL REACTOR SCRAM DUE TO FAILURE OF SAFETY RELIEF VALVE "G" TO CLOSE DURING STARTUP TESTING.
EVENT DATE: 031390 REPORT DATE: 100190 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 219636) AT 0536 ON MARCH 13, 1990, A MANUAL SCRAM WAS INITIATED DUE TO THE FAILURE OF SAFETY/RELIEF VALVE (SRV) B21-F013G TO CLOSE DURING STARTUP TESTING; REACTOR POWER WAS APPROXIMATELY 7% AND REACTOR PRESSURE WAS APPROX. 250 PSIG. THE 11 UNIT SRV'S WERE BEING CYCLED IN ACCORDANCE WITH PLANT PROCEDURES TO VERIFY OPERABILITY PER TECH SPEC 3.5.2. TEN OF THE ELEVEN SRV'S HAD BEEN SUCCESSFULLY TESTED PRIOR TO THIS FAILURE. A NORMAL SCRAM RECOVERY WAS CONDUCTED PER PLANT PROCEDURES AND NO AUTOMATIC SAFETY ACTUATIONS OR ISOLATIONS OCCURRED. REACTOR VESSEL LEVEL WAS MAINTAIN USING THE FEEDWATER SYSTEM. THE INVESTIGATION DETERMINED THAT THE SOLENOID VALVE WHICH ALLOWS REMOTE MANUAL OPERATION OF B21-F013G WAS INOPERABLE. THE SOLENOID WOULD ALLOW THE SRV TO BE OPENED, BUT WOULD NOT ALLOW TIMELY CLOSURE OF THE SRV. THE SOLENOID VALVE WAS REPLACED. THE UNIT RETURNED TO THE REQUIRED TESTING CONDITIONS AND THE SRV WAS SUCCESSFULLY TESTED. THE SOLENOID VALVE WAS SENT TO WYLE LABORATORY AND THE ROOT CAUSE WAS DETERMINED TO BE FAILURE OF THE SOLENOID DISC TO PROPERLY REALIGN WITH ITS SEAT AFTER DE-ENERGIZATION. A POTENTIAL CAUSE OF THE FAILURE TO RESEAT IS "DIRT" WHICH WAS EMBEDDED IN THE RUBBER PAD LOCATED ON THE BACKSEAT. AN INVESTIGATION INTO THE SOURCE OF THE MATERIAL IS CONTINUING. A SUPPLEMENT WILL BE ISSUED BY MARCH 19 1991, TO PROVIDE THE RESULTS OF THIS INVESTIGATION.

[21] BRUNSWICK 2 DOCKET 50-324 LER 90-006 REV 01
UPDATE ON HYDRAULIC PERTURBATION OF REACTOR VESSEL LEVEL INSTRUMENTATION.
EVENT DATE: 060490 REPORT DATE: 100490 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 1 (BWR)

(NSIC 219731) UNIT 2 WAS IN COLD SHUTDOWN(MODE 4). AT 1603 ON 6-4-90, WHILE VENTING A REACTOR PRESSURE INSTRUMENT FOLLOWING MAINTENANCE, A HYDRAULIC PERTURBATION OF A SHARED REFERENCE LEG RESULTED IN ISOLATION OF THE REACTOR WATER CLEANUP SYSTEM, AN AUTOMATIC INITIATION OF THE STANDBY GAS TREATMENT SYSTEM, AND AN ISOLATION OF THE REACTOR BUILDING VENTILATION SYSTEM. THIS EVENT POSES NO SAFETY SIGNIFICANCE SINCE THE AFFECTED SYSTEMS FUNCTIONED AS REQUIRED AND THE UNIT WAS IN COLD SHUTDOWN. MAINTENANCE GENERATED AN INTERIM POLICY WHICH PROVIDED FOR A DATA GATHERING PHASE TO DETERMINE LONG TERM CORRECTIVE ACTIONS. AS A RESULT, IT HAS BEEN DETERMINED THAT CAUTION NOTES SHOULD BE ADDED TO THE PROCESS INSTRUMENT CALIBRATION PROCEDURE FOR THE VALVING EVOLUTION. THE REVISIONS ARE EXPECTED TO BE COMPLETED BY OCTOBER 30, 1990. OTHER PERTURBATIONS HAVE BEEN REPORTED IN LERS 1-90-006, 2-89-017, 1-87-017, AND 2-86-020.

[22] BRUNSWICK 2 DOCKET 50-324 LER 90-009
REACTOR PROTECTION SYSTEM TRIP WHILE PERFORMING A SURVEILLANCE TEST ON CONDENSER LOW VACUUM INSTRUMENTATION AND ISOLATION LOGIC.
EVENT DATE: 081990 REPORT DATE: 091890 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 1 (BWR)
VENDOR: FISHER CONTROLS CO.
GENERAL ELECTRIC CO.
TARGET ROCK CORP.

(NSIC 219542) WHILE TESTING THE MAIN CONDENSER LOW VACUUM INSTRUMENTATION AND ISOLATION LOGIC, AN ISOLATION SIGNAL WAS GENERATED WHICH CLOSED THE MAIN STEAM ISOLATION VALVES. CLOSURE OF THE MAIN STEAM ISOLATION VALVES INITIATED A UNIT 2 REACTOR SCRAM AT 2154 ON 8/19/90. REACTOR PRESSURE PEAKED AT APPROXIMATELY 1133 PSIG AND THE MINIMUM WATER LEVEL REACHED WAS APPROXIMATELY 112 INCHES. NO SAFETY LIMITS WERE EXCEEDED IN THE PLANT RESPONSE TO THE LEVEL AND PRESSURE TRANSIENTS. THE OPERATING CREW WAS ABLE TO CONTROL THE PLANT BY USING REDUNDANT EQUIPMENT OR ALTERNATE METHODS. THE SAFETY RELIEF VALVE OPENING SEQUENCE AND ACTUATION PATTERN WERE QUESTIONED. THE ACTUATION PATTERN IS NOT A CONCERN. HOWEVER, SAFETY RELIEF VALVE, B21-F013C (SETPOINT 1105 PSIG) DID NOT OPEN. THE PILOT VALVE ASSEMBLIES WERE REPLACED ON SAFETY RELIEF VALVES B21-F013A, B21-F013C, B21-F013G, B21-F013H, AND B2L-F013K. SIMILAR PROBLEMS ENCOUNTERED DURING THIS SCRAM HAVE BEEN REPORTED IN LERS 2-88-005, 2-88-019, 2-87-004, 1-87-011, 2-86-001, 2-86-013, 2-86-017, 2-85-003, 2-85-011, AND 1-85-033.

[23] BRUNSWICK 2 DOCKET 50-324 LER 90-011
REACTOR WATER CLEAN-UP SYSTEM GROUP III ISOLATION ON SYSTEM HIGH DIFFERENTIAL FLOW ALARM.
EVENT DATE: 083090 REPORT DATE: 092890 NSSS: GE TYPE: BWR

(NSIC 219637) ON 8/30/90 AT 0204 THE REACTOR WATER CLEANUP (RWCU) SYSTEM (G31) WAS MANUALLY ISOLATED FOLLOWING THE RECEIPT OF THE "RWCU LEAK HI' AND "RWCU LEAK HI HI" ANNUNCIATORS. AN AUTOMATIC ISOLATION SIGNAL WAS RECEIVED FOLLOWING THE OPERATORS ACTION TO CLOSE THE G31-F001 AND G31-F004 VALVES PRIOR TO THE VALVES REACHING THE FULL CLOSED POSITION. AFTER VERIFYING THAT AN ACTUAL LEAK HAD NOT OCCURRED THE RWCU SYSTEM WAS PLACED IN SERVICE. SMALL REJECT FLOW OSCILLATIONS WERE OBSERVED WITH THE RWCU LEAK DIFFERENTIAL FLOW INSTRUMENTATION SHOWING DOWNSCALE. THE RWCU DIFFERENTIAL FLOW INSTRUMENTATION WAS THEN DECLARED INOPERABLE AND THE RWCU SYSTEM ISOLATED. SUBSEQUENT MAINTENANCE INCLUDED FILLING AND VENTING THE 3 DELTA FLOW INSTRUMENT TRANSMITTERS. TECHNICIANS IDENTIFIED AIR IN THE SENSING LINES OF THE 2-G31-FT-N012 INSTRUMENT. THE RWCU SYSTEM WAS RETURNED TO SERVICE AND THE RWCU DELTA FLOW INSTRUMENTATION WAS DECLARED OPERABLE. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL. AN INVESTIGATION INTO THE AIR INTRUSION WILL BE CONDUCTED AND THE RESULTS WILL BE REPORTED IN A SUPPLEMENT TO THIS LER DATED 12/05/90.

[24] BRUNSWICK 2 DOCKET 50-324 LER 90-012
REACTOR SCRAM DURING REACTOR START UP.
EVENT DATE: 083090 REPORT DATE: 092890 NSSS: GE TYPE: BWR
VENDOR: FISHER CONTROLS CO.

(NSIC 219638) ON AUGUST 30, 1990, UNIT 2 REACTOR START-UP WAS IN PROGRESS. THE REACTOR WAS AT APPROXIMATELY 8% POWER AND 300 PSIG. THE EMERGENCY CORE COOLING SYSTEMS WERE OPERABLE IN STANDBY LINE UP EXCEPT FOR THE HIGH PRESSURE COOLANT INJECTION SYSTEM WHICH WAS INOPERABLE DUE TO NOT PERFORMING SURVEILLANCE TEST PT-09.2. AT 1656 THE START-UP LEVEL CONTROL VALVE (SULCV) FAILED CLOSED RESULTING IN A LEVEL TRANSIENT. AT 1657 THE REACTOR PROTECTION SYSTEM (RPS) LOW LEVEL #1 SETPOINT (165") WAS REACHED CAUSING A REACTOR SCRAM. PRIMARY CONTAINMENT ISOLATION SYSTEM GROUPS 2, 6 AND 8 ALSO RECEIVED AN ISOLATION SIGNAL AND ACTUATED PER DESIGN. SCRAM RECOVERY WAS IN ACCORDANCE WITH THE EMERGENCY FLOWCHARTS AND PROCEDURES. APPROXIMATELY 20 MINUTES AFTER THE SCRAM LEVEL WAS STABILIZED. AN INVESTIGATION INTO THE FAILURE OF THE SULCV IS CONTINUING. THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL. LEVEL WAS RECOVERED WITHOUT THE NEED FOR SAFETY SYSTEM INJECTION AND THE UNIT IS DESIGNED FOR A LEVEL TRANSIENT FROM FULL POWER. A SUPPLEMENT TO THIS WILL BE ISSUED BY NOVEMBER 1, 1990.

[25] BRUNSWICK 2 DOCKET 50-324 LER 90-014
RWCU ISOLATION ON DIFFERENTIAL FLOW DUE TO A BLOWN FUSE.
EVENT DATE: 090890 REPORT DATE: 100490 NSSS: GE TYPE: BWR

(NSIC 219733) ON SEPTEMBER 8, 1990, AT 1711, WITH UNIT 2 AT 95% POWER, THE INBOARD AND OUTBOARD REACTOR WATER CLEAN-UP (RWCU) ISOLATION VALVES AUTOMATICALLY CLOSED ON A DIFFERENTIAL FLOW HIGH, HIGH SIGNAL. AN INVESTIGATION DETERMINED THAT THE EVENT WAS CAUSED BY A BLOWN FUSE IN THE POWER SUPPLY TO A SQUARE ROOT CONVERTER IN THE FLOW CIRCUITRY AND THAT AN ACTUAL HIGH DIFFERENTIAL FLOW CONDITION DID NOT EXIST. THE CAUSE OF THE BLOWN FUSE HAS NOT BEEN DETERMINED AND AN INVESTIGATION IS CONTINUING. THROUGHOUT THE EVENT THE EMERGENCY CORE COOLING SYSTEMS WERE OPERABLE AND IN STANDBY READINESS. PAST SIMILAR EVENTS HAVE BEEN REPORTED IN LERS 2-90-06 & 11; 2-89-03, 07, 12, 16; 2-88-03 & 10. THIS EVENT HAD MINIMAL SAFETY SIGNIFICANCE IN THAT A HIGH DIFFERENTIAL FLOW DID NOT ACTUALLY OCCUR AND THE EQUIPMENT OPERATED IN ACCORDANCE WITH ITS DESIGN. THE FUSE WAS REPLACED AND THE RWCU SYSTEM WAS RETURNED TO SERVICE. A SUPPLEMENT TO THIS REPORT WILL BE SUBMITTED BY NOVEMBER 25, 1990.

[26] BYRON 1 DOCKET 50-454 LER 90-012
AUXILIARY FEEDWATER DISCHARGE ISOLATION VALVES DESIGN INADEQUACY.
EVENT DATE: 090690 REPORT DATE: 100590 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 2 (PWR)

(NSIC 219765) ON 9/6/90, WITH UNIT 1 IN MODE [illegible], 100% REACTOR POWER AND UNIT 2 IN MODE 5, IT WAS DETERMINED BY THE NUCLEAR ENGINEERING DEPARTMENT THAT THE AUXILIARY FEEDWATER (AF) (BA) DISCHARGE HEADER ISOLATION VALVES 1/2 AF013 COULD NOT BE RELIED UPON TO FULLY CLOSE DURING A MAIN STEAM LINE BREAK INSIDE CONTAINMENT ACCIDENT. BASED ON THE RESULTS OF CALCULATIONS THAT HAD BEEN PERFORMED IN RESPONSE TO NRC GENERIC LETTER 89-10, IT HAD BEEN CONCLUDED THAT THE VALVE OPERATORS POTENTIALLY COULD NOT DEVELOP ADEQUATE THRUST. THIS COULD POTENTIALLY AFFECT THE ASSUMPTIONS OF THE UPDATED FSAR FOR A MAIN STEAMLINE BREAK INSIDE CONTAINMENT. UFSAR ASSUMED THAT THE AF FLOW TO A FAULTED STEAM GENERATOR (SG) WOULD BE ISOLATED WITHIN TEN MINUTES. METHODOLOGY SPECIFIED IN THE EMERGENCY PROCEDURES FOR THIS FUNCTION DEPENDED EXCLUSIVELY ON THE USE OF THE ASSOCIATED AF013 VALVES FOR THE AFFECTED SG. EMERGENCY PROCEDURES FOR UNIT 1 WERE IMMEDIATELY TEMPORARILY REVISED TO PROVIDE ALTERNATIVE MEASURES TO ISOLATE AF FLOW IF AF013 VALVES FAILED TO FULLY CLOSE. AN ON-SITE REVIEW (90-221) WAS COMPLETED ON 9/7/90 WHICH CONCLUDED THAT THE AF SYSTEM CONTINUED TO BE OPERABLE AND THAT APPLICABLE TECH SPECS AND UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) BASES CONTINUE TO BE SATISFIED. RECENTLY, MOTOR OPERATED VALVE PERFORMANCE MODELS HAVE BEEN IMPROVED TO CONSIDER ADDITIONAL FACTORS WHICH COULD DEGRADE VALVE PERFORMANCE.

[27] BYRON 2 DOCKET 50-455 LER 90-005
PRE-OUTAGE MODIFICATION WORK INITIATED WITHOUT PROPER OPERABILITY REVIEW DUE TO PERSONNEL ERROR.
EVENT DATE: 081190 REPORT DATE: 091490 NSSS: WE TYPE: PWR

(NSIC 219559) ON 8/17/90 AT APPROX. 1500, WITH UNIT 2 IN MODE 1 AT 49% POWER, IT WAS DISCOVERED THAT SUPPORTS ON THE 2A AUXILIARY FEEDWATER PUMP (AF)(BA) PIPING HAD BEEN WORKED ON WITHOUT OPERATING DEPARTMENT CONCURRENCE. LIMITING CONDITION FOR OPERATION ACTION REQUIREMENT 2BOS 4.10-1A WAS IMMEDIATELY ENTERED AND AN ENGINEERING EVALUATION WAS INITIATED. THE CURRENT CONDITION OF THE SYSTEM WAS FOUND ACCEPTABLE AND THE LCOAR WAS EXITED. ON 8/23/90 DURING A WALKDOWN TO VERIFY ASSUMPTIONS MADE IN DETERMINING OPERABILITY DURING THE SUPPORT WORK, SCAFFOLDING WAS DISCOVERED SUPPORTED BY THE AF LINE IN QUESTION. THE SCAFFOLDING WAS IMMEDIATELY REMOVED. THE LOAD BEARING NUTS ON ONE OF THE AFFECTED SUPPORTS WERE ALSO FOUND LOOSE AT THIS TIME. THEY WERE PROMPTLY TIGHTENED AND TESTED. THE ROOT CAUSE OF THE SUPPORTS BEING AFFECTED WAS A PERSONNEL ERROR ON THE PART OF A CONTRACTED MODIFICATION FOREMAN. THE CONTRACTOR DID NOT REQUEST AN OUT-OF-SERVICE PRIOR TO PERFORMING WORK ON EXISTING OPERATIONAL EQUIPMENT. THE ROOT CAUSE OF THE SCAFFOLDING BEING IMPROPERLY ATTACHED TO THE SYSTEM IS DUE TO A LACK OF CONTRACTOR AWARENESS OF THE IMPACT ON THE SYSTEM. THE LOOSE FASTENERS ARE SUSPECTED TO BE A RESULT OF THE WORK SEQUENCE. AS PREVENTIVE ACTION, THE DAILY CONSTRUCTION WORK AUTHORIZATION SHEET HAS BEEN FORMALIZED AND WILL SPECIFY THE SCOPE OF THE WORK STEP BY STEP DURING NON-OUTAGE PERIODS.

[28] BYRON 2 DOCKET 50-455 LER 90-006
INADVERTENT TRAIN A SAFETY INJECTION SIGNAL DUE TO MISCOMMUNICATION AND PROCEDURAL DEFICIENCY.
EVENT DATE: 090390 REPORT DATE: 100390 NSSS: WE TYPE: PWR
VENDOR: GOULDS CO.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 219766) ON 9/3/90 AT APPROXIMATELY 0805, WITH UNIT 2 IN MODE 5 (COLD SHUTDOWN) 2BOS 3.2.1.1.A-1, "UNIT TWO TRAIN A MANUAL SAFETY INJECTION INITIATION AND MANUAL PHASE A INITIATION SURVEILLANCE," WAS BEING PERFORMED PER THE REFUELING OUTAGE SCHEDULE. AFTER PERFORMING THE NORMAL SAFETY INJECTION (SI) (JE), IT WAS NOTED THAT THE 2C REACTOR CONTAINMENT FAN COOLER (RCFC) LOW SPEED FAN BREAKER DID NOT CLOSE. ATTEMPTS TO CLOSE THE BREAKER WERE UNSUCCESSFUL. AT 0820, THE 480 VOLT BUS THAT FEEDS THE BREAKER WAS DE-ENERGIZED TO ALLOW REMOVAL OF THE BREAKER. AT 0850, WHILE STRIPPING THE BUS OF ITS ALTERNATING CURRENT LOAD, INSTRUMENT INVERTERS 211 AND 213 WERE DE-ENERGIZED DUE TO A COMMUNICATIONS BREAKDOWN. WHEN THE INSTRUMENT BUSES WERE DE-ENERGIZED, THE PRESSURIZER PRESSURE LOW SI AND STEAMLINE PRESSURE LOW SI BLOCKS WERE LOST ON TRAIN A. THE UNIT REACTOR OPERATOR WAS UNAWARE THAT THE BLOCKS HAD BEEN LOST AND THE SURVEILLANCE DID NOT CONTAIN AN EMERGENCY EXIT SECTION TO PROVIDE RESTORATION GUIDANCE. AT 0902, THE TRAIN A REACTOR TRIP BREAKER WAS CLOSED PER THE SURVEILLANCE AND A SAFETY INJECTION SIGNAL RESULTED DUE TO A LOSS OF THE REACTOR TRIP INTERLOCK (P-4) WHILE CYCLING THE REACTOR TRIP BREAKER. CORRECTIVE ACTIONS INCLUDE A PROCEDURE REVISION TO THE MANUAL SI SURVEILLANCE TO INCLUDE AN EMERGENCY EXIT SECTION. THIS EVENT WILL ALSO BE INCLUDED IN OPERATOR REQUIRED LISTENING.

[29] CALVERT CLIFFS 1 DOCKET 50-317 LER 90-002
FUEL REPAIR ACTIVITIES PERFORMED ON MORE THAN ONE ASSEMBLY DURING THE SAME TIME PERIOD IN VIOLATION OF FUEL HANDLING INCIDENT SAFETY ANALYSIS.
EVENT DATE: 011690 REPORT DATE: 021490 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 2 (PWR)

(NSIC 219671) ON 1/16/90, WITH UNIT 1 IN MODE 5 AND UNIT 2 DEFUELED, A CONDITION WAS DETERMINED TO BE REPORTABLE AS A CONDITION OUTSIDE OF THE PLANT DESIGN BASIS AS DESCRIBED IN THE UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR). FUEL HANDLING PROCEDURES USED TO GOVERN FUEL RECONSTITUTION ACTIVITIES PERFORMED PRIOR TO UNIT 2 END OF CYCLE 8 DID NOT CONTAIN ADEQUATE CONTROLS TO PREVENT DAMAGE TO MORE THAN ONE FUEL ASSEMBLY IN THE EVENT OF A FUEL HANDLING ACCIDENT. THE UFSAR ASSUMES THAT ONLY ONE FUEL ASSEMBLY COULD BE DAMAGED IN THE EVENT OF A FUEL HANDLING ACCIDENT. A SUPPLEMENTAL LER WILL BE SUBMITTED TO DISCUSS THE SAFETY SIGNIFICANCE OF THIS CONDITION, ANY BROADER IMPLICATIONS AND ANY ADDITIONAL CORRECTIVE ACTIONS. FUEL HANDLING PROCEDURE, FH-48 HAS BEEN REVISED TO REQUIRE THAT ONLY ONE ASSEMBLY AT A TIME BE PLACED IN A DESIGNATED LOCATION FOR RECONSTITUTION. THIS CHANGE ENSURES THAT NO MORE THAN ONE ASSEMBLY COULD BE DAMAGED IN A FUEL HANDLING INCIDENT.

[30] CALVERT CLIFFS 1 DOCKET 50-317 LER 90-024
FAILURE TO TEST FIRE DETECTION CIRCUIT SUPERVISION DUE TO INADEQUATE CONTROLS.
EVENT DATE: 080390 REPORT DATE: 100390 NSSS: CE TYPE: PWR

(NSIC 219725) ON 8/31/90 IT WAS DISCOVERED THAT SURVEILLANCE TEST PROCEDURE (STP) M- 496, "SUPERVISORY TEST OF FIRE DETECTION INSTRUMENTS," HAD NOT BEEN PERFORMED ON THE SMOKE DETECTION SYSTEMS IN THE SERVICE WATER AND COMPONENT COOLING WATER PUMP ROOMS PRIOR TO THE MAXIMUM SURVEILLANCE INTERVAL BEING EXCEEDED. BOTH SYSTEMS HAD BEEN DISABLED WHEN THE STP WAS DUE TO BE PERFORMED. ALTHOUGH ADMINISTRATIVE TAGS WERE HUNG TO ALERT PERSONNEL THAT THE STP NEEDED TO BE PERFORMED, IN SEPARATE INSTANCES, EACH SYSTEM WAS RETURNED TO SERVICE WITHOUT THE SURVEILLANCE CRITERIA BEING MET. THE ROOT CAUSE OF THE FIRST EVENT WAS SOLE RELIANCE ON THE ADMINISTRATIVE TAG TO ENSURE THE STP WAS PERFORMED PRIOR TO RESTORING THE EQUIPMENT TO OPERABLE STATUS. THE ROOT CAUSE OF THE SECOND EVENT WAS PERSONNEL ERROR. WE ARE CONDUCTING A REVIEW OF OUR ADMINISTRATIVE AND PROCEDURAL CONTROLS FOR ENSURING THAT SURVEILLANCE TESTS ARE NOT MISSED BECAUSE EQUIPMENT IS UNAVAILABLE DURING THE SCHEDULED STP PERFORMANCE. THE GOAL OF THIS REVIEW IS TO IDENTIFY AND IMPLEMENT APPROPRIATE ENHANCEMENTS TO THESE CONTROLS. THE DETAILS OF THIS EVENT WILL BE REVIEWED WITH ALL LICENSED OPERATIONS PERSONNEL.

[31] CATAWBA 2 DOCKET 50-414 LER 90-012
TECHNICAL SPECIFICATION VIOLATION DUE TO PRESSURIZER HEATUP LIMIT EXCEEDED FOLLOWING RESIDUAL HEAT REMOVAL PUMP TEST.
EVENT DATE: 090190 REPORT DATE: 092790 NSSS: WE TYPE: PWR

(NSIC 219650) ON SEPTEMBER 1, 1990, WITH UNIT 2 IN MODE 5, COLD SHUTDOWN, FOLLOWING PERFORMANCE OF AN IWP TEST ON RESIDUAL HEAT REMOVAL (ND) SYSTEM PUMP 2B, A TEMPERATURE TRANSIENT OF THE REACTOR COOLANT (NC) SYSTEM PRESSURIZER (PZR) OCCURRED WHICH RESULTED IN THE TECHNICAL SPECIFICATION (T/S) HEATUP LIMIT BEING EXCEEDED AT APPROXIMATELY 0740 HOURS. WITH ND TRAIN A OPERATING TO PROVIDE DECAY HEAT REMOVAL CAPABILITY, AND CHEMICAL AND VOLUME CONTROL (NV) SYSTEM TRAIN A OPERATING TO PROVIDE NC SYSTEM CHARGING CAPABILITY, ND TRAIN B WAS ALIGNED PER THE PERFORMANCE ND PUMP 2B TEST PROCEDURE VALVE LINEUP. CONTROL ROOM OPERATORS (CROS) THEN ISOLATED BOTH ND TRAINS LETDOWN TO NV AND STARTED THE ND PUMP IN MINI-FLOW TO PERFORM THE TEST. CROS WERE CLOSELY MONITORING PZR LEVEL INDICATIONS AND NOTICED A PZR COOLDOWN WHICH APPROACHED BUT DID NOT EXCEED THE T/S PZR COOLDOWN RATE LIMIT. CROS REESTABLISHED ND LETDOWN TO SECURE THE COOLDOWN TRANSIENT AND ABORTED THE TEST. SUBSEQUENTLY, WHILE ATTEMPTING TO RECOVER FROM THE COOLDOWN, A HEATUP OF THE PZR OCCURRED WHICH EXCEEDED THE T/S HEATUP RATE LIMIT DUE TO TEMPERATURE STRATIFICATION WITHIN THE PZR. THIS INCIDENT IS ATTRIBUTED TO AN INADVERTENT ACTION AND A DEFECTIVE PROCEDURE. CORRECTIVE ACTIONS TAKEN INCLUDED SECURING A PZR HEATUP AND RECOVERING TEMPERATURE.

[32] COMANCHE 1 DOCKET 50-445 LER 90-024
PERSONNEL ERROR RESULTING IN FAILURE TO SATISFY TECHNICAL SPECIFICATION STAGGERED TEST BASIS REQUIREMENT.
EVENT DATE: 082490 REPORT DATE: 092490 NSSS: WE TYPE: PWR

(NSIC 219625) ON 8/24/90, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 WAS IN MODE 1, POWER OPERATIONS, WITH REACTOR POWER AT 100%. WHILE PREPARING TO PERFORM SURVEILLANCE TESTING ON CONTAINMENT PURGE AND HYDROGEN PURGE ISOLATION VALVES, TEST DEPARTMENT PERSONNEL DISCOVERED THAT TESTING ACTIVITIES WERE NOT BEING PERFORMED ON A STAGGERED TEST BASIS AS SPECIFIED BY THE ASSOCIATED TECHNICAL SPECIFICATION. THE EVENT WAS CAUSED BY PERSONNEL ERROR DURING INITIAL SURVEILLANCE PROGRAM DEVELOPMENT. THE INDIVIDUAL RESPONSIBLE FOR INPUTTING DATA TO THE SCHEDULING DATABASE OVERLOOKED THE REQUIREMENT. CORRECTIVE ACTIONS INCLUDED TESTING, PROGRAM REVIEW, AND PROCEDURAL ENHANCEMENT.

[33] COMANCHE 1 DOCKET 50-445 LER 90-025
REACTOR TRIP DUE TO THE FAILURE OF A FEEDWATER FLOW CONTROL VALVE LINKAGE ARM NUT.
EVENT DATE: 082590 REPORT DATE: 092490 NSSS: WE TYPE: PWR
VENDOR: BAILEY METER COMPANY

COPES-VULCAN, INC.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 219626) AT APPROXIMATELY 0037 ON 8/25/90, STEAM GENERATOR (SG) NUMBER 2 FEEDWATER FLOW CONTROL VALVE (FCV) FAILED FULL OPEN. A REACTOR OPERATOR ATTEMPTED TO CLOSE THE VALVE FROM THE MAIN CONTROL BOARD BY REDUCING THE DEMAND SIGNAL; HOWEVER, AT 0038, A TURBINE TRIP SIGNAL AND FEEDWATER ISOLATION SIGNAL WAS GENERATED DUE TO PROTECTION SYSTEM INTERLOCK P-14, HI-HI LEVEL IN SG NUMBER 2. THE REACTOR TRIPPED AT 0038 DUE TO THE TURBINE TRIP SINCE REACTOR POWER WAS ABOVE THE P-9 SETPOINT OF 50%. THE CAUSE OF THE FEEDWATER FCV FAILURE IS ATTRIBUTED TO THE FEEDBACK LINKAGE ARM FROM THE VALVE STEM TO THE VALVE POSITIONER SEPARATING DUE TO FLOW INDUCED OSCILLATIONS. CORRECTIVE ACTIONS INCLUDE THE INSTALLATION OF A LOCK WASHER ON THE FEEDBACK LINKAGE ARM AND A DESIGN MODIFICATION TO MODIFY THE VALVE INTERNALS TO REDUCE FLOW INDUCED OSCILLATIONS.

[34] COMANCHE 1 DOCKET 50-445 LER 90-026
MISSED SURVEILLANCE DUE TO INADEQUATE PROCEDURE REQUIREMENTS.
EVENT DATE: 082890 REPORT DATE: 092790 NSSS: WE TYPE: PWR

(NSIC 219661) ON 6/15/90, THE RESIDUAL HEAT REMOVAL PUMP-01 (RHRP-01) QUARTERLY INSERVICE TEST (IST) WAS SATISFACTORILY PERFORMED. ON 7/2/90, POST-TEST DATA REVIEW DETERMINED THAT RHRP-01 WAS IN ALERT STATUS DUE TO LOW DIFFERENTIAL PRESSURE, AS DEFINED BY AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND PRESSURE VESSEL CODE, SECTION XI. AS A RESULT, THE TEST FREQUENCY FOR RHRP-01 WAS INCREASED TO ONCE PER 46 DAYS. ON 7/25/90, A SURVEILLANCE WORK ORDER (SWO) WAS MANUALLY PRINTED IN ACCORDANCE WITH THE INCREASED TEST FREQUENCY REQUIREMENT. HOWEVER, THE TEST FREQUENCY FOR THIS ACTIVITY HAD NOT BEEN REVISED IN THE MANAGED MAINTENANCE COMPUTER PROGRAM SURVEILLANCE ACTIVITY DATA BASE. AS A RESULT, THE ACTUAL DUE DATE AND VIOLATION DATE WAS NOT REFLECTED ON THE SWO. ON 8/12/90, THE REQUIRED SURVEILLANCE EXCEEDED THE VIOLATION DATE. ON 8/14/90, THE REQUIRED SURVEILLANCE WAS PERFORMED SATISFACTORILY. ON 8/28/90, WHILE COMPILING TEST DATA FOR SEVERAL IST COMPONENTS, THE MISSED SURVEILLANCE WAS DISCOVERED. THE ROOT CAUSE WAS DETERMINED TO BE INADEQUATE MANUAL SURVEILLANCE SCHEDULING METHOD. CORRECTIVE ACTIONS INCLUDE REVISIONS TO STATION PROCEDURES.

[35] COMANCHE 1 DOCKET 50-445 LER 90-028
AUTOMATIC REACTOR TRIP CAUSED BY LIGHTNING STRIKE.
EVENT DATE: 090890 REPORT DATE: 100990 NSSS: WE TYPE: PWR

(NSIC 219779) AT 1428 ON SEPTEMBER 8, 1990, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 WAS AUTOMATICALLY TRIPPED FROM 38 PERCENT POWER. A LIGHTNING STRIKE IS BELIEVED TO HAVE CAUSED A SURGE IN THE INPUT POWER RESULTING IN THE DE-ENERGIZATION OF POWER SUPPLIES IN THE ROD DRIVE SYSTEM, CAUSING THE RODS CONTROLLED BY ONE ROD CONTROL CABINET TO DROP INTO THE CORE. THIS RESULTED IN THE REACTOR TRIP FORM HIGH NEGATIVE FLUX RATE. NO SPECIFIC COMPONENT OR SYSTEM FAILURES WERE IDENTIFIED AS THE CAUSE OF THIS EVENT. THE INSTALLATION OF SURGE SUPPRESSORS IN THE INPUT SUPPLY TO ROD DRIVE POWER SUPPLIES WILL PROVIDE ADDITIONAL ASSURANCE THAT POWER SUPPLIES REMAIN AVAILABLE DURING LIGHTNING STRIKES.

[36] COMANCHE 1 DOCKET 50-445 LER 90-029
TURBINE TRIP/REACTOR TRIP CAUSED BY FAILURE OF PERSONNEL TO ASSURE DRAIN PATH IS AVAILABLE.
EVENT DATE: 091090 REPORT DATE: 101090 NSSS: WE TYPE: PWR
VENDOR: MAGNETROL, INC.

(NSIC 219780) ON SEPTEMBER 10, 1990, AT 0917 CDT, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 EXPERIENCED A REACTOR TRIP FROM MODE 1, POWER OPERATIONS. THE REACTOR TRIP WAS CAUSED BY A TURBINE TRIP WHICH RESULTED FROM A HIGH LEVEL CONDITION IN THE B MOISTURE SEPARATOR REHEATER (MSR). THE HIGH LEVEL IN THE MSR OCCURRED WHEN AN OPERATOR ATTEMPTED TO RESTORE A SEPARATOR DRAIN TANK DRAIN VALVE TO SERVICE WHILE THE DRAIN VALVE WAS STILL ISOLATED. THE EVENT WAS CAUSED BY THE

FAILURE OF OPERATORS TO ADEQUATELY VERIFY COMPONENT STATUS PRIOR TO RETURNING THE COMPONENT TO SERVICE. CORRECTIVE ACTIONS TAKEN INCLUDE REVIEW OF THE LESSONS LEARNED PACKAGE BY OPERATORS AND INDIVIDUAL COUNSELING.

[37] CONNECTICUT YANKEE DOCKET 50-213 LER 90-016
DESIGN DEFICIENCY IDENTIFIED IN AUXILIARY FEEDWATER AUTO ACTUATION SYSTEM.
EVENT DATE: 082090 REPORT DATE: 091990 NSSS: WE TYPE: PWR

(NSIC 219535) ON AUGUST 20, 1990, AT 1515 HOURS WITH THE PLANT IN MODE 1 AT 9.5 PERCENT POWER, AN ENGINEERING EVALUATION REVEALED THAT THE AUTOMATIC ACTUATION PORTION OF THE AUXILIARY FEEDWATER (AFW) SYSTEM DID NOT MEET THE DESIGN BASIS REQUIREMENTS NECESSARY TO DECLARE IT OPERABLE. IT WAS DETERMINED THAT THE CALCULATED FLOW RATE ACHIEVED BY AUTOMATIC INITIATION OF THE AFW SYSTEM ALONE IS NOT SUFFICIENT TO ASSURE THAT THE CRITERIA OF THE DESIGN BASIS LOSS OF FEEDWATER ANALYSIS ARE MET. ALSO, RECENT TEST RESULTS REVEALED THAT THE AFW PUMPS COULD TRIP ON AN OVERSPEED CONDITION IF A SUDDEN LOSS OF CONTROL AIR OCCURRED. THE CAUSE IS ATTRIBUTED TO ERRORS IN THE ASSUMPTIONS AND CALCULATIONS USED FOR AUTOMATIC INITIATION OF THE AFW SYSTEM. IMMEDIATE CORRECTIVE ACTION WAS TO REMAIN BELOW 10 PERCENT POWER WHERE THE AUTOMATIC INITIATION FEATURE IS NOT REQUIRED. A CHANGE TO THE TECHNICAL SPECIFICATIONS WAS APPROVED TO ALLOW CONTINUE OPERATION FOR ONE CYCLE. LONG TERM CORRECTIVE ACTIONS INCLUDE MODIFICATIONS TO THE AFW SYSTEM. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B), AND 10CFR50.73(A)(2)(V)(B) AND (D).

[38] CONNECTICUT YANKEE DOCKET 50-213 LER 90-017
AUXILIARY FEEDWATER SYSTEM ACTUATED WHILE TROUBLESHOOTING FEED FLOW RECORDER.
EVENT DATE: 090290 REPORT DATE: 100190 NSSS: WE TYPE: PWR

(NSIC 219652) ON 9/2/90, AT 0930 HOURS, WITH THE PLANT IN MODE 1 AT 80% POWER, AN AUTOMATIC ACTUATION OF THE AUXILIARY FEEDWATER SYSTEM OCCURRED. PRIOR TO THE EVENT, INSTRUMENT AND CONTROLS TECHNICIANS WERE SENT TO INVESTIGATE A PROBLEM INVOLVING THE LOOP 1 FEEDWATER FLOW CHART RECORDER. WHEN THE CURRENT LOOP FOR LOOP 1 FEED FLOW WAS OPENED BY THE TECHNICIANS DURING TESTING, A STEAM FLOW/FEED FLOW MISMATCH OCCURRED CAUSING THE #1 FEEDWATER REGULATING VALVE TO GO FULL OPEN. THIS VALVE POSITION CAUSED LOW SUCTION PRESSURE ON EACH OF THE TWO MAIN FEEDWATER PUMPS, WHICH AUTOMATICALLY TRIPPED EACH PUMP. THE MOMENTARY LOSS OF BOTH FEED PUMPS INITIATED AUTOMATIC ACTUATION OF THE AUXILIARY FEEDWATER SYSTEM. CONTROL ROOM OPERATORS TOOK MANUAL CONTROL OF THE FEEDWATER REGULATING VALVE AND RESTORED STEAM GENERATOR LEVEL TO NORMAL. THE AUXILIARY FEEDWATER SYSTEM WAS SUBSEQUENTLY RESET AT THE CONTROL BOARD AND RETURNED TO NORMAL STATUS. THE ROOT CAUSE OF THE EVENT WAS DETERMINED TO BE PERSONNEL ERROR IN THAT THE SCOPE OF THE WORK AUTHORIZED WAS UNINTENTIONALLY EXCEEDED. CORRECTIVE ACTIONS INCLUDED PERSONNEL COUNSELING AND REEMPHASIS ON THE WORK ORDER PROCESS. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(IV) SINCE IT RESULTED IN AUTOMATIC ACTUATION OF AN ENGINEERED SAFETY FEATURE.

[39] CONNECTICUT YANKEE DOCKET 50-213 LER 90-018
MANUAL PLANT TRIP DUE TO FEEDWATER CONTROL VALVE FAILING OPEN.
EVENT DATE: 090390 REPORT DATE: 100190 NSSS: WE TYPE: PWR
VENDOR: BINGHAM PUMP CO.
CRANE PACKING CO.
FISHER CONTROLS CO.

(NSIC 219663) ON 9/3/90, AT 0457 HOURS, WITH THE PLANT IN MODE 1 AT 80% POWER, THE MAIN FEEDWATER REGULATING VALVE FOR THE NO. 1 STEAM GENERATOR FAILED OPEN. IMMEDIATE OPERATOR ACTION CONSISTED OF MANUALLY TRIPPING THE PLANT, CLOSING THE MAIN STEAM TRIP VALVE AND ISOLATING FEEDWATER TO THE NO. 1 STEAM GENERATOR. THIS EVENT ALSO RESULTED IN AUTOMATIC ACTUATION OF THE AUXILIARY FEEDWATER SYSTEM. SUBSEQUENT DISASSEMBLY AND INSPECTION OF THE VALVE REVEALED THAT THE PLUG HAD SEPARATED FROM THE VALVE STEM. IT WAS ALSO NOTED THAT FRAGMENTS OF THE VALVE STEM WERE CARRIED DOWNSTREAM IN THE FEEDWATER SYSTEM. AN ENGINEERING EVALUATION DETERMINED THAT THE LOOSE PARTS WERE NOT A CONCERN FOR ONE OPERATING CYCLE. THE ROOT CAUSE WAS IMPROPER PART FABRICATION AT THE FACTORY. CORRECTIVE ACTION

CONSISTED OF MODIFYING THE STEM-PLUG ASSEMBLIES ON ALL FOUR FEEDWATER REGULATING VALVES WITH A WELDED JOINT WHERE THE PLUG ATTACHES TO THE STEM. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(IV) SINCE IT RESULTED IN MANUAL ACTUATION OF THE REACTOR PROTECTION SYSTEM AND AUTOMATIC ACTUATION OF AN ENGINEERED SAFETY FEATURE.

[40] COOK 1 DOCKET 50-315 LER 90-009
ACCESS TO AN EXTREME HIGH RADIATION AREA NOT CONTROLLED IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS.
EVENT DATE: 082190 REPORT DATE: 091790 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)

(NSIC 219610) ON AUGUST 21, 1990 AT 1315 HOURS IT WAS DISCOVERED THAT THE OUTER DOOR TO THE 587' ELEVATION DRUMMING ROOM WAS UNLATCHED AND THE LOCK ON THE INTERNAL EXTREME HIGH RADIATION AREA GATE WAS FOUND UNLOCKED. THE SUBJECT GATE CONTROLS ACCESS TO THE HIGH LEVEL WASTE STORAGE AREA. THERE WERE NO INOPERABLE STRUCTURES, COMPONENTS OR SYSTEMS THAT CONTRIBUTED TO THIS EVENT. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO A COGNITIVE ERROR BY AN UNKNOWN INDIVIDUAL WHO LEFT THE GATE UNLOCKED. UPON DISCOVERY OF THE DEVIANT CONDITION THE GATE WAS IMMEDIATELY LOCKED TO PREVENT UNAUTHORIZED ENTRY. THE NRC WAS NOTIFIED VIA THE EMERGENCY NOTIFICATION SYSTEM (ENS) AT APPROXIMATELY 1315 HOURS ON AUGUST 21, 1990.

[41] COOK 2 DOCKET 50-316 LER 90-007 REV 01
UPDATE ON CONTAINMENT TYPE B AND C LEAKAGE EXCEEDS L.C.O. VALUE DUE TO DEGRADATION OF ISOLATION VALVE SEATING SERVICES.
EVENT DATE: 071790 REPORT DATE: 101190 NSSS: WE TYPE: PWR
VENDOR: CLOW CORP.

(NSIC 219724) THIS SUPPLEMENTAL REPORT IS BEING SUBMITTED TO PROVIDE ADDITIONAL INFORMATION REGARDING THE TYPE B AND C LEAKRATE TESTING INITIALLY REPORTED ON AUGUST 8, 1990. WITH THE REACTOR COOLANT SYSTEM IN MODE 5 (COLD SHUTDOWN), THE ACCUMULATED LEAKAGE FOUND USING THE MAXIMUM PATHWAY METHODOLOGY FOR THE TYPE B AND C LEAK RATE TESTS ON CONTAINMENT PENETRATIONS WAS 0.74 LA. THIS EXCEEDED THE L.C.O. VALUE (0.60 LA) OF TECHNICAL SPECIFICATION 3.6.1.2.B. THE CONTAINMENT PRESSURE RELIEF TRAIN-A CONTAINMENT ISOLATION VALVE, 2-VCR-107 (EIIS:ISV/BD), IS THE MAJOR CONTRIBUTING FACTOR FOR EXCEEDING THE TECHNICAL SPECIFICATION LIMIT. 2-VCR-107 EXHIBITED A LEAKAGE RATE OF 49,000 SCCM. THIS IS 60 PERCENT OF THE TOTAL LEAKAGE OBTAINED FROM ALL TESTED PENETRATIONS. OTHER CONTAINMENT ISOLATION VALVES THAT EXHIBIT LEAK RATES IN EXCESS OF GUIDELINE ACCEPTANCE CRITERIA WERE REPAIRED AND RETESTED TO ENSURE THE LEAK RATES ARE WITHIN ALLOWABLE LIMITS. THE FINAL AS-LEFT TYPE B AND C LEAKAGE RATE WAS 0.17 LA.

[42] COOK 2 DOCKET 50-316 LER 90-008
DEGRADATION OF DIVIDER BARRIER SEAL LOCATED BETWEEN CONTAINMENT WALL AND CRANE WALL.
EVENT DATE: 080790 REPORT DATE: 091790 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 1 (PWR)
VENDOR: UNIROYAL, INC.

(NSIC 219611) ON AUGUST 7, 1990, WITH UNIT 2 IN MODE 6 (REFUELING), WHILE REMOVING SAMPLES OF THE DIVIDER BARRIER SEAL, THE SAMPLES WOULD SPLIT WHEN REMOVED FROM THE SEATING SURFACES OR SEPARATE LONGITUDINALLY ALONG CRACKS NOT VISIBLE PRIOR TO REMOVAL. SURVEILLANCE 4.6.5.9.A REQUIRES THAT SAMPLES OF THE DIVIDER BARRIER SEAL TO BE REMOVED FOR LABORATORY ANALYSIS. TEST COUPONS OF THE DIVIDER BARRIER SEAL MATERIAL WERE DEPLETED AND PORTIONS OF THE ACTUAL BARRIER SEAL HAD TO BE REMOVED. IT COULD NOT BE CONCLUSIVELY DETERMINED WHAT CAUSED THE DIVIDER BARRIER SEAL MATERIAL TO DEGREE IN SUCH A MANNER. SAFETY EVALUATIONS FOR BOTH UNITS WERE PERFORMED TO DETERMINED THE IMPACT OF THE DIVIDER BARRIER SEAL DEGRADATION. THE EVALUATIONS INCLUDED THAT THE BARRIER SEALS WOULD HAVE PERFORMED THEIR DESIGNED SAFETY FUNCTION EVEN WHEN DEGRADED, AND DID NOT REPRESENT A SIGNIFICANT THREAT TO PUBLIC SAFETY. THE ENTIRE UNIT 2 DIVIDER BARRIER SEAL IS BEING REPLACED DURING THE CURRENT REFUELING OUTAGE. THE UNIT 1

ORIGINAL DIVIDER BARRIER SEAL (3807 MODEL) WILL BE REPLACED DURING THE UPCOMING REFUELING OUTAGE (OCTOBER - NOVEMBER, 1990).

[43] COOPER DOCKET 50-298 LER 90-008 REV 01
UPDATE ON SURVEILLANCE PROCEDURE NOT PERFORMED WITHIN REQUIRED INTERVALS DUE TO DEFICIENT COMPUTER SCHEDULING PROGRAM AND PERSONNEL ERROR.
EVENT DATE: 070690 REPORT DATE: 100590 NSSS: GE TYPE: BWR

(NSIC 219718) ON 7/6/90, DURING A REVIEW OF SURVEILLANCE RECORDS, IT WAS DISCOVERED THAT SURVEILLANCE PROCEDURES (SP) 6.3.10.22, DRYWELL SUMP ACCUMULATOR CHECK VALVE EXERCISE TEST, AND 6.4.5.6, FIRE DETECTION SYSTEM CIRCUITRY OPERABILITY, HAD NOT BEEN PERFORMED WITHIN REQUIRED SURVEILLANCE INTERVALS. SP 6.3.10.22 WAS PERFORMED 15 DAYS LATE, AND SP 6.4.5.6 WAS PERFORMED 23 DAYS LATE. ON 9/11/90, DURING FOLLOWUP ACTIONS FOR LER 90-008, IT WAS DETERMINED THAT BOTH SP 6.3.17.4, CONTROL ROOM EMERGENCY FAN HEPA FILTER LEAK TEST, AND 6.3.17.5, CONTROL ROOM EMERGENCY FAN CHARCOAL LEAK, CHARCOAL SAMPLING, AND FAN CAPACITY TEST, HAD BEEN PERFORMED 13 DAYS LATE. PLANT OPERATION VARIED FROM REFUELING THROUGH 100% POWER. ALL EQUIPMENT FUNCTIONED AS REQUIRED DURING TESTING, INDICATING IT WAS FUNCTIONAL THROUGHOUT THE SURVEILLANCE INTERVALS. ROOT CAUSES OF THESE EVENTS ARE A DEFICIENCY IN COMPUTER BASED SCHEDULING SYSTEM AND PERSONNEL ERROR. COMPUTER BASED SCHEDULING SYSTEM DID NOT CONTINUE TO LIST MISSED SURVEILLANCES BEYOND THE WEEK IN WHICH PERFORMANCE WAS SCHEDULED. COGNITIVE PERSONNEL ERROR BY SURVEILLANCE COORDINATOR RESULTED IN SP 6.3.10.22 AND 6.4.5.6 NOT BEING PROVIDED TO THE PERFORMING ORGANIZATION WHEN SCHEDULED. A COGNITIVE PERSONNEL ERROR BY SYSTEM ENGINEER RESULTED IN SP 6.3.17.4 AND 6.3.17.5 NOT BEING PERFORMED AS SCHEDULED.

[44] CRYSTAL RIVER 3 DOCKET 50-302 LER 88-007 REV 01
UPDATE ON UNKNOWN CAUSE RESULTS IN REACTOR BUILDING SPRAY PUMP OPERATING BELOW ITS DESIGN FLOW.
EVENT DATE: 012788 REPORT DATE: 101290 NSSS: BW TYPE: PWR

(NSIC 219680) AS A RESULT OF THE REEVALUATION OF THE EMERGENCY DIESEL GENERATOR LOADING ISSUE, THE UTILITY CONDUCTED ADDITIONAL PUMP FLOW TESTING TO DETERMINE THE ACTUAL KW LOAD OF MAJOR ENGINEERED SAFEGUARDS PUMPS. DURING A SUBSEQUENT NRC REVIEW, THE INSPECTOR NOTED A DISCREPANCY BETWEEN THE MANUFACTURER'S PUMP CURVE (HEAD-FLOW) FOR BUILDING SPRAY PUMP (BSP-1A) AND THE TEST DATA FOR THAT PUMP. AT THAT TIME, CRYSTAL RIVER UNIT 3 WAS OPERATING IN MODE 1 (POWER OPERATIONS) AT 99% POWER. TESTING WAS PERFORMED AND ENGINEERING CALCULATIONS SHOWED THAT BSP-1A WOULD PROVIDE A FLOW OF 1460 GPM AT 375 FEET OF HEAD, WHICH IS BELOW ITS DESIGN RATING OF 1500 GPM. DURING THE REFUEL 7 OUTAGE, THE PUMP WAS DISASSEMBLED FOR INSPECTION. AT THAT TIME, THE IMPELLER WAS DISCOVERED TO NOT MATCH THE ORIGINAL IMPELLER. IT WAS DETERMINED THAT A MANUFACTURING OVERSIGHT HAD CAUSED ONE FABRICATION STEP TO BE OMITTED. THIS CHANGED THE OPERATING CHARACTERISTICS OF THE PUMP. THE IMPELLER HAS BEEN REPAIRED AND THE VENDOR HAS TAKEN CORRECTIVE ACTION THAT IS ACCEPTABLE TO FPC.

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

(NSIC 219635) ON JULY 9, 1990, WHILE OPERATING AT 98% POWER, FLORIDA POWER CORPORATION DISCOVERED THE QUARTERLY CALIBRATION OF THE HYDROGEN CHANNEL OF THE WASTE GAS DECAY TANK (WGDT) EXPLOSIVE GAS MONITORING INSTRUMENTATION HAD NOT BEEN PERFORMED AS REQUIRED BY TECHNICAL SPECIFICATION 4.3.3.10 PRIOR TO DECLARING THE SYSTEM OPERABLE ON JUNE 2, 1990. THE IMMEDIATE CAUSE OF THIS NON CONFORMANCE WAS PERSONNEL ERROR. CRYSTAL RIVER UNIT 3 TECHNICIANS WERE NOT ADEQUATELY NOTIFIED OF THE NEED TO PERFORM THE QUARTERLY SURVEILLANCE. ADDITIONALLY, THE CALIBRATION THAT WAS PERFORMED WAS INCORRECTLY COMMUNICATED TO THE OPERATING SHIFT AS A COMPLETE CALIBRATION. TO PREVENT RECURRENCE, SURVEILLANCE PROCEDURES (SPS), OR OTHER TESTS USED TO RETURN EQUIPMENT TO SERVICE, WILL BE REVIEWED BY THE

RESPONSIBLE PROCEDURE SUPERVISOR PRIOR TO RETURNING THE EQUIPMENT TO OPERABLE STATUS.

[46] DIABLO CANYON 2 DOCKET 50-323 LER 88-023 REV 01
UPDATE ON VITAL BATTERIES INOPERABLE DUE TO LOW BATTERY AVERAGE ELECTROLYTE TEMPERATURE.
EVENT DATE: 122988 REPORT DATE: 092790 NSSS: WE TYPE: PWR
VENDOR: JOHNSON CONTROLS INC.

(NSIC 219681) ON DECEMBER 29, 1988 AT 1022 PST, UNIT 2 ENTERED TECHNICAL SPECIFICATION (TS) ACTION STATEMENT 3.0.3 WHEN THE VITAL BATTERIES WERE DECLARED INOPERABLE IN ACCORDANCE WITH TS SURVEILLANCE REQUIREMENT 4.8.3.1.B.3 DUE TO BATTERY 2-1 AVERAGE ELECTROLYTE TEMPERATURE (BAET) BEING BELOW 60 DEGREES FAHRENHEIT (F). THIS EVENT OCCURRED WHEN BATTERY 2-1 WAS EXCESSIVELY COOLED BY THE VENTILATION SYSTEM. TS SURVEILLANCE REQUIREMENT 4.8.3.1.B.3 REQUIRES VERIFICATION THAT THE BAET IS ABOVE 60 F. SURVEILLANCE TEST PROCEDURE M-11B WAS PERFORMED ON DECEMBER 28, 1988, AND BATTERY 2-1 BAET WAS FOUND ABOVE 60F. ON DECEMBER 29, 1988, UNIT 2 BATTERIES WERE CHECKED AND BATTERY 2-1 WAS DETERMINED TO HAVE A BAET OF 59 F. THE BATTERY WAS PUT ON EQUALIZING CHARGE AND ROOM DOORS OPENED TO ALLOW THE BAET TO WARM ABOVE 60 F. THIS EVENT WAS CAUSED BY EXCESSIVE FLOW OF COLD AIR FROM THE VENTILATION SYSTEM DUE TO FAILURE OF A THERMOSTAT WHICH CAUSED THE SUPPLY FAN TO STAY IN HIGH SPEED. TO PREVENT RECURRENCE, THE THERMOSTAT HAS BEEN REPAIRED AND ADDED TO THE PREVENTATIVE MAINTENANCE PROGRAM AND APPROPRIATE PROCEDURES HAVE BEEN REVISED.

[47] DIABLO CANYON 2 DOCKET 50-323 LER 90-006 REV 01
UPDATE ON VIOLATION OF TECHNICAL SPECIFICATIONS BECAUSE OF INOPERABLE STEAM FLOW TRANSMITTERS DUE TO PERSONNEL ERROR.
EVENT DATE: 042390 REPORT DATE: 100590 NSSS: WE TYPE: PWR
VENDOR: ROSEMOUNT, INC.

(NSIC 219730) ON 4/23/90 AT 0510 PDT, UNIT 2 ENTERED MODE 3 (HOT STANDBY) WITH TWO INOPERABLE STEAM GENERATOR STEAM FLOW CHANNELS (FLOW TRANSMITTER (FT) 512 FOR LOOP 1 AND FT-533 FOR LOOP 3), WHICH DID NOT MEET THE REQUIREMENTS OF TECHNICAL SPECIFICATIONS 3.3.1 AND 3.3.2. ON 4/30/90, AT 0933 PDT, UNIT 2 WAS PARALLELED TO THE GRID FOLLOWING COMPLETION OF THE THIRD REFUELING OUTAGE. AT 0959 PDT, STEAM GENERATOR 2-1 AND 2-3, STEAM FLOW CHANNELS 512 AND 533, RESPECTIVELY, WERE DECLARED INOPERABLE AFTER CONTROL ROOM OPERATORS NOTED NO FLOW INDICATION. SUBSEQUENT INVESTIGATION ON 4/30/90, DETERMINED THAT THE HIGH AND LOW PRESSURE SENSING LINES TO EACH TRANSMITTER HAD BEEN REVERSED WHEN THE TRANSMITTERS WERE REPLACED DURING THE OUTAGE. THE ROOT CAUSE WAS DETERMINED TO BE PERSONNEL ERROR IN THAT THE TECHNICIAN DID NOT USE GOOD CONFIGURATION CONTROL PRACTICES IN PERFORMING THE WORK. CORRECTIVE ACTIONS TO PREVENT RECURRENCE INCLUDE ISSUING A MAINTENANCE BULLETIN TO ALL MAINTENANCE AND MAINTENANCE PLANNING DEPARTMENTS, ISSUING AN I&C POLICY REQUIRING THE USE OF CONFIGURATION CONTROL SHEETS, REVISING ADMINISTRATIVE PROCEDURE C-4S3, "CONTROL OF LIFTED CIRCUITRY AND JUMPERS DURING MAINTENANCE," TO REQUIRE THE USE OF CONFIGURATION CONTROL IN SENSING LINE APPLICATIONS, AND DEVELOPING A PROCEDURE TO PROVIDE SPECIFIC DETAIL FOR ROSEMOUNT TRANSMITTER REPLACEMENT.

[48] FERMI 2 DOCKET 50-341 LER 90-007
CONTROL CENTER VENTILATION SHIFTS TO RECIRCULATION DUE TO BLOWN FUSE.
EVENT DATE: 082990 REPORT DATE: 092890 NSSS: GE TYPE: BWR

(NSIC 219658) ON 8/29/90, THE DIVISION 2 CONTROL CENTER HEATING VENTILATION AND AIR CONDITIONING (CCHVAC) HAD BEEN PLACED IN THE RECIRCULATION MODE FOR THE PERFORMANCE OF A SURVEILLANCE TEST. AT 1257 HOURS, THE CCHVAC DIVISION 2 CONTROL CENTER RADIATION MONITOR ALARMED AND THE CONTROL ROOM DIVISION 2 CCHVAC DAMPER (DMP) INDICATIONS WERE LOST. THE DIVISION 1 CCHVAC AUTOMATICALLY SHIFTED TO THE RECIRCULATION MODE IN RESPONSE TO THE LOSS OF POWER TO THE RADIATION MONITOR. A SEVEN DAY LIMITING CONDITION FOR OPERATION (LCO) WAS ENTERED. SUBSEQUENT TROUBLESHOOTING DETERMINED THAT A FAULTY LAMP FOR THE CONTROL CENTER KITCHEN EXHAUST OUTBOARD DAMPER INDICATION HAD CAUSED FUSE F9 IN PANEL H21-P296B TO OPEN.

THE LAMP AND FUSE F9 WERE REPLACED. THE LCO WAS CLEARED FOLLOWING THE INVESTIGATION AND REPAIR ON 8/30/90, AT 1706 HOURS. THE ROOT CAUSE WAS A DESIGN DEFICIENCY IN THE DIVISION 1 AND 2 RADIATION MONITOR TRIP CIRCUITRY. THE SAME POWER SUPPLY AND COMMON FUSE WERE UTILIZED FOR BOTH RADIATION MONITOR TRIP RELAY LOGIC AND INDICATION LIGHTING. A DESIGN CHANGE WILL BE IMPLEMENTED WHICH WILL ELIMINATE THE RADIATION MONITOR TRIP AND RESULTING CCHVAC RECIRCULATION MODE ACTUATION UPON A LOSS OF POWER DUE TO A SHORTED INDICATING LAMP.

[49] FERMI 2 DOCKET 50-341 LER 90-008
HPCI INADVERTENT ISOLATION DUE TO SPURIOUS HIGH STEAM FLOW SIGNAL.
EVENT DATE: 090590 REPORT DATE: 100590 NSSS: GE TYPE: BWR

(NSIC 219744) ON SEPTEMBER 5, 1990, AT 1026 HOURS THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM AUTOMATICALLY ISOLATED DUE TO A SPURIOUS HIGH STEAM FLOW SIGNAL ON ONE OF TWO CHANNELS. THE SYSTEM WAS IN STANDBY AT THE TIME OF THE EVENT. IN COMPLIANCE WITH TECHNICAL SPECIFICATION ACTION STATEMENT REQUIREMENTS, THE ISOLATION CHANNEL WAS PLACED IN A TRIPPED CONDITION, THE SYSTEM WAS DECLARED INOPERABLE, AND THE APPROPRIATE LCO WAS ENTERED. AS A RESULT OF THE INVESTIGATION, THE ROOT CAUSE OF THE HPCI ISOLATION IS BELIEVED TO HAVE BEEN A TRANSIENT PROCESS NOISE DISTURBANCE THAT BECAME SUPERIMPOSED ON AN ALREADY NOISY TRANSMITTER LONG ENOUGH TO ACTUATE THE HPCI ISOLATION OUTPUT LOGIC. HPCI WAS SUBSEQUENTLY DECLARED OPERABLE BASED ON THE FOLLOWING: (1) THE TROUBLESHOOTING PERFORMED, (2) NO TRIP SIGNAL PRESENT, (3) SUCCESSFUL TESTING OF SYSTEM FUNCTION AND (4) RELAY REPLACEMENT. AN ENGINEERING DESIGN CHANGE WAS IMPLEMENTED WHICH FILTERED OUT THE EXISTING BACKGROUND NOISE FROM THE TRANSMITTER'S OUTPUT, THUS DECREASING THE SUSCEPTIBILITY TO TRANSIENT DISTURBANCES. SUBSEQUENT TESTING VERIFIED THE EFFECTIVENESS OF THE CORRECTIVE ACTION.

[50] FITZPATRICK DOCKET 50-333 LER 89-004 REV 01
UPDATE ON DESIGN DEFICIENCY POTENTIALLY AFFECTS BOTH DIVISIONS OF SAFETY-RELATED SYSTEMS DUE TO VALVES FAILING CLOSED ON LOSS OF AIR.
EVENT DATE: 030989 REPORT DATE: 101090 NSSS: GE TYPE: BWR

(NSIC 219685) ON 3/9/89 DURING NORMAL OPERATION AT 100% POWER, ENGINEERING PERSONNEL IDENTIFIED A DESIGN DEFICIENCY WHICH ORIGINATED DURING PLANT CONSTRUCTION. THE DEFICIENCY WOULD RESULT IN LOSS OF AREA COOLING FOR PARTS OF BOTH SAFETY DIVISIONS OF SAFETY-RELATED AND NON-SAFETY-RELATED ELECTRICAL DISTRIBUTION SYSTEMS AS A RESULT OF LOSS OF INSTRUMENT AIR (LD) TO THE COOLING SYSTEM TEMPERATURE CONTROL VALVES. INVESTIGATION REVEALED THAT SIMILAR DEFICIENCIES DID NOT EXIST FOR COOLING OF OTHER SAFETY-RELATED EQUIPMENT. MANUAL BYPASS VALVES FOR THE AFFECTED VALVES WERE TAGGED OPEN TO ASSURE COOLING IN THE EVENT OF LOSS OF AIR. A MODIFICATION OF THE TEMPERATURE CONTROL VALVES TO CAUSE THE VALVE TO FAIL-OPEN UPON LOSS OF AIR WILL BE COMPLETED LATER. FAILURE OF THE VALVES IN THE CLOSED POSITION WOULD HAVE RESULTED IN A TEMPERATURE INCREASE REQUIRING INVESTIGATION AND MANUAL OPENING OF BYPASS VALVES. THERE HAVE NOT BEEN ANY SIMILAR DESIGN DEFICIENCY EVENTS RESULTING IN THE INCORRECT FAILURE MODE AT THIS FACILITY.

[51] FITZPATRICK DOCKET 50-333 LER 89-008 REV 01
UPDATE ON HIGH PRESSURE COOLANT INJECTION AND REACTOR CORE ISOLATION COOLING MADE INOPERABLF DUE TO PROCEDURE DEFICIENCY CAUSING MISSED SURVEILLANCE.
EVENT DATE: 051789 REPORT DATE: 092890 NSSS: GE TYPE: BWR

(NSIC 219686) ON 5/17/89 AT 1125 HOURS DURING NORMAL OPERATION AT 100% POWER IT WAS FOUND THAT SURVEILLANCE TESTING OF HIGH PRESSURE COOLANT INJECTION (HPCI) (BJ) AND REACTOR CORE ISOLATION COOLING (RCIC) (BN) AUTO ISOLATION SYSTEM (JE) TIME DELAY DEVICES HAD NOT BEEN COMPLETED AS REQUIRED BY TECHNICAL SPECIFICATION TABLE 4.2-2 SINCE INITIAL PLANT STARTUP. HPCI AND RCIC STEAM SUPPLY ISOLATION VALVES WERE DECLARED INOPERABLE WHEN REQUIRED TO BE OPERABLE OR CLOSED BY TECHNICAL SPECIFICATION 3.7.D. INVESTIGATION ALSO REVEALED THAT HIGH ENERGY LINE BREAK (HELB) ANALYSES DONE TO SUPPORT ENVIRONMENTAL QUALIFICATION HAD ASSUMED NO TIME DELAY. CIRCUITS WERE MODIFIED TO REMOVE TIME DELAY DEVICES AND THE ISOLATION VALVES WERE MADE OPERABLE AT 0103 HOURS ON 5/18/89. CONSERVATIVE

ANALYSIS TECHNIQUES AND OTHER ISOLATION SIGNALS (WHICH ARE NOT DELAYED) LIMITED POTENTIAL EFFECTS OF TIME DELAY DEVICES WHICH WERE NOT ASSUMED TO EXIST IN HELB ANALYSES. TECHNICAL SPECIFICATION TABLE 4.2-2 WILL BE CORRECTED. TRAINING AND ORGANIZATION CHANGES WILL REDUCE THE POTENTIAL FOR RECURRENCE. LER-86-001, 86-002, 87-071, 87-022, 88-006, AND 89-007 ARE ALSO MISSED SURVEILLANCE EVENTS.

[52] FITZPATRICK DOCKET 50-333 LER 90-003 REV 01
UPDATE ON CORE OVERPOWER EVENTS DUE TO FEED FLOW TRANSMITTER CALIBRATION.
EVENT DATE: 012990 REPORT DATE: 092590 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 219620) TWO NON-CONSERVATIVE ERRORS IN FEED (SJ) FLOW MEASUREMENT WERE DISCOVERED DURING A SITE REVIEW BEGUN IN OCTOBER 1989. FEED FLOW TRANSMITTERS WERE REPLACED ON 10/3/88 WITH A MODEL REQUIRING STATIC PRESSURE COMPENSATION. THE COMPENSATION WAS OMITTED DUE TO AN INCOMPLETE REVIEW OF THE VENDOR MANUAL. THE CALIBRATION WAS CORRECTED AND PERFORMED ON 11/14/89. SECONDLY, THE REVIEW ALSO RAISED QUESTIONS ABOUT A VENDOR SUPPLIED CORRELATION FOR THE FEED FLOW VENTURIS. A CONSERVATIVE LICENSEE CORRELATION WAS USED TO RECALIBRATE THE TRANSMITTERS AND THE RESULTING INDICATED POWER EXCEEDED LICENSED THERMAL POWER. POWER WAS PROMPTLY REDUCED TO THE LICENSED VALUE. FLOW ELEMENT VENDOR INPUT ERRORS HAVE SINCE BEEN IDENTIFIED AND CORRECTED. ADDITIONAL CORRECTIVE ACTIONS WILL INCLUDE ADDITIONAL TRAINING, MODIFICATION CONTROL PROCEDURE CHANGES, INCREASED CALIBRATION FREQUENCY, ENHANCED PERFORMANCE MONITORING OF THE FEED FLOW, AND ENHANCED CALIBRATION PROCEDURES. RELATED LERS: 82-002 AND 82-034

[53] FITZPATRICK DOCKET 50-333 LER 90-022
ENGINEERED SAFETY FEATURE ACTUATION DUE TO LOW VOLTAGE TRIP OF POWER SUPPLY TO REACTOR PROTECTION SYSTEM BUS DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 091090 REPORT DATE: 101090 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219742) THE PLANT WAS OPERATING AT FULL POWER ON 9/10/90. AT 9:38 A.M. A LOW VOLTAGE CONDITION ON THE OUTPUT FROM THE MOTOR GENERATOR (MG) POWER SUPPLY TO THE B REACTOR PROTECTION SYSTEM (RPS) (JC) RESULTED IN TRIPPING OF THE RPS BUS ELECTRICAL PROTECTIVE ASSEMBLY (EPA) UNDER VOLTAGE RELAY AND DEENERGIZING OF THE RPS BUS. AN AUTOMATIC HALF-SCRAM RESULTED, TOGETHER WITH ISOLATION OF THE REACTOR WATER CLEAN-UP SYSTEM (CE), REACTOR BUILDING (NG) VENTILATION (VA), CONTAINMENT AIR DILUTION (CAD) (LK), DRYWELL EQUIPMENT AND FLOOR DRAIN PUMPS (WK), AND CONTAINMENT SAMPLE (KN) SYSTEMS. THE RPS WAS SWITCHED TO ITS ALTERNATE POWER SUPPLY AND THE ISOLATIONS WERE RESET. THE LOW VOLTAGE CONDITION RESULTED FROM DRIFT OF THE MG SET VOLTAGE REGULATOR. THE DOWNWARD DRIFT OF THE OUTPUT VOLTAGE TO THE RANGE OF THE TRIP SETPOINT WAS NOT RECOGNIZED BY THE OPERATORS DURING THEIR SHIFT SURVEILLANCE BECAUSE A DEFICIENCY IN THE DAILY SURVEILLANCE TEST PROCEDURE SPECIFIED AN EXCESSIVELY WIDE RANGE FOR ACCEPTABLE OUTPUT VOLTAGE. CORRECTIVE ACTION INCLUDED REPLACEMENT OF VOLTAGE REGULATOR BOARDS, THE EPA ASSEMBLY CIRCUIT BREAKER, AND IMPROVEMENTS TO THE SURVEILLANCE PROCEDURES WHICH DESCRIBE THE MONITORING OF OUTPUT VOLTAGE.

[54] FT. CALHOUN 1 DOCKET 50-285 LER 90-021
INADVERTENT REACTOR PROTECTIVE SYSTEM ACTUATION.
EVENT DATE: 082990 REPORT DATE: 092890 NSSS: CE TYPE: PWR

(NSIC 219634) ON AUGUST 29, 1990, WITH THE PLANT IN COLD SHUTDOWN, THE REACTOR PROTECTION SYSTEM AT FORT CALHOUN STATION WAS INADVERTENTLY ACTUATED. THE ACTUATION OCCURRED WHILE AN OPERATOR WAS CHANGING THE POWER SOURCE FOR CONTROL ROD CLUTCH POWER SUPPLIES IN PREPARATION FOR ELECTRICAL MAINTENANCE ON AN INSTRUMENT BUS TRANSFORMER. PRIOR TO THIS INCIDENT, ALL CONTROL RODS WERE FULLY INSERTED IN THE CORE. THEREFORE, NO ACTUAL ROD MOVEMENT OCCURRED. THE CAUSE OF THIS EVENT WAS FAILURE BY THE OPERATOR INVOLVED TO FOLLOW PROCEDURE. A CONTRIBUTING FACTOR WAS THE FAILURE TO NOTIFY THE SHIFT SUPERVISOR WHEN UNEXPECTED RESULTS WERE OBTAINED DURING THE PERFORMANCE OF THIS ACTIVITY. CORRECTIVE ACTIONS INCLUDE INDIVIDUAL COUNSELING OF THE OPERATOR INVOLVED, AND

ADDITIONAL INSTRUCTION TO OPERATIONS PERSONNEL THAT A QUESTIONING ATTITUDE MUST BE MAINTAINED AT ALL TIMES AND UNANTICIPATED RESULTS REPORTED TO SUPERVISION.

[55] GRAND GULF 1 DOCKET 50-416 LER 90-015
CONTROL BUILDING PENETRATION NOT PROPERLY SEALED.
EVENT DATE: 082390 REPORT DATE: 092190 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: GRAND GULF 2 (BWR)

(NSIC 219616) DURING THE PENETRATION FOR THE IMPLEMENTATION OF A MINOR CHANGE PACKAGE, A FIRE RATED ASSEMBLY PENETRATION WAS DISCOVERED NOT PROPERLY SEALED. IT IS BELIEVED THAT THE PENETRATION WAS OPENED IN 1983 BY THE IMPLEMENTATION OF A DESIGN CHANGE PACKAGE AND WAS NOT PROPERLY RESEALED. THE FIRE RATED ASSEMBLY SEPARATES THE UNIT 2 CONTROL BUILDING, WHICH CONTAINS SAFETY-RELATED EQUIPMENT ASSOCIATED WITH UNIT 1 OPERATIONS, FROM THE UNIT 2 TURBINE BUILDING. AFTER COMPREHENSIVE REVIEWS OF WORK DOCUMENTS AND PLANT MODIFICATION PACKAGES, THE ROOT CAUSE COULD NOT POSITIVELY BE DETERMINED. UPON IDENTIFYING THE NONCONFORMANCE, A NONCONFORMANCE REPORT WAS WRITTEN AND A WORK ORDER WAS INITIATED TO SEAL THE PENETRATION. THIS BEING THE FIRST FIRE RATED ASSEMBLY PENETRATION FOUND NOT PROPERLY SEALED, ENTERGY OPERATIONS HAS REASONABLE ASSURANCE THAT THIS IS AN ISOLATED CASE AND THE ADMINISTRATIVE CONTROLS CURRENTLY IN PLACE ARE ADEQUATE TO PRECLUDE ANY UNCONTROLLED WORK OF FIRE RATED ASSEMBLY PENETRATIONS.

[56] INDIAN POINT 2 DOCKET 50-247 LER 90-007
TOXIC GAS MONITOR ALARMED RESULTING IN ISOLATION OF CENTRAL CONTROL ROOM VENTILATION SYSTEM.
EVENT DATE: 082590 REPORT DATE: 092490 NSSS: WE TYPE: PWR
VENDOR: WISCONSIN BRIDGE & IRON

(NSIC 219631) ON AUGUST 25, 1990, AT ABOUT 2335 HOURS, WITH REACTOR POWER AT 96.5%, THE HYDROGEN CYANIDE (HCN) TOXIC GAS MONITOR CHANNEL 2 ALARMED, RESULTING IN THE TRANSFER OF THE CENTRAL CONTROL ROOM (CCR) VENTILATION SYSTEM FROM THE NORMAL MODE TO THE INCIDENT MODE. RELATED ACTUATIONS OF THE HCN TOXIC GAS MONITOR CHANNELS OCCURRED SUBSEQUENTLY WITHIN THE PERIOD OF AUGUST 27-29. AS DESIGNED, THE DETECTION OF HCN BY EITHER CHANNELS 1 OR 2 OF THE TOXIC GAS MONITORS WILL GENERATE AN ALARM IN THE CCR AND ISOLATE THE CCR VENTILATION SYSTEM. THE TOXIC GAS MONITORING SYSTEM IS CLASSIFIED AS AN ENGINEERED SAFETY FEATURE (ESF). NO TECHNICAL SPECIFICATION OR NRC LIMITS WERE EXCEEDED.

[57] INDIAN POINT 3 DOCKET 50-286 LER 90-005
PLANT SHUTDOWN AND UNUSUAL EVENT DUE TO EMERGENCY DIESEL GENERATOR OPERABILITY CONCERNS FOLLOWING MODIFICATIONS.
EVENT DATE: 080990 REPORT DATE: 091190 NSSS: WE TYPE: PWR
VENDOR: ALCO ENGINE DIVISION, WHITE IND.

(NSIC 219538) ON AUGUST 9, 1990, WITH THE REACTOR AT 100 PERCENT POWER, ONE EMERGENCY DIESEL GENERATOR WAS OUT OF SERVICE AND THE OPERABILITY OF A SECOND EMERGENCY DIESEL GENERATOR BECAME SUSPECT. THE LIMITING CONDITION FOR OPERATION ACTION STATEMENT OF TECHNICAL SPECIFICATIONS FOR TWO INOPERABLE EMERGENCY DIESEL GENERATORS WAS ENTERED. A PLANT SHUTDOWN WAS STARTED AND AN UNUSUAL EVENT DECLARED. THE AUTHORITY HAS IDENTIFIED THE ROOT CAUSES AS POOR MAINTENANCE PRACTICES AND DOCUMENTATION INSUFFICIENT. ALL THREE EMERGENCY DIESEL GENERATORS WERE RESTORED OR CONFIRMED TO BE OPERABLE WITHIN THE SAME DAY. THE PLANT SHUTDOWN CONTINUED TO THE HOT SHUTDOWN CONDITION. THE UNUSUAL EVENT WAS TERMINATED AT 2035 HOURS ON AUGUST 9, 1990. THE REACTOR WAS BROUGHT CRITICAL AT 2020 HOURS ON AUGUST 10, 1990 AND FULL POWER OPERATIONS REACHED ON AUGUST 12, 1990 AT 1200 HOURS.

[58] KEWAUNEE DOCKET 50-305 LER 90-002
PROCEDURAL ERROR RESULTS IN CLOSURE OF 5 CONTAINMENT ISOLATION VALVES AND STARTS TRAIN "A" AUXILIARY BUILDING SPECIAL VENTILATION FAN.
EVENT DATE: 031090 REPORT DATE: 040990 NSSS: WE TYPE: PWR

(NSIC 219670) ON 3/10/90, AT 0028, WITH THE PLANT IN REFUELING SHUTDOWN, TRAIN "A" AUX. BLDG. SPECIAL VENTILATION FAN STARTED AND CONTAINMENT ISOLATION VALVES RC-507, MG(R)-509, RBV-1, RBV-4 AND MU-1010-1 CLOSED. THE FAN AND VALVES ARE ENGINEERED SAFETY FEATURES (ESF). ACTUATIONS OCCURRED DURING IMPLEMENTATION OF DESIGN CHANGE REQUEST (DCR) PROCEDURE 2392-1. THE PROCEDURE INCORRECTLY DIRECTED THE OPERATOR TO OPEN THE SUPPLY BREAKER TO BRA-103, A 125V DC DISTRIBUTION CABINET. WHEN THE BREAKER WAS OPENED, CONTROL POWER WAS LOST TO THE FAN AND 3 CONTAINMENT ISOLATION VALVES. THE FAN AND THE VALVES ASSUMED THEIR POST-ACCIDENT POSITIONS. THIS EVENT WAS CAUSED BY AN ERROR IN THE PROCEDURE. CONTRIBUTING TO THIS EVENT WAS A MISUNDERSTANDING ON THE PART OF THE AUTHOR OF THE PROCEDURE. THE AUTHOR BELIEVED THAT BRA-103 WAS GOING TO BE CONNECTED TO AN ALTERNATE POWER SUPPLY PRIOR TO IMPLEMENTATION OF 2392-1. THIS MISUNDERSTANDING OCCURRED BECAUSE IN ADDITION TO DCR 2392, DCR 2393 IS ALSO SCHEDULED TO BE COMPLETED DURING THE CURRENT REFUELING OUTAGE. DCR 2393 WILL RELOCATE THE NON-SAFETY RELATED LOADS FROM BRA-103 TO A NEW NON-SAFETY RELATED BATTERY. DURING IMPLEMENTATION OF DCR 2393, BRA-103 WILL BE SUPPLIED WITH AN ALTERNATE POWER SOURCE. THE PROCEDURES ASSOCIATED WITH DESIGN CHANGE 2392 WERE RE-REVIEWED AND REVISED.

[59] LA SALLE 2 DOCKET 50-374 LER 90-004 REV 01
UPDATE ON LOCAL LEAK RATE TEST MINIMUM PATHWAY LEAKAGE OF GREATER THAN 0.6 LA LIMITS DURING THIRD REFUEL OUTAGE.
EVENT DATE: 032090 REPORT DATE: 092590 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 219659) ON 3/20/90, WHILE UNIT 2 WAS IN OPERATIONAL CONDITION 5 (REFUEL), FOR ITS THIRD REFUELING OUTAGE, LOCAL LEAK RATE TESTS HAD BEEN PERFORMED ON 2G33-F001 AND 2G33-F004, REACTOR WATER CLEANUP (RWCU) SUCTION ISOLATION VALVES, AND VARIOUS OTHER PRIMARY CONTAINMENT VALVES. THESE TESTS RESULTED IN A MINIMUM PATHWAY LEAKAGE WHICH EXCEEDED 0.6 LA LIMITS. TABLE 1 IS INCLUDED IN THE TEST OF THE REPORT TO SUMMARIZE THE CAUSES AND CORRECTIVE ACTIONS OF ALL LOCAL LEAKRATE FAILURES AND THE PRIMARY CONTAINMENT TYPE-A TEST "AS FOUND," INTEGRATED LEAKRATE TEST FAILURE THAT OCCURRED DURING THE UNIT 2, THIRD REFUELING OUTAGE. THIS EVENT IS BEING REPORTED PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(II).

[60] LA SALLE 2 DOCKET 50-374 LER 90-010
REACTOR SCRAM ON GENERATOR LOCKOUT DURING SURVEILLANCE TESTING DUE TO SHORT TO GROUND ON "B" PHASE CURRENT TRANSFORMER IN THE 2E MAIN POWER TRANSFORMER.
EVENT DATE: 091290 REPORT DATE: 101190 NSSS: GE TYPE: BWR

(NSIC 219755) ON 9/12/90 AT 0305 HOURS, WITH UNIT 2 IN OPERATIONAL CONDITION 1 (RUN) AT 99.9% POWER, THE UNIT 2 NUCLEAR STATION OPERATOR (NSO, LICENSED REACTOR OPERATOR) WAS PERFORMING THE TURBINE GENERATOR WEEKLY SURVEILLANCE (LOS-TG-W1). AFTER THE NSO HAD PERFORMED THE GENERATOR REGULATOR MODE TRANSFER SWITCH PORTION OF THIS SURVEILLANCE, HE NOTICED A SEVERE TRANSIENT ON SEVERAL GENERATOR OUTPUT INDICATIONS. A FEW SECONDS AFTER THIS, THE 2E MAIN POWER TRANSFORMER "B" PHASE DIFFERENTIAL RELAY ACTUATED WHEN THE CURRENT TRANSFORMER FEEDING THIS RELAY SHORTED OUT TO GROUND. AS A RESULT, THE UNIT 2 MAIN GENERATOR LOCKED OUT CAUSING THE MAIN TURBINE TO TRIP. THIS IMMEDIATELY CAUSED THE UNIT 2 REACTOR TO SCRAM. FIVE SAFETY RELIEF VALVES ACTUATED TO CONTROL REACTOR PRESSURE. THE INITIAL CAUSE OF THE TRANSIENT COULD NOT BE DETERMINED. THE "B" PHASE CURRENT TRANSFORMER THAT FEEDS THE "B" PHASE DIFFERENTIAL RELAY SCHEME WAS REPLACED. ALL AFFECTED CIRCUITS ASSOCIATED WITH THE GENERATOR'S EXCITER (ALTEREX) HAVE BEEN TESTED PRIOR TO TURBINE ROLL UP AND ONCE AGAIN AFTER THE UNIT WAS ON LINE. NO PROBLEMS COULD BE FOUND. UNIT 2 WAS RESTARTED AND THE GENERATOR WAS SYNCHRONIZED TO THE GRID AT 0430 HOURS ON 9/21/90. THIS EVENT IS REPORTABLE PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV) DUE TO THE ACTUATION OF AN ENGINEERED SAFETY FEATURE SYSTEM.

[61] LIMERICK 1 DOCKET 50-352 LER 90-014
REATOR WATER CLEAN-UP SYSTEM ISOLATION DUE TO HIGH AREA REGENERATIVE HEAT EXCHANGER TEMPERATURE.
EVENT DATE: 070590 REPORT DATE: 080690 NSSS: GE TYPE: BWR

(NSIC 219673) ON 7/5/90, AT 0502 HOURS, A GROUP III PRIMARY CONTAINMENT AND

REACTOR VESSEL ISOLATION CONTROL SYSTEM (PCRVICS) ISOLATION SIGNAL OCCURRED, AN ENGINEERED SAFETY FEATURE (ESF), INITIATING A REACTOR WATER CLEANUP (RWCU)(EIIS:CE) SYSTEM ISOLATION. THE PCRVICS ISOLATION SIGNAL WAS INITIATED WHEN THE STEAM LEAK DETECTION SYSTEM (DIVISION 1 AND 4)(EIIS:IJ) DETECTED HIGH TEMPERATURES IN THE REGENERATIVE HEAT EXCHANGER ROOM ABOVE THE 122F TRIP SETPOINT. THE HIGH AREA ROOM TEMPERATURE WAS A RESULT OF A COMBINATION OF SEVERAL FACTORS. FIRST, HIGH REACTOR ENCLOSURE (RE) SECONDARY CONTAINMENT AIR TEMPERATURES (APPROXIMATELY 83F) COMBINED WITH NORMAL VENTILATION BEING SECURED FOR SEVENTEEN MINUTES (FOR SURVEILLANCE TESTING) PRIOR TO THE ISOLATION CREATED A HIGH AREA TEMPERATURE CONDITION. SECONDLY, RWCU SYSTEM VALVES 44-1053 AND 44-1052, WHICH ARE LOCATED IN THE REGENERATIVE HEAT EXCHANGER ROOM AND WERE CLOSED AT THE TIME OF THE EVENT, WERE IDENTIFIED TO BE LEAKING STEAM PAST THEIR SEATS WHILE IN THE CLOSED POSITION. SURVEILLANCE TEST, ST-6-076-250-1, "SGTS AND RERS FLOW TEST" WAS REVISED TO INCLUDE A PRECAUTION TO CONSIDER THE EFFECTS OF SECURING RE VENTILATION WITH HIGH OUTSIDE AMBIENT TEMPERATURES. IN ADDITION, RWCU SYSTEM VALVES 44-1052 AND 44-1053 WILL BE REPAIRED TO CORRECT THE LEAKAGE DURING THE NEXT OUTAGE OF SUFFICIENT DURATION.

[62] LIMERICK 1 DOCKET 50-352 LER 90-017
ENGINEERED SAFETY FEATURE ACTUATION OF THE PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM DUE TO PROBLEMS WITH TESTING AND PERSONNEL ERROR.
EVENT DATE: 082890 REPORT DATE: 092690 NSSS: GE TYPE: BWR

(NSIC 219643) ON AUGUST 28, 1990, AT 0917 HOURS, WHILE UNIT 1 WAS AT POWER, AN INSTRUMENTATION AND CONTROLS (I&C) TECHNICIAN INADVERTENTLY SHORTED THE POWER SUPPLY TO AN INSTRUMENT RACK WHICH RESULTED IN A BLOWN FUSE DURING THE INSTALLATION OF A TEST JACK. THIS LOSS OF POWER CAUSED BY THE BLOWN FUSE RESULTED IN AUTOMATIC PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM (PCRVICS) ACTUATIONS OF UNIT 1 ISOLATION VALVES, ENGINEERED SAFETY FEATURES. THE BLOWN FUSE WAS THEN REPLACED BY THE I&C TECHNICIAN. ALL PCRVICS ISOLATIONS WERE RESET AND NORMAL SYSTEMS OPERATIONS RESTORED BY THE MAIN CONTROL ROOM OPERATORS BY 0935 HOURS. THE CONSEQUENCES OF THIS EVENT WERE MINIMAL. THE UNIT 1 PCRVICS ISOLATION VALVES FUNCTIONED AS DESIGNED. THE CAUSES OF THIS EVENT WERE A DESIGN THAT DOES NOT FACILITATE TESTING AND A FAILURE TO TAKE PRECAUTIONARY MEASURES DURING TEST JACK INSTALLATION. SEVERAL CORRECTIVE ACTIONS WILL BE IMPLEMENTED TO MINIMIZE THE POSSIBILITY OF SIMILAR EVENTS.

[63] LIMERICK 1 DOCKET 50-352 LER 90-018
COMMON PLANT WATER AND STEAM BARRIERS WERE IN A DEGRADED CONDITION AND PLACED THE PLANT IN AN UNANALYZED CONDITION THAT MAY HAVE COMPROMISED PLANT SAFETY.
EVENT DATE: 083090 REPORT DATE: 100190 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 219751) ON 8/30/90, PLANT OPERATIONS PERSONNEL IDENTIFIED THAT TWO DIAMOND PLATE FLOOR SECTIONS LOCATED IN THE UNIT 1 REACTOR ENCLOSURE, WHICH SERVE AS WATER/STEAM BARRIERS FOR A POSTULATED HIGH ENERGY PIPE BREAK (HEPB) OR MODERATE ENERGY PIPE BREAK (MEPB) ACCIDENT HAD BEEN OPENED. WATER/STEAM BARRIERS WERE INCORPORATED INTO THE DESIGN OF THE PLANT TO PROTECT AND CONTROL AGAINST DIRECT OR INDIRECT INDUCED LOSS OF EQUIPMENT AND COMPONENTS NECESSARY TO ASSURE SAFE SHUTDOWN OF THE PLANT IN THE EVENT OF A PIPING FAILURE. UPON FURTHER REVIEW, OTHER DEGRADED WATER/STEAM BARRIERS WERE IDENTIFIED IN UNIT 1 AND COMMON PLANT AREAS. THE DEGRADED WATER/STEAM BARRIERS COMPROMISED THE ABILITY OF THE BARRIERS TO MITIGATE THE EFFECTS OF A POSTULATED HEPB OR MEPB ACCIDENT. IT WAS CONCLUDED THAT THE CONDITION MAY HAVE SIGNIFICANTLY COMPROMISED PLANT SAFETY. THE IDENTIFIED DEGRADED BARRIERS WERE CORRECTED BY 9/2/90. INTERIM HEPB/MEPB BARRIER CONTROLS HAVE BEEN ESTABLISHED AND ARE BEING IMPLEMENTED UNTIL A COMPREHENSIVE PROGRAM IS DEVELOPED. THIS PLAN IS EXPECTED TO BE DEVELOPED BY 3/31/91. THE PRIMARY CAUSE OF THE DEGRADED WATER/STEAM BARRIERS IS THE RESULT OF THE LACK OF A PROGRAM AND PROPER IMPLEMENTING PROCEDURES TO CONTROL AND MAINTAIN WATER/STEAM BARRIERS. A DETAILED CAUSE ANALYSIS WILL BE PROVIDED IN A SUPPLEMENT TO THIS REPORT.

[64] LIMERICK 2 DOCKET 50-353 LER 90-013
REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION DUE TO PERSONNEL ERROR RESULTING IN PROCEDURAL NON-COMPLIANCE.
EVENT DATE: 083190 REPORT DATE: 093090 NSSS: GE TYPE: BWR

(NSIC 219644) ON AUGUST 31, 1990, AT 1100 HOURS, DURING THE PERFORMANCE OF A SURVEILLANCE TEST (ST) PROCEDURE FOR THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM (ST PROCEDURE ST-2-049-614-2), AN UNEXPECTED RCIC STEAM LEAK DETECTION SYSTEM SIGNAL WAS GENERATED. THIS RESULTED IN AN AUTOMATIC ACTUATION OF THE PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM (PCRVICS), AN ENGINEERED SAFETY FEATURE (ESF). THE PCRVICS ACTUATION RESULTED IN CLOSURE OF THE RCIC SYSTEM INBOARD PRIMARY CONTAINMENT ISOLATION VALVE, HV-49-2F007, AND AN ACTUATION OF A RCIC TURBINE TRIP SIGNAL. THE CAUSE OF THIS EVENT WAS A PERSONNEL ERROR DUE TO A LACK OF ATTENTION TO DETAIL RESULTING IN PROCEDURAL NON-COMPLIANCE. THE INSTRUMENTATION AND CONTROLS (I&C) TECHNICIANS DID NOT DISCONNECT THE TEST EQUIPMENT AND TERMINATE THE FIELD WIRES AS DIRECTED BY THE ST PROCEDURE. THERE WERE NO ADVERSE CONSEQUENCES AND NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT. ADDITIONALLY, THE HIGH PRESSURE COOLANT INJECTION SYSTEM WAS OPERABLE AND AVAILABLE FOR HIGH PRESSURE INJECTION TO THE REACTOR VESSEL. THE MAIN CONTROL ROOM OPERATORS RESET THE ISOLATION LOGIC AND RETURNED THE RCIC SYSTEM TO SERVICE WITHIN TEN MINUTES OF THE ISOLATION. THE PCRVICS, STEAM LEAK DETECTION SYSTEM, AND RCIC SYSTEM OPERATED AS DESIGNED.

[65] LIMERICK 2 DOCKET 50-353 LER 90-014
ENGINEERED SAFETY FEATURE ACTUATION OF THE PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM DUE TO PERSONNEL ERROR IN DISCONNECTING A WIRE.
EVENT DATE: 090690 REPORT DATE: 100590 NSSS: GE TYPE: BWR

(NSIC 219752) ON SEPTEMBER 6, 1990, AT 2056 HOURS, A UTILITY EMPLOYED INSTRUMENTATION AND CONTROLS (I&C) TECHNICIAN DISCONNECTED THE WRONG WIRE WHILE PERFORMING A SURVEILLANCE TEST (ST) PROCEDURE. THIS CAUSED AN AUTOMATIC ACTUATION OF THE PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM (PCRVICS), AN ENGINEERED SAFETY FEATURE. THE PCRVICS ACTUATION RESULTED IN AN ISOLATION OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM. THE ST PROCEDURE BEING PERFORMED AFFECTED THE LOGIC OF THE RWCU OUTBOARD PRIMARY CONTAINMENT ISOLATION VALVE, WHILE THE I&C TECHNICIAN ERRONEOUSLY DISCONNECTED A WIRE TO THE RWCU INBOARD PRIMARY CONTAINMENT ISOLATION VALVE. THE CAUSE OF THIS EVENT WAS LACK OF ATTENTION TO DETAIL RESULTING IN PROCEDURAL NON-COMPLIANCE. FOLLOWING THE ISOLATION, THE I&C TECHNICIAN RECONNECTED THE WIRE. MAIN CONTROL ROOM OPERATORS RESET THE ISOLATION AND RESTORED THE RWCU SYSTEM TO NORMAL OPERATION BY 2106 HOURS. THE CONSEQUENCES OF THIS EVENT WERE MINIMAL. THE PCRVICS ISOLATION VALVE ON RWCU SYSTEM FUNCTIONED AS DESIGNED. REACTOR WATER CHEMISTRY WAS NOT ADVERSELY AFFECTED BECAUSE OF THE SHORT DURATION OF THE ISOLATION. THE I&C TECHNICIAN PERFORMING THE ST WAS COUNSELED REGARDING PROPER WORK PRACTICES AND A HIGHER LEVEL OF ATTENTION TO DETAIL.

[66] LIMERICK 2 DOCKET 50-353 LER 90-015
REACTOR SCRAM RESULTING FROM A SPURIOUS TRIP SIGNAL FROM A STEAM LEAK DETECTION SYSTEM TEMPERATURE SWITCH.
EVENT DATE: 091090 REPORT DATE: 101090 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 1 (BWR)
VENDOR: RILEY COMPANY, THE

(NSIC 219753) ON SEPTEMBER 10, 1990, A UNIT 2 REACTOR PROTECTION SYSTEM (RPS) ACTUATION OCCURRED RESULTING FROM AN ACTUATION OF THE PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM (PCRVICS) ENGINEERED SAFETY FEATURE (ESF) ACTUATIONS. A PCRVICS GROUP I MAIN STEAM ISOLATION VALVE (MSIV) ISOLATION SIGNAL OCCURRED DUE TO A SPURIOUS TRIP SIGNAL ON THE 'D' CHANNEL OF THE STEAM LEAK DETECTION SYSTEM (SLDS) WHEN A TEMPERATURE SWITCH MOMENTARILY SPIKED WHEN AN OPERATOR REPOSITIONED THE SWITCH WHILE INSTRUMENTATION AND CONTROL TECHNICIANS WERE SIMULTANEOUSLY TESTING THE 'A' CHANNEL OF THE SLDS. REACTOR PRESSURE VESSEL (RPV) PRESSURE INCREASED TO 1120 PSIG AND RPV LEVEL DECREASED FROM A NORMAL OPERATING LEVEL OF +35 INCHES TO -48 INCHES INSTRUMENT LEVEL. ADDITIONAL PCRVICS ACTUATION SIGNALS WERE RECEIVED AND APPROPRIATE VALVE AND DAMPER MOTION OCCURRED.

THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM RECEIVED AN INITIATION SIGNAL ON LOW LOW RPV LEVEL AND INJECTED INTO THE RPV. OPERATORS RESTORED RPV PRESSURE AND LEVEL TO NORMAL POST SCRAM OPERATING LEVELS BY 0941 HOURS. THE CAUSE OF THE EVENT WAS A SPURIOUS TRIP SIGNAL FROM A STEAM LEAK DETECTION SYSTEM TEMPERATURE SWITCH. THE ISOLATION SIGNALS WERE RESET BY 0948 HOURS. THE TEMPERATURE SWITCH WAS REPLACED AND THE FAILURE MECHANISM OF THE TEMPERATURE SWITCH IS UNDER INVESTIGATION.

[67] MCGUIRE 1 DOCKET 50-369 LER 89-010 REV 02
UPDATE ON MAIN FEEDWATER AND AUXILIARY FEEDWATER ISOLATION VALVES WERE POTENTIALLY INOPERABLE BECAUSE OF A MANUFACTURING DEFICIENCY.
EVENT DATE: 051589 REPORT DATE: 061489 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)
VENDOR: BORG-WARNER CORP.
LIMITORQUE CORP.
ROTORK INC.

(NSIC 219688) ON MAY 15, 1989, STATION MANAGEMENT PERSONNEL DETERMINED THAT VALVES CF-126B, 127B, 128B, AND 129B, MAIN FEEDWATER TO AUXILIARY FEEDWATER NOZZLE ISOLATION, AND VALVES CA-38B, 50B, 54AC, AND 66AC, AUXILIARY FEEDWATER PUMP TO STEAM GENERATOR, WERE POTENTIALLY INOPERABLE ON UNITS 1 AND 2 BECAUSE UNCORRELATED TESTING AND DESIGN ENGINEERING CALCULATIONS HAD SHOWN THAT THESE VALVES MAY NOT CLOSE UNDER DESIGN DIFFERENTIAL PRESSURE CONDITIONS. AT 1600, VALVES CA-126B, 127B, 128B, AND 129B WERE CLOSED BY OPERATIONS PERSONNEL AND POWER WAS REMOVED. ON MAY 16, 1989, OPERATIONS PERSONNEL CHANGED EMERGENCY PROCEDURES TO STRESS THAT MANUAL ACTIONS MAY BE NECESSARY TO CLOSE VALVES CA-38B, 50B, 45AC, AND 66AC. VALVES CF-126B, 127B, 128B, AND 129B WERE RETURNED TO OPERABLE STATUS AFTER A 95% TORQUE SWITCH BYPASS MODIFICATION WAS PERFORMED ON THE UNIT 2 VALVES ON JUNE 7, 1989 AND THE UNIT 1 VALVES ON JUNE 8, 1989. THE CAUSE OF THE VALVES TO FAIL TO CLOSE UNDER HIGH DIFFERENTIAL PRESSURE CONDITIONS HAS NOT BEEN DETERMINED BUT IS APPARENTLY A MANUFACTURER'S DESIGN DEFICIENCY. THIS EVENT IS ASSIGNED A CAUSE OF MANUFACTURING DEFICIENCY. UNIT 1 AND UNIT 2 WERE IN MODE 1, POWER OPERATION, AT 100% POWER WHEN THE OPERABILITY CONCERN WITH THESE VALVES WAS DETERMINED.

[68] MCGUIRE 1 DOCKET 50-369 LER 90-019 REV 02
UPDATE ON A DAILY CHANNEL CHECK FOR THE CONTAMINATED PARTS WAREHOUSE VENTILATION RADIATION MONITOR WAS MISSED BECAUSE OF INAPPROPRIATE ACTIONS.
EVENT DATE: 071190 REPORT DATE: 081090 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 219754) ON JUNE 1, 1990, A REVISION TO PROCEDURE PT/1/A/4600/03B, DAILY SURVEILLANCE ITEMS, WAS PUT INTO AFFECT. DURING THE REVISION PROCESS OF THIS PROCEDURE THE DAILY CHANNEL CHECK FOR EMF 53, RADIATION MONITOR FOR THE CONTAMINATED PARTS WAREHOUSE VENTILATION SYSTEM, WAS INADVERTENTLY DELETED FROM THE DAILY SURVEILLANCE ITEMS PROCEDURE CHECKLIST. NEITHER THE PREPARER OF THE PROCEDURE REVISION NOR THE QUALIFIED REVIEWER FOR THE PROCEDURE REVISION NOTICED THAT THE CHANNEL CHECK FOR EMF 53 WAS MISSING FROM THE CHECKLIST. ON JULY 11, 1990, DURING PERFORMANCE OF THE DAILY SURVEILLANCE ITEMS PROCEDURE, IT WAS DETERMINED THAT EMF 53 HAD BEEN INADVERTENTLY LEFT OFF THE CHECKLIST. THIS EVENT IS ASSIGNED CAUSES OF INAPPROPRIATE ACTIONS BECAUSE OF A LACK OF ATTENTION TO DETAIL BY OPERATIONS PERSONNEL. UNIT 1 AND UNIT 2 WERE IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER AT THE TIME OF DISCOVERY OF THIS EVENT. UNIT 1 VARIED IN POWER LEVELS FROM 5 PERCENT TO 100 PERCENT POWER WHILE THE TECHNICAL SPECIFICATION (TS) SURVEILLANCE WAS NOT BEING PERFORMED FOR EMF 53. UNIT 2 VARIED IN POWER FROM 93 PERCENT POWER TO 100 PERCENT POWER DURING THIS SAME TIME PERIOD. OPERATIONS PROCEDURE GROUP PERSONNEL IMMEDIATELY REVISED THE DAILY SURVEILLANCE ITEMS PROCEDURE BY ADDING THE DAILY CHANNEL CHECK FOR EMF 53 BACK ONTO THE CHECKLIST.

[69]   MCGUIRE 1   DOCKET 50-369   LER 90-025
SHUTDOWN BECAUSE OF UNIDENTIFIED REACTOR COOLANT SYSTEM LEAKAGE GREATER THAN TECHNICAL SPECIFICATION LIMITS.
EVENT DATE: 082790   REPORT DATE: 092690   NSSS: WE   TYPE: PWR
VENDOR: BORG-WARNER CORP.

(NSIC 219646) ON SEPTEMBER 27, 1990 AT 0554 HOURS, UNIT 1 BEGAN REDUCING LOAD TO COMPLY WITH THE UNIDENTIFIED LEAKAGE TECHNICAL SPECIFICATION. LEAKAGE CALCULATIONS INDICATED THAT UNIDENTIFIED LEAKAGE WAS GREATER THAN 1 GALLON PER MINUTE. AN UNUSUAL EVENT WAS DECLARED AT THIS TIME. UNIT 1 ENTERED MODE 3 (HOT STANDBY), AT 1013 HOURS. ENTRIES WERE MADE INTO CONTAINMENT AND PRESSURIZER PORV HEADER HI POINT VENT VALVE, 1NC-252, WAS FOUND TO HAVE A PACKING LEAK OF APPROXIMATELY 8 OUNCES PER HOUR. THE PACKING WAS ADJUSTED AND THE LEAK STOPPED. IN ADDITION, PRESSURIZER RELIEF ISOLATION VALVE, 1NC-33, WAS FOUND TO HAVE A PACKING LEAK AND THE VALVE STEM LEAKOFF LINE WAS SEPARATED FROM THE DRAIN PIPING. THE LEAKOFF LINE WAS REPLACED AND VALVE 1NC-33 WAS BACKSEATED TO STOP ITS LEAKAGE. THE PACKING ON VALVE 1NC-33 WAS TORQUED TO ITS MAXIMUM ALLOWED VALUE. THE POWER TO THE ASSOCIATED PRESSURIZER POWER OPERATED RELIEF VALVE SOLENOID WAS DEENERGIZED BECAUSE VALVE 1NC-33 IS TECHNICALLY INOPERABLE WHILE BACKSEATED. VALVE 1NC-33 WILL BE REPACKED DURING THE NEXT OUTAGE OF SUFFICIENT DURATION. THIS EVENT IS ASSIGNED A CAUSE OF EQUIPMENT FAILURE. UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER AT THE TIME THE LEAKAGE WAS DISCOVERED.

[70]   MILLSTONE 1   DOCKET 50-245   LER 90-009 REV 01
UPDATE ON HOUSE HEATING STEAM HIGH ENERGY LINE BREAK.
EVENT DATE: 051190   REPORT DATE: 100590   NSSS: GE   TYPE: BWR

(NSIC 219693) ON 5/1/90, WITH THE PLANT AT 100% POWER (530F AND 1030 PSIG), IT WAS DETERMINED THAT THE HOUSE HEATING STEAM SYSTEM, WHICH CLASSIFIED AS A HIGH ENERGY LINE BREAK (HELB) SYSTEM, COULD POTENTIALLY DEGRADE ENVIRONMENTALLY CLASSIFIED "EEQ MILD ENVIRONMENTS". UPON RECOGNITION OF THIS POTENTIALLY UNANALYZED CONDITION, THE HOUSE HEATING STEAM SYSTEM WAS REMOVED FROM SERVICE AT 1715 HOURS. A REPORTABILITY EVALUATION WAS IMMEDIATELY INITIATED TO DETERMINE IF A REPORTABLE CONDITION EXISTED. ON 5/11/90, PRELIMINARY RESULTS OF THE REPORTABILITY EVALUATION CONCLUDED THE CONSEQUENCES OF A HOUSE HEATING STEAM LINE RUPTURE IN "EEQ MILD ENVIRONMENTS" WOULD HAVE A POTENTIALLY NEGATIVE IMPACT ON CLASS 1E EQUIPMENT REQUIRED FOR SHUTDOWN. TO INSURE OTHER HIGH ENERGY SOURCES HAVE NOT BEEN OVERLOOKED RELATIVE TO THEIR POTENTIAL IMPACT ON EQUIPMENT REQUIRED FOR SAFE SHUTDOWN FOLLOWING A HELB, A MINI REVIEW OF THE 1973 HELB STUDY WAS IMMEDIATELY PERFORMED. THE REVIEW CONCLUDED THAT THERE WERE NO OTHER MAJOR PROBLEM AREAS AND THE AUXILIARY STEAM SYSTEM REPRESENTED AN ISOLATED CASE. THE REVIEW IDENTIFIED A CONCERN THAT THE HELB STUDY WAS A 1973 SNAPSHOT, RATHER THAN A LIVING DOCUMENT AGAINST WHICH PLANT MODIFICATIONS ARE REVIEWED AND DETERMINED ACCEPTABLE. WHILE THE STUDY, TO DATE, HAS NOT FOUND SPECIFIC INSTANCES OF PLANT MODIFICATIONS THAT ADVERSELY IMPACTED THE HELB STUDY RESULTS, ALL AREAS HAVING THIS POTENTIAL WILL BE VALIDATED.

[71]   MILLSTONE 2   DOCKET 50-336   LER 90-012
REACTOR TRIP DUE TO OPERATOR ERROR.
EVENT DATE: 082790   REPORT DATE: 092690   NSSS: CE   TYPE: PWR

(NSIC 219657) ON 8/27/90, AT 0100, WITH THE REACTOR PLANT IN MODE 1 (100% POWER, 575F, 2263 PSIG), WHILE PERFORMING SURVEILLANCE PROCEDURE SP 2601D "POWER RANGE SAFETY CHANNEL AND DELTA T POWER CHANNEL CALIBRATION" WITH 'A' RPS CHANNEL INOPERABLE (TRIPPED), THE PLANT OPERATOR DID NOT BYPASS THE SECOND CHANNEL BEFORE TESTING IT, WHICH CAUSED AN AUTOMATIC PLANT TRIP. OPERATORS THEN PERFORMED EMERGENCY OPERATING PROCEDURE EOP 2525. ALL EQUIPMENT RESPONDED AS EXPECTED AND THE UNIT WAS PLACED IN A STABLE CONDITION. THE CAUSE OF THE EVENT WAS OPERATOR ERROR IN THAT THE OPERATOR FAILED TO PERFORM THE PRESCRIBED STEPS WHEN INITIATING THE DAILY POWER RANGE SAFETY CHANNEL AND DELTA T POWER CHANNEL CALIBRATION. THE LICENSED REACTOR OPERATOR INVOLVED HAS BEEN INSTRUCTED IN THE PROPER SEQUENCE OF BYPASS SWITCH OPERATIONS AND THE ENTIRE OPERATIONS DEPARTMENT HAS DISCUSSED THE IMPORTANCE OF EVALUATING PLANT CONDITIONS AND SELF-VERIFICATION AT SUBSEQUENT DEPARTMENT MEETINGS. IN ADDITION, PROCEDURE SP 2601D HAS BEEN REVISED TO

INCORPORATE A SEPARATE SECTION ON PERFORMING THE CALIBRATIONS WITH ONE RPS CHANNEL INOPERABLE, INCLUDING A SIGNATURE REQUIREMENT FOR VERIFICATION THAT THE BYPASS LIGHTS ARE ENERGIZED ON THE CHANNEL TO BE TESTED. THIS EVENT IS BEING REPORTED PURSUANT TO THE REQUIREMENTS OF PARAGRAPH 50.73(A)(2)(IV), REPORTING ANY EVENT OR CONDITION THAT RESULTED IN MANUAL OR AUTOMATIC ACTUATION OF ANY ENGINEERED SAFETY FEATURE SYSTEM.

[72] MILLSTONE 2 DOCKET 50-336 LER 90-013
FAILURE TO RECORD SURVEILLANCE REQUIRED DATA.
EVENT DATE: 090590 REPORT DATE: 100490 NSSS: CE TYPE: PWR

(NSIC 219743) ON 9/5/90 AT 1400 HOURS, WHILE THE PLANT WAS AT 100% POWER, MODE 1, AN INTERNAL AUDIT IDENTIFIED TWO DATES THAT TECH SPEC REQUIRED SURVEILLANCE PROCEDURE 2602B WAS MISSED. TO PRECLUDE AN OVER-PRESSURE CONDITION IN THE STEAM GENERATORS, THIS SURVEILLANCE CALLS FOR RECORDING STEAM GENERATOR COOLANT TEMPERATURES AT LEAST ONCE PER HOUR WHEN PRESSURES IN THE STEAM GENERATOR ARE GREATER THAN 200 PSIG AND TAVE IS LESS THAN 200 DEGREES. AN INVESTIGATION HAS SINCE SHOWN ADDITIONAL DATES THAT THESE TEMPERATURES WERE NOT RECORDED, OR FAILED TO BE IDENTIFIED AS NOT MEETING THE ACCEPTANCE CRITERIA. THE ROOT CAUSE WAS PROCEDURE INADEQUACY AND OPERATOR INATTENTION TO DETAIL. INSUFFICIENT PROCEDURAL GUIDANCE WAS PROVIDED TO COMPLETE THE SURVEILLANCE. CORRECTIVE ACTION INCLUDED A PROCEDURE CHANGE TO PRECLUDE USE OF THE PLANT PROCESS COMPUTER TO COLLECT DATA. CHANGES ARE BEING EFFECTED TO MODIFY THE COMPUTER PROGRAM, TO ASSIST THE OPERATOR IN RECORDING THE REQUIRED DATA. ADDITIONALLY, ALL PROCEDURES THAT COULD PLACE THE PLANT IN THE CONDITIONS UNDER WHICH SURVEILLANCE 3.7.2.1 WOULD BE REQUIRED, ARE BEING CHANGED TO REFERENCE THIS SURVEILLANCE. DUE TO AN OUT OF SPECIFICATION CONDITION, A DETERMINATION OF STEAM GENERATOR ACCEPTABILITY FOR CONTINUED OPERATION IS ONGOING.

[73] MILLSTONE 3 DOCKET 50-423 LER 88-001 REV 01
UPDATE ON INADVERTENT SAFETY INJECTION DUE TO SENSITIVE EQUIPMENT.
EVENT DATE: 010588 REPORT DATE: 100190 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219682) ON 1/5/88 AT 1630 WITH THE PLANT AT 0% POWER, 110 DEGREES AND AMBIENT PRESSURE, AN INADVERTENT SAFETY INJECTION SIGNAL WAS RECEIVED ON TRAIN A. THE SIGNAL WAS NOT REQUIRED FOR SAFETY AND NO INJECTION TO THE CORE TOOK PLACE. THE IMMEDIATE CAUSE OF THE EVENT WAS AN IMPROPER SWITCH POSITION DURING RESTORATION FROM INSTRUMENT AND CONTROLS TESTING. THE MAIN BOARD OPERATOR WAS COGNIZANT OF THE RESTORATION BEING PERFORMED AND THE POTENTIAL FOR AN INADVERTENT SAFETY INJECTION. HIS ACTIONS, AND THOSE OF THE INSTRUMENT TECHNICIAN, WERE IN COMPLIANCE WITH THE RESTORATION PROCEDURE. IMMEDIATE ACTION WAS TO VERIFY THAT A SAFETY INJECTION WAS NOT REQUIRED AND RESET THE SIGNAL. NO ENGINEERED SAFETY FEATURES ACTUATIONS WERE REQUIRED FOR EXISTING PLANT CONDITIONS AND NONE OCCURRED. THE ROOT CAUSE OF THE EVENT IS A SENSITIVE SWITCH. AS AN INTERIM SOLUTION, THE OPERATING PROCEDURE HAS BEEN CHANGED TO VERIFY BY INDICATION THAT APPROPRIATE BLOCKS AND RESETS ARE IN PLACE PRIOR TO TAKING AN ACTION THAT COULD RESULT IN A SAFETY INJECTION. TO PREVENT RECURRENCE OF THIS PROBLEM, THIS SWITCH WILL BE REPLACED BY A LESS SENSITIVE DESIGN PRIOR TO THE END OF THE THIRD REFUELING OUTAGE (SCHEDULED FOR FEBRUARY, 1991). SIMILAR SWITCHES ON THE MAIN CONTROL BOARDS WILL BE EVALUATED AND ALSO CHANGED IF REQUIRED.

[74] MILLSTONE 3 DOCKET 50-423 LER 90-020 REV 01
UPDATE ON BOTH TRAINS OF QUENCH SPRAY AND HIGH PRESSURE SAFETY INJECTION INOPERABLE DUE TO DEFICIENT SURVEILLANCE PROCEDURE.
EVENT DATE: 061490 REPORT DATE: 100190 NSSS: WE TYPE: PWR

(NSIC 219770) ON 6/14/90, AT 1725 HOURS, WITH THE PLANT IN MODE 1, AT 80% POWER, THE "B" TRAIN RHR AREA COOLER UNIT, 3HVQ*ACUS1B, WAS FOUND TO BE DEGRADED DUE TO HEAT EXCHANGER FOULING. SUBSEQUENT ENGINEERING ANALYSIS DETERMINED THAT BOTH TRAINS OF HIGH PRESSURE SAFETY INJECTION AND QUENCH SPRAY WERE RENDERED INOPERABLE WHEN THE "A" TRAIN HIGH PRESSURE SAFETY INJECTION (HPSI) AND QUENCH SPRAY (QSS) PUMPS WERE TAKEN OUT OF SERVICE WHILE THE 3HVQ*ACUS1B UNIT WAS

INOPERABLE. THE "A" TRAIN HPSI AND QSS PUMPS WERE REMOVED FROM SERVICE WHILE THE EVALUATION OF THE OPERABILITY OF 3HVQ*ACUS1B WAS IN PROGRESS. RHR AREA COOLERS PROVIDE COOLING IN AN ACCIDENT CONDITION TO THE RHR, HPSI AND QSS PUMPS. BOTH TRAINS OF QSS WERE INOPERABLE FOR APPROXIMATELY 10 HOURS. BOTH TRAINS OF HPSI WERE INOPERABLE FOR APPROXIMATELY 11 HOURS. THE 3HVQ*ACUS1A UNIT WAS OPERABLE THE ENTIRE PERIOD OF TIME. ROOT CAUSE OF THE EVENT WAS A DEFICIENT SURVEILLANCE PROCEDURE. THE PROCEDURE REQUIRED AN ENGINEERING EVALUATION BE PERFORMED IF OPERABILITY WAS IN QUESTION WITHOUT ADMINISTRATIVELY CONTROLLING THE REMOVAL OF OPPOSITE TRAIN EQUIPMENT DURING THE EVALUATION. THE FOULED HEAT EXCHANGER WAS CLEANED AND PUT BACK IN SERVICE. CORRECTIVE ACTIONS INCLUDE CHANGING THE SURVEILLANCE PROCEDURE, CLEANING THE HEAT EXCHANGER AND VISUAL INSPECTION OF THE HEAT EXCHANGER TWICE EACH OPERATING CYCLE.

[75] MILLSTONE 3 DOCKET 50-423 LER 90-023 REV 01
UPDATE ON BOTH TRAINS OF CONTAINMENT RECIRCULATION PUMP AREA COOLERS INOPERABLE DUE TO AN INADEQUATE SURVEILLANCE PROCEDURE.
EVENT DATE: 061590 REPORT DATE: 100190 NSSS: WE TYPE: PWR
VENDOR: GRAHAM MFG CO.

(NSIC 219771) ON JULY 5, 1990 AT APPROX. 1005 HOURS, WHILE IN MODE 1 AT 80% POWER, 580F AND 2250 PSIA, A DETAILED ENGINEERING EVALUATION DETERMINED THAT BOTH TRAINS OF CONTAINMENT RECIRCULATION SPRAY HAD BEEN INOPERABLE SIMULTANEOUSLY, DUE TO FOULING OF THE ASSOCIATED AREA COOLER HEAT EXCHANGERS. ON 6/18/90 AT 1300 HOURS, THE "B" TRAIN CONTAINMENT RECIRCULATION PUMP AREA COOLER WAS FOUND TO BE DEGRADED WHEN DAMAGE AND FOULING WAS FOUND IN THE HEAT EXCHANGER HEADS. ON 6/15/90, SIMILAR DAMAGE HAD BEEN FOUND IN THE "A" TRAIN UNIT AND REPAIRED. SUBSEQUENT ENGINEERING ANALYSIS DETERMINED THAT BOTH TRAINS OF THE CONTAINMENT RECIRCULATION PUMP AREA COOLERS WOULD NOT HAVE BEEN ABLE TO SATISFY THEIR DESIGN BASIS ACCIDENT HEAT LOAD. THE ROOT CAUSE OF THE EVENT WAS AN INADEQUATE SURVEILLANCE PROCEDURE WHICH WAS NOT DESIGNED TO DETECT THE DAMAGE WHICH WAS DISCOVERED IN THIS EVENT. AS CORRECTIVE ACTION, THE DAMAGED HEAT EXCHANGERS WERE CLEANED AND REPAIRED. THE HEAT EXCHANGERS WILL BE VISUALLY INSPECTED TWICE EACH OPERATING CYCLE.

[76] MILLSTONE 3 DOCKET 50-423 LER 90-028
CONTROL BUILDING ISOLATION DUE TO RADIATION MONITOR DETECTOR FAILURE.
EVENT DATE: 090490 REPORT DATE: 100390 NSSS: WE TYPE: PWR
VENDOR: KAMAN SCIENCES CORP.

(NSIC 219772) AT 1240 HOURS ON 9/4/90 WHILE OPERATING IN MODE 1, 100% REACTOR POWER, 557F AND 2250 PSIA, A CONTROL BUILDING ISOLATION (CBI) SIGNAL WAS INITIATED BY THE "B" TRAIN CONTROL BUILDING INLET VENTILATION RADIATION MONITOR DUE TO EQUIPMENT FAILURE. THE "A" TRAIN MONITOR INDICATED ONLY NORMAL BACKGROUND RADIATION. CONTROL ROOM OPERATORS BLOCKED THE CBI AND INITIATED INVESTIGATIONS. OTHER EQUIPMENT AND SYSTEMS WERE NOT AFFECTED. RADIATION LEVELS IN THE CONTROL ROOM DID NOT INCREASE. ROOT CAUSE OF THE EVENT WAS DETECTOR FAILURE WHICH PRODUCED INDICATED RADIATION LEVELS WHICH EXCEEDED THE HIGH RADIATION ALARM SETPOINT AND INITIATED THE CBI SIGNAL. THE FAILED "B" TRAIN DETECTOR WAS REPAIRED. PERFORMANCE OF THIS MODEL DETECTOR IN OTHER APPLICATIONS WAS FOUND TO BE SATISFACTORY. THE DETECTOR SUPPLIER HAD NO RECORDS OF HIGH FAILURE RATES FOR THESE DETECTORS. AN NPRDS SEARCH REVEALED ONLY ONE OTHER FAILURE OF A DETECTOR FROM THIS MANUFACTURER WHICH COULD POSSIBLY BE THE SAME MODEL. A REPORT ON THIS EVENT WILL BE SUBMITTED ON NPRDS.

[77] MONTICELLO DOCKET 50-263 LER 90-012
LEAK FROM THREADED FITTING FATIGUE CRACK REQUIRES MANUAL ISOLATION OF REACTOR WATER CLEANUP SYSTEM.
EVENT DATE: 082290 REPORT DATE: 092190 NSSS: GE TYPE: BWR
VENDOR: SANDVIK SPECIAL METALS CORP.

(NSIC 219609) AT APPROXIMATELY 1400 CST ON AUGUST 22, 1990, WITH #11 REACTOR WATER CLEANUP RECIRCULATION PUMP IN SERVICE, AN OPERATOR DISCOVERED WATER LEAKING FROM A PENETRATION IN A WALL OF REACTOR WATER CLEANUP PUMP ROOM. WHILE AN

OPERATOR WAS PREPARING FOR ENTRY INTO THE #11 REACTOR WATER CLEANUP PUMP ROOM, THE REACTOR BUILDING HIGH AREA RADIATION ALARM FOR THE REACTOR BUILDING DRAIN TANK AREA WAS RECEIVED IN THE MAIN CONTROL ROOM. THIS ALARM CONDITION INITIATED ENTRY INTO EMERGENCY OPERATING PROCEDURES. THE #11 REACTOR WATER CLEANUP PUMP WAS SHUTDOWN AND REACTOR WATER CLEANUP PRIMARY CONTAINMENT ISOLATION VALVES WERE CLOSED TO PROVIDE EXPEDITIOUS ISOLATION OF THE LEAK. AT 1525 CST THE REACTOR BUILDING HIGH AREA RADIATION ALARMS CLEARED. THE EMERGENCY OPERATING PROCEDURES WERE EXITED. AT APPROXIMATELY 1535, THE REACTOR WATER CLEANUP SYSTEM WAS RESTORED TO SERVICE WITH #12 REACTOR WATER CLEANUP PUMP IN SERVICE. INSPECTION SHOWED THAT THE LEAK WAS FROM A CRACK IN #11 REACTOR WATER CLEANUP PUMP VENT LINE THREADED PIPE FITTING AT THE CONNECTION TO THE PUMP CASING. THE ROOT CAUSE OF THE VENT LINE FITTING FAILURE HAS BEEN DETERMINED TO BE VIBRATION INDUCED FATIGUE CRACKING. THE VENT LINE IS BEING MODIFIED TO REDUCE THE SUSCEPTIBILITY OF THE VENT LINE TO FATIGUE CRACKING.

[78] MONTICELLO DOCKET 50-263 LER 90-013
DISCOVERY OF NON-CONSERVATIVE ASSUMPTIONS IN ORIGINAL PLANT FLOODING ANALYSIS.
EVENT DATE: 091190 REPORT DATE: 101190 NSSS: GE TYPE: BWR

(NSIC 219698) ON SEPTEMBER 11, 1990, PLANT ENGINEERING STAFF QUESTIONED THE INTEGRITY OF FIRE PROTECTION PIPING IN THE DIVISION II EMERGENCY DIESEL GENERATOR ROOM. AT THE TIME, IT WAS BELIEVED THAT FAILURE OF THE PIPE COULD RESULT IN FLOODING THAT COULD MAKE BOTH EMERGENCY DIESEL GENERATORS INOPERABLE. AT THAT TIME, THE NRC WAS NOTIFIED PURSUANT TO 10CFR 50.72(B)(2)(III). FURTHER EVALUATION HAS SHOWN THAT THE PIPE WOULD NOT HAVE FAILED UNDER SAFE SHUTDOWN EARTHQUAKE CONDITIONS. HOWEVER, THIS LICENSEE EVENT REPORT (LER) IS BEING SUBMITTED VOLUNTARILY SINCE FOLLOW-UP INVESTIGATION INDICATED THAT NON-CONSERVATIVE ASSUMPTIONS WERE MADE IN THE ORIGINAL FLOODING ANALYSIS PERFORMED IN 1972. CORRECTIVE ACTIONS INCLUDED ISOLATION OF THE QUESTIONABLE FIRE HEADER, POSTING OF FIRE WATCHES, REVIEW OF THE REMAINDER OF THE FIRE PROTECTION SYSTEMS AND INSTALLATION OF A NEW PIPE SUPPORT PLANNED CORRECTIVE ACTION INCLUDES A DETAILED REVIEW OF THE DESIGN BASIS DOCUMENTATION FOR INTERNAL FLOODING TO ENSURE A CONSERVATIVE APPROACH WAS TAKEN.

[79] NINE MILE POINT 1 DOCKET 50-220 LER 88-014 REV 01
UPDATE ON FAILURE OF CORE SPRAY HIGH POINT VENT ISOLATION VALVE 40-30 TO MEET STROKE REQUIREMENT DUE TO PROCEDURAL DEFICIENCIES.
EVENT DATE: 051088 REPORT DATE: 100490 NSSS: GE TYPE: BWR
VENDOR: LIMITORQUE CORP.

(NSIC 219679) ON MAY 10, 1988, WITH NINE MILE POINT UNIT 1 (NMP1) IN A REFUELING OUTAGE, IT WAS IDENTIFIED THAT THE ACTUAL FULL-OPEN TO FUEL CLOSED STROKE TIME FOR CORE SPRAY HIGH POINT VENT ISOLATION VALVE (IV) 40-30 HAD EXCEEDED ITS TECHNICAL SPECIFICATION (T.S.) LIMIT. THIS WAS DISCOVERED WHILE CONDUCTING RESEARCH FOR SIGNIFICANT OPERATING EXPERIENCE REPORT (SOER) 86-2, "INACCURATE CLOSED POSITION INDICATION ON MOTOR-OPERATED VALVES". IN ACCORDANCE WITH SITE PROCEDURES, VALVE CLOSURE TIME IS MEASURED FROM THE TIME THE CONTROL SWITCH IS TAKEN TO THE CLOSED POSITION TO THE TIME THE RED INDICATING LAMP GOES OUT. FOR IV 40-30, THE RED LAMP WENT OUT WHEN THE VALVE WAS 89 PERCENT CLOSED. BASED ON A REVIEW OF TEST RESULTS FROM APRIL 30, 1986, THE ADDITIONAL 11 PERCENT OF VALVE TRAVEL RESULTED IN THE ACTUAL FULL OPEN TO FULL CLOSED STROKE TIME TO EXCEED THE T.S. REQUIREMENT OF 30 SECONDS. THE ROOT CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE PERSONNEL ERROR AS A RESULT OF A PROCEDURAL DEFICIENCY. ACTION TO CORRECT THE VALVE'S STROKE TIME WAS NOT INITIATED BECAUSE THE PROCEDURE DID NOT IDENTIFY T.S. CRITERIA. CORRECTIVE ACTIONS INCLUDE VALVE MAINTENANCE, PROCEDURAL REVISIONS, AND MODIFICATIONS TO MOTOR OPERATED VALVES TO ENSURE THE INDICATING LIGHTS ACCURATELY REPRESENT ACTUAL VALVE POSITION.

[80] NINE MILE POINT 1 DOCKET 50-220 LER 89-003 REV 01
UPDATE ON PROCEDURAL DEFICIENCY RESULTING IN TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 031189 REPORT DATE: 092490 NSSS: GE TYPE: BWR
VENDOR: BLISS COMPANY

(NSIC 219602) ON MARCH 11, 1989 WITH NINE MILE POINT UNIT 1 (NMP1) IN COLD SHUTDOWN, WITH THE CORE OFF-LOADED, IT WAS DETERMINED THAT THE VESSEL ISOLATION SIGNAL GENERATED FROM THE MAIN STEAM LINE (MSL) RADIATION MONITORS WAS NOT BEING TESTED IN ACCORDANCE WITH TECHNICAL SPECIFICATION (TECH SPEC) SURVEILLANCE REQUIREMENTS. THE CAUSE OF THIS EVENT WAS A PROCEDURAL DEFICIENCY. THE ROOT CAUSE OF THIS EVENT WAS AN INEFFECTIVE MANAGEMENT OF THE TECHNICAL REVIEW PROCESS OF SURVEILLANCE PROCEDURES IMPACTING TECH SPECS. CONTRIBUTING FACTORS INCLUDE MISINTERPRETATION OF THE TECH SPEC REQUIREMENTS, AND INADEQUATE TECHNICAL REVIEW OF PROCEDURES. INITIAL CORRECTIVE ACTION INVOLVED PERFORMING THE SURVEILLANCE TEST TO VERIFY THAT THE VESSEL ISOLATION SIGNAL FROM THE MSL RADIATION MONITORS WAS OPERABLE AND TO REVISE THE PROCEDURE TO DOCUMENT THIS FUNCTION. SUBSEQUENTLY, SURVEILLANCE PROCEDURES THAT MEET A TECH SPEC REQUIRED SURVEILLANCE (E.G. INSTRUMENT CHANNEL TEST, INSTRUMENT CHANNEL CALIBRATION, OR AUTOMATIC INITIATION) HAVE BEEN INDEPENDENTLY REVIEWED FOR TECHNICAL ACCURACY. THE DEFICIENCIES IDENTIFIED HAVE BEEN CORRECTED & WHEN NECESSARY THE TESTS WERE REPERFORMED. THIS SUPPLEMENT REPORTS THE DEFICIENCIES IDENTIFIED DURING THIS REVIEW.

[81] NINE MILE POINT 1 DOCKET 50-220 LER 90-007
FAILED SURVEILLANCE RESULTING FROM LEAK IN REVERSE FLOW DIRECTION OF THE CONTAINMENT SPRAY INLET CHECK VALVE.
EVENT DATE: 051590 REPORT DATE: 061590 NSSS: GE TYPE: BWR

(NSIC 219667) ON 5/15/90, AT 2215 HOURS, WITH NINE MILE POINT UNIT 1 IN A MAJOR REFUELING OUTAGE, THE CONTAINMENT SPRAY INLET CHECK VALVE #80-38 WAS DETERMINED TO LEAK IN THE REVERSE FLOW DIRECTION RESULTING IN THE UNSATISFACTORY PERFORMANCE OF SURVEILLANCE TEST PROCEDURE N1-ST-R10, "DRYWELL TO TORUS LEAK RATE TEST". AN INSPECTION WAS PERFORMED OF THE VALVE INTERNALS WHICH IDENTIFIED THAT A PLASTIC SAMPLE BOTTLE WAS WEDGED BETWEEN THE DISC AND SEAT OF THE VALVE, HOLDING THE VALVE OPEN. THE SOURCE OF THE BOTTLE BEING INTRODUCED IN TO THE SYSTEM IS UNKNOWN. IT APPEARS THAT THE BOTTLE MAY HAVE BEEN INTRODUCED INTO THE SYSTEM DURING ORIGINAL CONSTRUCTION, DURING THE INSTALLATION OF THE CONTAINMENT SPRAY RAW WATER HEAT EXCHANGERS OR DURING THE INSTALLATION OF THE CONTAINMENT SPRAY CROSS TIE PIPING, DURING THE 1984-1986 TIME FRAME. CORRECTIVE ACTIONS INCLUDED REMOVAL OF THE PLASTIC SAMPLE BOTTLE, A VISUAL INSPECTION OF THE OTHER CONTAINMENT SPRAY INLET CHECK VALVE AND SUCCESSFUL COMPLETION OF REVERSE FLOW TESTING AND DRYWELL TO TORUS LEAK RATE TESTING PER PROCEDURE N1-ST-R10.

[82] NINE MILE POINT 1 DOCKET 50-220 LER 90-020
TURBINE TRIP/MANUAL SCRAM DUE TO TURBINE VIBRATION DURING TORSIONAL TEST.
EVENT DATE: 081990 REPORT DATE: 091890 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219605) ON AUGUST 19, 1990, WITH THE REACTOR MODE SWITCH IN THE RUN POSITION, REACTOR POWER AT APPROXIMATELY 21.5 PERCENT, NINE MILE POINT UNIT 1 (NMP1) EXPERIENCED TURBINE VIBRATION PROBLEMS WHEN CONDUCTING POWER ASCENSION PROCEDURE N1-PAT-12-1, "MAIN TURBINE-GENERATOR ROTOR SYSTEM TORSIONAL SCREENING TEST". THE TURBINE WAS MANUALLY TRIPPED, SUBSEQUENTLY A MANUAL REACTOR SCRAM WAS INSERTED PRIOR TO BREAKING VACUUM TO PRECLUDE AN AUTOMATIC REACTOR SCRAM ON LOW CONDENSER VACUUM. CONSEQUENTLY, A HIGH PRESSURE COOLANT INJECTION (HPCI) (MODE OF FEEDWATER CONTROL) SIGNAL WAS RECEIVED DUE TO LOW REACTOR WATER LEVEL (53 INCHES) AND A MAIN STEAM ISOLATION VALVE ISOLATION OCCURRED ON DECREASING CONDENSER VACUUM (7 INCHES MERCURY HG). A ROOT CAUSE INVESTIGATION DETERMINED THAT THE MOST LIKELY CAUSE OF THE LOW PRESSURE (L.P.) TURBINE ROTOR VIBRATION WAS THAT A SLOW ACCELERATION RATE THROUGH AN L.P. TURBINE CRITICAL SPEED RANGE LED TO THE L.P. TURBINE ROTOR DEVELOPING AN INTERSTAGE PACKING RUB AND BOWED ROTOR. IMMEDIATE CORRECTIVE ACTION WAS TO TERMINATE N1-PAT-12-1. ADDITIONALLY, REVISION TO THE TURBINE TORSIONAL TEST IS BEING PURSUED BY ENGINEERING AND GENERAL ELECTRIC TO ADDRESS THE TECHNICAL PROBLEMS ENCOUNTERED DURING THE INITIAL RUN.

[83] NINE MILE POINT 2 DOCKET 50-410 LER 90-001 REV 01
UPDATE ON CONTROL ROOM SPECIAL FILTER TRAIN ACTUATION DUE TO BREAKER CYCLING.
EVENT DATE: 010390 REPORT DATE: 100890 NSSS: GE TYPE: BWR

(NSIC 219759) ON JANUARY 3, 1990, AT APPROXIMATELY 0928 HOURS, NINE MILE POINT UNIT 2 EXPERIENCED AN ACTUATION OF AN ENGINEERED SAFETY FEATURE. SPECIFICALLY, THE DIVISION 1 CONTROL ROOM SPECIAL FILTER TRAIN WAS STARTED AUTOMATICALLY BY A SPURIOUS TRIP OF THE DIVISION 1 CONTROL BUILDING VENTILATION RADIATION MONITORS. AT THE TIME OF THE EVENT THE PLANT WAS SHUTDOWN IN MODE 4 WITH THE VESSEL DEPRESSURIZED AND REACTOR COOLANT TEMPERATURE AT APPROXIMATELY 122 DEGREES FAHRENHEIT AND SHUTDOWN COOLING IN OPERATION. THE APPARENT CAUSE OF THE EVENT WAS THE ELECTRICAL INTERFERENCE ASSOCIATED WITH THE CYCLING OF A CONTROL BUILDING CHILLER BREAKER. BASED UPON RESEARCH AND TESTING OF THE SUBJECT MONITORS AND INFORMATION SECURED FROM ARIZONA PUBLIC SERVICE COMPANY, PALO VERDE NUCLEAR GENERATING STATION, THE MOST PROBABLE ROOT CAUSE FOR THESE ACTUATIONS HAS BEEN DETERMINED TO BE PARASITIC CAPACITANCE COUPLING OF THE RADIATION MONITOR BETA SCINTILLATOR'S GROUNDED OUTER HOUSING AND ITS PREAMPLIFIER'S SIGNAL GROUND. THE CORRECTIVE ACTIONS INCLUDE REPAIR OF THE CHILLER AND INITIATION OF PLANT CHANGE PC 2-0004-90 TO MODIFY THE DETECTOR GROUNDING SCHEME.

[84] NINE MILE POINT 2 DOCKET 50-410 LER 90-013
REACTOR SCRAM CAUSED BY TURBINE GENERATOR TRIP.
EVENT DATE: 090590 REPORT DATE: 100590 NSSS: GE TYPE: BWR

(NSIC 219760) ON SEPTEMBER 5, 1990, AT 0400 HOURS, WITH THE REACTOR MODE SWITCH IN "RUN" AND AT APPROXIMATELY 64% RATED THERMAL POWER (645 MWE), NINE MILE POINT UNIT 2 EXPERIENCED AN ENGINEERED SAFETY FEATURE ACTUATION. SPECIFICALLY, AN AUTOMATIC REACTOR SCRAM CAUSED BY A TURBINE GENERATOR TRIP WHICH WAS INITIATED BY A GENERATOR FIELD GROUND. THE ROOT CAUSE OF THE TURBINE GENERATOR FIELD GROUND IS BEING INVESTIGATED. THE IMMEDIATE CORRECTIVE ACTION WAS TO RESPOND TO THE REACTOR SCRAM AND TURBINE TRIP IN ACCORDANCE WITH PLANT PROCEDURES. A WORK REQUEST WAS ISSUED TO INVESTIGATE THE TURBINE GENERATOR FIELD GROUND. THE ROOT CAUSE AND THE LONG TERM CORRECTIVE ACTIONS WILL BE SUBMITTED AS A SUPPLEMENT TO THIS REPORT.

[85] OCONEE 1 DOCKET 50-269 LER 90-012
VALVE LIMIT SWITCH OPERATION, DUE TO AN UNKNOWN CAUSE, RESULTS IN CONDENSATE/FEEDWATER TRANSIENT AND REACTOR TRIP.
EVENT DATE: 082890 REPORT DATE: 092790 NSSS: BW TYPE: PWR
VENDOR: LIMITORQUE CORP.
VEEDER ROOT COMPANY

(NSIC 219653) ON 8/28/90, AT 14:27:26 HOURS, WHILE OPERATING AT 100% POWER, THE UNIT 1 REACTOR TRIPPED DUE TO HIGH REACTOR COOLANT SYSTEM (RCS) PRESSURE. THE UNIT WAS OPERATING AT A STEADY STATE PRIOR TO THE TRIP. ALL FOUR REACTOR PROTECTIVE SYSTEM (RPS) CHANNELS ACTUATED ON HIGH RCS PRESSURE DUE TO THE LOSS OF THE "B" CONDENSATE BOOSTER PUMP (CBP) AND THE SUBSEQUENT TRIP OF THE "A" FEEDWATER PUMP. THE UNIT WAS BROUGHT SAFELY TO HOT SHUTDOWN. THE "B" CBP TRIP WAS CAUSED BY A "NOT OPEN" SIGNAL FROM THE PUMP'S DISCHARGE VALVE 1C-84. PREVENTIVE MAINTENANCE HAD BEEN COMPLETED ON THE VALVE LESS THAN FIVE MINUTES PRIOR TO RECEIVING THE "NOT OPEN" SIGNAL. THE REASON FOR THE "NOT OPEN" SIGNAL COULD NOT BE DETERMINED, THEREFORE, THE ROOT CAUSE OF THIS EVENT IS CLASSIFIED AS "UNKNOWN". ADDITIONALLY, AT THE TIME OF THE "B" CBP TRIP, THE "A" CBP WAS IN STANDBY AND ALIGNED FOR AUTOMATIC START. THE "A" CBP FAILED TO START DUE TO FAULTY BEARING OIL PRESSURE INSTRUMENTATION. THEREFORE, A CONTRIBUTING CAUSE OF THIS TRIP IS "EQUIPMENT FAILURE/MALFUNCTION" DUE TO THE MALFUNCTION OF THE BEARING OIL PRESSURE INSTRUMENTATION WHICH PREVENTED THE STANDBY CBP FROM STARTING.

[86] OCONEE 1 DOCKET 50-269 LER 90-014
EQUIPMENT MALFUNCTIONS AND MANAGEMENT DEFICIENCY RESULT IN TECHNICAL SPECIFICATION VIOLATIONS ON CORE FLOOD TANK LEVEL.
EVENT DATE: 090790 REPORT DATE: 100890 NSSS: BW TYPE: PWR

VENDOR: BAILEY METER COMPANY
ROSEMOUNT, INC.

(NSIC 219700) ON SEPTEMBER 7, 1990, AT 1455 HOURS, WHILE UNIT 1 WAS OPERATING AT 100% FULL POWER, THE "B" CORE FLOOD TANK (CFT) WAS DETERMINED TO BE AT LESS THAN THE MINIMUM LEVEL REQUIRED BY TECHNICAL SPECIFICATIONS (TS). THIS WAS DISCOVERED AFTER CORRECTION OF PROBLEMS WITH BOTH STRINGS OF LEVEL INSTRUMENTATION, WHICH RESULTED IN BOTH STRINGS BEING OUT OF SERVICE SIMULTANEOUSLY. THIS ALSO VIOLATED TS. SPECIFICALLY, ONE STRING WAS FOUND OUT OF TOLERANCE, REPAIRED AND CALIBRATED, BUT RETURNED TO SERVICE WITH AN ERRONEOUS READING DUE TO INADEQUATE PROCEDURAL GUIDANCE. THEN THE SECOND STRING WAS REMOVED FROM SERVICE FOR CALIBRATION AND FOUND OUT OF TOLERANCE. THE ROOT CAUSES WERE EQUIPMENT FAILURE/MALFUNCTION AND MANAGEMENT DEFICIENCY FOR FAILURE TO PROVIDE ADEQUATE CORRECTIVE ACTIONS FROM A PREVIOUS EVENT. IMMEDIATE CORRECTIVE ACTIONS WERE TO FIX THE INTRUMENTS AND RAISE THE CFT LEVEL WITHIN TS LIMITS.

[87] PALISADES DOCKET 50-255 LER 90-015
INADEQUATE POST MAINTENANCE TESTING RESULTING IN AN INOPERABLE SERVICE WATER PUMP.
EVENT DATE: 070690 REPORT DATE: 101090 NSSS: CE TYPE: PWR

(NSIC 219697) ON 7/6/90 THE PLANT WAS OPERATING AT APPROXIMATELY 80% POWER. TECHNICAL SPECIFICATION 4.0.5 DESCRIBES SURVEILLANCE REQUIREMENTS FOR INSERVICE INSPECTION AND TESTING OF ASME CODE CLASS 1, 2, AND 3 COMPONENTS, AND REQUIRES THAT THE TESTING SHALL BE ACCOMPLISHED IN ACCORDANCE WITH SECTION XI OF THE ASME BOILER AND PRESSURE VESSEL CODE. SECTION XI, SUBSECTION IWP-3111, "EFFECT OF PUMP REPLACEMENT, REPAIR, AND ROUTINE SERVICING ON REFERENCE VALUES", REQUIRES THAT WHEN A REFERENCE VALUE MAY HAVE BEEN AFFECTED BY REPAIR OR ROUTINE SERVICING OF THE PUMP, A NEW REFERENCE VALUE OR SET OF VALUES SHALL BE DETERMINED, OR THE PREVIOUS VALUE RECONFIRMED BY AN INSERVICE TEST RUN PRIOR TO OR WITHIN 96 HOURS AFTER RETURN OF THE PUMP TO NORMAL SERVICE. DEVIATIONS BETWEEN THE PREVIOUS AND NEW SET OF REFERENCE VALUES SHALL BE IDENTIFIED, AND VERIFICATION THAT THE NEW VALUES REPRESENT ACCEPTABLE PUMP OPERATION SHALL BE PLACED IN THE RECORD OF TESTS. CONTRARY TO THESE REQUIREMENTS AN INSERVICE TEST WAS NOT RUN AND APPROPRIATE PUMP PERFORMANCE VALUES WERE NOT RECONFIRMED WITHIN THE SEVEN DAY LCO PERIOD OF TECHNICAL SPECIFICATION 3.4.2. APPROPRIATE PLANT PERSONNEL WILL BE BRIEFED ON THIS EVENT AND PLANT ADMINISTRATIVE PROCEDURES WILL BE REVIEWED AND REVISED AS APPROPRIATE TO CLARIFY POST-MAINTENANCE TESTING REQUIREMENTS.

[88] PALISADES DOCKET 50-255 LER 90-014
INADVERTENT START OF AN AUXILIARY FEEDWATER PUMP.
EVENT DATE: 082890 REPORT DATE: 092790 NSSS: CE TYPE: PWR

(NSIC 219632) AT 1342 HOURS ON AUGUST 28, 1990 WITH THE PLANT OPERATING AT APPROXIMATELY 80% POWER, THE CONFIGURATION CONTROL PROJECT (CCP) PERSONNEL PERFORMING ELECTRICAL WIRING DIAGRAM VERIFICATION ACTIVITIES WERE INSPECTING THE BACK OF CONTROL PANEL C-187B (BA; CAB), WHICH CONTAINS THE "B" AUXILIARY FEEDWATER ACTUATION (AFW) CHANNEL (JE). IT APPEARS THAT THE CCP PERSONNEL CAUSED A MOMENTARY GROUNDING OF THE MANUAL ACTUATION CIRCUIT FOR THE P-8C AUXILIARY FEEDWATER PUMP (BA; P), WHICH STARTED THE PUMP AND AUTOMATIC VALVE ACTUATION. THE AUXILIARY FEEDWATER PUMP START IS CONSIDERED AN ENGINEERED SAFETY FEATURE ACTUATION (JE). THE CONTROL ROOM OPERATORS VERIFIED NORMAL STEAM GENERATOR LEVELS, STOPPED THE AUXILIARY FEEDWATER PUMP, AND STOPPED WORK ACTIVITIES IN AND AROUND C-187B. THE CONFIGURATION CONTROL PROJECT ELECTRICAL WIRING DIAGRAM VERIFICATION WILL BE COMPLETED ON THE AUXILIARY FEEDWATER CONTROL PANELS (BA; CAB) WHEN THE SYSTEM IS OUT OF SERVICE DURING THE PRESENT REFUELING OUTAGE. THE SCOPE OF FUTURE CCP INSPECTIONS OF THE AUXILIARY FEEDWATER ACTUATION SYSTEM CONTROL PANELS WILL BE DISCUSSED BETWEEN THE I&C SUPERINTENDENT AND THE CCP PROJECT MANAGER. THE B CHANNEL OF AUXILIARY FEEDWATER ACTUATION SYSTEM (AFAS) (JE) WILL BE TESTED TO TRY TO DUPLICATE THE INADVERTENT ACTUATION AND DETERMINE ANY ADDITIONAL CORRECTIVE ACTION DURING THE 1990 REFOUT.

[89] PALO VERDE 1 DOCKET 50-528 LER 89-003 REV 02
UPDATE ON LOSS OF POWER TO ALTERNATE FUEL BUILDING EFFLUENT RADIATION MONITOR.
EVENT DATE: 021789 REPORT DATE: 092990 NSSS: CE TYPE: PWR

(NSIC 219690) ON FEBRUARY 17, 1989, AT APPROXIMATELY 1415 MST, PALO VERDE UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 100 PERCENT POWER WHEN A CHEMISTRY EFFLUENT TECHNICIAN (UTILITY, NON-LICENSED) DISCOVERED THAT THE PREPLANNED ALTERNATE SAMPLING SYSTEM (IL) FOR THE FUEL BUILDING WAS INOPERABLE. THE CIRCUIT BREAKER, WHICH SUPPLIES THE ELECTRICAL POWER (EC), HAD OPENED AND DEENERGIZED THE ALTERNATE SYSTEM. WITH THE ALTERNATE SAMPLING INOPERABLE, UNIT 1 OPERATED IN A CONDITION CONTRARY TO TECHNICAL SPECIFICATION (TS) 3.3.3.8. AT APPROXIMATELY 1425 MST, THE POWER TO THE PREPLANNED ALTERNATE SAMPLING SYSTEM POWER WAS RESTORED. NO SAFETY SYSTEM RESPONSES OCCURRED AND NONE WERE NECESSARY. THE ROOT CAUSE OF THE EVENT WAS TEMPORARY AND PERMANENT ELECTRICAL LOADS IN EXCESS OF CIRCUIT CAPACITY. IN RESPONSE, THE CIRCUIT BREAKER SUPPLYING THE ELECTRICAL POWER OPENED AND CAUSED THE LOSS OF ELECTRICAL POWER (EC)(BRK) TO THE ALTERNATE SAMPLING SYSTEM. A DESIGN CHANGE HAS BEEN ISSUED TO INSTALL DEDICATED ALTERNATE SAMPLE SYSTEMS TO RADIATION MONITORS RU-141, RU-143 AND RU-145. SIMILAR EVENTS WERE REPORTED IN LER 529/87-14, 529/88-13, AND 530/88-07.

[90] PALO VERDE 1 DOCKET 50-528 LER 90-002 REV 01
UPDATE ON UNQUALIFIED AIR REGULATORS IN ATMOSPHERIC DUMP VALVES CONTROL AIR SYSTEM.
EVENT DATE: 071390 REPORT DATE: 100690 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 2 (PWR)
PALO VERDE 3 (PWR)
VENDOR: MARSHALLTOWN MANUFACTURING INC.

(NSIC 219782) ON JULY 13, 1990, UNITS 1 AND 3 WERE IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER AND UNIT 2 WAS IN MODE 3 (HOT STANDBY) AT NORMAL OPERATING PRESSURE AND TEMPERATURE WHEN AN ENGINEERING EVALUATION WAS COMPLETED THAT DETERMINED THAT PREVIOUSLY INSTALLED UNQUALIFIED AIR REGULATORS IN THE ATMOSPHERIC DUMP VALVES (ADV) MANUAL REMOTE CONTROL AIR SYSTEM WERE REPORTABLE PURSUANT TO 10CFR21 AND CONSEQUENTLY REPORTABLE UNDER 10CFR50.72 AND 10CFR50.73. ON MAY 23, 1990, THE EQUIPMENT QUALIFICATION GROUP IDENTIFIED THAT THE MASONEILAN MODEL 77-4 AIR REGULATORS INSTALLED IN UNITS 1, 2, AND 3 ADV NITROGEN/AIR SUPPLY SYSTEM DID NOT HAVE ENVIRONMENTAL OR SEISMIC QUALIFICATION DOCUMENTATION. AN OPERABILITY EVALUATION WAS PERFORMED FOR ADVS REQUIRED TO BE OPERABLE AT THAT TIME AND DETERMINED THERE WERE NO IMMEDIATE OPERABILITY EFFECTS. QUALIFIED AIR REGULATORS WERE SUBSEQUENTLY OBTAINED AND INSTALLED IN ALL THREE UNITS. THE CONDITION WAS CAUSED BY A FAILURE OF THE VENDOR TO SUPPLY COMPONENTS THAT MET PROCUREMENT SPECIFICATIONS. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED PURSUANT TO 10CFR50.73.

[91] PALO VERDE 2 DOCKET 50-529 LER 87-004 REV 01
UPDATE ON REACTOR TRIP WHILE PERFORMING GROUND ISOLATION DUE TO INADEQUATE INFORMATION.
EVENT DATE: 041687 REPORT DATE: 100290 NSSS: CE TYPE: PWR

(NSIC 219678) AT 1346 MST ON APRIL 16, 1987, PALO VERDE UNIT 2 WAS OPERATING AT 100 PERCENT POWER WHEN A REACTOR TRIP AND TURBINE TRIP OCCURRED. THE TRIP WAS CAUSED BY HIGH LOCAL POWER DENSITY (LPD) TRIP SIGNALS GENERATED BY THE CORE PROTECTION CALCULATORS (CPC). PRIOR TO THE TRIP, CONTROL ROOM OPERATORS AND ELECTRICAL MAINTENANCE PERSONNEL WERE IN THE PROCESS OF LOCATING A GROUND ON THE "B" TRAIN 120 VAC - CLASS 1E BUS DISTRIBUTION PANEL 2E-PNB-D26. WHEN DISTRIBUTION BREAKER D2603 WAS OPENED AND RESHUT AN ERRONEOUS HIGH LPD TRIP SIGNAL WAS GENERATED ON ALL FOUR (4) PLANT PROTECTION SYSTEM CHANNELS RESULTING IN A REACTOR TRIP. THE ROOT CAUSE OF THIS EVENT WAS THE PARTIAL INTERRUPTION AND RESTORATION OF POWER TO THE CONTROL ELEMENT ASSEMBLY CALCULATOR (CEAC) RESULTING IN AN ERRONEOUS HIGH PENALTY FACTOR TRANSMISSION TO ALL CPC'S. THIS ANOMALY WAS KNOWN HOWEVER, THE ANOMALY WAS CONSIDERED TO BE A MINOR OPERATIONAL CONCERN. THE PROCESS USED FOR DETERMINING THE LOCATION OF GROUNDS WAS SYSTEMATIC AND IN ACCORDANCE WITH APPROVED PLANT PROCEDURES HOWEVER, OPERATIONS PERSONNEL WERE NOT AWARE OF THE INFORMATION CONCERNING THE ANOMALY. AS CORRECTIVE ACTION, THE

EFFECTS OF CYCLING EACH BREAKER ON THE CLASS 1E INSTRUMENT AC DISTRIBUTION PANELS WILL BE INCORPORATED, AS APPROPRIATE, INTO OPERATIONS GUIDELINES.

[92] PALO VERDE 2 DOCKET 50-529 LER 89-005 REV 02
UPDATE ON LOSS OF POWER TO ALTERNATE PLANT VENTILATION EFFLUENT RADIATION MONITOR.
EVENT DATE: 031089 REPORT DATE: 092990 NSSS: CE TYPE: PWR

(NSIC 219691) ON MARCH 10, 1989, AT APPROXIMATELY 2200 MST, PALO VERDE UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 100 PERCENT POWER WHEN A CHEMISTRY EFFLUENT TECHNICIAN (CONTRACTOR, NON-LICENSED) DISCOVERED THAT THE PREPLANNED ALTERNATE SAMPLING SYSTEM (IL) FOR THE PLANT VENTILATION WAS INOPERABLE. THE CIRCUIT BREAKER, WHICH SUPPLIES THE ELECTRICAL POWER (EC), HAD OPENED AND DEENERGIZED THE ALTERNATE SYSTEM. WITH THE ALTERNATE SAMPLING INOPERABLE, UNIT 2 OPERATED IN A CONDITION CONTRARY TO TECHNICAL SPECIFICATION (TS) 3.3.3.8. AT APPROXIMATELY 2210 MST, THE POWER TO THE PREPLANNED ALTERNATE SAMPLING SYSTEM POWER WAS RESTORED. NO SAFETY SYSTEM RESPONSES OCCURRED AND NONE WERE NECESSARY. THE CAUSE OF THE EVENT WAS ELECTRICAL LOADS IN EXCESS OF THE CIRCUIT CAPACITY RESULTING IN THE CIRCUIT BREAKER WHICH SUPPLIED ELECTRICAL POWER OPENING. A DESIGN MODIFICATION HAS BEEN ISSUED TO INSTALL DEDICATED ALTERNATE SAMPLE SYSTEMS TO RADIATION MONITORS RU-141, RU-143 AND RU-145. SIMILAR EVENTS WERE REPORTED IN LER 529/87-14, 529/88-13, 530/88-07, AND 528/89-03.

[93] PALO VERDE 2 DOCKET 50-529 LER 90-003 REV 01
UPDATE ON LOSS OF POWER TO ALTERNATE PLANT VENTILATION EFFLUENT RADIATION MONITOR.
EVENT DATE: 033190 REPORT DATE: 092990 NSSS: CE TYPE: PWR

(NSIC 219783) AT APPROXIMATELY 1230 MST ON MARCH 31, 1990, PALO VERDE UNIT 2 WAS IN A REFUELING OUTAGE WITH THE REACTOR CORE OFFLOADED TO THE SPENT FUEL POOL WHEN A CHEMISTRY EFFLUENT TECHNICIAN AND THE CONTROL ROOM SHIFT SUPERVISOR DISCOVERED THAT THE PREPLANNED ALTERNATE SAMPLING PROGRAM (PASP) PORTABLE SAMPLE CART FOR THE FUEL BUILDING VENTILATION HAD BEEN INOPERABLE. AT APPROXIMATELY 0620 MST ON MARCH 31, 1990, THE LOAD CENTER SUPPLYING POWER TO THE PASP PORTABLE SAMPLE CART HAD BEEN DEENERGIZED FOR A PREPLANNED ELECTRICAL OUTAGE. WHEN THE LOAD CENTER WAS DEENERGIZED, IT WAS NOT RECOGNIZED THAT IT SUPPLIED POWER TO THE PASP PORTABLE SAMPLE CART. AT APPROXIMATELY 0840 MST ON MARCH 31, 1990, THE PASP PORTABLE SAMPLE CART WAS RETURNED TO OPERABLE STATUS. WITH THE PASP PORTABLE SAMPLE CART INOPERABLE FOR APPROXIMATELY TWO HOURS AND TWENTY MINUTES, UNIT 2 OPERATED IN A CONDITION CONTRARY TO TECHNICAL SPECIFICATION 3.3.3.8. THE CAUSE OF THE EVENT WAS A PERSONNEL ERROR DUE TO INADEQUATE IDENTIFICATION OF THE LOADS ON THE LOAD CENTER PRIOR TO DEENERGIZATION. AS CORRECTIVE ACTION A DESIGN MODIFICATION HAS BEEN ISSUED TO INSTALL DEDICATED ALTERNATE SAMPLE SYSTEMS TO RADIATION MONITORS RU-141, RU-143 AND RU-145. SIMILAR EVENTS WERE REPORTED IN LERS 529/87-14, 529/88-13, 530/88-07, 530/89-03, AND 529/89-005.

[94] PEACH BOTTOM 2 DOCKET 50-277 LER 90-022
REACTOR WATER CLEAN-UP ISOLATION DUE TO MISCOMMUNICATION DURING TESTING.
EVENT DATE: 090990 REPORT DATE: 100990 NSSS: GE TYPE: BWR

(NSIC 219708) ON 9/9/90, 2025 HOURS, WITH THE REACTOR AT 40% POWER, A UNIT 2 GROUP II A PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) OCCURRED DURING THE PERFORMANCE OF A SURVEILLANCE TEST. THE ISOLATION RESULTED IN THE CLOSING OF THE REACTOR WATER CLEAN-UP (RWCU) INBOARD ISOLATION VALVE. THE RWCU ISOLATION OCCURRED BECAUSE THE I&C TECHNICIAN PERFORMING THE TEST MISSED A STEP IN THE PROCEDURE THAT WOULD HAVE INSTALLED A JUMPER TO PREVENT THE ISOLATION. THE TECHNICIAN FAILED TO COMMUNICATE IN BOTH THE PRE-JOB DISCUSSION, AND DURING THE TEST AS TO WHO WAS INSTALLING THE JUMPER; EACH ASSUMED THE OTHER HAD PERFORMED THE STEP. THE TECHNICIANS INVOLVED HAVE DISCUSSED THIS EVENT WITH PLANT MANAGEMENT AND WILL BE INVOLVED IN DISCUSSIONS WITH ALL I&C TECHNICIANS ABOUT THIS EVENT. NO SIGNIFICANT CHANGES OCCURRED TO REACTOR WATER CHEMISTRY DURING THE EVENT. ONE PREVIOUS SIMILAR LER WAS IDENTIFIED.

[95] PEACH BOTTOM 3 DOCKET 50-278 LER 90-012
MISCALIBRATION OF REACTOR LEVEL TRANSMITTERS RESULT IN TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 091190 REPORT DATE: 101190 NSSS: GE TYPE: BWR

(NSIC 219709) ON 9/11/90 IT WAS DISCOVERED DURING THE PERFORMANCE OF A SURVEILLANCE TEST THAT LEVEL TRANSMITTER (LT) 3-2-3-99D WAS OUT OF CALIBRATION CAUSING LEVEL INDICATING SWITCH (LIS) 3-2-3-99D TRIP SETPOINT TO EXCEED TECHNICAL SPECIFICATION LIMITS. ON 9/25/90, LT 3-2-3-99C WAS FOUND SIMILARLY OUT OF CALIBRATION CAUSING LIS 3-2-3-99C TRIP SETPOINT TO EXCEED TECHNICAL SPECIFICATION LIMITS. LT/LIS 3-2-3-99C AND D ARE TWO OF FOUR INSTRUMENT LOOPS WHICH PROVIDE A GROUP I PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) SIGNAL ON TRIPLE LOW REACTOR WATER LEVEL. THE OTHER TWO INSTRUMENT LOOPS WERE FUNCTIONAL. LT/LIS 3-2-3-99C AND D ARE BELIEVED TO HAVE BEEN OUT CALIBRATION SINCE THEIR LAST CALIBRATION DURING THE UNIT 3 SEVENTH REFUELING OUTAGE. THE CAUSE OF THESE MISCALIBRATIONS IS UNKNOWN. THE CALIBRATION ERROR OF LT 3-2-3-99C AND D WAS NOT DETECTED DURING SUBSEQUENT STAFF REVIEWS DUE TO UNCERTAINTIES IN REACTOR LEVEL INDICATORS ON LIS 3-2-3-99A, B, C, D, LESS THAN ADEQUATE ACCEPTANCE CRITERIA FOR WIDE RANGE LEVEL INSTRUMENT CHANNEL CHECKS, AND LACK OF CLEAR UNDERSTANDING BY OPERATIONS AND I&C ENGINEERING PERSONNEL OF THE DYNAMIC EFFECTS OF RECIRC FLOW ON THE WIDE RANGE LEVEL INSTRUMENTATION. OPERATIONS AND I&C PERSONNEL WILL RECEIVE TRAINING CONCERNING THE EFFECT OF FLOW ON REACTOR WATER WIDE RANGE LEVEL INSTRUMENTATION. THE TRANSMITTERS WERE RECALIBRATED.

[96] PERRY 1 DOCKET 50-440 LER 90-020
MAINTENANCE ACTIVITIES ON THE CONTROL ROOM EMERGENCY RECIRCULATION SYSTEM RESULTED IN TECHNICAL SPECIFICATION VIOLATION AND COMPROMISE OF A SAFETY SYSTEM.
EVENT DATE: 083190 REPORT DATE: 092890 NSSS: GE TYPE: BWR

(NSIC 219651) ON AUGUST 31, 1990, AT APPROXIMATELY 1430, MAINTENANCE ACTIVITIES ON A CONTROL ROOM HEATING, VENTILATION, AND AIR CONDITIONING (CRHVAC) SYSTEM BACKDRAFT DAMPER CAUSED THE INOPERABILITY OF BOTH TRAINS OF THE EMERGENCY RECIRCULATION MODE OF THE CRHVAC SYSTEM WHICH RESULTED IN A VIOLATION OF TECHNICAL SPECIFICATION 3.7.2 AND IN A CONDITION THAT ALONE COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF A SYSTEM THAT IS NEEDED TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT. THE CAUSE OF THIS EVENT IS PROGRAM DEFICIENCY. THE MAINTENANCE ACTIVITIES THAT WERE BEING PERFORMED ON THE COMMON PLENUM WERE NOT IDENTIFIED AS CAUSING BOTH TRAINS OF THE EMERGENCY RECIRCULATION MODE OF CRHVAC TO BE INOPERABLE. DUE TO INADEQUATE CONSIDERATION OF THE IMPLICATIONS OF THE DESIGN AND CONSTRUCTION OF THE CRHVAC SYSTEM COMMON PLENUM, MAINTENANCE ACTIVITIES WHICH REQUIRED OPENING ACCESS COVERS THAT COMPROMISED THE CONTROL ROOM ENVELOPE WERE ACCOMPLISHED DURING PLANT OPERATING CONDITIONS WHEN THE ACTION STATEMENT FOR TECHNICAL SPECIFICATION 3.7.2 REQUIRED ENTRY INTO TECHNICAL SPECIFICATION 3.0.3. TO PREVENT RECURRENCE, MAINTENANCE ACTIVITIES AND INSPECTIONS ARE BEING REVISED OR RESCHEDULED TO PRECLUDE VIOLATION OF THE CONTROL ENVELOPE DURING APPLICABLE PLANT OPERATING CONDITIONS.

[97] PERRY 1 DOCKET 50-440 LER 90-021
FAILURE OF TWO MAIN STEAM ISOLATION VALVES TO FAST CLOSE FOLLOWING SUCCESSFUL SLOW CLOSURES.
EVENT DATE: 090790 REPORT DATE: 100590 NSSS: GE TYPE: BWR
VENDOR: ALLEN-BRADLEY CO.
NORGREN

(NSIC 219775) ON SEPTEMBER 7, 1990, DURING COOLDOWN OF THE PERRY NUCLEAR POWER PLANT, UNIT 1 IN PREPARATION FOR THE SECOND REFUELING OUTAGE, DIRECTIONS WERE GIVEN TO CLOSE THE MAIN STEAM ISOLATION VALVES (MSIV'S) IN ORDER TO MAINTAIN CONTROL OF THE REACTOR COOLDOWN RATE. ALL EIGHT OF THE MSIV'S SLOW CLOSED PROPERLY, HOWEVER, TWO OF THEM (1B21-F022C AND -F028B) FAILED TO REMAIN CLOSED WHEN THEIR CONTROL SWITCHES WERE PLACED IN THE "CLOSE" POSITION. THESE VALVES LATER CLOSED ON THEIR OWN, WITH THE CONTROL SWITCH LEFT IN THE "CLOSE" POSITION. CENTERIOR ENERGY LETTER PY-CEI/OIE-0327 L TO THE NRC DATED 9/11/90 DOCUMENTED THIS EVENT. SIMILAR EVENTS OCCURRED ON 10/29/87, 11/3/87, AND 11/29/87. THESE EVENTS WERE CAUSED BY DEGRADED ETHYLENE PROPYLENE DIENE MONOMER (EPDM) ELASTOMERS

IN THE ASCO 3-WAY DUAL COIL SOLENOID VALVES (10/29/87 AND 11/03/87), AND BY A SLIVER OF EPDM INSIDE A SOLENOID CAUSING IT TO STICK (11/29/87). THE ELASTOMERS DEGRADED BECAUSE OF LOCALLY HIGH TEMPERATURES RESULTING FROM STEAM LEAKS. THE EPDM ELASTOMERS WERE REPLACED WITH VITON IN A COMPLETE SOLENOID VALVE CHANGEOUT DURING RF01. THE CAUSE(S) OF THE SEPTEMBER 7 EVENTS HAVE NOT YET BEEN DETERMINED. IT IS BELIEVED THAT THE FAILURES ARE ASSOCIATED WITH ASCO 3-WAY DUAL COIL SOLENOID VALVES.

[98] PERRY 1 DOCKET 50-440 LER 90-022
TWO REACTOR WATER CLEANUP CONTAINMENT ISOLATIONS OCCUR DURING PLANT SHUTDOWN DUE TO HIGH DIFFERENTIAL FLOW.
EVENT DATE: 090790 REPORT DATE: 100890 NSSS: GE TYPE: BWR

(NSIC 219776) ON SEPTEMBER 7, 1990 AT 0633 AND 1422 HOURS, TWO REACTOR WATER CLEANUP (RWCU) SYSTEM CONTAINMENT ISOLATIONS OCCURRED DUE TO HIGH DIFFERENTIAL FLOW. THE FIRST ISOLATION OCCURRED FOLLOWING A PLANNED, MANUAL SHUTDOWN OF THE PLANT. THE SECOND EVENT OCCURRED DURING THE SUBSEQUENT RWCU SYSTEM RESTORATION. AFTER AN UNSUCCESSFUL ATTEMPT AT 1422, THE RWCU SYSTEM WAS RETURNED TO SERVICE AT APPROXIMATELY 1500 ON SEPTEMBER 7, 1990. IN RESPONSE TO THE ISOLATIONS, PLANT OPERATORS VERIFIED THAT NO ACTUAL SYSTEM LEAKAGE EXISTED AND COMPLETED SECURING THE SYSTEM. THE ROOT CAUSE OF THESE EVENTS HAS NOT BEEN DETERMINED BUT MAY BE ATTRIBUTABLE TO A DESIGN DEFICIENCY ASSOCIATED WITH THE REDUCED FEEDWATER TEMPERATURE MODE OF OPERATION. CORRECTIVE ACTIONS WERE TAKEN TO VERIFY THAT NO ACTUAL SYSTEM LEAKAGE HAD OCCURRED. THE RWCU SYSTEM WAS SECURED AND SUBSEQUENTLY RETURNED TO SERVICE. AS A RESULT OF THESE ISOLATIONS, AN ENGINEERING EVALUATION WILL BE MADE TO DETERMINE THE ROOT CAUSE AND TO EVALUATE CHANGES TO THE SYSTEM DESIGN AND/OR OPERATING PROCEDURES. IN ORDER TO MINIMIZE THE RECURRENCE OF ISOLATIONS SIMILAR TO THE SECOND ISOLATION, A PROCEDURAL CHANGE IS BEING MADE TO SOI-G33 "REACTOR WATER CLEANUP SYSTEM" TO CLARIFY STARTUP REQUIREMENTS OF THE SYSTEM. THESE EVENTS WILL BE DISCUSSED WITH ALL LICENSED OPERATORS DURING CONTINUING TRAINING.

[99] PERRY 1 DOCKET 50-440 LER 90-023
RPS BUS DE-ENERGIZATION AND SHUTDOWN COOLING ISOLATION DUE TO PERSONNEL ERROR DURING SURVEILLANCE TESTING.
EVENT DATE: 090790 REPORT DATE: 100890 NSSS: GE TYPE: BWR

(NSIC 219777) ON SEPTEMBER 7, 1990 AT 2017 WHILE PERFORMING A SURVEILLANCE TEST, AN OPERATOR INADVERTENTLY DEENERGIZED THE REACTOR PROTECTION SYSTEM (RPS) DISTRIBUTION BUS A, RESULTING IN NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM BALANCE OF PLANT RESIDUAL HEAT REMOVAL A SHUTDOWN COOLING AND REACTOR WATER CLEANUP ISOLATIONS. THE OPERATORS RESPONDED IN ACCORDANCE WITH PLANT PROCEDURES TO RESTORE THESE SYSTEMS. THE CAUSES OF THIS EVENTS WERE PERSONNEL ERRORS, INATTENTION TO DETAIL AND FAILURE TO FOLLOW PROCEDURE. THE OPERATOR AND INSTRUMENT AND CONTROL (I&C) TECHNICIAN PERFORMING A SURVEILLANCE INSTRUCTION (SVI) FAILED TO IDENTIFY THE CORRECT ELECTRICAL PROTECTION ASSEMBLY (EPA) WHICH WAS BEING TESTED. ADDITIONALLY THE OPERATOR DID NOT RECOGNIZE A PROBLEM WHEN THE CIRCUIT BREAKER HE WAS TO "RESET AND THEN CLOSE" WAS ALREADY IN THE CLOSED POSITION. A PLANT ADMINISTRATIVE PROCEDURE DIRECTS THE PERFORMER TO NOTIFY SUPERVISION FOR FURTHER INSTRUCTION IF AN INSTRUCTION CANNOT BE PERFORMED AS WRITTEN. SUPERVISION WAS NOTIFIED, AND WHEN TESTING PROCEEDED THE INCORRECT EPA WAS TRIPPED AND THE RPS BUS WAS INADVERTENTLY DEENERGIZED. IN ORDER TO PREVENT RECURRENCE, THE OPERATOR AND I&C TECHNICIAN HAVE BEEN INVOLVED IN THE INVESTIGATION OF THIS EVENT AND HAVE BEEN MADE AWARE OF THE RAMIFICATIONS OF INATTENTION TO DETAIL WHEN PERFORMING A SVI.

[100] PERRY 1 DOCKET 50-440 LER 90-024
FAILURE TO PROPERLY CONTROL SURVEILLANCE TESTING RESULTED IN ALL SUPPRESSION POOL LEVEL INSTRUMENTATION BEING OUT OF SERVICE WITHOUT COMPENSATORY ACTIONS BEING TAKEN.
EVENT DATE: 090990 REPORT DATE: 100590 NSSS: GE TYPE: BWR

(NSIC 219778) ON SEPTEMBER 9, 1990 AT 0238, SURVEILLANCE TESTING RESULTED IN ALL

SUPPRESSION POOL LEVEL INSTRUMENTATION BEING REMOVED FROM SERVICE WITHOUT THE REQUIRED COMPENSATORY ACTIONS BEING TAKEN. THIS IS VIOLATION OF A TECHNICAL SPECIFICATION 3.5.3. THE CAUSES OF THIS EVENT WERE PROCEDURAL DEFICIENCY, INADEQUATE COMMUNICATIONS AND INATTENTION TO DETAIL. SURVEILLANCE INSTRUCTION "CONTAINMENT ATMOSPHERE MONITORING ISOLATION VALVES SEAT LEAKAGE AND POSITION INDICATION TEST" WAS DEFICIENT IN THAT IT DID NOT CONTROL THE SEQUENCE OF WORK AND DID NOT REQUIRE AN APPROVAL SIGNATURE AT THE START OF EACH SUBSECTION. MISCOMMUNICATIONS OCCURRED DURING SHIFT CHANGES AND BETWEEN LOCAL LEAK RATE TESTING (LLRT), INSTRUMENT AND CONTROL (I&C) AND OPERATIONS PERSONNEL ON THE SAME SHIFT. INATTENTION TO DETAIL WAS EVIDENT IN THAT COMPENSATORY ACTIONS REQUIRED BY TECHNICAL SPECIFICATIONS WERE NOT RECOGNIZED AS BEING REQUIRED IN THE RESULTING PLANT CONFIGURATION. THE ACTIONS TAKEN TO PREVENT RECURRENCE INCLUDE REVISING SVI-D23-T2002 TO PROHIBIT SIMULTANEOUS PERFORMANCE OF THE SUBSECTIONS AND REQUIRING A UNIT SUPERVISORS SIGNATURE IN ORDER TO BEGIN EACH SUBSECTION. THIS EVENT WILL BE ADDED TO THE LOCAL LEAK RATE TESTING TRAINING PROGRAM AS AN EXAMPLE OF INADEQUATE COMMUNICATION DURING TESTING.

[101] PILGRIM 1 DOCKET 50-293 LER 89-036 REV 01
UPDATE ON HIGH PRESSURE COOLANT INJECTION SYSTEM INOPERABLE DUE TO INOPERABLE GLAND SEAL CONDENSER BLOWER MOTOR.
EVENT DATE: 112289 REPORT DATE: 092090 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 219629) ON NOVEMBER 22, 1989 AT 0830 HOURS, THE HIGH PRESSURE COOLANT INJECTION HPCI SYSTEM WAS DECLARED INOPERABLE AND A SEVEN DAY TECHNICAL SPECIFICATION (3.5.C.2) LIMITING CONDITION FOR OPERATION (LCO) WAS ENTERED AT THAT TIME. THE SYSTEM WAS DECLARED INOPERABLE BECAUSE THE HPCI SYSTEM TURBINE GLAND SEAL CONDENSER BLOWER MOTOR DID NOT START WHEN THE BLOWER'S CONTROL SWITCH WAS MOVED TO THE START POSITION IN ACCORDANCE WITH PROCEDURE. THE BLOWER MOTOR DID NOT START BECAUSE OF WORN MOTOR BRUSHES, WHICH RESULTED FROM MOTOR SHAFT PLAY. THE MOTOR SHAFT PLAY WAS DUE TO AGE-RELATED WEAR OF THE MOTOR. THE BLOWER MOTOR, INSTALLED DURING ORIGINAL CONSTRUCTION (CIRCA 1970), WAS MANUFACTURED BY THE GENERAL ELECTRIC COMPANY AND IS A 250 VDC MOTOR, MODEL NUMBER 5BCJ56EA58. THE BLOWER MOTOR WAS REPLACED AND THE BLOWER WAS SUBSEQUENTLY TESTED WITH SATISFACTORY RESULTS. THE HPCI SYSTEM WAS DECLARED OPERABLE ON NOVEMBER 26, 1989 AT 1925 HOURS. THE PREVENTIVE MAINTENANCE PROGRAM HAS BEEN MODIFIED TO INCLUDE PERIODIC BRUSH REPLACEMENT FOR THE HPCI BLOWER MOTOR AND FOR OTHER MOTORS USED IN SIMILAR APPLICATIONS. THIS EVENT OCCURRED DURING POWER OPERATION WITH THE REACTOR MODE SELECTOR SWITCH IN THE RUN POSITION. THE REACTOR POWER LEVEL WAS 94 PERCENT. THE REACTOR VESSEL (RV) PRESSURE WAS 1024 PSIG WITH THE RV WATER TEMPERATURE AT 547 DEGREES FAHRENHEIT.

[102] PILGRIM 1 DOCKET 50-293 LER 89-037 REV 01
UPDATE ON PRIMARY CONTAINMENT/TRAVERSING IN-CORE PROBE BALL VALVE NOT CLOSED CONTRARY TO TECHNICAL SPECIFICATION.
EVENT DATE: 113089 REPORT DATE: 091890 NSSS: GE TYPE: BWR
VENDOR: CONSOLIDATED CONTROLS CORP.
GENERAL ELECTRIC CO.

(NSIC 219604) WHILE TROUBLE-SHOOTING THE TRAVERSING IN-CORE PROBE (TIP) BALL VALVE NO. 45-300A ON NOVEMBER 30, 1989 AT APPROXIMATELY 1200 HOURS, IT WAS DISCOVERED THAT THE BALL VALVE WAS ALMOST FULL OPEN WHEN IT WAS THOUGHT TO BE IN THE CLOSED POSITION. THE VALVE WAS BEING MANUALLY OPERATED AS PART OF TROUBLE-SHOOTING IN ACCORDANCE WITH STATION PROCEDURES. THE CAUSE OF THE BALL VALVE BEING OPEN WAS DAMAGE TO THE VALVE STEM. IT IS BELIEVED THAT STEM DAMAGE OCCURRED DURING PREVIOUS MANUAL MANIPULATION OF THE VALVE TO ALLOW TIP REMOVAL FOLLOWING SOLENOID FAILURE AND VALVE PARTIAL CLOSURE ON THE TIP CABLE . THE BALL VALVE AND ACTUATOR WERE REPLACED. OTHER CORRECTIVE ACTION INCLUDED INCORPORATING THE DETAILS OF THIS EVENT INTO THE INSTRUMENTATION AND CONTROL TECHNICIAN TRAINING PROGRAM. THE BALL VALVE (PLATE NUMBER 73110-2) WAS MANUFACTURED BY CONSOLIDATED CONTROLS, INC. THE SOLENOID ACTUATOR (PLATE NUMBER 112C2391P001-21) WAS MANUFACTURED BY GENERAL ELECTRIC CO. THE CONDITION WAS DISCOVERED DURING POWER OPERATION WITH THE REACTOR MODE SELECTOR SWITCH IN THE RUN POSITION. THE REACTOR VESSEL (RV) TEMPERATURE WAS APPROXIMATELY 540 DEGREES FAHRENHEIT AND THE

RV PRESSURE WAS 1025 PSIG. THE RV POWER LEVEL WAS APPROXIMATELY 94 PERCENT. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(I)(B) AND THE EVENT POSED NO THREAT TO THE HEALTH AND SAFETY OF THE PUBLIC.

[103] PILGRIM 1 DOCKET 50-293 LER 90-013
MANUAL REACTOR SCRAM DUE TO LOCKUP OF THE FEEDWATER REGULATING VALVES.
EVENT DATE: 090290 REPORT DATE: 100290 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING INDUSTRIES

(NSIC 219716) ON 9/2/90 AT 2233 HOURS, AN UNPLANNED MANUAL REACTOR SCRAM WAS INITIATED WITH REACTOR POWER AT 60%. OPERATORS MANUALLY SCRAMMED THE REACTOR DUE TO DIFFICULTIES EXPERIENCED IN CONTROLLING REACTOR VESSEL (RV) WATER LEVEL. A FUSE BLEW IN A FEEDWATER CONTROL CIRCUIT POWER SUPPLY CAUSING BOTH FEEDWATER REGULATING VALVES (FRVS) TO LOCKUP WITHOUT CONTROL ROOM INDICATION. A MODIFICATION WAS IMPLEMENTED WHILE SHUTDOWN WHICH IMPROVES THE RELIABILITY OF THE POWER SUPPLY AND PROVIDES CONTROL ROOM INDICATION OF A FRV LOCKUP FROM A LOSS OF CONTROL POWER. AFTER THE SHUTDOWN OTHER EQUIPMENT PROBLEMS WERE EXPERIENCED. STARTUP FRV FAILED OPEN DUE TO AIR LEAKS AND FAILURE OF ITS AIR BOOSTER RELAY THAT WAS LATER REPLACED. RCICS WAS DECLARED INOPERABLE DUE TO THE TURBINE TRIPPING ON THREE START ATTEMPTS. TURBINE TRIPS WERE ATTRIBUTED TO AN IMPROPER MANUAL START SEQUENCE SPECIFIED IN RCICS OPERATING PROCEDURE AND/OR LOOSENESS OF THE MECHANICAL OVERSPEED TRIP LINKAGE. THE RCICS SUCTION PIPING EXPERIENCED A PRESSURE TRANSIENT DUE TO THE INJECTION CHECK VALVE NOT FULLY SEATING AFTER THE SECOND START ATTEMPT. THE TURBINE OVERSPED AND AUTOMATICALLY RESET ON BOTH STARTS. AN EXACT CAUSE OF OVERSPEED TRIPS COULD NOT BE DETERMINED. A RESIDUAL HEAT REMOVAL SYSTEM/SHUTDOWN COOLING SUCTION ISOLATION VALVE WOULD NOT OPEN NORMALLY. CAUSE WAS CONTACT FAILURE ON THE VALVE'S SEAL-IN RELAY.

[104] PILGRIM 1 DOCKET 50-293 LER 90-014
AUTOMATIC CLOSING OF THE PRIMARY CONTAINMENT SYSTEM GROUP 3 ISOLATION VALVES WHILE SHUTDOWN.
EVENT DATE: 090390 REPORT DATE: 100190 NSSS: GE TYPE: BWR

(NSIC 219717) ON 9/3/90 AT 1450 HOURS, AN AUTOMATIC ACTUATION OF THE RESIDUAL HEAT REMOVAL SYSTEM (RHRS) PORTION OF THE PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM (PCIS) OCCURRED WHILE SHUTDOWN. THE ACTUATION OCCURRED WHEN THE RHRS WAS BEING STARTED FOR THE SHUTDOWN COOLING (SDC) MODE OF OPERATION IN ACCORDANCE WITH PROCEDURE. THE ACTUATION RESULTED IN THE AUTOMATIC CLOSING OF THE PRIMARY CONTAINMENT SYSTEM GROUP 3 (THREE)/SDC SUCTION PIPING ISOLATION VALVES. THE EVENT RESULTED FROM A PRESSURE TRANSIENT THAT ACTUATED THE PROTECTIVE HIGH PRESSURE (122 PSIG) SWITCHES FOR THE SDC SUCTION PIPING. THE CAUSE IS BELIEVED TO HAVE BEEN A COLLAPSING STEAM BUBBLE IN THE SDC SUCTION PIPING THAT IS CONNECTED TO THE RECIRCULATION SYSTEM PIPING. THE RELATED ACCESSIBLE PIPING WAS INSPECTED WITH NO DAMAGE NOTED AND THE PCIS LOGIC CIRCUITRY WAS RESET. THE RHRS WAS SATISFACTORILY PUT INTO SERVICE IN THE SDC MODE OF OPERATION AT 1733 HOURS. THE PROCEDURE WILL BE REVISED TO BACKFILL THE SDC SUCTION PIPING PRIOR TO PLACING THE RHRS INTO SDC SERVICE. OTHER REVISIONS TO THE RHRS PROCEDURE WILL ALSO BE MADE TO OPTIMIZE FLUSHING AND VENTING TO MINIMIZE THE FORMATION OF AIR POCKETS. THIS EVENT OCCURRED WHILE IN HOT SHUTDOWN WITH THE REACTOR MODE SELECTOR SWITCH IN THE SHUTDOWN POSITION. THE REACTOR POWER LEVEL WAS ZERO PERCENT. THE REACTOR VESSEL (RV) PRESSURE WAS APPROX. 6 PSIG, AND THE RV WATER TEMPERATURE WAS ABOUT 230F.

[105] POINT BEACH 1 DOCKET 50-266 LER 90-011
LOW NET POSITIVE SUCTION HEAD TO CONTAINMENT SPRAY PUMPS WITH ECCS IN RECIRCULATION MODE.
EVENT DATE: 082990 REPORT DATE: 092790 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: POINT BEACH 2 (PWR)

(NSIC 219633) ON AUGUST 29, 1990, AN ENGINEERING EVALUATION WAS COMPLETED WHICH INDICATES THAT, UNDER CERTAIN CONDITIONS, THE RESIDUAL HEAT REMOVAL PUMPS CANNOT PROVIDE ADEQUATE NET POSITIVE SUCTION HEAD (NPSH) TO THE CONTAINMENT SPRAY (CS) PUMPS WHEN THE EMERGENCY CORE COOLING SYSTEM IS IN THE RECIRCULATION MODE. CORRECTIVE ACTIONS INCLUDE TEMPORARY PROCEDURE CHANGES LIMITING USE OF THE CS

PUMPS TO CONDITIONS UNDER WHICH NPSH WILL BE ADEQUATE. PERMANENT PROCEDURE CHANGES WILL BE MADE AS FINAL CORRECTIVE ACTIONS.

[106] POINT BEACH 2 DOCKET 50-301 LER 89-010
UNANTICIPATED CONTAINMENT PRESSURE TRIP SIGNAL.
EVENT DATE: 100889 REPORT DATE: 101190 NSSS: WE TYPE: PWR

(NSIC 219684) ON OCTOBER 8, 1989, AT 1023 HOURS, A FIRST-OUT CONTAINMENT PRESSURE ALARM WAS RECEIVED BY CONTROL ROOM PERSONNEL AS A RESULT OF CONTAINMENT PRESSURE TRANSMITTER OPERATION DURING PERFORMANCE OF OPERATIONS REFUELING TEST, ORT 66A. AT THE TIME OF THE EVENT, UNIT 2 WAS IN THE COLD SHUTDOWN CONDITION WITH THE REACTOR TRIP BREAKERS RACKED OUT AND THEIR MOTOR-GENERATOR SETS POWERED OFF. THE ENGINEERED SAFEGUARDS FEATURE ACTUATION CIRCUITRY FOR CONTAINMENT PRESSURE HAD BEEN DISABLED PRIOR TO THE START OF THE TEST. THE PENETRATION FOR CONTAINMENT PRESSURE TRANSMITTER PT-945 WAS IN THE PROCESS OF BEING PRESSURIZED TO PERFORM THE SUBJECT REFUELING TEST WHEN THE FIRST-OUT HIGH CONTAINMENT PRESSURE SIGNAL WAS RECEIVED. INVESTIGATION REVEALED THAT CONTAINMENT PRESSURE TRANSMITTER PT-949A WAS IN THE TRIP POSITION BECAUSE THE BISTABLE HAD BEEN REMOVED FOR REFURBISHMENT. THUS, PRESSURIZATION OF THE PENETRATION FOR THE SECOND CONTAINMENT PRESSURE TRANSMITTER COMPLETED THE TWO-OUT-OF-THREE MATRIX REQUIRED FOR INITIATION OF THE TRIP SIGNAL.

[107] PRAIRIE ISLAND 1 DOCKET 50-282 LER 89-018 REV 03
UPDATE ON AUTO-STARTS OF AUXILIARY BUILDING SPECIAL VENTILATION SYSTEM AS A RESULT OF RADIATION MONITOR SPIKES.
EVENT DATE: 102489 REPORT DATE: 092690 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)
VENDOR: NUCLEAR MEASUREMENTS CORP.

(NSIC 219628) DURING THESE EVENTS, BOTH UNITS WERE OPERATING AT 100% POWER. THE CONTROL ROOM RECEIVED A HIGH RADIATION ALARM, WHICH INITIATED AN AUTOMATIC START OF THE AUXILIARY BUILDING SPECIAL VENTILATION SYSTEM ON FOUR SEPARATE OCCASIONS: AT 1320 HOURS ON OCTOBER 24, 1989; AT 1626 HOURS ON OCTOBER 25, 1989; AT 1210 HOURS ON OCTOBER 31, 1989; AND AT 0659 HOURS ON NOVEMBER 12, 1989. THESE WERE NON-ESF ACTUATIONS OF AN ESF SYSTEM. IN EACH CASE ONE OF THE RADIATION MONITORS WHICH ACTUATES THE AUXILIARY BUILDING SPECIAL VENTILATION SYSTEM, WAS FOUND TO BE IN ALARM WITH A NORMAL RESPONSE INDICATED BY THE METER LOCATED ON THE MONITOR. SINCE THERE WAS IN FACT NO HIGH RADIATION CONDITION IN THE AUXILIARY BUILDING, THE CONTROL ROOM OPERATOR RESET THE ALARM ON THE RADIATION MONITOR AND RETURNED THE AUXILIARY BUILDING SPECIAL VENTILATION SYSTEM TO THE NORMAL STANDBY CONDITION AND RETURNED THE AUXILIARY BUILDING NORMAL VENTILATION SYSTEM TO SERVICE. THESE RADIATION MONITOR MODULES WILL BE REPLACED WITH UPGRADED MONITOR MODULES.

[108] PRAIRIE ISLAND 1 DOCKET 50-282 LER 90-013
INADVERTENT MISPOSITIONING OF THE CONTROL SWITCH OF SHIELD BUILDING VENTILATION HEATER CONTROL.
EVENT DATE: 083090 REPORT DATE: 100190 NSSS: WE TYPE: PWR

(NSIC 219711) ON AUGUST 30, 1990, UNIT 1 WAS AT 100% POWER. SURVEILLANCE PROCEDURE SR1172, VENTILATION SYSTEM MONTHLY OPERATION WAS IN PROGRESS. DURING THE TEST, AT 1804, THE CONTROL ROOM OPERATOR NOTICED THAT THE MONITOR LIGHT INDICATING PROPER OPERATION OF NO. 11 SHIELD BUILDING VENT FILTER HEATER WAS NOT LIT. INVESTIGATION SHOWED THAT THE LOCAL CONTROL SWITCH FOR THE HEATER WAS IN THE OFF POSITION. THE SWITCH WAS IMMEDIATELY RETURNED TO THE ON POSITION. IT IS KNOWN THAT THE SWITCH WAS IN ITS PROPER POSITION ON AUGUST 22, SO THE HEATER COULD HAVE BEEN INOPERABLE FOR 8 DAYS. FROM THE INVESTIGATION OF THE EVENT, IT IS CONCLUDED THAT THE SWITCH WAS MOVED INADVERTENTLY AND UNKNOWINGLY BY A WORKMAN IN THE AREA. CORRECTIVE ACTION WILL INCLUDE THE INSTALLATION OF PROTECTIVE COVERS OVER THIS SWITCH AND THOSE WITH SIMILAR FUNCTION TO PREVENT INADVERTENT OPERATION.

[109] PRAIRIE ISLAND 2 DOCKET 50-306 LER 90-003 REV 01
UPDATE ON REACTOR TRIP WHILE TROUBLESHOOTING ROD CONTROL SYSTEM.
EVENT DATE: 031690 REPORT DATE: 091290 NSSS: WE TYPE: PWR

(NSIC 219540) ON MARCH 16, 1990, UNIT 2 WAS OPERATING AT 100% POWER. AT 0921 A REACTOR TRIP OCCURRED FOLLOWING THE RESETTING OF A ROD CONTROL SYSTEM URGENT FAILURE ALARM WHICH HAD BEEN GENERATED BY CONNECTING TEST EQUIPMENT TO THE ROD CONTROL SYSTEM. ELECTRONIC NOISE HAD BEEN OBSERVED ON TEMPORARY MONITORING EQUIPMENT INSTALLED ON UNIT 2 ROD CONTROL SYSTEM. IN AN ATTEMPT TO IDENTIFY OR ELIMINATE POSSIBLE SOURCES OF THIS NOISE, AN I&C TECHNICIAN PLANNED TO DISCONNECT AND RECONNECT VARIOUS INPUTS TO A RECORDER. THE TECHNICIAN CONNECTED AN OSCILLOSCOPE TO A POINT IN THE ROD CONTROL SYSTEM. WHEN THE OSCILLOSCOPE WAS CONNECTED, THE "V-REF" CONTROL SIGNAL WAS FORCED TO A LOW VALUE BY THE LOW INPUT IMPEDANCE OF THE OSCILLOSCOPE. THE "V-REF" SIGNAL FOR ONLY TWO RODS, E-03 AND I-11, WAS AFFECTED AND THEY DROPPED APPROXIMATELY 10 STEPS INTO THE CORE AS THE URGENT FAILURE CIRCUITRY RESPONDED TO PREVENT ROD MOTION BY APPLYING "HOLD" CURRENT TO THE ROD MECHANISMS. APPROXIMATELY 10 SECONDS LATER, THE URGENT FAILURE ALARM WAS RESET. WHILE THE RESET PUSHBUTTON WAS DEPRESSED, THE "HOLD" CURRENT WAS REMOVED FROM RODS E-03 AND I-11, AND THEY BEGAN TO DROP INTO THE CORE, RESULTING IN A HIGH NEGATIVE FLUX RATE REACTOR TRIP. RESPONSE TO THE TRIP WAS NORMAL AND CORRECT. AFTER EVALUATION OF THE EVENT, THE UNIT WAS RETURNED TO SERVICE AT 0128 THE NEXT DAY.

[110] PRAIRIE ISLAND 2 DOCKET 50-306 LER 90-004
AUTOMATIC START OF BOTH AUXILIARY FEEDWATER PUMPS CAUSED BY USE OF INADEQUATELY REVIEWED PROCEDURE.
EVENT DATE: 090590 REPORT DATE: 100590 NSSS: WE TYPE: PWR

(NSIC 219719) ON SEPTEMBER 5, 1990 UNIT 2 WAS AT 77% POWER COASTING DOWN TO A REFUELING OUTAGE PLANNED FOR THE FOLLOWING WEEK. INSTALLATION OF REPLACEMENT ATWS MITIGATING SYSTEM ACTUATING CIRCUITRY (AMSAC) EQUIPMENT WAS IN PROGRESS ON UNIT 2. DURING THE PERFORMANCE OF THE WORK, AT 1508, AN AMSAC ACTUATION SIGNAL WAS GENERATED, WHICH STARTED BOTH AUXILIARY FEEDWATER PUMPS AUTOMATICALLY. A TURBINE TRIP SIGNAL WAS ALSO GENERATED, BUT LEADS CONNECTING AMSAC TO TURBINE TRIP CIRCUITRY HAD BEEN LIFTED EARLIER IN THE WORK PROCESS, SO NO TURBINE TRIP OCCURRED. THE AUXILIARY FEEDWATER PUMPS' ACTUATION WAS DETERMINED TO BE SPURIOUS, AND THE PUMPS WERE SHUT DOWN AT 1523. THIS WAS A NON-ESF ACTUATION OF ESF EQUIPMENT. CAUSE OF THE EVENT WAS INADEQUATE REVIEW OF WORK PROCEDURES.

[111] PRAIRIE ISLAND 2 DOCKET 50-306 LER 90-005
AUTOMATIC START OF A TURBINE-DRIVEN AUXILIARY FEEDWATER PUMP DUE TO PERSONNEL OVERSIGHT IN PERFORMING UNIT SHUTDOWN PROCEDURE.
EVENT DATE: 091090 REPORT DATE: 101090 NSSS: WE TYPE: PWR

(NSIC 219720) ON SEPTEMBER 10, 1990 UNIT 2 WAS IN HOT SHUTDOWN, BEING READIED FOR A REFUELING SHUTDOWN. STEAM GENERATOR LEVEL CONTROL WAS BEING TRANSFERRED FROM THE MAIN FEEDWATER SYSTEM TO THE AUXILIARY FEEDWATER SYSTEM PER THE NORMAL SHUTDOWN PROCEDURE. AUXILIARY FEEDWATER PUMPS ARE NORMALLY USED DURING UNIT STARTUP AND SHUTDOWN OPERATIONS. NO. 21 MOTOR-DRIVEN AUXILIARY FEEDWATER PUMP WAS RUNNING, SUPPLYING FEEDWATER TO THE STEAM GENERATORS, AS WAS ONE MAIN FEEDWATER PUMP. WHEN THE MAIN FEEDWATER PUMP WAS SHUT DOWN AT 2020, NO. 22 TURBINE-DRIVEN AUXILIARY FEEDWATER PUMP STARTED AUTOMATICALLY. THE AUTOMATIC START OCCURRED BECAUSE THE SELECTOR SWITCH FOR NO. 22 TURBINE-DRIVEN AUXILIARY FEEDWATER PUMP WAS NOT IN THE POSITION CALLED FOR BY THE SHUTDOWN PROCEDURE. THIS WAS A NON-ESF ACTUATION OF ESF EQUIPMENT. CAUSE OF THE EVENT WAS A PERSONNEL OVERSIGHT. THE SHUTDOWN PROCEDURE CALLS FOR MOVING THE SELECTOR SWITCH FOR NO. 22 TURBINE-DRIVEN AUXILIARY FEEDWATER FROM AUTO TO SHUTDOWN AUTO BEFORE TAKING THE LAST MAIN FEEDWATER PUMP OUT OF SERVICE; THIS SWITCH POSITION, USED AT SHUTDOWN, DEFEATS THE AUTOMATIC AUXILIARY FEEDWATER PUMP START FROM "BOTH MAIN FEEDWATER PUMPS OFF."

[112] PRAIRIE ISLAND 2 DOCKET 50-306 LER 90-006
EXCESSIVE LEAKAGE THROUGH A CONTAINMENT ISOLATION VALVE DUE TO VALVE MALFUNCTION CAUSED BY INADEQUATE WORK INSTRUCTIONS.
EVENT DATE: 091290 REPORT DATE: 101290 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 1 (PWR)
VENDOR: ITT GRINNELL VALVE CO INC

(NSIC 219721) ON SEPTEMBER 12, 1990 UNIT 2 WAS IN COLD SHUTDOWN FOR REFUELING. DURING TESTING, IT WAS OBSERVED THAT THE POSITION INDICATING LIGHTS FOR CV-31736, REACTOR COOLANT DRAIN TANK PUMPS DISCHARGE CONTAINMENT ISOLATION VALVE, WERE BOTH LIT. THIS INDICATED THAT THE VALVE HAD NOT FULLY CLOSED AS DEMANDED. LOCAL LEAKAGE RATE TESTING SHOWED LEAKAGE THROUGH THE VALVE OF 193,000 STANDARD CUBIC CENTIMETERS PER MINUTE (SCCM), HIGHER THAN THE TECHNICAL SPECIFICATION LIMIT OF 103,200 SCCM. LEAKAGE THROUGH THE REDUNDANT VALVE IN THE PENETRATION, CV-31735, WAS 10 SCCM. INVESTIGATION SHOWED THAT THE ADJUSTING BUSHING, WHICH LIMITS VALVE CLOSURE, WAS SEPARATED FROM THE JAM NUT WHICH LOCKS IT IN PLACE. THE BUSHING HAD MOVED, PREVENTING FULL CLOSURE OF THE VALVE, BECAUSE IT WAS NOT ADEQUATELY LOCKED IN POSITION. AFTER THE ADJUSTING BUSHING WAS SET PROPERLY AND THE VALVE STROKED, LEAKAGE RATE TESTING SHOWED THE VALVE TO BE OPERABLE WITH LEAKAGE OF 80 SCCM. CAUSE OF THE EVENT WAS INADEQUATE WORK INSTRUCTIONS.

[113] QUAD CITIES 1 DOCKET 50-254 LER 90-018
OUTSIDE DESIGN SPECIFICATION FOR ELECTRICAL SEPARATION CRITERIA FOR TWO REDUNDANT SAFETY SYSTEMS DUE TO INADEQUATE ENGINEERING REVIEW.
EVENT DATE: 082990 REPORT DATE: 092890 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 219608) ON AUGUST 29, 1990, AT 1736 HOURS, UNITS ONE AND TWO WERE BOTH IN THE RUN MODE AT 100 PERCENT OF RATED CORE THERMAL POWER. ENGINEERING NOTIFIED THE STATION THAT BOTH UNITS WERE POTENTIALLY OUTSIDE THE DESIGN BASIS OF THE PLANT'S ELECTRICAL DIVISIONAL SEPARATION CRITERIA. TEST LEADS FOR THE EMERGENCY CORE COOLING SYSTEM (ECCS) SIMULATED AUTOMATIC ACTUATION AND DIESEL GENERATOR (DG) AUTO-START SURVEILLANCE WERE WIRED SUCH THAT TWO REDUNDANT SAFETY-RELATED DIVISIONS WERE LANDED ON ONE TERMINAL BLOCK WITHOUT AN ADEQUATE FIRE BARRIER. AN EMERGENCY NOTIFICATION SYSTEM (ENS) TELEPHONE CALL WAS COMPLETED IN ACCORDANCE WITH 10 CFR50.72(B)(1)(II). ALL TEST LEADS WERE REMOVED PRIOR TO THE EXPIRATION OF THE 24 HOUR LIMITING CONDITION FOR OPERATION (LCO). THE CAUSE OF THIS EVENT WAS LACK OF ADEQUATE ENGINEERING REVIEW WHEN THESE LEADS WERE INSTALLED. IMMEDIATE CORRECTIVE ACTIONS WERE TO REMOVE THE TEST LEADS. FURTHER CORRECTIVE ACTIONS ARE TO CHANGE THE TEST PROCEDURE TO INSTALL TEMPORARY LEADS FOR THE TEST AND REMOVE THEM IMMEDIATELY AFTERWARD. THIS REPORT IS SUBMITTED TO COMPLY WITH 10CFR50.73(A)(2)(II)(B).

[114] QUAD CITIES 2 DOCKET 50-265 LER 90-001
HPCI FIRE PROTECTION OUT OF SERVICE GREATER THAN 14 DAYS DUE TO CONSERVATIVE ACTION.
EVENT DATE: 122589 REPORT DATE: 012490 NSSS: GE TYPE: BWR

(NSIC 219668) ON 12/25/89, AT 0645 HOURS, UNIT TWO WAS IN THE RUN MODE AT 86% OF RATED CORE THERMAL POWER. THE NUMBER OF DAYS THE HIGH PRESSURE COOLANT INJECTION (HPCI) FIRE PROTECTION DELUGE SYSTEM HAD BEEN INOPERABLE HAD EXCEEDED THE 14-DAY REPORTING REQUIREMENT OF TECH SPEC 3.12.C.4. THE HPCI FIRE PROTECTION DELUGE SYSTEM HAD CONSERVATIVELY BEEN DECLARED INOPERABLE ON 12/11/89 WHEN IT WAS DECIDED THAT THE FUNCTION OF THE STEAM LEAK DETECTION HIGH TEMPERATURE, GROUP IV, ISOLATION OF THE HPCI TURBINE MAY BE DELAYED. A FIRE WATCH WAS INITIATED IN ACCORDANCE WITH TECH SPEC 3.12.C.3. THE CAUSE OF THIS EVENT IS A RESULT OF CONSERVATIVE ACTION DUE TO THE POSSIBILITY OF THE ACTUATION DELAY OF THE STEAM LEAK DETECTION. AS PART OF THE CORRECTIVE ACTIONS, THE HPCI FIRE SUPPRESSION SYSTEM WILL REMAIN INOPERABLE UNTIL A MODIFICATION IS INSTALLED TO CORRECT THE PROBLEM. IN THE INTERIM, THE FIRE WATCH FREQUENCY HAS BEEN INCREASED AND A TEMPORARY PROCEDURE HAS BEEN WRITTEN ON HOW TO MANUALLY OPERATE THE FIRE SUPPRESSION SYSTEM. THIS REPORT IS BEING PROVIDED TO COMPLY WITH TECH SPEC 3.12.C.4.

[115] QUAD CITIES 2 DOCKET 50-265 LER 90-005
GROUP II ISOLATION RESULTING FROM PERSONNEL ERROR.
EVENT DATE: 031990 REPORT DATE: 041890 NSSS: GE TYPE: BWR

(NSIC 219669) ON 3/19/90, AT 0329 HOURS, UNIT TWO WAS IN THE SHUTDOWN MODE. THE INSTRUMENT MAINTENANCE (IM) DEPARTMENT WAS PERFORMING MONTHLY SURVEILLANCE TEST QIS 6-2, HIGH DRYWELL PRESSURE SCRAM FUNCTION TEST (ST-6). BUS 29 HAD BEEN PREVIOUSLY REMOVED FROM SERVICE FOR MAINTENANCE RESULTING IN A HALF GROUP II ISOLATION. WHILE PERFORMING THE SURVEILLANCE TEST AND AS EXPECTED, HALF OF A GROUP II ISOLATION WAS PRODUCED, HOWEVER, THE RESULTANT ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF) WAS NOT EXPECTED. THE CAUSE OF THE EVENT WAS A PERSONNEL ERROR. THE IM FOREMAN DETERMINED THE TESTING COULD CONTINUE BY UTILIZING THE PROCEDURE AS WRITTEN. CONTRIBUTING CAUSES INCLUDE THE PROCEDURE PLACING THE RESPONSIBILITY SOLELY ON THE IM FOREMAN TO REVIEW ABNORMAL PLANT CONDITIONS AND THE PROCEDURE NOT REQUIRING DETERMINATION OF THE ACTUAL STATUS OF THE ALARM SIGNAL PRIOR TO TESTING. THE PROCEDURES WILL BE REVISED AS PART OF CORRECTIVE ACTIONS. DISCIPLINARY ACTION WAS ADMINISTERED. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV).

[116] SALEM 1 DOCKET 50-272 LER 90-026 REV 01
UPDATE ON ASME CODE 3 PIPING LEAKAGE CAUSED BY EQUIPMENT FAILURE.
EVENT DATE: 090690 REPORT DATE: 101090 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 219704) THIS LICENSEE EVENT REPORT (LER) ADDRESSES SEVERAL OCCURRENCES OF ASME CODE 3 PIPING LEAKAGE. BASED UPON DISCUSSION WITH THE NUCLEAR REGULATORY COMMISSION (NRC) NOTIFICATIONS WERE MADE FOR EACH OCCURRENCE AS AGREED IN ACCORDANCE WITH CODE OF FEDERAL REGULATIONS 10CFR 50.72. IN ALL CASES, SALEM UNIT 1 TECHNICAL SPECIFICATION 3.4.10.1 ACTION C AND SALEM UNIT 2 TECHNICAL SPECIFICATION 3.4.11.1 ACTION C WERE COMPLIED WITH. THE SALEM UNIT 1 AND UNIT 2 TECHNICAL SPECIFICATION FOR "STRUCTURAL INTEGRITY" ARE IDENTICAL EXCEPT FOR THEIR NUMBER (I.E., 3.4.10.1 VS. 3.4.11.1). THE ROOT CAUSE OF THE LISTED ASME CODE III COMPONENT LEAKAGE HAS BEEN ATTRIBUTED TO EQUIPMENT FAILURE. THE COMPONENT LEAKS WERE THE RESULT OF EROSION/CORROSION FACTORS. THE COMPONENTS WHICH EXHIBITED LEAKAGE WERE DECLARED INOPERABLE IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS. THE COMPONENTS WERE NOT DECLARED OPERABLE UNTIL COMPLETION OF REPAIRS, WHICH WERE DONE IN ACCORDANCE WITH THE ASME CODE FOR CLASS CODE III COMPONENTS. THE REQUIREMENTS OF THE TECHNICAL SPECIFICATIONS WERE COMPLIED WITH IN ALL CASES. AN ONGOING PROGRAM, AT SALEM GENERATING STATION, FOR THE UPGRADE OF SERVICE WATER SYSTEM PIPING IS CONTINUING. THE SCOPE AND PRIORITIZATION OF PIPE REPLACEMENT WILL BE REVIEWED AND MODIFIED, AS APPLICABLE, BASED UPON THE LEAKS IDENTIFIED IN THIS REPORT.

[117] SALEM 1 DOCKET 50-272 LER 90-030
REACTOR TRIP ON #13 S/G LOW-LOW LEVEL DUE TO PERSONNEL ERROR.
EVENT DATE: 091090 REPORT DATE: 100990 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 219705) ON 9/10/90 AT 1201 HOURS A REACTOR TRIP ON NO. 13 STEAM GENERATOR (S/G) LOW-LOW LEVEL OCCURRED. PRIOR TO THE EVENT, THE PIPE BETWEEN THE NO. 11TD900 VALVE AND THE MAIN STEAM LINE SHEARED PROVIDING A STEAM FLOW PATH TO ATMOSPHERE. TO REDUCE THE STEAM FLOW, THE NO. 11MS29 VALVE (MS GOVERNOR VALVE) WAS CLOSED. AT 80% POWER, THE 14MS29 VALVE IS CLOSED ("PARTIAL ARC CONTROL SCHEME"). BOTH THE NOS. 11&14MS29 VALVES DIRECT STEAM TO THE UPPER HALF OF THE TURBINE. THEREFORE, WITH BOTH VALVES CLOSED, A SIGNIFICANT DP ACROSS THE HP TURBINE DEVELOPED. CONTRIBUTING TO THIS DP WAS OPENING THE 11TD4 VALVE WHICH RESULTED IN BLEEDING STEAM AWAY FROM THE UPPER HALF OF THE HP TURBINE. THE TURBINE SHAFT DEFLECTED CREATING AN ELIPTICAL OSCILLATION RESULTING IN DESTRUCTION OF THE AUX. SPEED SENSOR WHICH GENERATED AN OVERSPEED SIGNAL CAUSING CLOSURE OF THE MS29 VALVES. CLOSURE OF THE MS29 VALVES LED TO THE TRIP ON NO. 13 S/G LOW-LOW LEVEL. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO PERSONNEL ERROR. OPS. DEP'T. MANAGEMENT DID NOT LAYOUT AN APPROVED PLAN OF ACTION IN ADDRESSING THE PIPE BREAK ASSOCIATED WITH THE 11TD900 VALVE. CONTRIBUTING FACTORS WERE PROCEDURAL INADEQUACY AND INADEQUATE TRAINING. THIS EVENT HAS BEEN REVIEWED

BY SENIOR MANAGEMENT. THOSE INDIVIDUALS INVOLVED IN THIS EVENT HAVE BEEN REVISED TO CLEARLY IDENTIFY THE CONCERNS WITH TURBINE VALVE TESTING BELOW 85% POWER.

[118] SALEM 1 DOCKET 50-272 LER 90-031
CONTROL ROOM VENTILATION SWITCHOVER DUE TO EQUIPMENT FAILURE.
EVENT DATE: 091790 REPORT DATE: 101090 NSSS: WE TYPE: PWR
VENDOR: LFE CORP.

(NSIC 219706) ON 9/17/90, DURING NORMAL PLANT OPERATION, THE CONTROL ROOM GENERAL AREA RADIATION MONITORING SYSTEM (RMS) MONITOR (1R1A) SPIKED HIGH. THIS RESULTED IN THE AUTOMATIC SWITCHING OF THE CONTROL ROOM VENTILATION FROM NORMAL OPERATION TO ITS ACCIDENT MODE OF OPERATION (100% RECIRCULATION). AT THE TIME OF THE CHANNEL SPIKE, A TECHNICIAN WAS PERFORMING A CHANNEL CALIBRATION OF THE 1R33 RMS CHANNEL (ION EXCHANGE FILTER MONITOR). THE SWITCHING OF THE CONTROL ROOM VENTILATION SYSTEM TO ITS EMERGENCY MODE OF OPERATION IS AN ENGINEERED SAFETY FEATURE (ESF). THE CAUSE OF THE ESF ACTUATION IS ATTRIBUTED TO EQUIPMENT FAILURE AS ASSOCIATED WITH INADEQUATE PREVENTIVE MAINTENANCE. THE PREVENTIVE MAINTENANCE PROGRAM FOR RMS EQUIPMENT DID NOT INCLUDE PERIODIC REPLACEMENT OF POWER SUPPLY CIRCUIT CAPACITORS. THE 1R33 CHANNEL CONTROL DRAWER IS LOCATED IN THE SAME CABINET AS THE 1R1A CHANNEL. WHEN THE TECHNICIAN CYCLED THIS CHANNEL'S POWER, THE 1R1A CHANNEL SPIKE OCCURRED DUE TO THE FAILURE OF THE VARIOUS FAILED FILTERING ELECTROLYTIC CAPACITORS IN THE 1R1A POWER SUPPLY CIRCUIT. IN TURNING THE 1R33 CHANNEL CONTROL CIRCUIT POWER ON/OFF, NORMAL EM/RFI (ELECTRO-MAGNETIC/RADIO FREQUENCY INTERFERENCE) WAS GENERATED BUT NOT FILTERED. THE 1R1A RADIATION MONITOR CHANNEL POWER SUPPLY CIRCUIT WAS REPAIRED BY REPLACEMENT OF THE VARIOUS ELECTROLYTIC CAPACITORS.

[119] SALEM 2 DOCKET 50-311 LER 90-036
TURBINE TRIP/REACTOR TRIP ON NO. 24 S/G HI-HI LEVEL DUE TO EQUIPMENT FAILURE.
EVENT DATE: 090490 REPORT DATE: 100490 NSSS: WE TYPE: PWR
VENDOR: HAGAN CONTROLS
MASONEILAN INTERNATIONAL, INC.
UNITED ENGINEERS & CONSTRUCTORS, INC.

(NSIC 219722) ON 9/4/90 AT 0226 HOURS, DURING NORMAL POWER OPERATION, NO. 21 STEAM GENERATOR FEED PUMP (SGFP) TRIPPED ON LOW SUCTION PRESSURE. A TURBINE RUNBACK TO 60% POWER AT 15%/MIN WAS INITIATED. AT THE COMPLETION OF THE RUNBACK, STEAM GENERATOR (S/G) LEVEL BEGAN INCREASING. AS PER PROCEDURE, THE FOUR BF19 VALVES (S/G FEEDWATER CONTROL VALVES) WERE PLACED IN MANUAL MODE TO GAIN CONTROL OF S/G LEVEL. HOWEVER, JUST AFTER THE 24BF19 VALVE WAS PUT IN MANUAL CONTROL, NO. 24 S/G REACHED ITS HIGH LEVEL SETPOINT INITIATING A TURBINE TRIP AT 0231 HOURS ON 9/4/90. WITH REACTOR POWER ABOVE PERMISSIVE P-9 (50% POWER), A REACTOR TRIP FOLLOWED. THE ROOT CAUSE OF THE REACTOR TRIP IS ATTRIBUTED TO EQUIPMENT FAILURE. THE 21BF19 AUTO/MAN CONTROLLER FAILED, CAUSING VALVE CLOSURE DURING THE RECOVERY OPERATION AFTER THE NO. 21 SGFP HAD TRIPPED. THE SETPOINT FOR THE SUCTION PRESSURE SWITCH HAD DRIFTED HIGH. WHEN THE 23HD15 VALVE (HEATER DRAIN PUMP DISCHARGE CONTROL VALVE) FAILED CLOSED (RUPTURED DIAPHRAGM) A SIGNIFICANT SGFP SUCTION PRESSURE DROP OCCURRED. THIS COUPLED WITH THE HIGH SETPOINT RESULTED IN THE NO. 21 SGFP TRIP. PRECEDING THE RUPTURE OF THE VALVE DIAPHRAGM, THE VALVE CONTROLLER FAILED. THE FAILED PRESSURE SWITCH, THE 23HD15 VALVE DIAPHRAGM, AND THE 23HD15 VALVE CONTROLLER'S PNEUMATIC RELAY WERE REPLACED. THE 21BF19 VALVE AUTO/MAN CONTROLLER WAS REPLACED.

[120] SAN ONOFRE 1 DOCKET 50-206 LER 89-010
INADEQUATE APPLICABILITY FOR TECHNICAL SPECIFICATIONS RELATING TO CORE POWER DISTRIBUTION MONITORING DUE TO ENGINEERING DEFICIENCIES.
EVENT DATE: 022289 REPORT DATE: 092190 NSSS: WE TYPE: PWR

(NSIC 219683) ON FEBRUARY 22, 1989, WITH UNIT 1 IN MODE 5, FOLLOWING AN SCE REVIEW OF BOTH THE CYCLE 10 RELOAD SAFETY EVALUATION REPORT AND RELATED CORRESPONDENCE FROM WESTINGHOUSE, THE POTENTIAL OPERATION OF UNIT 1 IN AN UNANALYZED CONDITION WAS IDENTIFIED. THE APPLICABILITY OF TECHNICAL SPECIFICATIONS (TSS) 3.10 AND 3.11 (MODE 1 > 90% POWER), WHICH PROVIDE

REQUIREMENTS FOR MONITORING OF CORE POWER DISTRIBUTION, HAD ALLOWED OPERATION OF THE CORE AT POWER LEVELS BELOW 90% WITHOUT MONITORING FOR POWER ASYMMETRY. WESTINGHOUSE COULD NOT DOCUMENT ANALYSES WHICH ASSURED ACCEPTABLE RESULTS FOR A DESIGN BASIS EVENT WITH INITIAL CONDITIONS RESULTING FROM AN UNLIKELY OCCURRENCE WHICH: 1) ADVERSELY AFFECTS CORE POWER DISTRIBUTION AND 2) REMAINS UNDETECTED. SCE BELIEVES THAT THIS CONDITION WAS THE RESULT OF PREVIOUSLY IDENTIFIED GENERAL ENGINEERING DEFICIENCIES, INCLUDING EXCESSIVE RELIANCE ON CONTRACTOR WORK. THE APPLICABILITY FOR TSS 3.10 AND 3.11 HAS BEEN CHANGED FROM "MODE 1 ABOVE 90% RATED THERMAL POWER" TO "MODE 1". THIS ENSURES THAT ASSUMPTIONS MADE IN THE SAFETY ANALYSES WILL REMAIN VALID BY REQUIRING THE CORE POWER DISTRIBUTION TO BE MONITORED ANY TIME THE UNIT IS AT POWER. AS DISCUSSED IN PREVIOUS SUBMITTALS TO THE NRC, SCE HAS TAKEN AGGRESSIVE CORRECTIVE ACTIONS ASSOCIATED WITH THE GENERAL ENGINEERING DEFICIENCIES MENTIONED ABOVE.

[121] SAN ONOFRE 2 DOCKET 50-361 LER 90-009
ATMOSPHERIC DUMP VALVE INOPERABLE DUE TO BACKUP NITROGEN.
EVENT DATE: 082990 REPORT DATE: 092890 NSSS: CE TYPE: PWR

(NSIC 219645) ON SEPTEMBER 1, 1990, AT 0350, WITH UNIT 2 IN MODE 1, ATMOSPHERIC DUMP VALVE (ADV) 2HV-8421 FAILED TO STROKE OPEN UTILIZING BACKUP NITROGEN (BN) DURING ROUTINE QUARTERLY INSERVICE TESTING. A PRESSURE REGULATOR IN THE BN SUPPLY TO THE VALVE WAS FOUND WITH A SET POINT WHICH WAS LOWER THAN THAT REQUIRED FOR VALVE OPERATION. THE PROPER SET POINT WAS RE-ESTABLISHED AND 2HV-8421 WAS SATISFACTORILY TESTED AND MADE FULLY OPERABLE AT 0615. OUR INVESTIGATION REVEALED THAT THE REGULATOR SET POINT WAS PROPERLY ESTABLISHED AND THE REGULATOR WAS INSTALLED DURING A MAINTENANCE ACTIVITY WHICH WAS CONCLUDED WITH RETURN TO SERVICE OF 2HV-8421 AT 1250 ON AUGUST 25, 1990, WITH THE PLANT IN MODE 4. THE PLANT ENTERED MODE 1 ON AUGUST 29, 1990. IF THE REGULATOR MISADJUSTMENT OCCURRED PRIOR TO RETURN TO SERVICE FROM THE MAINTENANCE ACTIVITY (A SPECULATION THAT SCE HAS BEEN UNABLE TO CONFIRM) THEN, 1) ENTRY INTO MODE 1 ON AUGUST 29 CONSTITUTES A CONDITION PROHIBITED BY TECHNICAL SPECIFICATION (TS) 3.0.4, AND 2) THE ACTION REQUIREMENTS OF TS 3 7.1.6, ATMOSPHERIC DUMP VALVES, WERE EXCEEDED AT 1518 ON AUGUST 30, 1990. THROUGHOUT THIS EVENT, THE VALVE WAS CAPABLE OF BEING OPERATED BOTH MANUALLY AND BY UTILIZING INSTRUMENT AIR, SUCH THAT ITS SAFETY FUNCTION CONTINUED TO BE MET.

[122] SEABROOK 1 DOCKET 50-443 LER 90-022
REACTOR TRIP DUE TO LOSS OF VOLTAGE ON THE ELECTROHYDRAULIC 24 VOLT DC BUS.
EVENT DATE: 082290 REPORT DATE: 092190 NSSS: WE TYPE: PWR

(NSIC 219624) ON 8/22/90 AT 9:19 A.M., EDT, WHILE IN MODE 1 AT 100% REACTOR POWER, A TURBINE-GENERATOR TRIP WITH REACTOR TRIP OCCURRED. THE TRIP WAS INITIATED BY AN APPARENT LOSS OF VOLTAGE ON THE ELECTROHYDRAULIC CONTROL (EHC) 24 VOLT DC BUS DURING TROUBLESHOOTING ACTIVITIES. A MAIN FEEDWATER ISOLATION ALSO OCCURRED SUBSEQUENT TO THE REACTOR TRIP. A WORK REQUEST WAS INITIATED TO PERFORM CIRCUIT CHECKS IN THE EARLY VALVE ACTUATION (EVA) CIRCUITRY DUE TO INCONSISTENT OPERATION OF THE EVA'S TEST INTERLOCK LIGHT LOCATED ON THE MAIN CONTROL BOARD (MCB). SUBSEQUENT TO INITIAL TESTING AT THE MCB, IT WAS DECIDED TO CONTINUE THE TESTING LOCALLY AT THE EHC CABINET. TWO TEST LEADS WERE USED TO SIMULATE THE TEST SIGNAL AND TO SUPPLY 24 VOLT DC POWER TO THE EVA CIRCUIT. AFTER THE SECOND APPLICATION OF THE TEST LEADS, A VOLTAGE DROP OCCURRED ON THE 24 VOLT DC TRIP BUS RESULTING IN A TURBINE-GENERATOR TRIP WITH REACTOR TRIP. THE ROOT CAUSE FOR THE LOSS OF VOLTAGE ON THE EHC 24 VOLT DC BUS COULD NOT BE CONCLUSIVELY DETERMINED, ALTHOUGH A CONTRIBUTING FACTOR WAS THE TROUBLESHOOTING ACTIVITY ASSOCIATED WITH THE EVA CIRCUIT. PERSONNEL ERROR IN APPLYING THE TEST LEADS HAS NOT BEEN RULED OUT BUT IS CONSIDERED UNLIKELY. NEW HAMPSHIRE YANKEE WILL CAREFULLY EVALUATE ALL FUTURE EHC MAINTENANCE ACTIVITIES PERFORMED DURING POWER OPERATION IN ORDER TO MINIMIZE CHALLENGES TO PLANT SYSTEMS.

[123] SEQUOYAH 1 DOCKET 50-327 LER 90-019
FAILURE TO UPDATE THE P-250 PLANT COMPUTER CONSTANTS RESULTED IN THE AXIAL FLUX DIFFERENCE MONITOR BEING INOPERABLE.
EVENT DATE: 082890 REPORT DATE: 092790 NSSS: WE TYPE: PWR

(NSIC 219640) ON AUGUST 28, 1990, AT APPROXIMATELY 1600 EASTERN DAYLIGHT TIME (EDT) WITH UNIT 1 IN MODE 1, IT WAS DETERMINED THAT UNIT 1 HAD OPERATED IN NONCOMPLIANCE WITH SURVEILLANCE REQUIREMENT 4.2.1.1 FOR PERIODS WHEN THE AXIAL FLUX DIFFERENCE (AFD) MONITOR ALARM WAS DETERMINED TO HAVE BEEN INOPERABLE. THE AFD MONITOR ALARM WAS DETERMINED TO HAVE BEEN INOPERABLE WHEN THE P-250 PLANT COMPUTER AFD MONITOR ALARM CONSTANTS HAD NOT BEEN UPDATED IN A TIMELY MANNER TO REFLECT THE MOST RECENT EXCORE NUCLEAR INSTRUMENTATION SYSTEM (NIS) CALIBRATION DATA. THE P-250 COMPUTER CONSTANTS WERE NOT UPDATED IN A TIMELY MANNER BECAUSE OF INADEQUATE PROCEDURES AND INAPPROPRIATE PERSONNEL ACTIONS. THE P-250 COMPUTER CONSTANTS WERE UPDATED ON AUGUST 28 WHEN IT WAS DISCOVERED THAT THEY DID NOT REFLECT THE AUGUST 5 NIS CALIBRATION DATA. INVESTIGATION REVEALED ANOTHER EXAMPLE OF NOT UPDATING THE P-250 COMPUTER CONSTANTS OCCURRED FROM JUNE 6 TO JULY 24, 1990 FOR UNIT 1. THE IMPORTANCE OF COMPLETING TEST PACKAGES IN A TIMELY MANNER HAS BEEN REVIEWED WITH THE TECHNICAL SUPPORT STAFF. PROCEDURE 0-PI-NXX-092-001.0 WILL BE REVISED TO PROVIDE ADDITIONAL DIRECTION AND ADMINISTRATIVE CONTROLS.

[124] SEQUOYAH 1 DOCKET 50-327 LER 90-020
CONTROL ROOM EMERGENCY VENTILATION SYSTEM WAS INOPERABLE DUE TO THE B TRAIN DIESEL GENERATOR BEING OUT OF SERVICE FOR MAINTENANCE AND THE A TRAIN MAIN CONTROL ROOM AIR HANDLING UNIT FAILING TO START.
EVENT DATE: 082990 REPORT DATE: 092690 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
VENDOR: DUAL SNAP

(NSIC 219641) ON AUGUST 29, 1990, AT 0935 EASTERN DAYLIGHT TIME (EDT) WITH UNIT 1 IN MODE 1 AT 99.5 PERCENT POWER AND UNIT 2 IN MODE 1 AT 72.2 PERCENT POWER, LIMITING CONDITIONS FOR OPERATIONS (LCOS) 3.0.5 AND 3.7.7 WERE ENTERED WHEN BOTH TRAINS OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEMS (CREVS) WERE DECLARED INOPERABLE. THE 1B-B EMERGENCY DIESEL GENERATOR (D/G) WAS OUT OF SERVICE FOR SCHEDULED SURVEILLANCE AND MAINTENANCE ACTIVITIES WHEN THE A TRAIN MAIN CONTROL ROOM (MCR) AIR HANDLING UNIT (AHU), WHICH IS ATTENDANT EQUIPMENT FOR THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM FAILED TO START. OPERATIONS' PERSONNEL DIRECTED EXPEDIENT RETURN TO SERVICE OF THE 1B-B D/G AND DIRECTED IMMEDIATE TROUBLESHOOTING ACTIVITIES ON THE A TRAIN MCR AHU. MAINTENANCE'S PERSONNEL DETERMINED THE ROOT CAUSE OF THE AHU FAILING TO START TO BE A PRESSURE-SWITCH WITH CONTACTS STUCK CLOSED. THE PRESSURE SWITCH WAS ADJUSTED AND CALIBRATED. THE AHU OPERATED PROPERLY WHEN TESTED AND WAS RETURNED TO SERVICE. LCOS 3.0.5 AND 3.7.7 WERE EXITED AT 1355 EDT ON AUGUST 29, 1990.

[125] SEQUOYAH 1 DOCKET 50-327 LER 90-018
REQUIRED SURVEILLANCE INSPECTION NOT PERFORMED ON FOUR FIRE DOOR DAMPERS AS A RESULT OF AN INADEQUATE PROCEDURE.
EVENT DATE: 083190 REPORT DATE: 100190 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 219639) ON AUGUST 31, 1990, WITH UNITS 1 AND 2 IN MODE 1, IT WAS DISCOVERED THAT SURVEILLANCE INSPECTIONS ON FOUR FIRE DAMPERS LOCATED IN FIRE DOORS IN THE AUXILIARY BUILDING HAD NOT BEEN PERFORMED WITHIN THE REQUIRED TIME INTERVAL. ALTHOUGH THE SUBJECT FIRE DOORS ARE INCLUDED IN THE APPROPRIATE SURVEILLANCE INSTRUCTION (SI), NO ACCEPTANCE CRITERIA ARE GIVEN FOR THE FIRE DAMPERS. THEREFORE, THERE IS NO RECORD OF THE FIRE DAMPERS HAVING BEEN INSPECTED. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO AN INADEQUATE PROCEDURE. UPON DISCOVERY OF THE PROBLEM, LIMITING CONDITION FOR OPERATION (LCO) 4.0.3 WAS ENTERED. A WORK REQUEST WAS PREPARED AND COMPLETED THE SAME DAY FOR INSPECTION OF EACH OF THE FIRE DAMPERS. ALL FOUR FIRE DAMPERS WERE VERIFIED TO BE FUNCTIONAL, AND LCO 4.0.3 WAS EXITED. THIS PROBLEM WAS VERIFIED TO BE LIMITED TO ONLY THE FOUR SUBJECT DAMPERS. THE SI WILL BE REVISED TO INCLUDE ACCEPTANCE CRITERIA FOR THE FIRE DAMPERS PRIOR TO THE NEXT SCHEDULED PERFORMANCE, BUT NO LATER THAN DECEMBER 1, 1991.

[126] SHEARON HARRIS 1 DOCKET 50-400 LER 90-018
LOSS OF CCW INVENTORY DUE TO LIFTED REACTOR COOLANT DRAIN TANK HEAT EXCHANGER VALVE CAUSED BY INCORRECT RELIEF VALVE SET POINT.
EVENT DATE: 082390 REPORT DATE: 092490 NSSS: WE TYPE: PWR

(NSIC 219623) AT 0357 ON 8/23/90, DURING THE PERFORMANCE OF THE QUARTERLY OPERABILITY VALVE TEST FOR THE COMPONENT COOLING WATER SYSTEM (CCW) ISOLATION VALVES SERVING THE EXCESS LETDOWN AND REACTOR COOLANT DRAIN TANK HEAT EXCHANGERS INSIDE CONTAINMENT, A RELIEF VALVE LIFTED. THE HEADER PRESSURE REMAINED ABOVE THE VALVE'S RESET POINT. CCW INVENTORY WAS LOST AT AN ESTIMATED 17 GPM RATE WITH A TOTAL LOSS OF 300 TO 350 GALLONS RELIEVED INTO THE CONTAINMENT SUMP. OPERATIONS PERSONNEL RECEIVED A LOW CCW SURGE TANK LEVEL ALARM AND INITIATED MAKEUP TO THE SURGE TANK FROM THE DEMINERALIZED WATER SYSTEM TO MAINTAIN NORMAL TANK LEVEL. THE OPERATORS SECURED ONE OF THE TWO CCW PUMPS WHICH REDUCED THE PRESSURE AND ALLOWED THE RELIEF VALVE TO RESET. IT WAS DETERMINED THAT THE CAUSE OF THE EVENT WAS THE LINEUP OF TWO CCW PUMPS IN SERVICE DURING TESTING WITH THE CONTAINMENT HEADER ISOLATED ON THE DISCHARGE SIDE, RESULTING IN HIGH PRESSURE IN THE HEADER. THIS SYSTEM ALIGNMENT COULD EXIST FOLLOWING AN ACCIDENT WITH A SINGLE FAILURE OF THE CONTAINMENT ISOLATION VALVE ON THE HEADER INLET (FAILS OPEN) AND HAS BEEN IDENTIFIED AS A SAFETY CONCERN. THIS POTENTIAL SINGLE FAILURE WOULD REQUIRE OPERATOR ACTION TO TERMINATE THE LOSS OF CCW INVENTORY. A DESIGN REVIEW OF THE CCW SYSTEM FOUND ONLY ONE SAFETY VALVE SUBJECT TO LIFTING WITH BOTH CCW PUMPS OPERATING.

[127] SHOREHAM DOCKET 50-322 LER 90-007
UNPLANNED ACTUATION OF ENGINEERED SAFETY FEATURE SYSTEMS DURING AN I&C SURVEILLANCE TEST.
EVENT DATE: 090790 REPORT DATE: 092890 NSSS: GE TYPE: BWR

(NSIC 219656) ON 9/7/90 AT 1037, AN UNPLANNED ACTUATION OF THE "A" TRAINS OF THE ENGINEERED SAFETY FEATURES REACTOR BUILDING STANDBY VENTILATION SYSTEM (RBSVS) AND CONTROL ROOM AIR CONDITIONING (CRAC) OCCURRED. THIS ACTUATION OCCURRED DURING THE PERFORMANCE OF SURVEILLANCE PROCEDURE SP44.650.16, REACTOR BUILDING DIFFERENTIAL PRESSURE - LOW, CHANNEL FUNCTIONAL TEST. DURING THIS TEST TWO PAIRS OF LEADS WERE REMOVED FROM TERMINAL STRIPS TO PREVENT THE ESF SYSTEM ACTUATIONS. HOWEVER, IN THIS EVENT THE TECHNICIAN ALSO TAPED EACH APRI OF LEADS TOGETHER WHICH ALLOWED THE RBSVS AND CRAC INITIATING RELAYS TO ENERGIZE DURING THE FUNCTIONAL TEST. THE TEST WAS STOPPED. THE RBSVS AND CRAC SYSTEMS WERE RESET AT 1118 AND PLANT MANAGEMENT PERSONNEL WERE NOTIFIED. THE NRC WAS NOTIFIED AT 1134 PER 10CFR50.72(B)(2)(II). CORRECTIVE ACTIONS TO PREVENT THIS FROM RECURRING INCLUDE PROCEDURE REVISIONS AND ADDING A REPORT OF THIS EVENT TO THE REQUIRED READING LIST FOR THE I&C SECTION.

[128] SHOREHAM DOCKET 50-322 LER 90-008
UNPLANNED ACTUATION OF ENGINEERED SAFETY FEATURE SYSTEMS DUE TO A LIGHTNING STRIKE.
EVENT DATE: 091590 REPORT DATE: 100590 NSSS: GE TYPE: BWR

(NSIC 219729) ON 9/15/90, AT 0732, A LIGHTNING STRIKE DURING A SEVERE THUNDERSTORM CAUSED THE UNPLANNED ACTUATION OF ENGINEERED SAFETY FEATURE SYSTEMS REACTOR BUILDING STANDBY VENTILATION SYSTEM, AND CONTROL ROOM AIR CONDITIONING AND THE CLOSURE OF CONTAINMENT ISOLATION VALVES FOR THE REACTOR WATER CLEANUP SYSTEM, THE REACTOR BUILDING FLOOR DRAINS AND THE REACTOR BUILDING EQUIPMENT DRAINS. REACTOR PROTECTION SYSTEM BUS "B" WAS DEENERGIZED DUE TO TRIPPING OF AN ELECTRICAL PROTECTION ASSEMBLY BREAKER. OTHER EQUIPMENT EFFECTS CAUSED BY THE LIGHTING STRIKE INCLUDE TRIPPING OF THE STATION AIR COMPRESSORS, LOSS OF THE PLANT PROCESS COMPUTER, A CONTROL ROOM VENTILATION MONITOR GOING INTO A FAILURE MODE AND BLOWN CIRCUIT CARDS IN SEVERAL REMOTE DATA ACQUISITION UNITS IN THE PLANT SECURITY SYSTEM. OPERATORS AND TECHNICIANS RESTARTED THE AFFECTED EQUIPMENT AND RESTORED SYSTEMS TO THEIR NORMAL LINEUPS. PLANT MANAGEMENT PERSONNEL WERE NOTIFIED OF THE EVENT AND THE NRC WAS NOTIFIED AT 1006 PER 10CFR50.72(B)(2)(II). THIS LICENSEE EVENT REPORT IS BEING SUBMITTED PER 10CFR50.73(A)(2)(IV).

[129] SOUTH TEXAS 1 DOCKET 50-498 LER 90-022
VIOLATION OF TECHNICAL SPECIFICATION DUE TO EXCEEDING THE ALLOWABLE TEMPERATURE IN THE RCS WITH ONE OF THE HIGH HEAD SAFETY INJECTION PUMPS INOPERABLE.
EVENT DATE: 091290 REPORT DATE: 101290 NSSS: WE TYPE: PWR

(NSIC 219781) ON SEPTEMBER 12, 1990, UNIT 1 WAS IN THE PROCESS OF PERFORMING A PLANT HEATUP. AT APPROXIMATELY 0714 HOURS, MODE 3 WAS ENTERED. AT APPROXIMATELY 0745 HOURS RCS COLD LEG TEMPERATURE EXCEEDED 375 DEGREES F. THE PLANT HEATUP CONTINUED WITH THE 1B HIGH HEAD SAFETY INJECTION (HHSI) PUMP OUT OF SERVICE. THIS WAS A VIOLATION OF TECHNICAL SPECIFICATION 3.5.2 WHICH REQUIRES ALL HHSI PUMPS TO BE OPERABLE PRIOR TO ONE OR MORE REACTOR COOLANT SYSTEM (RCS) COLD LEG TEMPERATURES EXCEEDING 375 DEGREES F. AT 0800 HOURS, THE CONDITION WAS DISCOVERED. THE PLANT HEATUP WAS SECURED AND THE 1B HHSI PUMP WAS RETURNED TO SERVICE AT 0807 HOURS. THE CAUSE OF THIS EVENT WAS LESS THAN ADEQUATE SUPERVISORY OVERSIGHT AND COMMUNICATION. APPROPRIATE ACTION WAS TAKEN WITH RESPECT TO THE UNIT SUPERVISOR INVOLVED. THE REACTOR OPERATOR INVOLVED IN THIS EVENT WAS COUNSELED IN REGARD TO HIS RESPONSIBILITY TO BE KNOWLEDGEABLE OF APPLICABLE PROCEDURAL REQUIREMENTS. EXPECTATIONS OF MANAGEMENT REGARDING ADHERENCE TO PROCEDURES AND ATTENTION TO DETAIL WILL BE REVIEWED WITH THE OPERATORS. THE REQUIREMENTS FOR CONTROL OF HHSI PUMPS DURING HEATUP WILL ALSO BE REVIEWED WITH THE OPERATORS.

[130] ST. LUCIE 1 DOCKET 50-335 LER 90-004
INADVERTENT PARTIAL ACTUATION OF "A" TRAIN CONTAINMENT ISOLATION AND CONTAINMENT SPRAY SYSTEMS DUE TO EQUIPMENT MALFUNCTION.
EVENT DATE: 022890 REPORT DATE: 032890 NSSS: CE TYPE: PWR

(NSIC 219672) ON 2/28/90, ST. LUCIE UNIT 1 WAS IN MODE 6 IN A REFUELING OUTAGE WHEN AN INADVERTENT PARTIAL ACTUATION OF THE ENGINEERED SAFEGUARDS FEATURES ACTUATION SYSTEM (ESFAS) WAS RECEIVED. TRAIN "A" OF THE CONTAINMENT SPRAY AND CONTAINMENT ISOLATION SYSTEMS WERE PARTIALLY ACTUATED. AT THE TIME OF THE EVENT, INSTRUMENTATION AND CONTROLS (I&C) PERSONNEL WERE INSTALLING THE NRC REQUIRED ANTICIPATED TRANSIENT WITHOUT A SCRAM (ATWS) PLANT MODIFICATION IN THE ACTUATION PORTION OF THE ESFAS CABINET. A JUMPER WAS BEING INSTALLED TO ENSURE CIRCUIT CONTINUITY WHEN THE CONTAINMENT SPRAY ACTUATION SIGNAL (CSAS) AND THE CONTAINMENT ISOLATION SIGNAL (CIS) TRAIN "A" ANNUNCIATORS WERE RECEIVED ON THE REACTOR TURBINE GENERATOR BOARD (RTGB). THE 1A EMERGENCY DIESEL GENERATOR STARTED AND THE INSTRUMENT AIR CONTAINMENT ISOLATION VALVE I-MV-18-1 CLOSED. THE ROOT CAUSE OF THE EVENT WAS EQUIPMENT MALFUNCTION. SEVERAL SCREWS ON THE TERMINAL BOARD BEING WORKED WERE FOUND TO BE LOOSE. ATTACHING THE JUMPER TO THE TERMINAL BOARD CAUSED THE CONNECTIONS TO BE JARRED LOOSE, BREAKING THE CIRCUIT, AND PARTIALLY INITIATING THE ESFAS. CORRECTIVE ACTIONS TAKEN WERE TO VERIFY THE PARTIAL ACTUATION RECEIVED WAS THE CORRECT RESPONSE OF THE SYSTEM AND INSPECTING AND TIGHTENING ALL CONNECTIONS IN THE ESFAS ACTUATION CABINETS.

[131] SURRY 1 DOCKET 50-280 LER 90-012
COMMON MODE FAILURE MECHANISM DUE TO MARINE GROWTH IDENTIFIED FOR EMERGENCY SERVICE WATER PUMPS.
EVENT DATE: 090990 REPORT DATE: 100590 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 219710) ON 9/6/90 WITH UNIT 1 AT 74% POWER AND UNIT 2 AT 100% POWER, THE EMERGENCY SERVICE WATER PUMP, 1-SW-P-1A, WAS DECLARED INOPERABLE WHEN ITS FLOW RATE WAS TOO LOW TO MEET THE ACCEPTANCE CRITERION OF 1-PT-25.3A. ON 9/9/90 WITH BOTH UNITS AT THE SAME CONDITIONS AS ABOVE, THE EMERGENCY SERVICE WATER PUMP, 1-SW-P-1C, WAS DECLARED INOPERABLE WHEN ITS FLOW RATE WAS TOO LOW TO MEET THE ACCEPTANCE CRITERION OF 1-PT-25.3C. IN BOTH CASES, THE CAUSE OF THE PROBLEM WAS DETERMINED TO BE MARINE GROWTH ON THE IMPELLERS AND PUMP SUCTIONS. FOLLOWING MAINTENANCE ACTIVITIES, BOTH PUMPS PASSED THEIR PERFORMANCE TESTS AND ARE BACK IN SERVICE. IN ACCORDANCE WITH 10CFR50.73(A)(2)(VII), ANY EVENT WERE A SINGLE CAUSE OR CONDITION THAT CAN CAUSE TWO INDEPENDENT TRAINS OR CHANNELS TO BECOME INOPERABLE IN A SINGLE SYSTEM DESIGNED TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT IS A REPORTABLE EVENT.

[132] SURRY 2 DOCKET 50-281 LER 90-004
MANUAL REACTOR TRIP FOLLOWING INADVERTENT GROUNDING OF THE "A" MAIN FEEDWATER REGULATING VALVE CONTROL SIGNAL DURING TESTING.
EVENT DATE: 082790 REPORT DATE: 092090 NSSS: WE TYPE: PWR
VENDOR: HAGAN CONTROLS

(NSIC 219655) ON 8/27/90 AT 0925 HOURS, WITH UNIT 2 OPERATING AT 100% POWER, THE "A" MAIN FEEDWATER REGULATING VALVE (MFRV) CLOSED FOLLOWING THE INADVERTENT GROUNDING OF ITS CONTROL SIGNAL DURING PERIODIC SURVEILLANCE TESTING. CLOSURE OF THE "A" MFRV RESULTED IN A MISMATCH BETWEEN "A" STEAM GENERATOR (S/G) FEEDWATER FLOW AND STEAM FLOW CAUSING SEVERAL ANNUNCIATORS TO ALARM. THE REACTOR OPERATOR IMMEDIATELY ATTEMPTED TO REOPEN THE "A" MFRV BY INCREASING THE CONTROLLER DEMAND IN MANUAL BUT THE VALVE DID NOT RESPOND. SINCE A LOW S/G LEVEL COINCIDENT WITH STEAM FLOW-FEEDWATER FLOW MISMATCH REACTOR TRIP WAS IMMINENT, THE REACTOR OPERATOR MANUALLY TRIPPED THE REACTOR AT APPROXIMATELY 27% LEVEL IN THE "A" S/G. THE OPERATORS FOLLOWED APPROPRIATE PLANT PROCEDURES AND QUICKLY STABILIZED THE UNIT FOLLOWING THE MANUAL TRIP. A FOUR HOUR NON-EMERGENCY REPORT WAS MADE TO THE NUCLEAR REGULATORY COMMISSION IN ACCORDANCE WITH 10CFR50.72.

[133] SUSQUEHANNA 1 DOCKET 50-387 LER 90-018
SAND INTRUSION RESULTED IN TWO DIESEL GENERATORS BECOMING INOPERABLE.
EVENT DATE: 083090 REPORT DATE: 092790 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)

(NSIC 219660) ON 8/30/90, WITH BOTH UNIT 1 AND UNIT 2 OPERATING AT 100% POWER, THE 'B' EMERGENCY DIESEL GENERATOR (EDG) WAS DECLARED INOPERABLE DUE TO ITS LUBRICATING OIL HAVING A HIGH CHROMIUM CONCENTRATION. FURTHER INVESTIGATION REVEALED SIGNIFICANT SCORING OF NUMEROUS CYLINDER LINERS AND PISTON RINGS. SIMILAR SCORING OF SEVERAL CYLINDER LINERS AND PISTON RINGS HAD BEEN FOUND EARLIER (ON 8/28/90) ON THE 'D' EDG WHICH HAD BEEN OUT OF SERVICE FOR MODIFICATIONS AND INSPECTIONS. THE 'E' EDG WAS SUBSTITUTING FOR THE 'D' EDG DURING THIS EVOLUTION. SAND (ALUMINUM OXIDE) WAS DISCOVERED IN THE COMBUSTION AIR MANIFOLDS OF BOTH 'D' & 'B' EDG'S. A ROOT CAUSE INVESTIGATION WAS COMMENCED. IT WAS DETERMINED THAT SAND HAD APPARENTLY BEEN INADVERTENTLY INTRODUCED TO THE ENGINES DURING RECENT CLEANING AND COATING OF THE TUBE SIDE OF THE ENGINE COMBUSTION AIR INTERCOOLERS. A DEFICIENT WORK PLAN FAILED TO SPECIFY ADEQUATE SEALING REQUIREMENTS TO PREVENT SAND INTRUSION INTO THE INTAKE AIR SIDE OF THE INTERCOOLERS DURING SANDBLAST CLEANING OF THE TUBE SIDE. FURTHER REVIEW REVEALED A NEAR-MISS SIMILAR INCIDENT HAD OCCURRED IN 1987. A SATISFACTORY UPGRADE OF THE WORK PLAN WAS NOT INCORPORATED AT THAT TIME. CORRECTIVE ACTIONS INCLUDE THE REWORK OF BOTH EDG'S AND REVISING THE INTERCOOLER CLEANING AND COATING WORK PLAN TO PROVIDE DETAILS NECESSARY TO ASSURE THAT INTERCOOLER AIR SIDE CLEANLINESS IS MAINTAINED.

[134] SUSQUEHANNA 1 DOCKET 50-387 LER 90-019
LCO 3.0.3 ENTRIES DUE TO NO CONTAINMENT RADIATION MONITOR ALIGNED TO DRYWELL.
EVENT DATE: 090690 REPORT DATE: 100990 NSSS: GE TYPE: BWR
VENDOR: NUCLEAR MEASUREMENTS CORP.

(NSIC 219756) ON 3 OCCASIONS BETWEEN 9/6-9/90, THE IN-SERVICE 'B' CONTAINMENT RADIATION MONITOR (CRM) SYSTEM WAS REMOVED FROM SERVICE TO ALLOW THE 'B' HYDROGEN OXYGEN (H2O2) ANALYZER TO BE SWAPPED FROM THE DRYWELL TO THE SUPPRESSION CHAMBER FOR OXYGEN SAMPLING. THE 'A' CRM SYSTEM WAS INOPERABLE DURING THIS TIME FRAME. THIS LEFT NO CRM IN SERVICE ALIGNED TO THE DRYWELL AS REQUIRED BY TECHNICAL SPECIFICATION 3.4.3.1. IT WAS NOT RECOGNIZED ON THE FIRST OCCURRENCE THAT LIMITING CONDITION FOR OPERATION (LCO) 3.0.3 HAD BEEN ENTERED. ON THE SUBSEQUENT 2 OCCURRENCES, LCO 3.0.3 WAS ENTERED TO ALLOW PERFORMANCE OF THE SUPPRESSION CHAMBER OXYGEN SAMPLING REQUIREMENT STIPULATED IN THE UNIT'S DAILY SURVEILLANCE OPERATING LOG. THE CAUSE FOR NOT RECOGNIZING THE FIRST ENTRY INTO LCO 3.0.3 WAS DUE TO PERSONNEL ERROR OF INATTENTION TO DETAIL IN RECORDING PLANT STATUS ON TURNOVER LOGS. THE CAUSE FOR INTENTIONALLY ENTERING LCO 3.0.3 THE OTHER TIMES WAS ATTRIBUTED TO ADHERENCE TO THE STATION'S POLICY OF PROCEDURAL COMPLIANCE AND NOT FULLY APPRECIATING THE NEED TO EXHAUST ALL OTHER ALTERNATIVES BEFORE ENTERING LCO 3.0.3. CORRECTIVE ACTIONS INCLUDED REPAIRING THE 'A' CRM SUCH THAT IT COULD

BE RESTORED TO OPERABLE STATUS AND COMMITTING TO TRAIN OPERATIONS PERSONNEL ON THE IMPORTANCE OF ATTENTION TO DETAIL IN LOG KEEPING AND IN THE IMPORTANCE OF NOT ENTERING LCO 3.0.3 UNLESS ALL OTHER POSSIBLE AVENUES HAVE BEEN EXHAUSTED.

[135] TROJAN DOCKET 50-344 LER 90-015 REV 01
UPDATE ON INADEQUATE ORIGINAL DESIGN OF CONTROL ROOM EMERGENCY VENTILATION SYSTEM COOLERS RESULTS IN PLANT OPERATION IN AN UNANALYZED CONDITION.
EVENT DATE: 051690 REPORT DATE: 101590 NSSS: WE TYPE: PWR

(NSIC 219746) ON MAY 16, 1990, THE TROJAN NUCLEAR PLANT WAS IN THE 1990 REFUELING OUTAGE. DURING A DESIGN REVIEW OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM, IT WAS DISCOVERED THAT THE SYSTEM'S CALCULATED COOLING CAPACITY WAS NOT ADEQUATE, AND THAT CONTROL ROOM TEMPERATURE COULD EXCEED THE DESIGN LIMIT OF 110 DEGREES F DURING A DESIGN BASIS ACCIDENT WHEN OFFSITE POWER REMAINED AVAILABLE. THE CONTROL ROOM HEAT LOAD IS HIGHER WHEN OFFSITE POWER IS AVAILABLE THAN WHEN IT IS LOST BECAUSE MORE EQUIPMENT AND LIGHTING IN THE CONTROL ROOM REMAIN ENERGIZED. THIS CONDITION WAS THE RESULT OF AN INADEQUATE ORIGINAL DESIGN. CORRECTIVE ACTIONS: THE CONTROL ROOM SUPPLEMENTAL COOLING SYSTEM WAS UPGRADED TO MEET SAFETY RELATED, SEISMIC CATEGORY I CRITERIA IN 1990; IT IS NOW CONSIDERED A REQUIRED PORTION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM. (NOTE: PORTIONS OF THE SYSTEM LOCATED ON THE CONTROL BUILDING ROOF ARE NOT DESIGNED TO WITHSTAND TORNADO WIND LOADINGS OR MISSILES. A PROBABILISTIC RISK ASSESSMENT WHICH SHOWS THESE ARE NOT CREDIBLE HAZARDS WAS SUBMITTED TO THE NRC DECEMBER 21, 1989.) CONSERVATIVE CONTROL ROOM HEAT UP CALCULATIONS, USING ACTUAL PLANT DATA WERE PERFORMED. THE RESULTS SHOW THE CONTROL ROOM TEMPERATURE WOULD HAVE INCREASED AND EXCEEDED 110 DEGREES F ABOUT SEVEN DAYS FOLLOWING ONSET OF AN ACCIDENT.

[136] TROJAN DOCKET 50-344 LER 90-027 REV 01
UPDATE ON INADEQUATE IMPLEMENTATION OF A PROGRAMMATIC CHANGE IN HOW A TECH SPEC SURVEILLANCE WAS TO BE MET RESULTS IN A MISSED SURVEILLANCE DUE TO AN INADEQUATE PROCEDURE.
EVENT DATE: 071790 REPORT DATE: 101190 NSSS: WE TYPE: PWR

(NSIC 219747) ON 7/17/90, PLANT WAS IN MODE 1 (POWER OPERATION) WITH A GENERATOR LOAD OF 1140 MWE. DURING A REVIEW OF A PLANT PROCEDURE, PLANT SYSTEMS ENGINEERING PERSONNEL IDENTIFIED THAT THE MONTHLY SURVEILLANCE REQUIRED BY TROJAN TECH SPEC (TTS) 4.6.1.1, "PRIMARY CONTAINMENT - CONTAINMENT INTEGRITY" DID NOT INCLUDE SIX VALVES ASSOCIATED WITH THE CONTAINMENT PENETRATION BOUNDARY FOR THE SG BLOWDOWN SYSTEM. ONE DRAIN VALVE HAD NOT BEEN INCLUDED IN THE PLANT PROCEDURES USED TO PERFORM THE REQUIRED SURVEILLANCE DUE TO INADEQUATE IMPLEMENTATION OF A 1988 PROGRAMMATIC CHANGE WHICH ADDED ALL VENT, TEST, AND DRAIN VALVES WITHIN THE CONTAINMENT PENETRATION BOUNDARY TO THE LIST OF VALVES REQUIRING TTS 4.6.1.1 SURVEILLANCE. THE OTHER 5 VALVES ARE LOCATED OUTSIDE OF THE FIRST MOTOR-OPERATED CONTAINMENT ISOLATION VALVE FOR A CLOSED SYSTEM INSIDE CONTAINMENT. CONTAINMENT ISOLATION DESIGN BASIS FOR THIS CLOSED SYSTEM IS ONE VALVE OUTSIDE CONTAINMENT, EXCEPT WHEN TWO VALVES ARE REQUIRED TO MEET SINGLE FAILURE CRITERIA. 5 VALVES SHOULD HAVE BEEN INCLUDED IN THE SURVEILLANCE SINCE THE MOTOR-OPERATED VALVE COULD FAIL TO OPERATE. CORRECTIVE ACTION WAS TO PLACE THESE 6 VALVES IN THE PROCEDURE WHICH IS USED TO PERFORM THE TTS 4.6.1.1 REQUIRED SURVEILLANCE. 2 ADDITIONAL VALVES WERE ADDED FOR HUMAN FACTORS CONSIDERATIONS.

[137] TROJAN DOCKET 50-344 LER 90-035
NORMAL VARIATION IN BACKGROUND ACTIVITY AND LOW MONITOR SETPOINT RESULT IN RADIATION MONITOR INITIATING CONTAINMENT VENTILATION ISOLATION.
EVENT DATE: 082290 REPORT DATE: 092190 NSSS: WE TYPE: PWR

(NSIC 219613) ON AUGUST 22, 1990, THE PLACE WAS IN MODE 1 (POWER OPERATION) AT 100 PERCENT RATED THERMAL POWER AND GENERATING 1140 MWE. THE CONTAINMENT HYDROGEN VENT SYSTEM WAS IN OPERATION TO REDUCE CONTAINMENT PRESSURE, WITH THE CONTAINMENT PROCESS RADIATION MONITORING (PRM) SYSTEM ALIGNED TO MONITOR THE EFFLUENT PATHWAY. AT 1231 HOURS, THE HIGH LEVEL NOBLE GAS MONITOR (PRM-1D) BACKGROUND HAD RISEN FROM AN INITIAL READING OF APPROXIMATELY 26 COUNTS PER MINUTE (CPM) TO APPROXIMATELY 50 CPM. THE CHANGE CAUSED A CONTAINMENT

VENTILATION ISOLATION SIGNAL WHICH CLOSED THE CONTAINMENT HYDROGEN VENT SYSTEM VALVES. THE CAUSE OF THIS EVENT WAS THE NORMAL VARIATION IN BACKGROUND COUNT RATE REACHED THE PRM-1D SETPOINT WHICH ACTUATED THE CONTAINMENT VENTILATION ISOLATION VALVES. A CONTRIBUTING CAUSE IS THAT A TROJAN TECHNICAL SPECIFICATION PROVISION WHICH ALLOWS PRM-1D TO BE INOPERABLE IF THE LOW LEVEL NOBLE GAS MONITOR IS ON SCALE WAS NOT IMPLEMENTED IN THE PLANT PROCEDURE FOR CONTAINMENT PRESSURE REDUCTIONS. THE PLANT PROCEDURE FOR CONTAINMENT PRESSURE REDUCTION WAS REVISED TO CLARIFY HOW TO DETERMINE BACKGROUND READINGS FOR PRM-1. THIS WAS DONE TO MINIMIZE THE CHANCE OF ESTABLISHING TOO LOW A SETPOINT WHEN THE NORMAL BACKGROUND IS LOW.

[138] TROJAN DOCKET 50-344 LER 90-036
INCONSISTENT INTERPRETATION OF TECH SPECS RESULTS IN INSERVICE TESTING OF VALVES DURING PLANT OPERATION WHICH WERE DESIGNATED NOT TESTABLE DURING OPERATION.
EVENT DATE: 080690 REPORT DATE: 100990 NSSS: WE TYPE: PWR

(NSIC 219748) ON AUGUST 24, 1990, THE TROJAN NUCLEAR PLANT WAS OPERATING AT 100 PERCENT RATED THERMAL POWER. DURING A REVIEW ASSOCIATED WITH POST MAINTENANCE TESTING OF A VALVE, A DISCREPANCY BETWEEN THE TROJAN TECHNICAL SPECIFICATIONS (TTS) AND THE TROJAN INSERVICE TESTING PROGRAM FOR PUMPS AND VALVES (IST PROGRAM) WAS IDENTIFIED. TTS TABLE 3.6-1, CONTAINMENT ISOLATION VALVES, DESIGNATES WHETHER OR NOT THE LISTED VALVES ARE TESTABLE DURING PLANT OPERATION. SEVERAL VALVES DESIGNATED NOT TESTABLE DURING PLANT OPERATION HAVE BEEN TESTED DURING PLANT OPERATION IN ACCORDANCE WITH THE IST PROGRAM. ADDITIONALLY, THE TTS FOR THE AUXILIARY FEEDWATER SYSTEM REQUIRES PERIODIC TESTING OF THE STEAM TURBINE DRIVEN AUXILIARY FEEDWATER PUMP. IN ORDER TO PERFORM THE TEST, TURBINE STEAM SUPPLY VALVES MUST BE OPENED. THESE VALVES ARE DESIGNATED NOT TESTABLE DURING PLANT OPERATION IN TTS TABLE 3.6-1. THE CAUSE OF THIS EVENT WAS DETERMINED TO BE INCONSISTENT INTERPRETATION OF THE MEANING OF TTS TABLE 3.6-1. A LICENSE CHANGE APPLICATION TO REVISE TTS TABLE 3.6-1 WAS APPROVED BY THE NUCLEAR REGULATORY COMMISSION ON SEPTEMBER 28, 1990. THIS CHANGE RESOLVED THE CONFLICT BETWEEN TTS TABLE 3.6-1 AND THE AUXILIARY FEEDWATER PUMP TEST REQUIREMENTS OF TTS 4.7.1.2.1.

[139] TROJAN DOCKET 50-344 LER 90-037
LIGHTNING STRIKE ON DISTRIBUTION SYSTEM GRID RESULTS IN MOMENTARY UNDERVOLTAGE ON SECONDARY SIDE OF STARTUP TRANSFORMERS AND CAUSES EMERGENCY DIESEL GENERATOR START.
EVENT DATE: 090790 REPORT DATE: 100890 NSSS: WE TYPE: PWR

(NSIC 219749) ON SEPTEMBER 7, 1990, THE TROJAN NUCLEAR PLANT WAS OPERATING AT 100 PERCENT RATED THERMAL POWER. THE GENERATOR LOAD WAS APPROXIMATELY 1142 MW. AT 1418 A DISTURBANCE ON THE TRANSMISSION SYSTEM GRID RESULTED IN A LOAD SWING AT TROJAN OF ABOUT 200 MW AND A MOMENTARY UNDERVOLTAGE CONDITION ON THE SECONDARY SIDE OF THE PLANT STARTUP TRANSFORMERS. IN RESPONSE TO THE LOAD SWING, THE TURBINE OUTPUT WAS AUTOMATICALLY REDUCED TO MATCH LOAD, AND THE CONTROL RODS AUTOMATICALLY STEPPED IN TO REDUCE REACTOR POWER. IN ADDITION, BOTH EMERGENCY DIESEL GENERATORS AUTOMATICALLY STARTED IN RESPONSE TO THE UNDERVOLTAGE CONDITION ON THE SECONDARY SIDE OF THE PLANT STARTUP TRANSFORMERS. THE TRANSMISSION SYSTEM GRID DISTURBANCE WAS CAUSED BY A LIGHTNING STRIKE APPROXIMATELY 125 MILES EAST OF THE TROJAN SITE. THE EMERGENCY DIESEL GENERATOR START WAS INITIATED BY PROTECTIVE RELAYS WHICH SENSED AN UNDERVOLTAGE CONDITION ON THE SECONDARY SIDE OF THE PLANT STARTUP TRANSFORMERS. VOLTAGE HISTORY DATA RELATED TO THE GRID DISTURBANCE WAS ANALYZED AND IT WAS DETERMINED THAT THE PLANT RESPONDED TO THE GRID DISTURBANCE IN ACCORDANCE WITH DESIGN. THIS EVENT HAD NO EFFECT UPON PUBLIC HEALTH AND SAFETY.

[140] TROJAN DOCKET 50-344 LER 90-038
INCOMPLETE PERFORMANCE OF SURVEILLANCES DUE TO PERSONNEL ERRORS IN INTERPRETATION OF TECH SPECS AND IN DEVELOPING PROCEDURE REVISION.
EVENT DATE: 090790 REPORT DATE: 100990 NSSS: WE TYPE: PWR

(NSIC 219750) ON SEPTEMBER 7, 1990 THE PLANT WAS IN MODE 1 (POWER OPERATION) AT A NOMINAL GENERATOR LOAD OF 1140 MWE. WHILE PERFORMING A DESIGN REVIEW, A DESIGN

ENGINEER DISCOVERED THAT A PORTION OF THE TURBINE TRIP ON REACTOR TRIP CIRCUIT WAS NOT TESTED BY EXISTING TROJAN TECHNICAL SPECIFICATION SURVEILLANCE PROCEDURES. THE PORTION OF THE CIRCUIT NOT TESTED CONTAINED WIRING AND WIRING CONNECTIONS, NOT ELECTRO-MECHANICAL COMPONENTS. THE EVALUATION ALSO DETERMINED THAT SURVEILLANCE TESTING OF THE FEEDWATER ISOLATION FUNCTION UPON A REACTOR TRIP WAS TESTED EVERY OTHER ANNUAL REFUELING, RATHER THAN ONCE PER 18 MONTHS AS REQUIRED BY TROJAN TECHNICAL SPECIFICATION 3/4.3.2, "ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION". THE PRIMARY CAUSES OF THESE EVENTS WERE AN INADEQUATE REVIEW OF PROCEDURES TO IDENTIFY NEEDED CHANGES TO IMPLEMENT A TROJAN TECHNICAL SPECIFICATION AMENDMENT AND AN INCORRECT INTERPRETATION OF THE REQUIRED SURVEILLANCE FREQUENCY FOR THE CHANNELS WHICH INITIATE FEEDWATER ISOLATION FUNCTION UPON A REACTOR TRIP. THE REQUIRED SURVEILLANCES WERE PERFORMED DURING THE SEPTEMBER 25, 1990 FORCED OUTAGE. CORRECTIVE ACTIONS TO BE TAKEN INCLUDE STRENGTHENING ADMINISTRATIVE CONTROLS FOR IDENTIFICATION OF PROCEDURE CHANGES NEEDED DUE TO TROJAN TECHNICAL SPECIFICATION AMENDMENT.

[141] TURKEY POINT 3 DOCKET 50-250 LER 90-003 REV 01
UPDATE ON SPENT FUEL POOL COOLING PUMP 3B SHAFT FAILURE RESULTED IN DAMAGE TO THE PUMP MECHANICAL SEAL AND RELEASE OF BORATED CONTAMINATED WATER TO THE SPENT FUEL POOL BUILDING.
EVENT DATE: 022090 REPORT DATE: 091490 NSSS: WE TYPE: PWR
VENDOR: INGERSOLL-RAND CO.

(NSIC 219552) AT 1120, ON 2/20/90, WITH UNIT 3 IN MODE 6 (REFUELING), CONTROL ROOM PERSONNEL WERE NOTIFIED THAT A MECHANICAL SEAL FAILURE HAD OCCURRED ON THE 3B SPENT FUEL POOL (SFP) COOLING PUMP (EIIS:DA, COMPONENT:P). APPROXIMATELY THREE INCHES OF BORATED CONTAMINATED WATER HAD ACCUMULATED IN THE UNIT 3 SFP HEAT EXCHANGER ROOM AND THE UNIT 3 CASK WASH AREA. THE 3B SFP COOLING PUMP SHAFT SHEARED AS THE RESULT OF FATIGUE FAILURE. ABNORMAL OPERATING STRESSES LED TO THE FATIGUE FAILURE. WHEN THE PUMP SHAFT FAILED, THE IMPELLER AND THAT PORTION OF THE PUMP SHAFT UP TO THE FRACTURE POINT CONTINUED TO ROTATE WITHOUT RADIAL OR AXIAL SUPPORT. THIS RESULTED IN FAILURE OF THE MECHANICAL SEAL. THE AUX. BLDG. DRAINS ARE DESIGNED TO ACCOMMODATE APPROXIMATELY 70 GALLONS PER MINUTE (GPM) TOTAL SYSTEM LEAKAGE. THE 3B SFP COOLING PUMP MECHANICAL SEAL FAILURE IS ESTIMATED TO HAVE CREATED AN 18 GPM LEAK. PARTIALLY CLOGGED DRAIN LINES IN THE AUXILIARY BUILDING LED TO THE WATER ACCUMULATION. NO RADIOACTIVE LIQUID EFFLUENT WAS RELEASED TO THE ENVIRONMENT OR TO AN UNRESTRICTED AREA. THE EFFECTS ON THE SFP AND UNIT 3 REFUELING CAVITY WATER LEVEL WERE NEGLIGIBLE. THE 3A SFP COOLING PUMP (GOULDS) HAS BEEN ALIGNED AS THE PRIMARY SFP COOLING PUMP.

[142] TURKEY POINT 4 DOCKET 50-251 LER 90-002 REV 01
UPDATE ON POST ACCIDENT CONTAINMENT VENT INOPERABLE DUE TO LOCAL LEAK RATE TESTS.
EVENT DATE: 022890 REPORT DATE: 092090 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 3 (PWR)

(NSIC 219606) THE UNIT 4 POST ACCIDENT CONTAINMENT VENT (PACV) IS ALLOWED TO BE INOPERABLE 7 DAYS PURSUANT TO TECH SPEC 4.4.6.B. CONTRARY TO THE TECH SPEC, WITH UNIT 4 IN MODE 1 THE PACV WAS INOPERABLE FOR 13 DAYS. ON FEBRUARY 15, 1990, DURING A UNIT 3 REFUELING OUTAGE, LOCAL LEAK RATE TESTS (LLRT) WERE SCHEDULED ON CONTAINMENT PENETRATIONS PRIOR TO FUEL MOVEMENT. CONTAINMENT INTEGRITY WAS ESTABLISHED AS REQUIRED PRIOR TO REFUELING OPERATIONS PREVENTING CONTINUATION OF LEAK RATE TESTING ON VARIOUS VALVES. CLEARANCE TAGS (FOR CLOSURE) INSTALLED ON TWO NORMALLY OPEN VALVES (HV-1 AND HV-2) WERE ALLOWED TO REMAIN ON THE TWO VALVES THROUGH FUEL MOVEMENT UNTIL AFTER THE COMPLETION OF THE LLRT ON UNIT 3 PACV PENETRATION 16 DUE TO A PERSONNEL ERROR. THE TWO CLOSED VALVES ALSO AFFECTED THE OPERABILITY OF THE UNIT 4 PACV. THE REFUELING PROCESS WAS COMPLETED ON FEBRUARY 22 AND THE LLRT ON PENETRATION 16 WAS COMPLETED ON FEBRUARY 26. THE UNIT 4 PACV SYSTEM WAS RETURNED TO SERVICE ON FEBRUARY 28 WHEN HV-1 AND HV-2 WERE REOPENED WITHIN 6 MINUTES OF DISCOVERY . CAUTIONS CONCERNING THE EFFECT OF THE UNIT 3 LLRT ON UNIT 4 PACV OPERABILITY HAVE BEEN ADDED TO THE UNIT 3 PACV PENETRATION 16 LLRT PROCEDURE. THE OCCURRENCE IS REPORTABLE IN ACCORDANCE WITH 10 CFR 50.73 (A)(2)(I)(B). FURTHER DOSE EVALUATIONS HAVE BEEN COMPLETED CONFIRMING THE ACCESSABILITY OF HV-1 AND HV-2 DURING POST ACCIDENT CONDITIONS.

[143] TURKEY POINT 4 DOCKET 50-251 LER 90-009
CONTAINMENT PERSONNEL AIR LOCK PRESSURE TEST NOT PERFORMED IN ACCORDANCE WITH PLANT TECHNICAL SPECIFICATIONS DUE TO WORK CONTROL DEFICIENCIES.
EVENT DATE: 071090 REPORT DATE: 092090 NSSS: WE TYPE: PWR

(NSIC 219607) ON JULY 10, 1990, WITH UNIT 4 IN MODE 1 (POWER OPERATION), AT APPROXIMATELY 100 PERCENT POWER, THE ACTION REQUIREMENT OF TECHNICAL SPECIFICATION (TS) 3.3.1 WAS NOT MET. CORRECTIVE MAINTENANCE PERFORMED ON THE PERSONNEL AIR LOCK INNER EQUALIZING VALVE ON JULY 10 MADE THE INNER BARRIER OF THE AIR LOCK TECHNICALLY INOPERABLE UNTIL A FULL PRESSURE TEST COULD BE PERFORMED. WITH THE OUTER DOOR INOPERABLE DUE TO A FAILED SURVEILLANCE TEST AND THE INNER BARRIER TECHNICALLY INOPERABLE, THE ACTION REQUIREMENT OF TS 3.3.1 WAS NOT MET. IN ADDITION, TS 3.3.4 REQUIRES A FULL PRESSURE TEST WITHIN 24 HOURS AND PRIOR TO DECLARING THE AIR LOCK OPERABLE. THEREFORE, THE PERSONNEL AIR LOCK LIMITING CONDITION FOR OPERATION (LCO) WAS NOT MET FROM JULY 11, 1990, UNTIL A FULL PRESSURE TEST WAS PERFORMED ON AUGUST 7, 1990. TO PREVENT RECURRENCE, WORK ORDERS ISSUED ON THE PERSONNEL AND EMERGENCY AIR LOCKS WILL BE PRINTED WITH A CAUTION STATEMENT. THE CAUTION STATES THAT, DURING TIME WHEN THE AIR LOCK IS REQUIRED TO BE OPERABLE AND MAINTENANCE IS REQUIRED ON ONE BARRIER (DOOR, VALVE, ETC.), NO MAINTENANCE SHALL BE PERFORMED ON THE OTHER "OPERABLE" BARRIER. IN ADDITION THE EVENT WAS REVIEWED WITH APPLICABLE PERSONNEL TO EMPHASIZE THE POST-MAINTENANCE TEST REQUIREMENTS.

[144] TURKEY POINT 4 DOCKET 50-251 LER 90-010
TECHNICAL SPECIFICATIONS 3.0.1 ENTRY DUE TO TRAIN "B" UNDERVOLTAGE PROTECTION CIRCUIT INOPERABLE.
EVENT DATE: 091090 REPORT DATE: 100190 NSSS: WE TYPE: PWR
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)

(NSIC 219696) AT 1515 ON SEPTEMBER 10, 1990, WITH UNIT 4 IN MODE 1 AT 100 PERCENT POWER, AN ALARM WAS RECEIVED ON CONTROL ROOM ANNUNCIATOR X-6/5: "4KV SYSTEM - BUS A AND B LOSS OF VOLTAGE FUSE FAILURE." AT 1605, ELECTRICAL MAINTENANCE PERSONNEL NOTIFIED THE PLANT SUPERVISOR - NUCLEAR (PSN) THAT ONE OF THE TWO FU-2 UNDERVOLTAGE PROTECTION FUSES IN THE 4B SEQUENCER RELAY CABINET WAS BLOWN. THE BLOWN FUSE RENDERED THE TRAIN B UNDERVOLTAGE PROTECTION CIRCUIT INOPERABLE. IN ACCORDANCE WITH TECHNICAL SPECIFICATION (TS) 3.5.1, UNIT 4 ENTERED TS 3.0.1 AT THIS TIME. THE SPECIFIC CAUSE FOR THE BLOWN FUSE CANNOT BE DETERMINED. TWO POSSIBLE CAUSES FOR THE BLOWN FUSE WERE POSTULATED BY THE FUSE VENDOR. FIRST, THE FUSE SENSED A LOW LEVEL FAULT CURRENT AND OPENED AS DESIGNED. FPL CONFIRMED THAT NO TESTING OR MAINTENANCE WAS BEING PERFORMED ON THE CIRCUIT PROTECTED BY THE FUSE. SECOND, THE FUSE EXPERIENCED CYCLIC EXPOSURE TO SMALL CURRENT SURGES OVER A PERIOD OF TIME WHICH LED TO DEGRADATION OF THE FUSE ELEMENT. THIS COULD NOT BE CONFIRMED THROUGH ANALYSIS OF THE BLOWN FUSE. AT 1715, THE FUSE WAS REPLACED IN THE 4B SEQUENCER RELAY CABINET AND THE TRAIN B UNDERVOLTAGE PROTECTION CIRCUIT WAS DECLARED OPERABLE. UNIT 4 EXITED TS 3.0.1 AT THIS TIME. A REPRESENTATIVE SAMPLE OF THE FU-1, FU-2 AND FU-3 FUSES USED IN THE UNIT 3 AND UNIT 4 UNDERVOLTAGE PROTECTION CIRCUITS WILL BE REPLACED.

[145] VERMONT YANKEE DOCKET 50-271 LER 89-005 REV 0
UPDATE ON INADVERTENT PRIMARY CONTAINMENT ISOLATION SYSTEM ACTUATION DUE TO AN INADEQUATE PROCEDURE.
EVENT DATE: 021489 REPORT DATE: 092190 NSSS: GE TYPE: BWR

(NSIC 219603) ON 2/14/89 AT APPROXIMATELY 1012 HOURS, WITH THE REACTOR IN COLD SHUTDOWN AND THE PLANT IN AN OUTAGE, A GROUP III PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) AND SUBSEQUENT STANDBY GAS TREATMENT SYSTEM (SBGTS) INITIATION OCCURRED. THE PCIS ISOLATION WAS INITIATED BY A HIGH RADIATION SIGNAL RECEIVED FROM THE REFUELING FLOOR MONITOR (EIIS=MON) WHEN THE REACTOR STEAM DRYER WAS BEING MOVED TO THE DRYER/SEPARATOR PIT AND WAS LIFTED OUT OF THE WATER. THE DRYER WAS PLACED IN THE DRYER/SEPARATOR PIT AND AT APPROXIMATELY 1017 HOURS, THE ISOLATION WAS RESET AND SYSTEMS WERE RETURNED TO NORMAL. THE ROOT CAUSE OF THIS EVENT IS AN INADEQUATE PROCEDURE. THE PROCEDURE USED TO DISSEMBLE THE REACTOR VESSEL DOES NOT HAVE A STEP TO ALERT THE OPERATORS OF THE EXPECTED INCREASED RADIATION LEVELS. THE PROCEDURE NEEDS TO ADDRESS AND DOCUMENT THE EXPECTED

ISOLATION AND REQUIRE APPROPRIATE NOTIFICATION TO THE CONTROL ROOM, PRIOR TO AND FOLLOWING STEAM DRYER MOVEMENT.

[146] VERMONT YANKEE DOCKET 50-271 LER 90-011
FULL REACTOR PROTECTION SYSTEM ACTUATION FROM SPIKE IN A SHARED LOCAL POWER RANGE MONITOR.
EVENT DATE: 090290 REPORT DATE: 100290 NSSS: GE TYPE: BWR

(NSIC 219654) ON 9/2/90 AT APPROXIMATELY 1830 HOURS WITH THE REACTOR SHUT DOWN AT 0% POWER, A HIGH-HIGH FLUX SIGNAL WAS RECEIVED FROM AVERAGE POWER RANGE MONITORS (APRMS) "C" AND "F" RESULTING IN A FULL SCRAM SIGNAL. AT APPROXIMATELY 1835 HOURS IT WAS DETERMINED THAT LOCAL POWER RANGE MONITOR (LPRM) 4B-32-25 WAS SPIKING HIGH AND IT WAS BYPASSED. THIS LPRM IS A SHARED INPUT TO BOTH APRM "F" AND APRM "C". AT THE TIME THIS OCCURRED ALL CONTROL RODS WERE INSERTED TO NORMAL FULL-IN (LATCHED) POSITION. PREVIOUS TO THE EVENT PERSONNEL HAD BEEN UNDER THE VESSEL REPOSITIONING NEUTRON MONITORING CABLES TO PREPARE FOR WORK ON THE CONTROL ROD DRIVEN (CRD) SYSTEM. IT IS BELIEVED THAT THE HANDLING OF THESE CABLES RESULTED IN A LOOSE OR BROKEN CONNECTOR WHICH CAUSED THE SPIKE. THIS LPRM WAS PUT INTO BYPASS AND THE SCRAM SIGNAL WAS RESET. A MAINTENANCE REQUEST HAS BEEN GENERATED TO INVESTIGATE AND REPAIR THE LPRM.

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

(NSIC 219702) ON 9/3/90 AND 9/6/90 WHILE PERFORMING TYPE C LEAK RATE TESTING WITH THE PLANT SHUTDOWN FOR THE 1990 REFUEL OUTAGE FEEDWATER CHECK VALVE FDW-96A (EIIS=SJ) AND PRIMARY CONTAINMENT ATMOSPHERIC CONTROL VALVE PCAC-6B (EIIS=BB) WERE FOUND TO HAVE SEAT LEAKAGE ABOVE THAT PERMITTED BY TECHNICAL SPECIFICATION 3.7.A.4. ON 9/3/90 THE SUM TOTAL LEAKAGE FOR TYPE B (PENETRATIONS) AND TYPE C (VALVES) EXCEEDED THAT ALLOWED BY 10 CFR 50 APPENDIX J. THE ASSIGNED PATHWAY LEAKAGE EXCEEDED THAT ALLOWED BY APPENDIX J AS A RESULT OF THE LEAKAGE THROUGH CHECK VALVE FDW-96A. APPENDIX J LIMITS THE TOTAL B AND C PENETRATION LEAKAGE TO 0.60 LA. VERMONT YANKEE IS PERFORMING MAINTENANCE ON THE VALVES THAT WERE FOUND TO BE LEAKING TO DETERMINE THE CAUSE OF THE FAILURE. THE VALVES WILL BE RETESTED TO VERIFY THAT SEAT LEAKAGE IS WITHIN ALLOWABLE LIMITS PRIOR TO PLANT STARTUP FOLLOWING THE 1990 REFUELING OUTAGE.

[148] VOGTLE 1 DOCKET 50-424 LER 90-017
INADEQUATE PROCEDURE LEADS TO INADEQUATE SURVEILLANCE TESTING.
EVENT DATE: 030590 REPORT DATE: 092090 NSSS: WE TYPE: PWR

(NSIC 219617) ON 8-23-90, THE SYSTEM ENGINEER RESPONSIBLE FOR THE ENGINEERED SAFETY FEATURE ACTUATION SYSTEM (ESFAS) WAS REVIEWING THE UNIT 2 ESFAS TEST PROCEDURE IN PREPARATION FOR FUTURE TESTING. HE FOUND THAT CERTAIN ELECTRICAL CIRCUITS WERE NOT IDENTIFIED FOR TESTING DURING PERFORMANCE OF THE PROCEDURE. THIS PROCEDURE IS USED TO SATISFY CERTAIN TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS WHICH MUST BE PERFORMED AT LEAST EVERY 18 MONTHS. A REVIEW OF UNIT 1 AND 2 OPERATIONS SURVEILLANCE PROCEDURES (OSP'S) FOUND THAT ALL OF THE CIRCUITS INVOLVED WERE BEING TESTED ON A QUARTERLY BASIS, SO THAT THE 18 MONTH SURVEILLANCE REQUIREMENT WAS BEING MET. HOWEVER, IT WAS FOUND THAT ON ONE OCCASION, A UNIT 1 QUARTERLY OSP HAD TAKEN CREDIT FOR THE ESFAS TESTING WHEN, IN FACT, THE AFFECTED CIRCUIT WAS NOT INCLUDED IN THE ESFAS TEST. THE UNIT SHIFT SUPERVISOR WAS ADVISED AND IT WAS DETERMINED THAT THE AFFECTED CIRCUIT HAD SUBSEQUENTLY BEEN SATISFACTORILY TESTED ON TWO OCCASIONS. THE CAUSES OF THIS EVENT WERE AN INADEQUATE ESFAS TEST PROCEDURE AND AN INADEQUATE REVIEW WHICH RESULTED IN THE INAPPROPRIATE USE OF THE 18 MONTH SURVEILLANCE PROCEDURE TO SATISFY THE QUARTERLY TESTING REQUIREMENTS. THE SYSTEM ENGINEER COMPLETED HIS REVIEW OF THE ESFAS TEST PROCEDURE WITH NO OTHER INADEQUATE SURVEILLANCE TESTS FOUND.

[149] VOGTLE 1 DOCKET 50-424 LER 90-018
INADEQUATE VERIFICATION OF OPERABILITY OF NUCLEAR SERVICE COOLING WATER CHECK VALVES.
EVENT DATE: 090790 REPORT DATE: 100590 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: VOGTLE 2 (PWR)

(NSIC 219773) ON 9-7-90, IT WAS DISCOVERED THAT THE ACCEPTANCE CRITERIA FOR VERIFYING OPERABILITY OF CHECK VALVES IN THE NUCLEAR SERVICE COOLING WATER, (NSCW) SUPPLY LINES TO THE COMPONENT COOLING WATER HEAT EXCHANGERS (CCW HX'S) WAS INCORRECT. THE SURVEILLANCE PROCEDURE REQUIRED A MINIMUM READING OF 45% (6750 GPM) TO BE OBTAINED FROM FLOW TRANSMITTERS LOCATED IN THESE LINES. HOWEVER, PER THE FINAL SAFETY ANALYSIS REPORT (FSAR), THE MINIMUM NSCW FLOW TO THE CCW HX'S IS 7950 GPM OR 53% OF INSTRUMENT SPAN. A REVIEW OF PRIOR PERFORMANCES OF THE SURVEILLANCE REVEALED THAT A READING LESS THAN 53% HAD BEEN OBTAINED ON SEVERAL OCCASIONS ON UNIT 1 AND ON ONE OCCASION ON UNIT 2. THEREFORE, A SURVEILLANCE REQUIRED BY TECHNICAL SPECIFICATION 4.0.5 AND ASME SECTION XI HAD BEEN INADEQUATELY PERFORMED. THE CAUSE OF THIS EVENT WAS PROCEDURE INADEQUACY. THE SURVEILLANCE PROCEDURE HAS BEEN CORRECTED AND THE OPERABILITY OF A CHECK VALVE THAT HAD A RECORDED FLOW OF LESS THAN 53% DURING ITS LAST SURVEILLANCE HAS BEEN REVERIFIED.

[150] VOGTLE 2 DOCKET 50-425 LER 90-011
CIRCUIT BOARD FAILURE LEADS TO CONTAINMENT VENTILATION ISOLATION.
EVENT DATE: 083090 REPORT DATE: 092490 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 219618) ON 8-30-90, AT 1011 CDT, THE REACTOR OPERATOR (RO) NOTICED THE INTERMEDIATE RADIATION LEVEL ALARM AND MONITOR CIRCUIT TROUBLE ALARM BEING INTERMITTENTLY DISPLAYED ON THE SAFETY RELATED DISPLAY CONSOLE (SRDC) IN THE CONTROL ROOM FOR CONTAINMENT AREA RADIATION MONITOR 2RE-0002. AT 1012 CDT, A HIGH ALARM WAS ALSO RECEIVED ON AN INTERMITTENT BASIS AND A CONTAINMENT VENTILATION ISOLATION (CVI) OCCURRED. TRAIN A AND TRAIN B EQUIPMENT ACTUATED AS DESIGNED. A REDUNDANT MONITOR (2RE-0003) WAS CHECKED, BUT IT SHOWED ONLY THE NORMAL BACKGROUND RADIATION LEVEL. CHEMISTRY PERSONNEL REPORTED THAT THE CONTAINMENT RADIATION LEVEL WAS NORMAL. THE 2RE-0002 DATA PROCESSING MODULE (DPM) WAS TAKEN TO THE BYPASS POSITION, BUT ALARMS CONTINUED TO OCCUR ON THE SRDC. THE LEADS WHICH SEND THE CVI ACTUATION SIGNAL FROM 2RE-0002 WERE LIFTED. AT 1142 CDT, THE CVI SIGNAL WAS RESET AND BY 1310 CDT, ALL VALVES AND DAMPERS AFFECTED BY THE CVI WERE RESTORED TO THEIR NORMAL STATUS. EXTENSIVE TESTING OF THE SYSTEM FOUND THAT A RANDOM ACCESS MEMORY (RAM) CIRCUIT BOARD IN THE DPM HAD FAILED. THIS FAILURE LED TO THE INTERMITTENT ALARMS AND THE CVI. THE FAILED RAM BOARD WAS REPLACED AND WILL BE RETURNED TO THE VENDOR FOR FAILURE ANALYSIS. ANY FURTHER CORRECTIVE ACTIONS WILL BE BASED ON THE RESULTS OF THE ANALYSIS.

[151] VOGTLE 2 DOCKET 50-425 LER 90-012
PERSONNEL ERRORS LEAD TO CONTAINMENT SPRAY PUMPS' DEACTIVATION.
EVENT DATE: 091690 REPORT DATE: 101590 NSSS: WE TYPE: PWR

(NSIC 219774) ON 9-16-90, WHILE IN MODE 4, THE SHIFT SUPERVISOR (SS) INADVERTENTLY ISSUED A MODE 5 CLEARANCE TO REMOVE THE TRAIN A AND TRAIN B CONTAINMENT SPRAY SYSTEM (CSS) PUMPS FROM SERVICE FOR MAINTENANCE. THE REACTOR OPERATOR (RO) HUNG CLEARANCE TAGS ON THE ACTUATION HANDSWITCHES AND PLACED THE HANDSWITCHES IN THE "PULL-TO-LOCK" POSITION AT 0440 CDT. THE SHIFT SUPERINTENDENT SAW THESE TAGS AND REALIZED THAT THE UNIT WAS OUT OF COMPLIANCE WITH THE TECHNICAL SPECIFICATIONS, WHICH REQUIRE BOTH PUMPS TO BE OPERABLE. SINCE THE LIMITING CONDITION FOR OPERATION (LCO) ACTION STATEMENT ADDRESSES ONLY ONE CSS BEING INOPERABLE, THE UNIT HAD ENTERED TS 3.0.3 BECAUSE IT WAS IN A CONDITION WHERE NO OTHER LCO ACTION STATEMENT WAS APPLICABLE. CONTROL ROOM OPERATORS RESTORED THE CSS PUMP HANDSWITCHES TO THE "AUTOMATIC" POSITION AT 0447 CDT. THIS ACTION RESTORED THE TRAIN B CSS, BUT THE TRAIN A PUMP MOTOR BREAKER WAS BEING RACKED OUT AT THE TIME. PLANT EQUIPMENT OPERATORS WERE NOTIFIED AND THE TRAIN A PUMP MOTOR BREAKER WAS RACKED IN. THE TRAIN A PUMP WAS STARTED TO VERIFY PROPER BREAKER OPERATION AND TRAIN A WAS RETURNED TO SERVICE AT 0602 CDT. THE

CAUSES OF THIS EVENT WERE COGNITIVE PERSONNEL ERRORS ON THE PART OF THE SS AND THE RO. BOTH HAVE BEEN COUNSELED.

[152] WATERFORD 3 DOCKET 50-382 LER 90-012
REACTOR TRIP DUE TO LIGHTNING STRIKE.
EVENT DATE: 082590 REPORT DATE: 092490 NSSS: CE TYPE: PWR

(NSIC 219647) AT 1802 HOURS ON AUGUST 25, 1990, WITH WATERFORD STEAM ELECTRIC STATION UNIT 3 AT 100% POWER, A SEVERE VOLTAGE TRANSIENT ON THE SOUTHEASTERN LOUISIANA 230 KV POWER TRANSMISSION GRID RESULTED IN A REACTOR TRIP. THE TRANSIENT WAS INITIATED WHEN A FAULT ATTRIBUTED TO A LIGHTNING STRIKE OCCURRED AT THE WATERFORD 230 KV SWITCHYARD. ONE CIRCUIT BREAKER EXPLODED AND BURNED. THE RAPID REDUCTION IN LOAD ON THE MAIN GENERATOR, COMBINED WITH AN INOPERABLE STEAM BYPASS CONTROL SYSTEM CAUSED REACTOR COOLANT SYSTEM TEMPERATURE AND PRESSURE TO INCREASE. A REACTOR TRIP OCCURRED DUE TO HIGH PRESSURIZER PRESSURE. THIS EVENT IS REPORTABLE AS AN AUTOMATIC REACTOR PROTECTION SYSTEM ACTUATION. AS A RESULT OF THIS SYSTEM TRANSIENT, VOLTAGE ON THE WATERFORD 230KV SWITCHYARD DECAYED TO 33 KV OVER A TWO SECOND PERIOD. AN INVESTIGATION CONCLUDED THAT THE IN-PLANT SAFETY BUS VOLTAGE LEVEL DID NOT DROP LOW ENOUGH FOR A SUFFICIENT DURATION TO ACTUATE THE UNDERVOLTAGE RELAYS ASSOCIATED WITH THE EMERGENCY DIESEL GENERATORS. THE ROOT CAUSE OF THIS EVENT WAS THE SEVERE GRID DISTURBANCE, WHICH IS ATTRIBUTED TO A LIGHTNING STRIKE. BECAUSE PLANT PROTECTIVE FEATURES FUNCTIONED AS DESIGNED, THIS EVENT DID NOT THREATEN THE HEALTH OR SAFETY OF THE GENERAL PUBLIC OR PLANT PERSONNEL.

[153] WATERFORD 3 DOCKET 50-382 LER 90-013
UNSAMPLED RELEASE FROM THE GASEOUS WASTE MANAGEMENT SYSTEM.
EVENT DATE: 090390 REPORT DATE: 100290 NSSS: CE TYPE: PWR

(NSIC 219648) AT 0525 HOURS ON SEPTEMBER 3, 1990, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS AT 100% POWER WHEN PLANT PERSONNEL COMMENCED A BATCH RELEASE OF GASEOUS WASTE MANAGEMENT (GWM) SYSTEM GAS DECAY TANK (GDT) 'B'. APPROXIMATELY 55 SECONDS AFTER THE RELEASE WAS COMMENCED, THE LOCAL GWM EFFLUENT RADIATION MONITOR WENT INTO HIGH ALARM AND AUTOMATICALLY SECURED THE BATCH RELEASE. THE MOST LIKELY CAUSE OF THIS EVENT IS INTERNAL LEAKAGE PAST DISCHARGE ISOLATION VALVES GWM-305C AND 3051C CAUSING A SMALL VOLUME OF UNSAMPLED GAS TO BE RELEASED FROM GDT 'C' DURING THE GASEOUS RELEASE FROM GDT 'B'. TECHNICAL SPECIFICATIONS (TS) REQUIRE THAT REPRESENTATIVE SAMPLES AND ANALYSES OF GASEOUS EFFLUENTS BE OBTAINED PRIOR TO RELEASE; THEREFORE, THIS EVENT IS REPORTABLE AS PLANT OPERATION PROHIBITED BY TS. THE GWM SYSTEM IS BEING EVALUATED TO DETERMINE WHICH VALVES ARE INTERNALLY LEAKING AND TO IDENTIFY LONG TERM CORRECTIVE MEASURES. SINCE CONSERVATIVE CALCULATIONS OF THE ESTIMATED INSTANTANEOUS DOSE RATES SHOWED THAT THE UNSAMPLED RELEASE WAS WELL WITHIN TS LIMITS, THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT.

[154] WPPSS 2 DOCKET 50-397 LER 90-017
HIGH PRESSURE CORE SPRAY SYSTEM INOPERABILITY AS A RESULT OF 125 VDC BATTERY AND DIESEL GENERATOR FUEL OIL LEVEL SWITCH INOPERABILITY.
EVENT DATE: 083090 REPORT DATE: 100190 NSSS: GE TYPE: BWR
VENDOR: C & D BATTERIES, DIV OF ELTRA CORP.

(NSIC 219757) THREE RELATED INSTANCES OF INOPERABILITY OF THE HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) OCCURRED OVER A TWENTY DAY PERIOD. ON AUGUST 30, 1990, AT 1234 HOURS, DURING PERFORMANCE OF WEEKLY TECHNICAL SPECIFICATION-REQUIRED SURVEILLANCE TESTS, HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) 125 VDC BATTERY CELL NUMBER NINE WAS DISCOVERED TO CONTAIN A CRACK IN THE CELL JAR. THE HPCS 125 VDC BATTERY WAS DECLARED INOPERABLE ALONG WITH THE HPCS SYSTEM. ON AUGUST 31, 1990, AT 0515 HOURS, WHILE THE HPCS SYSTEM WAS STILL INOPERABLE, DURING PERFORMANCE OF HPCS DIESEL GENERATOR (HPCS DG) OPERABILITY SURVEILLANCE TESTS, IT WAS NOTED THAT DO-LS-21, FUEL OIL LEVEL SWITCH FOR THE HPCS FUEL OIL DAY TANK (DO-TK-3C) WAS EXHIBITING QUESTIONABLE PERFORMANCE. AS A RESULT, AT 1132 HOURS, THE HPCS DIESEL GENERATOR WAS DECLARED INOPERABLE. ON SEPTEMBER 6, 1990, AT 1350 HOURS, WHILE PERFORMING THE WEEKLY TECHNICAL SPECIFICATION BATTERY CHECKS, AN ELECTRICAL

MAINTENANCE TECHNICIAN DISCOVERED THAT THE HPCS 125 VDC BATTERY VOLTAGE WAS BELOW THE VALUE ALLOWED IN TECHNICAL SPECIFICATIONS. THE HPCS BATTERY AND THE HPCS SYSTEM WERE DECLARED INOPERABLE. IT WAS DISCOVERED THAT BATTERY VOLTAGE HAD BEEN 0.75 VDC BELOW THE MINIMUM VALUE LISTED IN TECHNICAL SPECIFICATIONS SINCE AUGUST 31, 1990.

[155] WPPSS 2 DOCKET 50-397 LER 90-018
ENGINEERED SAFETY FEATURE ACTUATION OF CONTAINMENT INSTRUMENT AIR CAUSED BY RELIEF VALVE BLOWDOWN.
EVENT DATE: 090490 REPORT DATE: 100490 NSSS: GE TYPE: BWR

(NSIC 219758) AT APPROXIMATELY 1020 HOURS ON SEPTEMBER 4, 1990, A PRESSURE DECREASE OCCURRED IN THE CONTAINMENT INSTRUMENT AIR (CIA) SYSTEM AS A RESULT OF AN INADVERTENT MANUAL RELIEF VALVE ACTUATION. THIS PRESSURE DECREASE CAUSED THE NON-SAFELY-RELATED PART OF THE CIA SYSTEM TO BE ISOLATED FROM THE SAFETY RELATED PART AND AUTOMATICALLY PLACED THE SAFETY RELATED BOTTLED NITROGEN SOURCE INTO SERVICE. THIS ACTION IS CONSIDERED AN ENGINEERED SAFETY FEATURE ACTUATION. THE ROOT CAUSE OF THIS EVENT WAS A DEFICIENCY IN THE EQUIPMENT DESIGN. THIS ALLOWED A LIGHTING FIXTURE TO BE IN A POSITION TO BE PUSHED INTO AND ACTUATE A RELIEF VALVE WHICH RESULTED IN A LOSS OF PRESSURE IN THE SAFETY-RELATED PORTION OF THE CIA SYSTEM. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO PREVENT RECURRENCE BY TEMPORARILY REMOVING THE MANUAL ACTUATOR ON THE RELIEF VALVE AND SECURING THE LIGHTING FIXTURE. THE EVENT POSED NO THREAT TO THE HEALTH AND SAFETY OF EITHER THE PUBLIC OR PLANT PERSONNEL.

[156] ZION 1 DOCKET 50-295 LER 90-011
UNCONTROLLED REACTOR HEAD VENT RELEASE.
EVENT DATE: 041090 REPORT DATE: 100490 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 2 (PWR)

(NSIC 219713) ON 4/10/90, WHILE A REACTOR HEAD VENT WAS IN PROGRESS, IT WAS REALIZED THAT IF ANY CHANNEL OF THE CONTAINMENT PURGE MONITOR 1(2)R-PR09 WAS INOPERABLE, THEN A REACTOR VESSEL HEAD VENT WAS AN UNCONTROLLED RELEASE, BECAUSE THE HEAD VENT RIG BYPASSES THE CONTAINMENT ATMOSPHERE MONITOR 1(2)R-PR40 SAMPLE POINT. THE RELEASE IS STILL MONITORED, AS THE VENTILATION STACK RADIATION MONITORS ARE DOWNSTREAM OF 1(2)R-PR09, BUT THE STACK MONITORS HAVE NO CONTROL FUNCTION. THIS REPORT IS BEING SUBMITTED AS A VOLUNTARY LER. THE ROOT CAUSE OF THIS EVENT IS A FAILURE TO REALIZE THE ROUTING OF THE TYGON VENT RIG BYPASSED THE 1(2)R-PR40 MONITOR SAMPLE POINT. THIS WAS DUE IN PART TO MODIFICATIONS TO THE RADIATION MONITORING SYSTEM AND CHANGES IN TECHNICAL SPECIFICATIONS AFTER USE OF THE HEAD VENT RIG HAD BECOME STANDARD PRACTICE. THE HEAD NOBLE GASES ARE SAMPLED AND THE ACTIVITY IS QUANTIFIED PRIOR TO RELEASE. AT NO TIME WERE ANY DOSE RATE LIMITS EXCEEDED. THERE WAS THEREFORE NO SAFETY SIGNIFICANCE TO THIS EVENT. THE CORRECTIVE ACTIONS FOR THIS EVENT ARE TO IMPOSE STRICTER PROCEDURAL CONTROLS ON THE ACTIONS REQUIRED FOR HEAD VENT RELEASES.

[157] ZION 1 DOCKET 50-295 LER 90-018
VENT STACK SYSTEM PARTICULATE IODINE NOBLE GAS MONITOR FOUND IN FLUSH MODE DUE TO PERSONNEL ERROR.
EVENT DATE: 082490 REPORT DATE: 092490 NSSS: WE TYPE: PWR

(NSIC 219621) ON 8/24/90 AT APPROXIMATELY 0305, THE NUCLEAR STATION OPERATOR (NSO) PERFORMING NIGHTLY PERIODIC TEST (PT)-0 APPENDIX N (RADIATION MONITOR CHECK SHEET) FOUND THE VENT STACK SPING (SYSTEM PARTICULATE IODINE NOBLE GAS) MONITOR (1RIA-PR49) IN FLUSH MODE AT THE MONITOR CONSOLE. BEING IN THE FLUSH MODE RENDERS THE MONITOR INOPERABLE. THE REQUIRED SAMPLES WERE NOT TAKEN. THE ROOT CAUSE OF THIS EVENT WAS A PERSONNEL ERROR. NO SAFETY CONSEQUENCES RESULTED, AS REDUNDANT MONITORS SHOWED NO ABNORMAL ACTIVITY. CORRECTIVE ACTIONS INCLUDE SHIFTLY CHECKS TO ENSURE PROPER MONITOR OPERATION, AND A LONG TERM CORRECTIVE ACTION OF MAKING THE MONITOR MUCH MORE RELIABLE.

[158] ZION 1 DOCKET 50-295 LER 90-019
MISSED SURVEILLANCE ON PRESSURIZER LEVEL CHANNEL DUE TO PROGRAMMATIC DEFICIENCY.
EVENT DATE: 082690 REPORT DATE: 092590 NSSS: WE TYPE: PWR

(NSIC 219622) THE UNIT 1 FUNCTIONAL TEST 1L-459, WHICH IS THE QUARTERLY TEST OF A PRESSURIZER LEVEL INSTRUMENT CHANNEL, WAS ON THE INSTRUMENT MAINTENANCE DEPARTMENT (IMD) SURVEILLANCE SCHEDULE FOR THE MONTH OF AUGUST. THE DUE DATE WAS 8/3/90, WITH A CRITICAL DATE (WHICH ALLOWS FOR THE 25% GRACE PERIOD) OF 8/26/90. THE FUNCTIONAL TEST FOR 1L-459 WAS NOT PERFORMED UNTIL 8/27/90, WHICH WAS PAST THE CRITICAL DATE. THUS, THE TECHNICAL SPECIFICATION SURVEILLANCE, WHICH IS REQUIRED TO BE PERFORMED QUARTERLY WAS MISSED. THE ROOT CAUSE OF THIS EVENT WAS A PROGRAMMATIC DEFICIENCY. NO SAFETY CONSEQUENCES RESULTED, AS THE SURVEILLANCE FOUND THE CHANNEL TO BE IN CALIBRATION. CORRECTIVE ACTIONS INCLUDE CHANGES IN THE SCHEDULING PROGRAM, AND HEIGHTENED MANAGEMENT AND WORKER AWARENESS OF SURVEILLANCES IN DANGER OF BEING OVERDUE.

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

(NSIC 219714) ON 9/15/90 AT 1226, WHILE PERFORMING PERIODIC TEST (PT) 5B, "REACTOR PROTECTION LOGIC REACTOR AT NORMAL OPERATION CONDITONS", SECTION 4 OF THE TEST FAILED. SECTION 4 TESTS THE TRAIN A 2/4 REACTOR COOLANT PUMP (RCP) BUS UNDERVOLTAGE REACTOR TRIP LOGIC BY DE-ENERGIZING LOGIC RELAYS TO SIMULATE RCP BUS UNDERVOLTAGE. TROUBLESHOOTING EFFORTS IDENTIFIED THAT THE BUS 144 UNDERVOLTAGE LOGIC RELAY, 27-2/XA, WAS DEFECTIVE. DURING THE REPLACEMENT OF THE RELAY, 1A AUXILIARY FEEDWATER (AFW) PUMP STARTED AUTOMATICALLY. DETERMINATION OF THE RELAY CONTACTS OPENED THE NORMALLY ENERGIZED 2/4 LOGIC CIRCUIT WHICH AUTOSTARTED THE AFW PUMP. FOLLOWING RELAY REPLACEMENT, THE "RCP BUS UV RX TRIP" ANNUNCIATOR DID NOT ACTUATE AS REQUIRED DURING TESTING DUE TO A WIRING ERROR ON RELAY 27-2/XA. THE WIRING DISCREPANCY WAS CORRECTED AND SECTION 4 OF THE TEST COMPLETED SATISFACTORILY. THERE WAS MINIMAL EFFECT ON PLANT SAFETY SINCE TRAIN "B" OF REACTOR PROTECTION WAS ALWAYS AVAILABLE.

[160] ZION 2 DOCKET 50-304 LER 90-010
TRIP DUE TO CONDENSER BOOT FAILURE.
EVENT DATE: 090790 REPORT DATE: 100990 NSSS: WE TYPE: PWR
VENDOR: DEARBORN RUBBER CORP.

(NSIC 219715) ON 9/7/90 ZION UNIT 2 TRIPPED ON A SUDDEN LOSS OF CONDENSER VACUUM. INSPECTIONS FOUND A FAILED CONDENSER EXPANSION BOOT IN THE A-BAY OF THE CONDENSER, A FAILED LOW PRESSURE TURBINE RUPTURE DISC, AND THAT BOTH OF THE FEED WATER PUMPS' RUPTURE DISCS HAD BEEN DEFORMED. INTERNAL INSPECTION OF THE CONDENSER DETERMINED THAT THE INITIATING EVENT HAD BEEN A FAILURE OF THE BOOT. THE ROOT CAUSE ANALYSIS DETERMINED THAT THE CONDENSER EXPANSION BOOT HAD A LIFE EXPECTANCY OF 5 TO 7 YEARS BUT HAD BEEN IN SERVICE FOR 9 YEARS. AGGRAVATING THE AGED CONDITION WAS IMPROPER TORQUING AND EXPOSURE TO EXCESSIVE TEMPERATURE. ALL SAFETY SYSTEMS RESPONDED AS DESIGNED. CORRECTIVE ACTION INCLUDED REPLACEMENT OF THE DAMAGED COMPONENTS, ADDING THE BOOTS TO THE PREVENTIVE MAINTENANCE PROGRAM TO ENSURE THAT THEY ARE REPLACED AT 5 YEAR INTERVALS, AND TO FOLLOW VENDOR INSTALLATION RECOMMENDATIONS.

COMPONENT INDEX

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

COMPONENT INDEX

SYSTEM INDEX

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

SYSTEM INDEX

SYSTEM INDEX

KEYWORD INDEX

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

KEYWORD INDEX

KEYWORD INDEX

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KEYWORD INDEX

VENDOR CODE INDEX

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BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any): NUREG/CR-2000, Vol. 9, No. 11; ORNL/NSIC-200

2. TITLE AND SUBTITLE: Licensee Event Report (LER) Compilation For month of November 1990

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR
December	1990

5. AUTHOR(S): Prepared by Oak Ridge National Laboratory

6. DATE REPORT ISSUED

MONTH	YEAR
December	1990

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): November 1990

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

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

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS

Licensee Event Report
Sequence Coding and Search System
Reactor, PWR
Reactor, BWR
Systems
Components
Operating Experience
Event Compilation

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT: Unlimited

16. SECURITY CLASSIFICATION
(This page) Unclassified
(This report) Unclassified

17. NUMBER OF PAGES

18. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

FIRST CLASS MAIL
POSTAGE & FEES PAID
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PERMIT No. G-67

120555139531 1 1AN1CV1NJ11M1
US NRC-OADM
DIV FOIA & PUBLICATIONS SVCS
TPS PDR-NUREG
P-223
WASHINGTON DC 20555

COMPILATION

DECEMBER 1990