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Vol. 8, No. 11

Licensee Event Report (LER) Compilation

For month of November 1989

Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory
Commission

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

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Abstract

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

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

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[1] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 89-029
PERSONNEL ERROR RESULTS IN OPERATION OF THE SERVICE WATER SYSTEM IN A CONFIGURATION IN WHICH A SINGLE COMPONENT FAILURE COULD RENDER THE SYSTEM INOPERABLE DURING EMERGENCY OPERATING CONDITIONS.
EVENT DATE: 082589 REPORT DATE: 092289 NSSS: BW TYPE: PWR

(NSIC 215361) ON 8/25/89, PLANT ENGINEERING PERSONNEL DETERMINED THAT THE SERVICE WATER SYSTEM (SW) OPERATING PROCEDURE ALLOWED OPERATION OF THE SYSTEM IN A CONFIGURATION THAT, UNDER CERTAIN CONDITIONS, COULD RESULT IN THE SYSTEM BEING RENDERED INOPERABLE BY A SINGLE COMPONENT FAILURE. THE PROCEDURE PROVIDED FOR NORMAL OPERATION OF THE SYSTEM WITH TWO SW PUMPS IN SERVICE AND ALL OF THE LOOP CROSSTIE VALVES OPEN. IF AN ENGINEERED SAFEGUARDS (ES) ACTUATION WERE TO OCCUR CONCURRENT WITH A LOSS OF OFFSITE POWER WHILE PUMP P4A (LOOP 1) WAS OUT OF SERVICE AND P4B (ALIGNED TO "RED" ES POWER) AND P4C ("GREEN" ES POWER) WERE SUPPLYING SW AND AUXILIARY COOLING SYSTEM LOADS, A FAILURE OF THE "GREEN" EMERGENCY DIESEL GENERATOR TO START WOULD RESULT IN ONLY ONE PUMP (P4B) REMAINING IN SERVICE SUPPLYING BOTH LOOPS OF SW AND THE ACW LOADS. SINCE ONE SW PUMP IS NOT CAPABLE OF SUPPLYING ADEQUATE FLOW TO BOTH SW LOOPS AND ACW, THE SYSTEM WOULD BE INOPERABLE. ORIGINALLY, THE SW OPERATING PROCEDURE CALLED FOR THE CROSSTIE VALVES TO BE CLOSED. HOWEVER, IT WAS COMMON PRACTICE TO OPERATE WITH THEM OPEN TO ALLOW RUNNING THREE SW PUMPS DURING HOT WEATHER. THE CAUSE OF THIS CONDITION WAS PERSONNEL ERROR IN THAT PLANT PERSONNEL FAILED TO ADEQUATELY EVALUATE THE POTENTIAL CONSEQUENCES OF OPERATING WITH THE CROSSTIE VALVES OPEN.

[2] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 89-032
INADEQUATE PROCEDURAL GUIDANCE RESULTS IN THE FAILURE TO PERFORM THE REACTOR BUILDING AREA RADIATION MONITORS MONTHLY SURVEILLANCE REQUIRED BY TECH SPECS.
EVENT DATE: 090689 REPORT DATE: 100689 NSSS: BW TYPE: PWR

(NSIC 215555) ON 9/6/89, IT WAS DISCOVERED THAT THE MONTHLY FUNCTIONAL TEST FOR THE REACTOR BUILDING AREA RADIATION MONITORS WAS NOT PERFORMED AS REQUIRED BY TECH SPEC. A NEW PROCEDURE FOR TESTING OF THESE MONITORS HAD BEEN WRITTEN AND THE MONITORS HAD BEEN DELETED FROM THE ORIGINAL TEST PROCEDURE. A MASTER TEST CONTROL LIST (MTCL), MAINTAINED TO TRACK THE TESTING REQUIREMENTS ASSOCIATED WITH TECH SPECS, SHOULD BE REVISED WHENEVER A PROCEDURE THAT MAY AFFECT THE MTCL IS CHANGED. CURRENTLY, THERE IS NO PROCEDURAL GUIDANCE GIVEN TO ENSURE THIS IS ACCOMPLISHED. A REVISION TO THE MTCL WAS SUBMITTED WITH THE REVISIONS TO THE AREA RADIATION MONITOR PROCEDURES, HOWEVER, DUE TO AN ERROR ON THE MTCL REVISION REQUEST IT WAS NOT APPROVED AT THE SAME TIME THE AREA RADIATION MONITOR TEST PROCEDURES WERE APPROVED AND IMPLEMENTED ON 6/26/89. THE TIME LAPSE ASSOCIATED WITH FINAL APPROVAL OF THE MTCL REVISION RESULTED IN THE REACTOR BUILDING AREA RADIATION MONITORS NOT BEING TESTED AS REQUIRED. THE SUBSEQUENT SATISFACTORY COMPLETION OF THE SURVEILLANCE INDICATED THE MONITORS WERE OPERABLE. TO ENSURE THAT A REQUIRED TECH SPEC SURVEILLANCE IS PROPERLY IDENTIFIED ON THE MTCL, A REVISION TO THE PROCEDURE REVISION REQUEST FORM, WHICH ACCOMPANIES EACH PROCEDURE CHANGE, AND SPECIFIC PROCEDURAL GUIDANCE CONCERNING THE MTCL HAS BEEN INITIATED.

[3] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 89-028
WIRING ERROR DUE TO INADEQUATE CONFIGURATION CONTROL RESULTS IN TWO INOPERABLE SERVICE WATER PUMPS.
EVENT DATE: 091289 REPORT DATE: 101289 NSSS: BW TYPE: PWR

(NSIC 215668) ON 09/12-13/89, AN EXTRA CONTACT WAS FOUND IN THE CONTROL CIRCUITS FOR SERVICE WATER PUMPS P4A AND P4C WHICH WAS NOT SHOWN ON THE PUMPS' CONTROL CIRCUIT SCHEMATICS. UNDER CERTAIN REQUIRED STARTING CONDITIONS (E.G., AN ENGINEERED SAFEGUARDS ACTUATION SIGNAL PRIOR TO OR WITHOUT A MAIN GENERATOR LOCKOUT, CAUSING A "SLOW" TRANSFER TO OFF-SITE POWER), THE EXTRA CONTACTS WOULD HAVE CAUSED THE "ANTI-PUMP" CIRCUIT IN THE POWER SUPPLY BREAKERS FOR THE PUMPS TO LOCKOUT THE CLOSE SIGNAL TO THE BREAKERS. THIS WOULD PREVENT AN AUTOMATIC (AS

DESIGNED) OR MANUAL CONTROL ROOM START OF THE PUMPS. BOTH P4A AND P4C WERE DECLARED INOPERABLE UNTIL THE CIRCUIT WAS MODIFIED AND TESTED ON 09/15/89. THE EVENT IS CONSIDERED SAFETY SIGNIFICANT DUE TO THE POTENTIAL FOR LOSS OF THE SERVICE WATER SYSTEM. THE WIRING ERROR WAS DETERMINED TO BE THE RESULT OF INADEQUATE CONFIGURATION CONTROL DURING THE DESIGN, CONSTRUCTION, AND STARTUP PHASES OF ANO-1. A PROGRAM OF WIRING INSPECTIONS WAS INITIATED ON BOTH ANO-1 AND ANO-2 TO PROVIDE ASSURANCE THAT NO ADDITIONAL SAFETY SIGNIFICANT WIRING DISCREPANCIES EXISTED. ALTHOUGH ADDITIONAL DISCREPANCIES HAVE BEEN IDENTIFIED, NONE HAVE BEEN FOUND TO BE SAFETY SIGNIFICANT. FURTHER LONG-TERM ACTIONS WILL RESOLVE THE IDENTIFIED DISCREPANCIES BY REVISING DRAWINGS AND/OR EQUIPMENT AS NECESSARY.

[4] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 87-010
PLANT PROTECTION SYSTEM PANELS SEISMIC QUALIFICATION COMPROMISED DUE TO LOOSE FASTENERS CAUSED BY PERSONNEL ERROR.
EVENT DATE: 052287 REPORT DATE: 060189 NSSS: CE TYPE: PWR

(NSIC 215396) ON 5/22/87, DURING A QUALITY ASSURANCE SURVEILLANCE, IT WAS IDENTIFIED THAT AFTER THE PERFORMANCE OF SURVEILLANCE TESTING, INSTRUMENTATION AND CONTROLS MAINTENANCE PERSONNEL TIGHTENED ONLY TWO OF THE FOURTEEN FASTENERS WHICH SECURE EACH OF THE FOUR PLANT PROTECTION SYSTEM (PPS) POWER SUPPLY PANELS TO THE FRAMES OF THE MAIN CABINETS. ALSO, SOME OF THE FASTENERS WERE MISSING. THE PPS CABINETS WERE DESIGNED AND QUALIFIED TO WITHSTAND A SEISMIC EVENT WITHOUT COMPROMISE TO THE STRUCTURE OR SUBASSEMBLIES VITAL TO SYSTEM OPERATION. THE POWER SUPPLY PANELS ARE HELD IN PLACE BY A FULL LENGTH HINGE ON ONE SIDE AND FOURTEEN SCREWS ON THE OTHER SIDE AND BOTTOM. THE SCREWS MUST BE REMOVED TO PERFORM TESTING OR MAINTENANCE. IT WAS DETERMINED THAT IT HAD BEEN COMMON PRACTICE TO TIGHTEN ONLY TWO OF THE PANEL FASTENERS AFTER TESTING OR MAINTENANCE. SINCE ONLY TWO OF FOURTEEN SCREWS WERE TIGHTENED PER PANEL, THE SEISMIC QUALIFICATION OF THE PANELS WAS JEOPARDIZED. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR IN THAT THE TECHNICIANS DID NOT CONSIDER THE SEISMIC QUALIFICATION OF THE EQUIPMENT WHEN ELECTING NOT TO TIGHTEN ALL THE SCREWS. THE SIGNIFICANCE OF THIS EVENT IS CONSIDERED MINIMAL BASED ON THE FAIL-SAFE DESIGN OF THE PSS. THE EXISTING PANEL SCREWS WERE TIGHTENED AND THE MISSING SCREWS WERE REPLACED.

[5] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 89-016
LACK OF CONTINGENCIES ESTABLISHED TO ENSURE CONTINUOUS FIRE WATCHES REQUIRED BY TECH SPECS COULD BE MAINTAINED IN NONFUNCTIONAL FIRE BARRIERS IF HIGH AIRBORNE RADIOACTIVITY WERE TO OCCUR.
EVENT DATE: 090389 REPORT DATE: 100389 NSSS: CE TYPE: PWR

(NSIC 215522) ON 9/3/89 AND ON 9/18/89, THE CONTINUOUS FIRE WATCH PERSONNEL STATIONED AS REQUIRED BY TECH SPECS TO MONITOR NONFUNCTIONAL FIRE BARRIERS, WERE REMOVED DUE TO HIGH AIRBORNE RADIOACTIVITY. FOLLOWING THE FIRST EVENT THE CONTROL ROOM SHIFT SUPERVISOR MET WITH THE FIRE WATCH SUPERVISOR AND HP TECHNICIANS TO ESTABLISH A CONTINGENCY OF RELIEVING FIRE WATCHES IN AFFECTED AREAS WITH A ROVING FIRE WATCH WHO HAD DONNED RESPIRATORY PROTECTION UNTIL THE PERSONNEL POSTED AT SPECIFIC LOCATIONS COULD OBTAIN THE APPROPRIATE RESPIRATORY EQUIPMENT AND RETURN TO THEIR POSTS. DURING THE TIMES FIRE WATCHES WERE NOT MAINTAINED AT REQUIRED AREAS, THE FIRE DETECTION INSTRUMENTATION AND ASSOCIATED CONTROL ROOM ANNUNCIATION AND FIRE WATER SUPPRESSION SYSTEMS FOR AFFECTED AREAS WERE VERIFIED TO BE OPERATIONAL ENSURING IF A FIRE WERE TO OCCUR IT COULD HAVE BEEN PROMPTLY DETECTED AND SUPPRESSED. OPERATIONS PERSONNEL ALSO EVALUATED THE PLANT EVOLUTIONS IN PROGRESS AT THE TIME OF INCREASED AIRBORNE RADIOACTIVITY TO ATTEMPT TO DETERMINE THE CAUSE OF THE AIRBORNE ACTIVITY AND COORDINATED WITH FIRE WATCH PERSONNEL IF AIRBORNE ACTIVITY WERE EXPECTED TO OCCUR. WHEN AN EVOLUTION KNOWN TO RESULT IN HIGH AIRBORNE RADIOACTIVITY WAS SCHEDULED FIRE WATCHES IN AFFECTED AREAS WERE EQUIPPED WITH RESPIRATORY EQUIPMENT.

[6] ARNOLD DOCKET 50-331 LER 89-011 REV 01
UPDATE ON TURBINE CONTROL VALVE FAST CLOSURE TRIP RESULTS IN REACTOR SCRAM WHILE PERFORMING TESTING.
EVENT DATE: 082689 REPORT DATE: 102589 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215750) ON AUGUST 26, 1989, AT 1642 HOURS WITH THE PLANT OPERATING AT 100% POWER, OPERATIONS PROCEDURE, "POWER/LOAD UNBALANCE AND RELAY CIRCUITS TEST" WAS IN PROGRESS. THIS TEST IS PERFORMED FOR CONTINUED RELIABLE OPERATION OF THE MAIN TURBINE. CONTRARY TO WHAT WAS EXPECTED, A TRIP OF THE MAIN TURBINE CONTROL VALVES AND SUBSEQUENT REACTOR SCRAM OCCURRED AT 1643 HOURS. SUBSEQUENT DETAILED INVESTIGATIONS IDENTIFIED BRIDGING OF A MERCURY-WETTED RELAY IN THE POWER/LOAD UNBALANCE CIRCUITRY AS THE MOST PROBABLE ROOT CAUSE FOR THE TURBINE TRIP AND SUBSEQUENT REACTOR SCRAM. APPROXIMATELY FIVE MINUTES FOLLOWING THE SCRAM, PROBLEMS WERE ENCOUNTERED ON THE "B" ESSENTIAL AND NON-ESSENTIAL BUSSES. SUBSEQUENT INVESTIGATION REVEALED THE ROOT CAUSE TO BE A FAILED TRIP COIL ON AN ASSOCIATED BREAKER. THE PLANT WAS BROUGHT TO A NORMAL SAFE SHUTDOWN CONDITION AND THE APPROPRIATE NOTIFICATIONS WERE MADE. THERE WAS NO EFFECT ON THE SAFE OPERATION OF THE PLANT.

[7] ARNOLD DOCKET 50-331 LER 89-012
LOSS OF SECONDARY CONTAINMENT DUE TO DEGRADED VENT SHAFT AND INADEQUATE TEST METHODS.
EVENT DATE: 092089 REPORT DATE: 101989 NSSS: GE TYPE: BWR

(NSIC 215733) ON 9/20/89 WITH THE PLANT IN COLD SHUTDOWN FOR MID-CYCLE TESTING OF THE MAIN STEAM ISOLATION VALVES AND MAINTENANCE ACTIVITIES, A HOLE WAS DISCOVERED IN THE REACTOR BUILDING VENTILATION DUCTWORK. THIS HOLE ALLOWED THE MAIN PLANT EXHAUST FANS TO EXHAUST AIR FROM THE REACTOR BUILDING EVEN UNDER SECONDARY CONTAINMENT ISOLATION CONDITIONS. THE SECONDARY CONTAINMENT OPERABILITY TEST WAS NOT CAPABLE OF DETECTING A FAILURE OF THIS NATURE. THE FAILURE WAS DETERMINED TO BE DUE TO THE COVER OF THE REACTOR BUILDING EXHAUST VENT SHAFT NOT BEING INSTALLED IN ACCORDANCE WITH THE ORIGINAL DESIGN SPECIFICATIONS. THE DUCTWORK WAS MODIFIED AND REINSTALLED WITH AN IMPROVED SUPPORTING STRUCTURE. THE SECONDARY CONTAINMENT OPERABILITY TEST WAS CHANGED TO ENSURE DEGRADATION OF SECONDARY CONTAINMENT IS DETECTED. AS A RESULT OF THE TESTING FOLLOWING REPAIR OF THE VENT SHAFT, SEVERAL SMALL HOLES, CRACKS AND PENETRATION SEALS WERE REPAIRED. FOLLOWING THESE REPAIRS THE SECONDARY CONTAINMENT OPERABILITY TEST WAS PERFORMED SUCCESSFULLY.

[8] BEAVER VALLEY 1 DOCKET 50-334 LER 89-008
REACTOR TRIP AND FEEDWATER ISOLATION DURING SHUTDOWN.
EVENT DATE: 090189 REPORT DATE: 100289 NSSS: WE TYPE: PWR
VENDOR: COPES-VULCAN, INC.

(NSIC 215520) ON 9/1/89, UNIT 1 WAS PERFORMING A PLANNED SHUTDOWN IN PREPARATION FOR THEIR SEVENTH REFUELING OUTAGE. THE REACTOR WAS SUBCRITICAL WITH CONTROL ROD INSERTION IN PROGRESS. AT 2210 HOURS, THE "A" MAIN FEEDWATER REGULATING VALVE, WHICH HAD BEEN CLOSED EARLIER DURING THE SHUTDOWN, WAS OBSERVED TO BE OPEN. DESPITE OPERATOR ACTIONS (SHUTTING DOWN THE MAIN FEEDWATER PUMP), AT 2211 HOURS, "A" STEAM GENERATOR LEVEL INCREASED TO ITS HIGH-HIGH LEVEL SETPOINT, INITIATING AN AUTOMATIC FEEDWATER ISOLATION (FWI). OPERATORS VERIFIED ALL REQUIRED FWI VALVES CLOSED. AT 2213 HOURS, A REACTOR TRIP WAS INITIATED DUE TO SOURCE RANGE HIGH FLUX. OPERATORS HAD NOT INSTALLED THE SOURCE RANGE INSTRUMENT POWER FUSES ON CHANNEL N32 DURING THE SHUTDOWN AS THE DETECTOR WAS OUT OF SERVICE DUE TO AN EXPIRED CALIBRATION. WHEN INTERMEDIATE RANGE FLUX DECREASED TO THE POINT WHERE THE SOURCE RANGES AUTOMATICALLY ENERGIZED, CHANNEL N32 INITIATED A TRIP SIGNAL DUE TO LACK OF INSTRUMENT POWER. ALL CONTROL RODS FULLY INSERTED IN RESPONSE TO

THE TRIP SIGNAL. THERE WERE NOT SAFETY IMPLICATIONS DUE TO THIS EVENT. ALL REQUIRED SAFETY FUNCTIONS ACTUATED AS DESIGNED.

[9] BEAVER VALLEY 1 DOCKET 50-334 LER 89-009
ENGINEERED SAFETY FEATURES ACTUATION DUE TO VENTILATION REALIGNMENT.
EVENT DATE: 090289 REPORT DATE: 100289 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215511) ON 9/02/89 AT 1500 HOURS, WITH THE UNIT IN HOT SHUTDOWN, A PLANT SHUTDOWN FOR REFUELING WAS IN PROGRESS. A SURVEILLANCE TEST INVOLVING A LEAK CHECK ON THE RESIDUAL HEAT REMOVAL (RHR) INLET MOTOR OPERATED VALVES WAS IN PROGRESS. DURING THIS TEST, ANY LEAKAGE PAST THE RHR INLET VALVES, IS DIRECTED TO THE REACTOR PLANT SAMPLE PANEL. AT 1802, A HIGH-HIGH ALARM WAS RECEIVED ON THE TRAIN A AUXILIARY BUILDING VENTILATION RADIATION MONITOR, RM-VS-102A. THIS ALARM CAUSED A REALIGNMENT OF THE SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM (SLCRS) DAMPERS TO THE MAIN FILTER BANKS. THIS IS REPORTABLE AS AN ENGINEERED SAFETY FEATURES ACTUATION. THE CAUSE FOR THE HIGH ALARM WAS DUE TO A LARGE VOLUME OF REACTOR COOLANT IN THE SAMPLE PANEL DUE TO LEAKAGE OF AN ISOLATION VALVE WHICH PREVENTED DEPRESSURIZATION OF THE RHR SYSTEM. ADDITIONALLY, THE PHYSICAL PROXIMITY OF THE SAMPLE PANEL EXHAUST CONNECTION TO THE PRIMARY AUXILIARY BUILDING VENTILATION SYSTEM AND THE PRIMARY AUXILIARY BUILDING RADIATION MONITOR ACCOUNTS FOR A VERY SMALL DILUTION OF THE SAMPLE PANEL EXHAUST. THE ISOLATION VALVE, IDENTIFIED AS LEAKING, WILL BE REPAIRED DURING THE REFUELING OUTAGE. THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC AS RESULT OF THIS EVENT. THE SLCRS ACTUATED AS DESIGNED TO DIRECT THE ACTIVITY TO THE MAIN FILTER BANKS FOR ABSORPTION/FILTERING PRIOR TO RELEASE TO THE PUBLIC.

[10] BEAVER VALLEY 1 DOCKET 50-334 LER 89-011
PRESSURIZER SURGE LINE RUPTURE RESTRAINTS OUTSIDE THE DESIGN BASIS.
EVENT DATE: 092089 REPORT DATE: 102089 NSSS: WE TYPE: PWR

(NSIC 215684) ON 9/20/89, WITH THE UNIT IN REFUELING (OPERATING MODE 6), DURING AN INSPECTION OF THE REACTOR COOLANT SYSTEM (RCS) PRESSURIZER (PZR) SURGE LINE, IN ACCORDANCE WITH INSPECTION AND ENFORCEMENT BULLETIN 88-01, ENGINEERING DETERMINED THAT THE RUPTURE RESTRAINT GAPS WERE OPENED TO GREATER THAN THE ORIGINAL DESIGN ANALYSIS. THE CAUSE FOR THIS EVENT WAS DUE TO AN ALIGNMENT PERFORMED DURING HOT FUNCTIONAL TESTING WHICH ALLOWED FREE THERMAL MOVEMENT OF THE SURGE LINE DUE TO THE PIPING MOVEMENT CAUSED BY PLANT HEATUPS AND COOLDOWNS. TO CORRECT THIS DEFICIENCY, A DESIGN CHANGE MODIFICATION, WHICH WILL READJUST THE RESTRAINT GAP SETTINGS WILL BE PERFORMED DURING THE SEVENTH REFUELING OUTAGE. ADDITIONALLY, ADMINISTRATIVE OPERATIONAL LIMITS HAVE BEEN PLACED ON THE TEMPERATURE DIFFERENTIAL BETWEEN THE PZR SURGE LINE AND THE RCS HOT LEG, TO MAINTAIN THIS DIFFERENTIAL LESS THAN 200 DEGREE FAHRENHEIT DURING THE FORMATION OR COLLAPSING OF THE STEAM BUBBLE. THERE WERE MINIMAL SAFETY IMPLICATIONS AS A RESULT OF THIS EVENT. IN THE EVENT OF A SURGE LINE RUPTURE, THE SURGE LINE RESTRAINTS MAY NOT HAVE CONTAINED THE SURGE LINE. NO OTHER PLANT RUPTURE RESTRAINTS WERE FOUND TO BE AFFECTED.

[11] BEAVER VALLEY 2 DOCKET 50-412 LER 89-025
AUXILIARY FEEDWATER INTRA-SYSTEM RECIRCULATION DUE TO UNANALYZED CONDITION.
EVENT DATE: 091089 REPORT DATE: 101089 NSSS: WE TYPE: PWR
VENDOR: ATWOOD & MORRILL CO., INC.
WALWORTH COMPANY

(NSIC 215579) ON 9/10/89 AT 1500 HOURS, OPERATIONS PERSONNEL IDENTIFIED HIGH TEMPERATURES ON THE "B" AND "C" STEAM GENERATOR (SGB AND SGC) AUXILIARY FEEDWATER (AFW) LINES. PYROMETER READINGS (250 DEGREES FAHRENHEIT (F) TO 328 F) INDICATED RECIRCULATION WITHIN THE SGB AND SGC AFW LINES, APPARENTLY THE RESULT OF AFW

CHECK VALVES NOT PROPERLY SEATED. OPERATIONS PERSONNEL INITIATED A PROCEDURE TO VERIFY AND CORRECT THIS CONDITION BY FLUSHING THE AFW LINE WITH AN AFW PUMP TO RESEAT THE CHECK VALVES. DUE TO THERMAL CONSIDERATIONS AND POTENTIAL WATERHAMMER, ENGINEERING SUPPORT WAS REQUESTED PRIOR TO PERFORMANCE OF THIS PROCEDURE. THE TRAIN "B" HYDRAULIC CONTROL VALVE IN THE SGB AFW LINE WAS CLOSED, AT 1640 HOURS, TO LIMIT THE BACKLEAKAGE. ENGINEERING RECOMMENDED THAT FLUSHING BE PERFORMED AFTER THE PIPING WAS BELOW 200 F. AT 2003 HOURS, AFTER THE PIPING TEMPERATURE WAS LESS THAN 200 F, THE "B" AFW LINE WAS FLUSHED FOR TWO MINUTES, REDUCING THE PIPING TEMPERATURE TO 85 F AND RESEATING THE CHECK VALVES. A PERIODIC TREND OF AFW PIPING TEMPERATURE WAS INITIATED TO VERIFY NO FURTHER LEAKAGE OCCURRED. THERE WERE NO SAFETY IMPLICATIONS AS A RESULT OF THIS EVENT. AN ENGINEERING EVALUATION SHOWED THAT THE ELEVATED TEMPERATURES WOULD NOT HAVE IMPACTED THE AFW SYSTEM'S ABILITY TO PERFORM ITS DESIGN FUNCTION.

[12] BRAIDWOOD 1 DOCKET 50-456 LER 89-008
AUXILIARY FEEDWATER PUMP SUCTION PRESSURE SWITCHES FOUND OUT OF CALIBRATION DUE TO FAILURE TO CONSIDER HEAD CORRECTION.
EVENT DATE: 090589 REPORT DATE: 100589 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BRAIDWOOD 2 (PWR)

(NSIC 215594) ON 8/29/89 BRAIDWOOD STATION WAS INFORMED THAT BYRON STATION WAS HEAD CORRECTING THE AUXILIARY FEEDWATER PUMP (AF) SUCTION PRESSURE TRANSMITTERS TO THE ELEVATION OF THE PROCESS TAP IN RESPONSE TO A SARGENT & LUNDY ENGINEERS (S&L) RECOMMENDATION. THESE TRANSMITTERS PROVIDE INPUT TO THE SWITCHES THAT SWITCH THE AF PUMP WATER SUPPLY FROM THE CONDENSATE STORAGE TANK (CST) TO THE ESSENTIAL SERVICE WATER SYSTEM AND THE SWITCHES THAT PROVIDE LOW PUMP SUCTION PRESSURE TRIP AND ALARM. ON 8/30/89 THIS RECALIBRATION WAS PERFORMED FOR BOTH AF PUMPS ON UNITS 1 AND 2 AT BRAIDWOOD. ON 9/5/89 AN S&L EVALUATION DETERMINED THAT WITH THE UNCORRECTED SETPOINTS THE ALLOWABLE VALUE SPECIFIED IN THE TECH SPEC WOULD HAVE BEEN EXCEEDED. ON 9/13/89 S&L RECOMMENDED NEW SUCTION PRESSURE SWITCH SETPOINTS. ON 9/14/89 THE SETPOINT CHANGES WERE MADE FOR BOTH UNIT 2 AF PUMPS. AN ADMINISTRATIVE MINIMUM LIMIT ON THE CST OF 70% WAS IMPOSED. THE CAUSE OF THIS EVENT WAS UNCLEAR DESIGN DOCUMENTS. A DETAILED ANALYSIS OF AF SETPOINTS IS BEING PERFORMED. A REVIEW OF OTHER INSTRUMENTS WITH AUTOMATIC ESF FUNCTIONS WILL BE CONDUCTED. THERE HAVE BEEN NO PREVIOUS OCCURRENCES.

[13] BRAIDWOOD 1 DOCKET 50-456 LER 89-010
REVERSE OPERATION OF HYDROGEN ANALYZER CONTAINMENT ISOLATION VALVE DUE TO A PRESERVICE WIRE LABELING ERROR.
EVENT DATE: 091589 REPORT DATE: 101389 NSSS: WE TYPE: PWR
VENDOR: VALCOR ENGINEERING CORP.

(NSIC 215656) AT 0400 ON 9/15/89, THE LEAKAGE RATE SURVEILLANCE (LLRT) FOR 1PS229B, OB HYDROGEN ANALYZER CONTAINMENT ISOLATION VALVE, WAS INITIATED. THE MEASURED LEAKRATE WAS LARGER WITH THE VALVE INDICATING CLOSED. THE CORRECT VALVE STEM TRAVEL COULD NOT BE MADE BY DIRECT OBSERVATION BECAUSE THE VALVE AND COIL ASSEMBLY ARE ENCAPSULATED. SEVERAL ADDITIONAL LLRTS WERE PERFORMED ON THE VALVE. EACH TIME THE RESULTS INDICATED REVERSE OPERATION, BUT WERE INCONCLUSIVE. THE WIRING WAS CHECKED AND FOUND TO BE CORRECT. THE VALVE WAS REMOVED AND BENCH TESTED. THE LEADS FROM THE ENCAPSULATED COIL WERE FOUND TO BE IMPROPERLY LABELED. THE VALVE ALSO DRIFTED TO MID-POSITION WHEN THE CLOSED COIL WAS DEENERGIZED. NO WORK ACTIVITIES WERE IDENTIFIED THAT WOULD HAVE REQUIRED RE-LABELING. THE LABELS ON THE LEADS WERE COMPARED TO A NEW COIL ASSEMBLY. THE NEW LABELS WERE SIMILAR BUT HAD A PLASTIC COATING WHICH 1PS229B DID NOT HAVE. THE FAILURE OF THE VALVE TO REMAIN IN THE CLOSED POSITION WHEN THE CLOSING COIL WAS DENERGIZED MADE DETECTION OF THE ERROR VIRTUALLY IMPOSSIBLE DURING NORMAL OPERATION. AN INVESTIGATION TO DETERMINE THE MODE OF FAILURE AND WHEN THE MIS-LABELING OCCURRED IS STILL IN PROGRESS. 1PS229B IS BEING REPLACED WITH A

DIFFERENT MODEL VALVE WHICH WILL BE TESTED IN ACCORDANCE WITH THE STATION MODIFICATION PROGRAM.

[14] BRAIDWOOD 1 DOCKET 50-456 LER 89-011
BORATION FLOWPATH VALVE INOPERABLE DUE TO PROGRAMMATIC DEFICIENCY.
EVENT DATE: 092089 REPORT DATE: 102089 NSSS: WE TYPE: PWR

(NSIC 215657) THE 1A DIESEL GENERATOR (DG) AND THE HIGH HEAD SAFETY INJECTION VALVE, 1SI8801B HAD BEEN TAKEN OUT OF SERVICE. THE ISI8801B HAS THE 1B DG AS AN EMERGENCY POWER SOURCE. AT 0224 ON 9/20/89 CORE ALTERATIONS FOR THE REFUELING OUTAGE WERE INITIATED. THESE ALTERATIONS CONTINUED UNTIL 1315. AT 1720 THE UNIT ONE BORON INJECTION FLOWPATH MONTHLY SURVEILLANCE WAS PERFORMED. THE ACCEPTANCE CRITERIA REQUIRES ONE OF THE REDUNDANT HIGH HEAD SAFETY INJECTION VALVES TO BE CAPABLE OF BEING POWERED FROM AN OPERABLE EMERGENCY POWER SOURCE. THE ISI8801B WAS OUT OF SERVICE. HIGH HEAD SAFETY INJECTION VALVE, 1SI8801A, WAS NOT CAPABLE OF BEING POWERED BY AN OPERABLE EMERGENCY POWER SOURCE BECAUSE THE 1A DG WAS OUT OF SERVICE. THE CAUSE OF THIS EVENT WAS A PROGRAMMATIC DEFICIENCY. THE RECENTLY IMPLEMENTED COMPUTER OUT OF SERVICE (OOS) PROGRAM WAS NOT PROPERLY STRUCTURED FOR OOS'S PREPARED IN ADVANCE FOR A MAJOR OUTAGE. THE COMPUTER WOULD AUTOMATICALLY PRINT THE INITIALS OF THE SUPERVISOR ON THE FORM. FOR THE REFUELING OUTAGE THE OOS FORMS WERE PRINTED IN ADVANCE. WHEN THE OOS FOR THE 1SI8801B WAS FORWARDED TO THE CONTROL ROOM, THE SCRE INITIALS WERE ALREADY COMPLETED. BASED ON THE COMPLETED FORM THE SCRE CONCLUDED THAT NO ADDITIONAL REVIEW WAS NECESSARY. A POLICY STATEMENT HAS BEEN ISSUED. THE COMPUTER OOS PROGRAM WILL BE REVISED TO PROVIDE FOR PRINTING AN OOS WITHOUT INITIALS ON THE FORM.

[15] BRAIDWOOD 2 DOCKET 50-457 LER 89-004
REACTOR TRIP AS A RESULT OF LIGHTNING INDUCED VOLTAGE TRANSIENT AFFECTING ROD CONTROL SYSTEM.
EVENT DATE: 090789 REPORT DATE: 100389 NSSS: WE TYPE: PWR

(NSIC 215593) AT APPROXIMATELY 2000 HOURS, 9/7/89 A SEVERE THUNDERSTORM WAS IN THE AREA OF BRAIDWOOD STATION. A VIDEO RECORDER HAD BEEN SET UP TO MONITOR THE EFFECTS OF ATMOSPHERIC EVENTS. FROM 2029 TO 2036 SIXTY-THREE LIGHTNING FLASHES WERE RECORDED BY THE CAMERA. FOUR OF THESE LIGHTNING STRIKES HIT STATION STRUCTURES. THE UNIT 2 AUX. BLDG. VENT STACK WAS STRUCK TWICE. THE BRAIDWOOD STATION SWITCHYARD WAS STRUCK. AT 2031:44 THE UNIT 2 CONTAINMENT WAS STRUCK. AT 2032 ALL TEN ROD CONTROL SYSTEM (RD) POWER CABINET OVERVOLTAGE PROTECTION DEVICES ACTUATED. THIS CAUSED THE STATIONARY GRIPPER COILS OF THE CONTROL RODS TO DEENERGIZE AND THE RODS DROPPED INTO THE CORE. THE RAPID FLUX DECREASE WAS SENSED BY THE NUCLEAR INSTRUMENTATION WHICH GENERATED A POWER RANGE FLUXRATE HIGH REACTOR TRIP. THE REACTOR TRIP BREAKERS OPENED, THE TURBINE TRIPPED, AND FEEDWATER ISOLATION OCCURRED. THE SHRINK EFFECT ON STEAM GENERATOR LEVEL INSTRUMENTATION RESULTED IN AN AUTO START OF THE AUXILIARY FEEDWATER PUMPS ON LOW WATER LEVEL. THE CAUSE OF THIS EVENT WAS A LIGHTNING INDUCED VOLTAGE TRANSIENT. THE RD OVERVOLTAGE PROTECTION DEVICES WERE RESET. THE RD SYSTEM WAS TESTED. A TIME DELAY HAS BEEN ADDED TO THE OVERVOLTAGE PROTECTION DEVICES. RECOMMENDATIONS ON ADDITIONAL CORRECTIVE MEASURES ARE BEING EVALUATED. THERE HAVE BEEN TWO PREVIOUS OCCURRENCES. PREVIOUS CORRECTIVE ACTIONS WERE NOT APPLICABLE.

[16] BROWNS FERRY 1 DOCKET 50-259 LER 89-014 REV 01
UPDATE ON UNPLANNED DIESEL GENERATOR STARTS, AN ENGINEERED SAFETY FEATURE ACTUATION CAUSED BY PERSONNEL ERROR, AND PROCEDURAL INADEQUACY.
EVENT DATE: 060689 REPORT DATE: 100289 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215343) ON 6/6/89, AT 0403 HOURS, DIESEL GENERATORS (DGS) 1B AND 1D AUTO-STARTED DURING THE PERFORMANCE OF 0-SI-4.9.A.3.A. THE DGS STARTED BECAUSE A UTILITY ELECTRICIAN INADVERTENTLY ALLOWED A WIRE HE WAS ATTEMPTING TO RELAND ON TERMINAL 13 OF THE COMMON ACCIDENT SIGNAL A-2 (CASA-2) RELAY TO CONTACT TERMINAL 11 WHICH WAS ENERGIZED. THIS ENERGIZED RELAY CASA-1 WHICH IN TURN ENERGIZED CASA-5 AND CASA-6 RESULTING IN THE AUTO-START OF THE DGS. THE DGS WERE IMMEDIATELY SHUTDOWN AND PLACED IN STANDBY READINESS. THIS EVENT WAS CAUSED BY PERSONNEL ERROR. CONTRIBUTING FACTORS TO THE EVENT WERE A CONGESTED WORK AREA, POOR LIGHTING, AND AN INADEQUATE PROCEDURE. SURVEILLANCE PROCEDURE 0-SI-4.9.A.3.A WAS IMMEDIATELY REVISED TO INCLUDE A PRECAUTIONARY NOTE TO INSULATE ADJACENT TERMINALS PRIOR TO LIFTING WIRES. DURING THE EVENT'S INVESTIGATION, IT WAS DETERMINED THAT THE WIRE LIFT WHICH RESULTED IN THE AUTO-START OF THE DGS WAS NOT NECESSARY TO MEET SURVEILLANCE REQUIREMENTS AND WAS DELETED FROM THE SURVEILLANCE INSTRUCTION (SI). THE PROCEDURE VERIFICATION REVIEW CHECKLIST CONTAINED IN SITE DIRECTOR STANDARD PRACTICE-7.4 REQUIRES THAT WIRE LIFTS BE MINIMIZED AND ARE THE OPTIMUM METHOD FOR TEMPORARY ALTERATIONS. THE ELECTRICIANS INVOLVED WERE COUNSELED TO FURTHER EMPHASIZE THE NEED TO EXERCISE CAUTION.

[17]   BROWNS FERRY 1   DOCKET 50-259   LER 89-020
CORE SPRAY SYSTEM MINIMUM FLOW BYPASS VALVES NOT QUALIFIED DUE TO DESIGN ERROR IN ORIGINAL ANALYSIS FOR TORUS HYDRODYNAMIC MOTIONS.
EVENT DATE: 090889   REPORT DATE: 100589   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
                      BROWNS FERRY 3 (BWR)
VENDOR: VELAN VALVE CORP.

(NSIC 215536) ON 9/8/89, ENGINEERING ANALYSIS REVEALED THAT THE VALVE YOKES FOR THE CORE SPRAY (CS) SYSTEM MINIMUM FLOW BYPASS VALVES FOR UNITS 1, 2, AND 3 WOULD EXCEED ALLOWABLE YIELD STRESS WHEN SUBJECTED TO LOADINGS FROM TORUS HYDRODYNAMIC MOTIONS. YIELDING OF THE VALVE YOKE COULD PREVENT THE CS MINIMUM FLOW BYPASS VALVES AND, THEREFORE, THE CS SYSTEM FROM PERFORMING ITS INTENDED SAFETY FUNCTION. THIS DISCREPANCY WAS IDENTIFIED DURING COMPONENT QUALIFICATION ANALYSIS WHICH WAS BEING PERFORMED FOR A DESIGN CHANGE. ROOT CAUSE OF THIS EVENT IS A DESIGN ERROR IN ORIGINAL ANALYSIS USED TO QUALIFY THE CS SYSTEM MINIMUM FLOW BYPASS VALVES FOR TORUS HYDRODYNAMIC MOTIONS. AS THE RESULT OF AN INADEQUATE ENGINEERING JUDGMENT, THE CS SYSTEM MINIMUM FLOW BYPASS VALVES WERE INCORRECTLY DETERMINED TO BE QUALIFIED FOR THE ADDITIONAL TORUS HYDRODYNAMIC LOADINGS. FOLLOWING THIS EVENT, A REVIEW OF TORUS ATTACHED PIPING ACTIVE VALVES WITH MOTOR OR AIR ACTUATORS REQUIRED FOR UNIT 2 OPERATION WAS PERFORMED TO IDENTIFY THIS TYPE VALVE. ADDITIONALLY, A VALVE IDENTICAL TO THE CS SYSTEM MINIMUM FLOW BYPASS VALVES WAS IDENTIFIED ON UNIT 1. EVALUATION OF THE UNIT 1 VALVE IS STILL IN PROGRESS. ADDITIONAL EVALUATION FOR ALL 3 UNITS IS UNDERWAY. APPROPRIATE CORRECTIVE ACTIONS WILL BE IMPLEMENTED. THE AFFECTED FLOW BYPASS VALVES WILL BE MODIFIED/REPLACED.

[18]   BROWNS FERRY 1   DOCKET 50-259   LER 89-026
FAILURE TO SAMPLE ALL SEVEN-DAY FUEL OIL STORAGE TANKS FOR THE DIESEL GENERATORS RESULTING IN A VIOLATION OF TECH SPECS.
EVENT DATE: 091289   REPORT DATE: 101189   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
                      BROWNS FERRY 3 (BWR)

(NSIC 215537) AS A RESULT OF A REVIEW OF THE DIESEL GENERATORS (DGS), ON SEPTEMBER 12, 1989, TVA DETERMINED THAT FUEL OIL QUALITY FOR ALL THE SEVEN-DAY STORAGE TANKS FOR THE DGS SHOULD BE SAMPLED ONCE A MONTH IN ORDER TO MEET THE REQUIREMENT OF TECHNICAL SPECIFICATION (TS) 4.9.A.1.E. BFN HAD HISTORICALLY INTERPRETED TS 4.9.A.1.E TO MEAN THE SEVEN-DAY STORAGE TANKS SHALL BE SAMPLED ONCE A MONTH ON A STAGGERED BASIS. BFN SUBSEQUENTLY RECONSIDERED THE ISSUE AND

WILL SAMPLE ALL SEVEN-DAY TANKS EACH MONTH. FAILURE TO SAMPLE ALL SEVEN-DAY FUEL OIL STORAGE TANKS FOR THE DGS HAS BEEN DETERMINED TO BE A VIOLATION OF TSS AND IS REPORTABLE IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(I). DURING THE EVENT, THE UNIT 2 REACTOR WAS IN THE COLD SHUTDOWN CONDITION WITH IRRADIATED FUEL IN THE REACTOR. UNITS 1 AND 3 WERE DEFUELED. THE CAUSE OF THE EVENT IS ATTRIBUTED TO AN INCORRECT TS INTERPRETATION AND THE FACT THAT THE TS SURVEILLANCE REQUIREMENTS ARE NOT SPECIFIC REGARDING THE FREQUENCY FOR SAMPLING EACH DIESELS FUEL OIL SUPPLY. BFN HAS SINCE TAKEN THE POSITION THAT SAMPLING ALL SEVEN-DAY TANKS ON A MONTHLY BASIS IS THE INTENT OF TS 4.9.A.1.E. THE SURVEILLANCE INSTRUCTION (SI) PROCEDURE WILL BE REVISED TO REFLECT SAMPLING AND ANALYZING THE FUEL OIL QUALITY OF ALL SEVEN-DAY TANKS ONCE A MONTH AND A CHANGE REQUEST WILL BE SUBMITTED TO THE TSS FOR DIESEL FUEL OIL SAMPLING FREQUENCY.

[19]   BROWNS FERRY 2   DOCKET 50-260   LER 89-008 REV 03
UPDATE ON ELECTRICAL FAULT ON TRANSFORMER CAUSES ENGINEERED SAFETY FEATURE ACTUATIONS.
EVENT DATE: 030989   REPORT DATE: 092989   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 1 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 215506) AT 1032 HOURS ON MARCH 9, 1989, AN ESF ACTUATION OCCURRED DUE TO AN ELECTRICAL FAULT ON THE UNIT STATION SERVICE TRANSFORMER (USST) 2B. THE FAULT LED TO THE LOSS OF SHUTDOWN BUS 2 (THE ALTERNATE FEED WAS TAGGED OUT FOR MAINTENANCE). AS A RESULT, 4KV SHUTDOWN BOARDS C AND D SENSED THE UNDERVOLTAGE CONDITION AND AUTOMATICALLY STARTED DIESEL GENERATORS C AND D. EMERGENCY EQUIPMENT COOLING WATER PUMPS CL AND D3 ALSO AUTO STARTED PER DESIGN DUE TO THE DIESEL GENERATOR STARTS. THE TRANSFORMER FAULT OCCURRED BECAUSE OF 1) INADEQUATE INSULATION ABOVE THE BUS JOINT 2) THE DESIGN OF THE BUS DUCT ALLOWED COLLECTION OF CONDENSATION, AND 3) VENDOR RECOMMENDED PREVENTIVE MAINTENANCE WAS NOT PERFORMED. DURING THE POST EVENT RESTORATION OF POWER, ADDITIONAL ESF ACTUATIONS OCCURRED WHEN REACTOR PROTECTION SYSTEMS CIRCUIT PROTECTORS TRIPPED ON UNITS 1 AND 2. AS IMMEDIATE CORRECTIVE ACTION, STABLE ELECTRICAL POWER WAS RESTORED AND SAFETY SYSTEMS WERE RETURNED TO NORMAL. PREVENTIVE MAINTENANCE PRACTICES WERE REVIEWED. DESIGN CHANGES TO THE BUS CONNECTION AND BUS DUCT WERE INITIATED ON THE 2B USST. ADDITIONALLY, COMPREHENSIVE CORRECTIVE ACTION PLANS WERE DEVELOPED TO INCORPORATE LESSONS LEARNED FROM THIS EVENT. LONG TERM CORRECTIVE ACTION INCLUDES INSPECTION AND TESTING OF OTHER TRANSFORMERS OF THIS TYPE. UNITS 1 AND 3 WERE DEFUELED AND UNIT 2 WAS IN COLD SHUTDOWN DURING THIS EVENT.

[20]   BROWNS FERRY 2   DOCKET 50-260   LER 89-025
REMOVAL OF FIRE HOSE COMPENSATORY MEASURE DUE TO PERSONNEL ERROR RESULTS IN A CONDITION PROHIBITED BY TECH SPECS.
EVENT DATE: 091989   REPORT DATE: 101989   NSSS: GE   TYPE: BWR

(NSIC 215693) ON SEPTEMBER 19, 1989 AT 0630, A QUALITY ASSURANCE EVALUATOR OBSERVED THAT THE TECHNICAL SPECIFICATION COMPENSATORY MEASURE WAS NOT IN PLACE FOR TWO INOPERABLE UNIT 2 HOSE STATIONS. FIRE PROTECTION WAS NOTIFIED AND A GATED WYE, ADDITIONAL FIRE HOSE, AND SIGN WERE INSTALLED AT OPERABLE FIRE HOSE STATION 2-26-281. DURING THIS EVENT, UNIT 2 WAS IN COLD SHUTDOWN WITH IRRADIATED FUEL IN THE REACTOR VESSEL. THE CAUSE OF THIS EVENT IS PERSONNEL ERROR. IT HAS BEEN DETERMINED THAT THE GATED WYE AND ADDITIONAL FIRE HOSE WERE REMOVED FROM FIRE HOSE STATION 2-26-281 ON AUGUST 28, 1989, COINCIDENT WITH THE REMOVAL OF COMPENSATORY MEASURES FOR OTHER FIRE HOSE STATIONS. THE SIGN AT FIRE HOSE STATION 2-26-281 INDICATING THE COMPENSATORY MEASURE REQUIRED FOR THE TWO INOPERABLE FIRE HOSE STATIONS WAS REMOVED BY UNKNOWN PERSONNEL. CONTRIBUTING CAUSES ARE THAT THERE ARE NO REQUIREMENTS FOR PERIODIC VERIFICATION OF FIRE PROTECTION COMPENSATORY MEASURES. FIRE PROTECTION PERSONNEL HAVE REVIEWED A DESCRIPTION OF THIS EVENT. FIRE PROTECTION PROCEDURES WILL BE REVISED TO REQUIRE A PERIODIC VERIFICATION OF THE COMPENSATORY MEASURE.

[21] BROWNS FERRY 3 DOCKET 50-296 LER 89-004
PERSONNEL ERROR RESULTS IN AN UNPLANNED START OF A DIESEL GENERATOR.
EVENT DATE: 090689 REPORT DATE: 100589 NSSS: GE TYPE: BWR

(NSIC 215553) ON SEPTEMBER 6, 1989 AT 0315 HOURS, WITH UNIT 3 IN A DEFUELED CONDITION, AN UNPLANNED START OF THE 3D DIESEL GENERATOR (DG) OCCURRED DURING THE PERFORMANCE OF POSTMAINTENANCE WORK INSTRUCTIONS. THE DG WAS IMMEDIATELY STOPPED, AND WORK WAS SUSPENDED. WORK CONTINUED AFTER THE CAUSE OF THE DG START WAS DETERMINED, AND THE 3D DG WAS SUBSEQUENTLY RETURNED TO AN OPERABLE CONDITION. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR IN THAT THE ELECTRICAL PLANNER DID NOT INCLUDE THE APPROPRIATE PREREQUISITES IN THE POSTMAINTENANCE WORK INSTRUCTION TO PREVENT AN UNPLANNED START OF THE DIESEL GENERATOR. THE DG IS PART OF THE ENGINEERED SAFETY FEATURES AND ITS UNPLANNED START IS REPORTABLE IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(IV). THE EVENT HAD NO SIGNIFICANT SAFETY CONCERN SINCE THE DG PERFORMED AS DESIGNED AND ITS OPERATION WOULD HAVE PROVIDED BACKUP POWER TO ESSENTIAL EQUIPMENT, IF NEEDED. THE ELECTRICAL PLANNER RESPONSIBLE FOR THE EVENT WAS COUNSELED. A DESCRIPTION OF THIS EVENT WAS DISCUSSED WITH ELECTRICAL PLANNERS, WHO WERE INSTRUCTED TO REVIEW APPROPRIATE DOCUMENTS TO IDENTIFY SYSTEM ACTUATIONS POSSIBLE DURING MAINTENANCE WORK AND TO REVIEW THE PREREQUISITES OF THE APPLICABLE PROCEDURE NECESSARY TO PREVENT SYSTEM ACTUATIONS.

[22] BRUNSWICK 1 DOCKET 50-325 LER 89-001 REV 02
UPDATE ON DEENERGIZATION OF UNITS 1 AND 2 COMMON EMERGENCY BUS E1 RESULTING IN UNIT 1 GROUPS 2, 3, AND 6 ISOLATIONS AND SUBSEQUENT FAILURE TO MEET TECH SPECS.
EVENT DATE: 011289 REPORT DATE: 102789 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)
VENDOR: ALLEN-BRADLEY CO.

(NSIC 215731) AT 1756 HOURS ON 1/12/89, POWER TO THE UNITS 1 AND 2 COMMON EMERGENCY 4160-VOLT BUS E1 WAS LOST WHEN THE EMERGENCY DIESEL GENERATOR (DG NO. 1) SUPPLYING THE BUS LOST CONTROL POWER. THE LOSS OF E1 CAUSED THE UNITS' COMMON CONTROL BUILDING EMERGENCY VENTILATION SYSTEM TO ACTUATE, AND UNIT 1 PRIMARY CONTAINMENT GROUPS 2, 3, AND 6 ISOLATIONS. DG NO. 1 WAS SUPPLYING E1 DURING ROUTINE MAINTENANCE ON THE BALANCE-OF-PLANT 4160-VOLT BUS 1D (NORMAL FEED TO 1E) WITH UNIT 1 IN A REFUEL/MAINTENANCE OUTAGE AND UNIT 2 AT 100%. AT 2203 HOURS, POWER TO E1 FROM BUS 1D WAS RESTORED. AFTER THE 31 LOSS, LIMITING CONDITIONS FOR OPERATION (LCO) WERE EXCEEDED FOR TECH SPECS 3.3.5.7 AND 3.3.5.9. THIS EVENT HAD MINIMAL IMPACT UPON PLANT SAFETY. DG NO. 1 "AUTO START" INDICATING LAMP SOCKET HAD SHORTED, CAUSING THE DG CONTROL POWER FUSES TO BLOW. THE FUSES WERE 15 AMP VERSUS REQUIRED 30 AMPS. TECH SPEC 3.3.5.7 WAS EXCEEDED DUE TO FAILURE TO RECOGNIZE THE LOSS OF FIRE DETECTION IN THE UNITS' AUGMENTED OFF-GAS BUILDING, AND TECH SPEC 3.3.5.9 WAS EXCEEDED DUE TO THE OPERATOR FAILING TO PROPERLY RESTORE PLANT EQUIPMENT. THE CORRECT FUSES AND THE SOCKET WERE REPLACED IN DG NO. 1, AND IT WAS RETURNED TO STANDBY READINESS. AFTER THE FIRE DETECTION WAS OPERABLE, IT WAS RECOGNIZED THAT TECH SPEC 3.3.5.7 HAD BEEN EXCEEDED.

[23] BRUNSWICK 1 DOCKET 50-325 LER 89-019
FAILURE OF THE SERVICE WATER SYSTEM TO MEET DESIGN REQUIREMENTS.
EVENT DATE: 091489 REPORT DATE: 101389 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215563) AS A RESULT OF SERVICE WATER (SW) SYSTEM DESIGN CONCERNS RAISED BY THE NRC DET INSPECTION CONDUCTED APRIL 10 THROUGH MAY 5, 1989, IT WAS DETERMINED ON 9/14/89 THAT, SINCE INITIAL OPERATION OF THE PLANT, THE SW SYSTEM MAY NOT HAVE MET ITS DESIGN REQUIREMENTS UNDER CERTAIN WORST CASE CONDITIONS. THE CONCERNS FELL INTO THREE MAJOR CATEGORIES AND APPLIED TO BOTH UNIT 1 AND UNIT 2. ROOT CAUSE OF THE EVENT WAS DETERMINED TO BE PRIMARILY A RESULT OF INADEQUATE INITIAL SYSTEM AND COMPONENT DESIGN. ENGINEERING EVALUATIONS, SYSTEM AND COMPONENT

TESTING, INTERIM OPERATING RESTRICTIONS AND SYSTEM MODIFICATIONS WERE PERFORMED TO ENSURE CONTINUED OPERABILITY OF THE SYSTEM. AS A RESULT OF MODIFICATIONS, TESTING AND INTERIM OPERATING RESTRICTIONS, THE SW SYSTEM IS CURRENTLY OPERABLE AND CAPABLE OF PERFORMING ITS INTENDED DESIGN FUNCTION. CONTINUING CORRECTIVE MODIFICATIONS, ASSESSMENTS, AND EVALUATIONS WILL BE COMPLETED BY THE END OF THE UPCOMING UNIT 1 1990 REFUEL OUTAGE. THIS EVENT IS CONSIDERED TO HAVE POSSIBLY HAD A MAJOR SAFETY IMPACT.

[24] BRUNSWICK 2 DOCKET 50-324 LER 89-008 REV 01
UPDATE ON AUTO INITIATION WITHOUT INJECTION OF LOW PRESSURE COOLANT INJECTION CORE SPRAY AND RESIDUAL HEAT REMOVAL PUMPS DUE TO LOCA SIGNAL DURING SURVEILLANCE TESTING.
EVENT DATE: 060589 REPORT DATE: 102589 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 1 (BWR)
VENDOR: AGASTAT RELAY CO.
ROSEMOUNT, INC.

(NSIC 215749) AT 1352 HOURS ON 6/5/89, AN UNPLANNED UNIT 2 LOCA INITIATION SIGNAL OCCURRED DURING ROUTINE MAINTENANCE SURVEILLANCE TESTING OF EMERGENCY CORE COOLING SYSTEM (ECCS) DIVISION 11 REACTOR LEVEL (B21) TRIP UNITS B21-LTS-N031B-3 AND N031B-4. THIS RESULTED IN AUTO STARTING (WITHOUT REACTOR INJECTION) OF THE A AND B CORE SPRAY AND A, B, C, AND D RESIDUAL HEAT REMOVAL (RHR) LOW PRESSURE COOLANT INJECTION PUMPS AND AUTO STARTING OF UNITS 1 AND 2 EMERGENCY DIESEL GENERATORS NOS. 1-4. UNIT 2 REMAINED AT 100% POWER. THE LOCA SIGNAL WAS DETERMINED TO BE INVALID, OPERATING PARAMETERS WERE VERIFIED WITHIN NORMAL RANGE, AND AFFECTED PLANT SYSTEMS WERE RETURNED TO NORMAL. TECHNICAL SPECIFICATION 3.0.3 WAS ENTERED DUE TO SECURING THE LPCI PUMPS WITH THE LOCA SIGNAL SEALED IN, AND WAS EXITED AT 1425 HOURS. THIS EVENT HAD MINIMAL EFFECT UPON PLANT SAFETY. THE LOCA SIGNAL RESULTED FROM A SPURIOUS TRIP OF REDUNDANT DIVISION 2 ECCS TRIP UNIT N031D-4. THE TRIP UNIT WAS REPLACED AND THE TRIP LOGIC WAS RETURNED TO NORMAL. THE RESULTS OF BENCH TESTING THE REMOVED TRIP UNIT TO DETERMINE THE INVOLVED FAILURE MODE WERE INCONCLUSIVE. FUTURE ACTION WILL BE BASED UPON EFFORTS BY THE VENDOR AND THE INDUSTRY WHICH ARE BEING FOLLOWED BY THE NRC FROM A GENERIC STANDPOINT.

[25] BRUNSWICK 2 DOCKET 50-324 LER 89-012
MOTOR FAILURE OF REACTOR PROTECTION SYSTEM (RPS) MOTOR GENERATOR SET 2B RESULTING IN B LOGIC AUTO-SCRAM SIGNAL AND GROUPS 2, 3, AND 6 ISOLATIONS.
EVENT DATE: 083089 REPORT DATE: 092989 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO (SAN JOSE MOTOR PLANT)

(NSIC 215364) AT 2106 HOURS ON 8/30/89, AN EMERGENCY (E) BUS E8 "SUB (SUBSTATION) E8 XFMR (TRANSFORMER) TEMP (TEMPERATURE) HIGH-480V (VOLT) GROUND" ALARM ANNUNCIATION OCCURRED, AND APPROXIMATELY 45-60 SECONDS LATER, THE UNIT 2 DIVISION II (B LOGIC) REACTOR PROTECTION SYSTEM (RPS) BUS DEENERGIZED, COINCIDENT WITH CLEARING OF THE SUBJECT ANNUNCIATION. PER DESIGN, AN AUTO-SCRAM SIGNAL ON RPS DIVISION II AND PRIMARY CONTAINMENT GROUPS 1 (B LOGIC), 2, 3, 6, AND 8 (SUBJECT ISOLATION VALVE ALREADY CLOSED), ISOLATION SIGNALS OCCURRED. THE REACTOR BUILDING HEATING, VENTILATING, AIR CONDITIONING SYSTEM AUTO-ISOLATED AND THE STANDBY GAS TREATMENT SYSTEM AUTO-STARTED. THE CONTROL OPERATOR BECAME AWARE OF THIS EVENT THROUGH CONTROL ROOM INDICATION AND ALARM ANNUNCIATION. AN AUXILIARY OPERATOR DETERMINED THE RPS BUS MOTOR GENERATOR (MG) SET HAD TRIPPED DUE TO FAILURE OF THE MG SET MOTOR. THE ALTERNATE POWER SUPPLY WAS ALIGNED TO THE BUS AND AFFECTED SYSTEMS WERE RETURNED TO NORMAL BY 2200 HOURS. DISASSEMBLY OF THE MOTOR REVEALED FAILURE OF THE WINDINGS; HOWEVER, A ROOT CAUSE FOR THE FAILURE WAS NOT DETERMINED. THE MOTOR WAS REBUILT, REINSTALLED, AND AT 0817 HOURS ON 9/3/89, THE MG SET WAS ALIGNED TO POWER THE RPS BUS. NO FARTHER ACTION IS PLANNED.

[26] BRUNSWICK 2 DOCKET 50-324 LER 89-013
FAILURE OF THE HPCI AUXILIARY OIL PUMP SEAL.
EVENT DATE: 090989 REPORT DATE: 100689 NSSS: GE TYPE: BWR
VENDOR: TUTHILL PUMP COMPANY

(NSIC 215560) WHILE PERFORMING OVERSPEED TESTING OF THE HPCI TURBINE DURING PLANT SHUTDOWN FOR THE PENDING REFUEL OUTAGE FOR UNIT 2, THE HPCI TURBINE AUXILIARY OIL PUMP DEVELOPED A SHAFT SEAL LEAK OF APPROXIMATELY 1 GPM THE TURBINE WAS SECURED AND EVENT INVESTIGATION BEGUN. REVIEW OF THE EVENT WITH THE PUMP VENDOR HAS IDENTIFIED THREE POTENTIAL FAILURE MODES: OVERPRESSURIZATION OF THE TURBINE DURING THE OVERSPEED TESTING, INCOMPATIBILITY OF THE SEAL MATERIAL WITH THE TURBINE LUBE OIL AND NORMAL WEAR A FAILURE ANALYSIS OF THE SEAL IS BEING PERFORMED FINAL RESULTS OF THE ANALYSIS, ALONG WITH ADDITIONAL TESTING OF THE SYSTEM DURING OVERSPEED TESTING DURING STARTUP FOLLOWING COMPLETION OF THE OUTAGE WILL BE DESCRIBED IN A SUPPLEMENT TO THIS LER. THE SUPPLEMENT WILL BE ISSUED WITHIN 30 DAYS FOLLOWING THE END OF THE OUTAGE AND SUBSEQUENT OVERSPEED TESTING. OUTAGE COMPLETION IS CURRENTLY SCHEDULED FOR 2/09/90.

[27] BRUNSWICK 2 DOCKET 50-324 LER 89-014
FAILURES OF DRYWELL HEAD OUTER SEAL, MAIN STEAM ISOLATION VALVES B21-F028C, F028D, AND MAIN STEAM DRAIN ISOLATION VALVE(S) B21-F016 AND/OR F019 DURING LOCAL LEAK RATE TESTING.
EVENT DATE: 091089 REPORT DATE: 100589 NSSS: GE TYPE: BWR
VENDOR: ANCHOR VALVE CO.
ANCHOR/DARLING VALVE CO.

(NSIC 215561) DURING THE UNIT 2 1989-1990 REFUEL/MAINTENANCE OUTAGE, PRIMARY CONTAINMENT (PC) LOCAL LEAK RATE TESTING (LLRT), INCLUDING TYPES B AND C TESTING, AND MAIN STEAM ISOLATION VALVE TESTING REVEALED LEAKAGE WHICH DEGRADED THE PC SAFETY BARRIER. IDENTIFIED FAILURES, WITH DATES INCLUDE: THE OUTER SEAL OF THE DRYWELL HEAD FLANGE DOUBLE O-RING SEAL ON 9/10/89, MAIN STEAM DRAIN LINE (INBOARD AND/OR OUTBOARD VALVE) ON 9/11/89, MAIN STEAM 1INES "C" (OUTBOARD VALVE) AND "D" (OUTBOARD VALVE) ON 9/12 AND 9/13/89, AND THE "A" AND "B" FEEDWATER LINES (INBOARD VALVES) ON 9/20 AND 9/23/89. BASED UPON THE MAXIMUM PATHWAY ANALYSIS METHOD FOR ANALYZING CONTAINMENT LEAKAGE, A CALCULATED PC LEAKAGE RATE OF <0.60LA, REFERENCE TECHNICAL SPECIFICATION (T/S) 3.6 1.2B, COULD NOT BE ACHIEVED. IN ADDITION, A LEAKAGE RATE OF </ = 11.5 STANDARD CUBIC FEET PER HOUR, REFERENCE T/S 3.5.1.2C, COULD NOT BE ACHIEVED. THE SUBJECT LLRT-IDENTIFIED PROBLEMS ARE UNDER INVESTIGATION FOR ROOT CAUSE DETERMINATION AND CORRECTIVE ACTION. LLRT IS CONTINUING WITH A SUPPLEMENT TO THIS REPORT APPROPRIATELY ADDRESSING THESE PROBLEMS AND ANY OTHER LLRT-REPORTABLE PROBLEMS TO BE SUBMITTED BY 3/30/90. PRIOR LLRT-IDENTIFIED FAILURES HAVE BEEN REPORTED IN LERS 1-85-016, 1-87-005, 1-88-025, 2.84-001, 2.86-005, AND 2-88-002.

[28] BRUNSWICK 2 DOCKET 50-324 LER 89-015
PCIS/SDC SUCTION VALVE, 2-E11-F008, ISOLATED AFTER ACTION TAKEN TO PRECLUDE THE ISOLATION.
EVENT DATE: 091189 REPORT DATE: 101189 NSSS: GE TYPE: BWR

(NSIC 215562) ON SEPTEMBER 11, 1989, AT 0321, THE UNIT 2, OUTBOARD SUCTION VALVE FOR THE SHUTDOWN COOLING (SDC) SYSTEM, 2-EIL-F008, AUTOMATICALLY ISOLATED DURING A MAINTENANCE SURVEILLANCE TEST (MST). THE ISOLATION OCCURRED BECAUSE ACTION TO PREVENT IT WAS TAKEN ON THE WRONG SUCTION VALVE (I E., INBOARD VALVE, 2-E11-F009). THIS ERROR WAS THE RESULT OF AN INADEQUATE REVIEW, BECAUSE OF A PRECONCEIVED IDEA THAT 2-E11-F009 WAS THE VALVE INVOLVED, ON THE PART OF THE SHIFT TEST DIRECTOR, A SENIOR REACTOR OPERATOR (SRO). THIS IDEA RESULTED FROM INACCURATE INFORMATION RECEIVED FROM HIS TURNOVER. UNIT 2 WAS IN COLD SHUTDOWN IN THE THIRD DAY OF THE SCHEDULED RECIRCULATION PIPE REPLACEMENT/REFUEL OUTAGE. REACTOR COOLANT TEMPERATURE WAS APPROXIMATELY 125 DEGREES FAHRENHEIT (F) AND

INCREASED APPROXIMATELY 14 DEGREES. THE CORE SPRAY SYSTEM WAS OPERABLE IN STANDBY READINESS AND THE RHR SYSTEM WAS OPERABLE WITH B LOOP LINED UP FOR LOW PRESSURE COOLANT INJECTION AND A LOOP IN SDC. THE INVOLVED SRO HAS BEEN COUNSELLED AND APPROPRIATE PERSONNEL WILL RECEIVE TRAINING ON THIS EVENT. APPROPRIATE PROCEDURES WILL BE REVISED. THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL AS THE LOSS OF SDC WAS PART OF THE MST AND THE CONDENSER WAS AVAILABLE AS A BACKUP MEANS OF DECAY HEAT REMOVAL. PAST SIMILAR EVENTS INCLUDE LERS 1.88-01 AND 29; 1.89.04 AND 15.

[29] BRUNSWICK 2 DOCKET 50-324 LER 89-016
UNEXPECTED ISOLATION OF 2-G31-F004 VALVE DURING SP-89-51 DUE TO INADEQUATE CLEARANCE REVIEW.
EVENT DATE: 092089 REPORT DATE: 102089 NSSS: GE TYPE: BWR

(NSIC 215730) ON 9/20/89, AT 1317, THE UNIT 2 REACTOR WATER CLEANUP (RWCU) OUTBOARD ISOLATION VALVE, 2-G31-F004, UNEXPECTEDLY ISOLATED WHEN POWER WAS INTERRUPTED TO ITS ISOLATION LOGIC. A TEST WAS BEING PERFORMED TO VERIFY THAT DIESEL GENERATOR (DG) 3 WOULD ENERGIZE ITS ASSOCIATED EMERGENCY BUS WITHIN 10 SECONDS. IT REQUIRED THAT THE RWCU SYSTEM BE REMOVED FROM SERVICE. THE DECISION WAS MADE TO LEAVE RWCU IN SERVICE AND A CLEARANCE WAS PLACED ON THE DIVISION I RWCU INBOARD ISOLATION VALVE, 2-G31-F001, TO PREVENT IT FROM CLOSING (DG-3 IS A DIVISION I POWER SUPPLY). THE OPERATIONS STAFF FAILED TO RECOGNIZE THAT THE NONREGENERATIVE HEAT EXCHANGER OUTLET TEMPERATURE HIGH ISOLATION SIGNAL IS POWERED FROM A DIVISION I POWER SUPPLY AND CAUSES THE 2-G31-F004 TO ISOLATE. THE CAUSE OF THIS EVENT WAS FAILURE TO PERFORM AN ADEQUATE REVIEW PRIOR TO CHANGING THE PROCEDURAL STEP AND ESTABLISHING THE CLEARANCE. AT THE TIME OF THIS EVENT, UNIT 2 WAS IN COLD SHUTDOWN IN A SCHEDULED OUTAGE. APPROPRIATE PERSONNEL WILL BE COUNSELED AND RECEIVE TRAINING ON THIS EVENT. THIS EVENT HAD NO SAFETY SIGNIFICANCE AS THE PRIMARY CONTAINMENT ISOLATION SYSTEM FUNCTION OF THE RWCU VALVES IS ONLY REQUIRED DURING MODES 1, 2, AND 3 AND THE INVOLVED ISOLATION WAS NOT THE RESULT OF AN ENGINEERED SAFETY FEATURE ACTUATION. SIMILAR EVENTS INCLUDE 1-89-15, 04, 1-88-29, 01 AND 2-89-15.

[30] CALLAWAY 1 DOCKET 50-483 LER 89-010 REV 01
UPDATE ON ENGINEERED SAFETY FEATURE ACTUATIONS DUE TO A FAILED POWER SUPPLY AND AN AWF ACTUATION WITH SWAPOVER TO ESSENTIAL SERVICE WATER DUE TO IMPROPER OPERATOR ACTION.
EVENT DATE: 090589 REPORT DATE: 100389 NSSS: WE TYPE: PWR
VENDOR: CONSOLIDATED CONTROLS CORP.

(NSIC 215584) ON 9/5/89 AT 1531 CDT, ENGINEERED SAFETY FEATURES (ESF) CONTAINMENT PURGE ISOLATION, CONTROL ROOM VENTILATION ISOLATION AND FUEL BUILDING VENTILATION ISOLATION ACTUATIONS WERE RECEIVED WHEN A 15VDC POWER SUPPLY FAILED IN THE ESF CABINET FOR 'B' TRAIN. AT 1939, WHILE ATTEMPTING TO RESTORE THE ESF CABINET TO SERVICE FOLLOWING THE POWER SUPPLY REPLACEMENT, A 'B' TRAIN AUXILIARY FEEDWATER ACTUATION (AFAS) WITH SWAPOVER TO ESSENTIAL SERVICE WATER (ESW) WAS RECEIVED. THE PLANT WAS IN MODE 1 - POWER OPERATION AT 100% POWER. THE SECOND ESF ACTUATIONS WERE CAUSED BY IMPROPER OPERATOR ACTION. THE RESTORATION OF ESF PROCEDURE REQUIRES TRIPPED LOGIC TO BE RESET PRIOR TO PLACING RELAY POWER SUPPLY TO ON. WHEN THE LICENSED REACTOR OPERATOR WAS UNABLE TO RESET THE TRIPPED LOGIC HE MISINTERPRETED THE PROCEDURE AND TURNED ON THE POWER SUPPLY THUS ACTUATING THE AFAS AND VALVE SWAPOVER TO ESW. PLANT PROCEDURES REQUIRED A PLANT SHUTDOWN TO MODE 2 - START UP TO RESTORE STEAM GENERATOR CHEMISTRY. MODE 1 WAS RESUMED AT 2304 ON 9/7/89. THE FAILED POWER SUPPLY WAS REPLACED. THE IMPORTANCE OF CREW COMMUNICATION AND CONSERVATIVE JUDGEMENT WAS DISCUSSED WITH THOSE INVOLVED AND WILL BE COVERED IN TRAINING. THE FAILURE OF THE TRIPPED LOGIC TO RESET WAS EVALUATED. A REVIEW FOR SIMILAR INDUSTRY EXPERIENCE WILL BE PERFORMED.

[31] CALVERT CLIFFS 1 DOCKET 50-317 LER 89-012 REV 01
UPDATE ON SWITCHGEAR ROOM HALON SYSTEM INOPERABLE DUE TO LACK OF PROCEDURE FOR DISABLING MASTER SOLENOIDS RESULTING IN CONDITIONS PROHIBITED BY TECH SPECS.
EVENT DATE: 072089 REPORT DATE: 102789 NSSS: CE TYPE: PWR

(NSIC 215748) AT APPROXIMATELY 0200 HOURS ON JULY 20, 1989, WITH UNIT 1 SHUTDOWN, A TECHNICIAN DISCOVERED A MASTER SOLENOID TO THE SWITCHGEAR ROOM HALON SYSTEM TO BE DISCONNECTED. UPON DISCOVERY, THE HALON SYSTEM WAS IMMEDIATELY DECLARED INOPERABLE AND AN HOURLY FIRE WATCH WAS ESTABLISHED. THE SOLENOID WAS RECONNECTED AND THE SOLENOID WAS VERIFIED OPERABLE BY A FUNCTIONAL TEST. FURTHER INVESTIGATION INDICATES THAT THE SOLENOID WAS LAST TAKEN OUT-OF-SERVICE ON JUNE 29, 1989 AND THEN INADVERTENTLY LEFT DISCONNECTED. THE ROOT CAUSE OF THIS EVENT IS PERSONNEL ERROR RESULTING FROM THE LACK OF A WRITTEN PROCEDURE FOR PERFORMING THIS TASK. CONTRIBUTING CAUSES INCLUDE AN INADEQUATE PROCEDURE FOR ADDRESSING FIRE SYSTEM IMPAIRMENTS AND THE FAILURE TO APPLY TEMPORARY MODIFICATIONS AND SAFETY TAGGING PROCEDURES TO THIS TASK. CORRECTIVE ACTIONS INCLUDE: REVISING A PROCEDURE TO APPLY TEMPORARY MODIFICATIONS AND SAFETY TAGGING TO FIRE SYSTEMS; REVISING A SURVEILLANCE TEST PROCEDURE FOR VERIFYING PLACEMENT OF SOLENOIDS; INSTALLING IDENTIFICATION TAGS ON SOLENOIDS; INSTALLING WARNING SIGNS ON HALON SYSTEMS; CONDUCTING A QUALITY ASSURANCE SURVEILLANCE ON APPLICABILITY AND WORKING KNOWLEDGE OF PROCEDURES; ESTABLISHING A WRITTEN PROCEDURE FOR DISABLING SOLENOIDS; AND CONDUCTING TRAINING.

[32] CALVERT CLIFFS 1 DOCKET 50-317 LER 89-014
SALTWATER HEADER NOT SEISMICALLY QUALIFIED DUE TO SPOOL TACK WELDS.
EVENT DATE: 072389 REPORT DATE: 100389 NSSS: CE TYPE: PWR

(NSIC 215510) ON JULY 23, 1989, WHILE UNIT 1 WAS IN COLD SHUTDOWN, IT WAS DETERMINED THAT THE NO. 12 SALTWATER HEADER WAS NOT CAPABLE OF WITHSTANDING A SEISMIC EVENT INTACT. IF THE NO. 12 SALTWATER HEADER FAILED, IT WOULD HAVE RESULTED IN THE FAILURE OF THE NO. 12 SERVICE WATER HEAT EXCHANGER AND THE NO. 12 COMPONENT COOLING WATER HEAT EXCHANGER. THE FAILURE WAS DETERMINED TO BE CAUSED BY INADEQUATE WELDING OF THE BLIND SPOOL PIECES IN THE PIPE. A VOLUNTARY LER IS BEING SUBMITTED DUE TO THE ON-GOING INVESTIGATIONS INTO ADDITIONAL SYSTEMS. WALKDOWNS ARE CONTINUING ON THE BLIND SPOOL PIECES IN THE SALTWATER SYSTEM, SERVICE WATER SYSTEM, AND THE COMPONENT COOLING WATER SYSTEM TO DETERMINE IF THEY HAVE BEEN ADEQUATELY WELDED. THE SPOOL PIECES IN THE SALT WATER SYSTEM HAVE BEEN REPAIRED WITH CONTINUOUS WELDS. ANALYSIS SHOWS THAT THE PIPING WILL NOW REMAIN INTACT DURING A SEISMIC EVENT.

[33] CALVERT CLIFFS 1 DOCKET 50-317 LER 89-017
INCORRECT SURVEILLANCE TEST CRITERIA.
EVENT DATE: 090789 REPORT DATE: 100989 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 2 (PWR)

(NSIC 215558) ON SEPTEMBER 7, 1989, THE SPENT FUEL POOL VENTILATION SYSTEM ENGINEER DETERMINED THAT A DISCREPANCY IN THE ACCEPTANCE CRITERIA OF SURVEILLANCE TEST PROCEDURE M.452.0 RESULTED IN A FAILURE TO FULLY COMPLY WITH THE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.9 12. ACTION A. THE NUMBER 11 SPENT FUEL EXHAUST FAN WAS DECLARED OPERABLE AFTER MEETING THE ACCEPTANCE CRITERIA OF M.542.0, YET IT DID NOT SATISFY THE MINIMUM TECHNICAL SPECIFICATION FLOW REQUIREMENT. THE ROOT CAUSE OF THE PROCEDURE ERROR CAN NOT BE DETERMINED. EXISTING RECORDS DID NOT INDICATE WHY THE ACCEPTANCE CRITERIA WAS CHANGED, HOWEVER IT APPEARS TO HAVE BEEN A TYPOGRAPHICAL ERROR. THE LACK OF SUFFICIENT SUBSEQUENT PROCEDURE REVIEWS IS A SECONDARY CAUSE FOR THIS EVENT. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDE: 1. A MAINTENANCE REQUEST WAS INITIATED TO DETERMINE THE CAUSE AND CORRECTIVE ACTIONS FOR THE LOW FLOW CONDITION ON THE NUMBER 11 FAN. 2. A PROCEDURE CHANGE WAS MADE TO CORRECT THE MINIMUM FLOW ACCEPTANCE CRITERIA IN STP M.542.0. 3. A DETAILED REVIEW OF TEST PROCEDURES

USED TO SATISFY TECHNICAL SPECIFICATION REQUIREMENTS WILL BE PERFORMED. 4. AN UPGRADED STP PROGRAM HAS BEEN IMPLEMENTED. 5. THE QUALITY ASSURANCE TECHNICAL SPECIFICATION AUDIT PROCESS HAS BEEN IMPROVED.

[34] CALVERT CLIFFS 1 DOCKET 50-317 LER 89-016
RTDS NOT ENVIRONMENTALLY QUALIFIED DUE TO UNSEALED HOUSING.
EVENT DATE: 090889 REPORT DATE: 101089 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 2 (PWR)

(NSIC 215557) ON SEPTEMBER 8, 1989, WHILE UNIT 1 WAS IN COLD SHUTDOWN AND UNIT 2 WAS DEFUELED, IT WAS DETERMINED THAT THE AS-FOUND CONDITION OF THE RESISTANCE TEMPERATURE DETECTORS (RTDS) DID NOT MATCH THE TESTED CONFIGURATION. THIS CONDITION INVALIDATES THE ENVIRONMENTAL QUALIFICATION OF THE RTDS. THE RTDS PROVIDE INPUT TO THE POST-ACCIDENT MONITORING INSTRUMENTATION AND ARE GOVERNED BY TECH SPEC 3.3.3 6 BECAUSE THE CONDITION EXISTED DURING MODE 1 OPERATION, THE ASSUMED INABILITY OF THESE INSTRUMENTS TO FUNCTION UNDER POST-ACCIDENT CONDITIONS CONSTITUTES A VIOLATION OF THE TECHNICAL SPECIFICATIONS. A SUPPLEMENTAL LER WILL BE SUBMITTED TO DISCUSS THE ROOT CAUSE OF THE EVENT AND ANY ADDITIONAL CORRECTIVE ACTIONS. PRIOR TO RESTARTING EITHER UNIT, THE RTDS WILL BE SEALED IN ACCORDANCE WITH ENVIRONMENTAL QUALIFICATION REQUIREMENTS.

[35] CALVERT CLIFFS 2 DOCKET 50-318 LER 89-007 REV 01
UPDATE ON EVIDENCE LEAKAGE FROM UNIT 2 PRESSURIZER HEATER PENETRATIONS DUE TO INTERGRANULAR STRESS CORROSION CRACKING CAUSED BY RESIDUAL FABRICATION STRESS.
EVENT DATE: 050589 REPORT DATE: 110389 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 1 (PWR)
VENDOR: COMBUSTION ENGINEERING, INC.

(NSIC 215777) AT 0820 HOURS MAY 5, 1989, AN IN-SERVICE INSPECTION OF THE UNIT 2 PRESSURIZER DISCOVERED EVIDENCE OF REACTOR COOLANT LEAKAGE FROM AN UNKNOWN NUMBER OF THE 120 PRESSURIZER VESSEL HEATER PENETRATIONS AND ONE PRESSURIZER VESSEL PRESSURE/LEVEL PENETRATION. UNIT 2 WAS IN A REFUELING OUTAGE (MODE 6) AT THE TIME OF THE DISCOVERY. AT 0430 HOURS ON MAY 6, 1989, UNIT 1 WAS SHUTDOWN FROM 100% POWER (MODE 1) TO ALLOW INSPECTION OF ITS PRESSURIZER. NO SIGNS OR EVIDENCE OF LEAKAGE WERE FOUND ON THE UNIT 1 PRESSURIZER HEATER PENETRATIONS OR PRESSURE/LEVEL PENETRATIONS. ADDITIONAL INSPECTIONS, INCLUDING DYE PENETRANT AND EDDY CURRENT TESTS, OF 28 UNIT 2 AND 12 UNIT 1 HEATER SLEEVES WERE CONDUCTED. THREE SLEEVES FROM UNIT 2 WERE DESTRUCTIVELY EXAMINED. THE CAUSE OF LEAKAGE WAS INTERGRANULAR STRESS CORROSION CRACKING OF INCONEL 600. ALL CRACKS WERE AXIAL AND DETERMINED TO HAVE MINIMAL SAFETY SIGNIFICANCE. REAMING AND REPAIR OPERATIONS ASSOCIATED WITH FABRICATING THE UNIT 2 PRESSURIZER APPEAR TO HAVE CONTRIBUTED TO THE CAUSE. ALL UNIT 2 PENETRATIONS USING J-WELDS AND INCONEL 600 WERE VISUALLY INSPECTED. ALL UNIT 1 PRESSURIZER PENETRATIONS WERE VISUALLY INSPECTED. NO UNIT 1 PENETRATIONS SHOWED ANY SIGNS OF LEAKAGE. CORRECTIVE ACTIONS WILL BE DISCUSSED IN A SUPPLEMENTAL LER.

[36] CATAWBA 1 DOCKET 50-413 LER 89-026
UNEXPECTED HYDROGEN SKIMMER FAN BREAKER TRIP DUE TO A DEFECTIVE WESTINGHOUSE HFB BREAKER.
EVENT DATE: 082389 REPORT DATE: 100489 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215591) ON 11/11/88, THE NRC ISSUED BULLETIN 88-10. IN RESPONSE TO THIS NRC BULLETIN, 25 CIRCUIT BREAKERS WERE IDENTIFIED AT CATAWBA TO BE NON-TRACEABLE AND WERE REPLACED WITH EITHER A TESTED CIRCUIT BREAKER OR A TRACEABLE CIRCUIT BREAKER. ON 6/21/89, WITH UNIT 2 IN MODE 1, POWER OPERATION, UNDER AN INVESTIGATE AND REPAIR (I/R) WORK REQUEST (W/R), THE BREAKER FOR THE HYDROGEN

SKIMMER FAN 2A (HSF-2A) WAS REPLACED WITH A TESTED BREAKER. ON AUGUST 3, WITH UNITS 1 AND 2 IN MODE 1, SUPPLEMENT 1 TO NRC BULLETIN 88-10 WAS ISSUED. PER THIS SUPPLEMENT, TESTED BREAKERS COULD ONLY BE INSTALLED AS DIRECT REPLACEMENT FOR NON-TRACEABLE BREAKERS. IN RESPONSE TO THE SUPPLEMENT, ON AUGUST 23, WITH UNIT 2 IN MODE 1, THE TESTED BREAKER WAS REPLACED WITH A TRACEABLE BREAKER. SUBSEQUENTLY DURING POST-MAINTENANCE TESTING, THE BREAKER TRIPPED AND A W/R WAS WRITTEN TO I/R HSF-2A TRIPPING. ON 8/24, AT 1940 HOURS, WITH UNIT 2 IN MODE 1 AT 98% POWER, THE DECISION WAS MADE TO DECREASE POWER AT 10% PER HOUR IN ANTICIPATION OF NOT MEETING THE 72 HOUR ACTION STATEMENT FOR HSF-2A. ON 8/25, AT 0445 HOURS, WITH THE TURBINE/GENERATOR OFF-LINE AND REACTOR POWER AT 6%, HSF-2A SATISFACTORILY PASSED POST-MAINTENANCE TESTING. THIS INCIDENT HAS BEEN ATTRIBUTED TO A MANUFACTURING DEFICIENCY. THIS REPORT IS BEING PROVIDED IN REGARD TO 10CFR PART 21.

[37] CATAWBA 1 DOCKET 50-413 LER 89-016
FOUR CHANNELS OF POWER RANGE INSTRUMENTATION INOPERABLE FOLLOWING UNIT RUNBACK AS A RESULT OF FAILURE OF A GENERATOR BREAKER AIR PRESSURE GAUGE.
EVENT DATE: 091389 REPORT DATE: 101189 NSSS: WE TYPE: PWR

(NSIC 215580) ON SEPTEMBER 13, 1989, AT 0541 HOURS, UNIT 1 WAS IN MODE 1, 100% POWER OPERATION. GENERATOR 1B POWER CIRCUIT BREAKER (PCB) OPENED CAUSING UNIT RUNBACK TO 55% POWER. FOUR OUT OF FOUR CHANNELS OF POWER RANGE NUCLEAR INSTRUMENTATION (PRNI) DISPLAYED GREATER THAN THE 5% ALLOWABLE MISMATCH BETWEEN RATED THERMAL POWER (RTP) AND NUCLEAR POWER, IN THE NON- CONSERVATIVE DIRECTION. AT 0550 HOURS, TECHNICAL SPECIFICATION 3.0.3 WAS ENTERED AND WORK REQUEST 4099 SWR WAS ISSUED TO COMPLETE CALIBRATION OF THE PRNIS. THE UNIT WAS STABLE AT 55% POWER AT 0630 HOURS AND THE CALIBRATIONS WERE PERFORMED. FOLLOWING THE REQUIRED CALIBRATIONS OF THE PRNI, THE UNIT EXITED TECHNICAL SPECIFICATION 3.0.3. THE PNEUMATIC GAUGE WAS SUBSEQUENTLY REPLACED, AND GENERATOR PCB 1B WAS RESTORED TO SERVICE. UNIT POWER INCREASES COMMENCED AT 1003 HOURS ON SEPTEMBER 13, 1989. ALL REQUIRED PRNI CALIBRATIONS WERE COMPLETED TO WITHIN 2% OF RTP BY 1525 HOURS. UNIT POWER REACHED 97% POWER AT 1815 HOURS. AT 2100 HOURS, UNIT REACTOR POWER REACHED 100%. THE POWER RANGE MISMATCH WAS CONSIDERED TO BE AN EXPECTED PHENOMENON FOLLOWING A UNIT RUNBACK. THIS INCIDENT HAS BEEN ATTRIBUTED TO EQUIPMENT FAILURE DUE TO THE FAILURE OF THE PRESSURE GAUGE ON THE PCB WHICH CAUSED THE UNIT RUNBACK.

[38] CATAWBA 1 DOCKET 50-413 LER 89-023
BOTH TRAINS OF CONTROL ROOM AREA VENTILATION INOPERABLE DUE TO AN INCOMPLETE TESTING PROCEDURE.
EVENT DATE: 091589 REPORT DATE: 101189 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 215581) ON SEPTEMBER 15, 1989, AT 1315 HOURS, WITH UNITS 1 AND 2 IN MODE 1, POWER OPERATION, TECHNICAL SPECIFICATION 3.0.3 WAS ENTERED DUE TO BOTH TRAINS OF THE CONTROL ROOM AREA VENTILATION (VC) SYSTEM BEING INOPERABLE. TRAIN B OF THE VC SYSTEM WAS ALREADY INOPERABLE FOR MAINTENANCE. TRAIN A OF VC WAS DECLARED INOPERABLE FOLLOWING THE UNSATISFACTORY PERFORMANCE OF A CONTROL ROOM POSITIVE PRESSURE TEST WITH ONLY ONE OF THE TWO OUTSIDE AIR INTAKES OPEN. THE CONTROL ROOM RETURN AIR DAMPER WAS ADJUSTED ON TRAIN A AND THE CONTROL ROOM POSITIVE PRESSURE TEST WAS PERFORMED WITH ACCEPTABLE RESULTS IN ALL ALIGNMENTS. BOTH TRAINS WERE RETURNED TO OPERABILITY ON SEPTEMBER 16, FOLLOWING SUCCESSFUL TESTING. THIS INCIDENT IS ATTRIBUTED TO INCOMPLETE TESTING DURING PRE-OPERATIONAL TESTING OF THE VC SYSTEM. ALL OTHER APPROPRIATE VENTILATION SYSTEMS' PRE-OPERATIONAL TESTS ARE BEING REVIEWED TO ASSURE COMPLETE TESTING WAS PERFORMED.

[39] CATAWBA 1 DOCKET 50-413 LER 89-024
INOPERABLE FIRE DOOR DUE TO LATCH FAILURE AND INADEQUATE POLICY REGARDING CONTROLLED ACCESS FIRE DOORS.
EVENT DATE: 092089 REPORT DATE: 102089 NSSS: WE TYPE: PWR
VENDOR: HARTY, R.V. DIV

(NSIC 215713) ON SEPTEMBER 20, 1989, AT 1530 HOURS, WITH UNITS 1 AND 2 IN MODE 1, POWER OPERATION, A CATAWBA SAFETY REVIEW GROUP STAFF MEMBER EN ROUTE TO THE DIESEL GENERATOR ROOMS IDENTIFIED COMMITTED FIRE DOOR S102A AS BEING A POSSIBLY INADEQUATE FIRE BOUNDARY DOOR. THIS DOOR IS EQUIPPED WITH A MANUAL LATCHBOLT (WHICH WAS FOUND BROKEN), A KEY OPERATED LOCK, AND A CONTROLLED ACCESS DOOR (CAD) MECHANISM WHICH IS ACTUATED BY A SECURITY BADGE ACCESS KEY. A FIRE WATCH WAS ESTABLISHED, PENDING A DESIGN ENGINEERING (DE) OPERABILITY EVALUATION. ON SEPTEMBER 21, DE DETERMINED THAT THE CAD MECHANISM FOR THIS DOOR WAS INADEQUATE AS A FIRE PROTECTION LATCHING DEVICE. DOOR S102A WAS REPAIRED ON OCTOBER 4, AND THE FIRE WATCH WAS TERMINATED. THE LATCH IS KNOWN TO HAVE BEEN IN A FAILED CONDITION BETWEEN APRIL 27, 1988 AND SEPTEMBER 20, 1989, WITH NO FIRE WATCH POSTED, THEREBY VIOLATING TECHNICAL SPECIFICATIONS. THIS INCIDENT IS ATTRIBUTED TO A DESIGN DEFICIENCY, FOR THE SELECTION OF A DOOR NOT CAPABLE OF WITHSTANDING FREQUENT USE, RESULTING IN A FAILED LATCH, AND TO AN INADEQUATE POLICY REGARDING CONTROLLED ACCESS FIRE DOORS, RESULTING IN THE BELIEF THAT THE CAD MECHANISM WOULD MEET THE INTENT OF TECHNICAL SPECIFICATIONS WHEN A FIRE DOOR LATCH WAS FAILED. CORRECTIVE ACTIONS INCLUDED ESTABLISHMENT OF A FIRE WATCH AND REPAIRING THE DOOR, AND WILL INCLUDE A POLICY STATEMENT FROM DE.

[40] CATAWBA 1 DOCKET 50-413 LER 89-025
TECH SPEC 3.0.3 ENTERED ON BOTH UNITS FOR INOPERABLE POWER RANGE NUCLEAR INSTRUMENTATION DUE TO POWER REDUCTION DURING HURRICANE HUGO.
EVENT DATE: 092289 REPORT DATE: 102089 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 215714) ON SEPTEMBER 22, 1989, UNITS 1 AND 2 WERE IN MODE 1, POWER OPERATION, AT 100% AND 98% POWER, RESPECTIVELY. AT 0547 HOURS, 1HTA, 13.8 KV AUXILIARY SWITCHGEAR, DEENERGIZED RESULTING IN THE LOSS OF POWER TO THE CONDENSER CIRCULATING WATER COOLING TOWER FANS. THIS REQUIRED THE OPERATORS TO REDUCE REACTOR POWER IN ORDER TO MAINTAIN CONDENSER VACUUM. DURING THE POWER REDUCTIONS, UNITS 1 AND 2 ENTERED TECHNICAL SPECIFICATION 3.0.3 AT 0600 HOURS AND 0632 HOURS, RESPECTIVELY, DUE TO GREATER THAN 5 PERCENT MISMATCH BETWEEN THERMAL POWER BEST ESTIMATE AND THE POWER RANGE NUCLEAR INSTRUMENTATION (PRNI). FOLLOWING RECALIBRATION OF THE PRNIS, UNITS 1 AND 2 EXITED TECHNICAL SPECIFICATION 3.0.3 AT 0655 HOURS AND 0640 HOURS, RESPECTIVELY. THE PRNI MISMATCH IS CONSIDERED TO BE AN EXPECTED PHENOMENON DURING A POWER REDUCTION. THE POWER REDUCTION IS ATTRIBUTED TO UNUSUAL WEATHER CONDITIONS CAUSED BY THE HIGH WINDS AND RAINFALL DELIVERED BY HURRICANE HUGO WHICH CAUSED WATER FROM THE SERVICE BUILDING ROOF TO LEAK ON THE 1HTA SWITCHGEAR, THUS TRIPPING THE SWITCHGEAR. ALTHOUGH A SUBSEQUENT INSPECTION OF THE ROOF DID NOT IDENTIFY ANY DEFECTS, THE ROOF IS SCHEDULED TO BE REWORKED IN THE NEAR FUTURE.

[41] CLINTON 1 DOCKET 50-461 LER 89-034
LACK OF TRAINING, PERSONNEL ERROR AND INADEQUATE COMMUNICATIONS RESULT IN FAILURE TO VERIFY PROCESS RADIATION MONITOR OPERABILITY AND TO MEET TECH SPEC REQUIREMENTS.
EVENT DATE: 100489 REPORT DATE: 110189 NSSS: GE TYPE: BWR

(NSIC 215784) ON OCTOBER 4, 1989, A RADIATION PROTECTION SHIFT SUPERVISOR (RPSS) DISCOVERED THAT THE IN-SERVICE STATION HEATING, VENTILATING AND AIR CONDITIONING (HVAC) EXHAUST STACK PROCESS RADIATION MONITOR (PRM), ORIX-PR001 HAD NOT BEEN VERIFIED AS OPERABLE. THIS RESULTED IN A FAILURE TO MEET THE LIMITING CONDITION FOR OPERATION FOR TECHNICAL SPECIFICATION (TS) 3.3.7.12. THIS TS REQUIRES THAT

ONE STATION HVAC EXHAUST STACK PRM BE OPERABLE AT ALL TIMES. AT 0826 HOURS, PRM ORIX-PR002 WAS REMOVED FROM SERVICE AND PRM ORIX-PR001 WAS PLACED IN SERVICE. THE RADIATION PROTECTION (RP) TECHNICIAN (TECH) TRANSFERRING THE MONITORS DID NOT PERFORM ALL OF THE CHECKS REQUIRED TO VERIFY THAT PRM ORIX-PR001 WAS OPERABLE AFTER PLACING IT IN SERVICE. SPECIFICALLY, THE RP TECH PERFORMED A CHANNEL CHECK ON THE MONITOR BUT DID NOT VERIFY FLOW. THEREFORE, THE PRM WAS INOPERABLE. THIS EVENT WAS CAUSED BY LACK OF TRAINING, PERSONNEL ERROR AND INADEQUATE COMMUNICATIONS. THE RP TECH WAS NOT FULLY QUALIFIED AND THE RPSS FAILED TO DIRECT THE RP TECH TO USE APPLICABLE PROCEDURES WHEN PLACING PRM ORIX-PR001 IN SERVICE. CORRECTIVE ACTIONS INCLUDE REMINDING APPROPRIATE RP PERSONNEL OF THE NEED TO USE AND FOLLOW PROCEDURES, AND COUNSELING RP SHIFT SUPERVISORS ON THE NEED TO ENSURE TECHS ARE QUALIFIED AND ON THE NEED TO REVIEW AND STATUS ACTIVITIES AFFECTING RP.

[42] CONNECTICUT YANKEE DOCKET 50-213 LER 89-014
A&B SERVICE WATER PUMP FLOW DETERMINED INADEQUATE DURING TESTING.
EVENT DATE: 090489 REPORT DATE: 100389 NSSS: WE TYPE: PWR

(NSIC 215505) ON SEPTEMBER 4, 1989, AT APPROXIMATELY 1230 HOURS, WITH THE PLANT IN MODE 5 (COLD SHUTDOWN) AN IN-SERVICE INSPECTION TEST IDENTIFIED THE 'A' AND 'B' SERVICE WATER PUMP CAPACITIES AS BEING LESS THAN THE DESIGN BASIS (AS ANALYZED) FOR MODES 1, 2, 3, AND 4. SERVICE WATER PUMP PERFORMANCE BELOW THE DESIGN BASIS COULD REDUCE THE PLANT'S CAPABILITY TO RESPOND TO A VARIETY OF ACCIDENT SCENARIOS. THE REDUCED PUMP PERFORMANCE WAS IDENTIFIED DURING TESTING IN THE HIGH FLOW PORTION OF THE PUMP CURVE. THE ROOT CAUSE OF THIS EVENT IS THE FAILURE TO ACCURATELY PREDICT SERVICE WATER PUMP PERFORMANCE IN THE HIGH FLOW PORTION OF THE PUMP CURVE. THE REQUIRED CORRECTIVE ACTIONS TO ACHIEVE FLOW RATES GREATER THAN DESIGN BASIS REMAIN UNDER VIEW BY ENGINEERING. THE RESULTS OF THESE ACTIONS WILL BE FORWARDED IN A SUPPLEMENTAL REPORT. THE 'C' AND 'D' SERVICE WATER PUMPS WERE NOT AFFECTED BY THIS EVENT AND WOULD HAVE BEEN AVAILABLE TO PERFORM THEIR INTENDED SAFETY FUNCTION. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(V) BECAUSE A CONDITION EXISTED THAT ALONE COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF A SYSTEM NEEDED TO REMOVE RESIDUAL HEAT AND MITIGATE THE CONSEQUENCES OF AN ACCIDENT.

[43] CONNECTICUT YANKEE DOCKET 50-213 LER 89-015
EG-2A EMERGENCY DIESEL GENERATOR ROOM FIRE DOOR INOPERABLE.
EVENT DATE: 090589 REPORT DATE: 100389 NSSS: WE TYPE: PWR

(NSIC 215519) ON 9/5/89, AT APPROXIMATELY 1200 HOURS, WITH THE PLANT IN MODE 5 (COLD SHUTDOWN) THE EG-2A EMERGENCY DIESEL GENERATOR ROOM WAS DEVITALIZED, FROM A SECURITY STANDPOINT, TO ALLOW UNRESTRICTED ACCESS TO THE ROOM TO SUPPORT MAINTENANCE ACTIVITIES. IT WAS NOT RECOGNIZED THAT THIS CONDITION RENDERED THE DIESEL ROOM TECH SPEC FIRE DOOR INOPERABLE AND A FIRE WATCH WAS NOT ESTABLISHED IN ACCORDANCE WITH THE TECH SPECS. AT APPROXIMATELY 1630 HOURS THE SAME DAY THE CONDITION WAS IDENTIFIED DURING OPERATOR ROUNDS AND AN HOURLY FIRE WATCH PATROL WAS IMMEDIATELY ESTABLISHED. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. ALL PERSONNEL INVOLVED IN THE EVENT HAVE BEEN MADE AWARE OF THE VIOLATION AND APPROPRIATE PROCEDURES WILL BE REVISED TO PREVENT RECURRENCE. THIS EVENT IS REPORTABLE PER 10CFR50.73(A)(2)(I)(B) SINCE IT INVOLVED A CONDITION PROHIBITED BY THE PLANT'S TECH SPECS.

[44] CONNECTICUT YANKEE DOCKET 50-213 LER 89-016
POTENTIAL RATING DEFICIENCY IDENTIFIED IN MOLDED CASE CIRCUIT BREAKERS.
EVENT DATE: 092289 REPORT DATE: 102089 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215676) ON SEPTEMBER 22, 1989, AT APPROXIMATELY 1200 HOURS, WITH THE PLANT

IN MODE 5 (COLD SHUTDOWN), AN ENGINEERING ANALYSIS IDENTIFIED 15 WESTINGHOUSE 480-V CIRCUIT BREAKERS IN MOTOR CONTROL CENTER (MCC) 5 WHICH HAVE FAULT-INTERRUPTING CAPABILITY LESS THAN THE POTENTIALLY AVAILABLE (WORST CASE) FAULT CURRENTS. IT IS POSTULATED THAT IF A BOLTED THREE-PHASE FAULT WERE TO OCCUR ACROSS THE LOAD TERMINALS OF ONE OF THESE BREAKERS, IT COULD CREATE AN OVER TRIP CONDITION. THIS OVER TRIP CONDITION COULD CAUSE THE ENTIRE MOTOR CONTROL CENTER TO ISOLATE FROM THE 480-V AC POWER SYSTEM RENDERING CERTAIN RESIDUAL HEAT REMOVAL (RHR) AND SAFETY INJECTION (SI) VALVES INOPERATIVE. THE ROOT CAUSE OF THIS EVENT WAS A DESIGN CONTROL DEFICIENCY. CORRECTIVE ACTION WILL BE TO REPLACE THE IDENTIFIED CIRCUIT BREAKERS WITH CIRCUIT BREAKERS OR MOTOR CIRCUIT PROTECTORS WHICH HAVE THE APPROPRIATE FAULT INTERRUPTING CAPABILITIES. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(V) SINCE A CONDITION EXISTED WHICH ALONE COULD HAVE PREVENTED THE FULFILLMENT OF A SAFETY FUNCTION OF A SYSTEM NEEDED TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT.

[45]   CONNECTICUT YANKEE   DOCKET 50-213   LER 89-018
CONTAINMENT PENETRATION FAILS TYPE C LOCAL LEAK RATE TEST.
EVENT DATE: 092989   REPORT DATE: 102789   NSSS: WE   TYPE: PWR
VENDOR: NIBCO COMPANY

(NSIC 215721) CONTAINMENT PENETRATION LOCAL LEAK RATE TESTING (TYPE B AND C) IS BEING CONDUCTED DURING THE 1989 REFUELING OUTAGE IN ACCORDANCE WITH 10CFR50 APPENDIX J AND TECH SPEC 4.4.II. ON 9/29/89, AT APPROXIMATELY 2215, WITH THE PLANT SHUTDOWN IN MODE 6, CONTAINMENT PENETRATION P-65 (AIR MONITOR SAMPLE TO CONTAINMENT) ISOLATION VALVE VS-CV-1104 FAILED ITS AS FOUND TYPE C LOCAL LEAK RATE TEST. THE ROOT CAUSE OF THE FAILURE HAS NOT BEEN DETERMINED. THE VALVE WILL BE REMOVED AND INSPECTED AND THE RESULTS OF THE INSPECTION AND CORRECTIVE ACTION WILL BE FORWARDED IN A SUPPLEMENTAL REPORT. THIS EVENT IS REPORTABLE PER 10CFR50.73(A)(2)(I)(B) SINCE IT INVOLVES A CONDITION PROHIBITED BY THE PLANTS TECH SPECS.

[46]   COOK 1   DOCKET 50-315   LER 89-011
INCOMPLETE SURVEILLANCE/RETEST SCHEDULE RESULTS IN FAILURE TO CONDUCT TECH SPEC REQUIRED SNUBBER RETESTING.
EVENT DATE: 090189   REPORT DATE: 092989   NSSS: WE   TYPE: PWR
VENDOR: ITT GRINNELL

(NSIC 215509) ON SEPTEMBER 1, 1989, WITH UNIT 1 OPERATING AT 100 PERCENT THERMAL POWER, A TECHNICAL SPECIFICATION SNUBBER WAS IDENTIFIED WHICH HAD NOT BEEN TESTED AS REQUIRED FOLLOWING A PREVIOUS FAILURE TO MEET PLANT LOCKUP VELOCITY CRITERIA. SNUBBER 1-GRC-S519, INSTALLED ON THE PRESSURIZER SPRAY LINE, HAD BEEN TESTED IN AUGUST, 1987. THE AS-FOUND LOCKUP VELOCITY WAS 16.3 INCHES/MINUTE COMPARED TO THE PLANT ESTABLISHED MAXIMUM OF 15.1 INCHES/MINUTE. TECHNICAL SPECIFICATION 3/4.7.8 REQUIRED RE-TEST DURING THE UNIT 1 REFUELING OUTAGE COMPLETED JUNE 20, 1989. THE SNUBBER IS INACCESSIBLE FOR INSPECTION DURING POWER OPERATION. THE PROCEDURE CONTROLLING RE-TESTING DID NOT CLEARLY DELINEATE BETWEEN NORMAL INSPECTIONS AND MANDATED RE-TESTS AND THE COMMITMENT WAS MISSED. AFTER DISCOVERY, AND BASED ON THE MANUFACTURER'S ENGINEERING EVALUATION, IT WAS DETERMINED THAT THE ORIGINAL TEST CRITERIA WERE CONSERVATIVE AND THAT THE SNUBBER WOULD HAVE FUNCTIONED PER DESIGN. A WAIVER FROM TECHNICAL SPECIFICATION 3/4.7.8 WAS REQUESTED FROM THE NRC SEPTEMBER 1, 1989, AND GRANTED ON THAT DATE.

[47]   COOK 1   DOCKET 50-315   LER 89-012
SOLID STATE PROTECTION SYSTEM SURVEILLANCE TESTING PERFORMED ON THE B TRAIN WHILE THE A TRAIN SAFETY INJECTION PUMP WAS INOPERABLE DUE TO PERSONNEL ERROR.
EVENT DATE: 091389   REPORT DATE: 100589   NSSS: WE   TYPE: PWR

(NSIC 215556) AT 0852 ON 9/13/89, THE B TRAIN OF THE SOLID STATE PROTECTION

SYSTEM (SSPS) WAS MADE INOPERABLE FOR SSPS AND REACTOR TRIP BREAKER SURVEILLANCE TESTING AFTER RECEIVING PERMISSION FROM THE SHIFT SUPERVISOR AND UNIT SUPERVISOR. DUE TO DESIGN OF THE CIRCUITRY, THE B TRAIN OF SAFETY INJECTION IS ALSO MADE INOPERABLE DURING THE TESTING. THE A TRAIN SAFETY INJECTION PUMP WAS ALSO INOPERABLE DURING THE TESTING DUE TO BEING ISOLATED FOR LEAK REPAIRS. THEREFORE, BOTH SI PUMPS WERE INOPERABLE FROM 0852 UNTIL THE B TRAIN OF SSPS WAS RETURNED TO OPERABILITY AT 1000 ON 9/13/89. THIS VIOLATION OF TECH SPEC 3.5.2 WAS RECOGNIZED BY A SECOND UNIT SUPERVISOR WHEN REVIEWING PLANT STATUS IN PREPARATION FOR THE A TRAIN SSPS AND REACTOR TRIP BREAKER TESTING. THE PRIMARY CAUSE OF THIS EVENT WAS FAILURE OF THE SHIFT SUPERVISOR AND UNIT SUPERVISOR TO RECOGNIZE THAT SSPS TESTING SHOULD NOT HAVE BEEN DONE WITH THE OPPOSITE TRAIN SI PUMP INOPERABLE. ALSO CONTRIBUTING TO THE EVENT WAS THE JOB PLANNING PROCESS WHICH DID NOT CONSIDER THE SURVEILLANCE SCHEDULE AND LACK OF GUIDANCE IN THE SURVEILLANCE PROCEDURE TO ENSURE THAT OPPOSITE TRAIN EQUIPMENT WAS OPERABLE. PREVENTIVE ACTIONS TAKEN TO PREVENT RECURRENCE INCLUDE REVISION OF SURVEILLANCE PROCEDURE AND INCLUSION OF THE SURVEILLANCE SCHEDULE IN THE JOB PLANNING PROCESS.

[48]   COOPER   DOCKET 50-298   LER 89-025
UNPLANNED MAIN TURBINE TRIP AND SUBSEQUENT REACTOR SCRAM CAUSED BY A SPURIOUS MAIN TURBINE HYDRAULIC CONTROL OIL RESERVOIR LOW LEVEL SIGNAL.
EVENT DATE: 092889   REPORT DATE: 102789   NSSS: GE   TYPE: BWR
VENDOR: MAGNETROL, INC.

(NSIC 215767) ON SEPTEMBER 28, 1989, AT 11:36 AM, A MAIN TURBINE TRIP OCCURRED, FOLLOWED IMMEDIATELY BY A REACTOR SCRAM. THE REACTOR VESSEL WATER LEVEL TRANSIENT THAT RESULTED FROM THE SCRAM CAUSED CONTAINMENT ISOLATION GROUPS 2 (PRIMARY CONTAINMENT), 3 (REACTOR WATER CLEANUP) AND 6 (SECONDARY CONTAINMENT) TO OCCUR. THE TURBINE TRIP SIGNAL THAT INITIATED THE EVENT WAS TURBINE HYDRAULIC CONTROL OIL RESERVOIR LOW LEVEL, WHICH OCCURRED APPROXIMATELY TWO MINUTES AFTER THE CONTROL OIL PUMPS WERE SHIFTED. IT WAS LATER VERIFIED THAT RESERVOIR LEVEL HAD REMAINED IN THE NORMAL RANGE THROUGHOUT THE EVENT. THE EXACT CAUSE FOR THE TURBINE HYDRAULIC CONTROL OIL RESERVOIR LOW LEVEL TRIP SIGNAL COULD NOT BE POSITIVELY IDENTIFIED. IT WAS CONCLUDED THE TRIP SIGNAL WAS CAUSED BY SPURIOUS ACTUATION OF THE LEVEL SWITCH DUE TO EQUIPMENT VIBRATION AS A RESULT OF SHIFTING CONTROL OIL PUMPS. THE IMMEDIATE ACTIONS TAKEN WERE TO STABILIZE THE PLANT FOLLOWING THE SCRAM. TO MINIMIZE THE POSSIBILITY OF RECURRENCE, A TEMPORARY INSTRUCTION WAS ISSUED TO LIMIT CONTROL OIL PUMP SHIFTING, AND A CONTROL OIL SYSTEM FLUSH WAS SCHEDULED FOR THE NEXT OUTAGE. ADDITIONALLY, SEVERAL SYSTEM RELIABILITY IMPROVEMENTS WILL BE EVALUATED.

[49]   CRYSTAL RIVER 3   DOCKET 50-302   LER 85-034 REV 01
UPDATE ON FAILURE TO VERIFY EQUIPMENT OPERABILITY PRIOR TO ENTERING TECH SPEC APPLICABILITY CONDITIONS.
EVENT DATE: 080985   REPORT DATE: 102789   NSSS: BW   TYPE: PWR

(NSIC 215768) TECH SPEC 3.5.1 REQUIRES THE CORE FLOOD TANK ISOLATION VALVES (CFV-5 AND CFV-6) TO BE OPERABLE WHEN IN MODES 1, 2 AND 3 WITH SYSTEM PRESSURE GREATER THAN 750 PSIG. TECH SPEC SURVEILLANCE 4.5.1.D REQUIRES THE "VALVE NOT OPEN" ALARM ANNUNCIATOR TO BE TESTED AT A SYSTEM PRESSURE ABOVE 750 PSIG. TECH SPEC SURVEILLANCE 4.5.1.D REQUIRES THE "VALVE NOT OPEN" ALARM ANNUNCIATOR TO BE TESTED AT A SYSTEM PRESSURE ABOVE 750 PSIG. IN ADDITION TECH SPEC 4.0.4 PROHIBITS MODE ASCENSION OR CHANGES IN SPECIFIED APPLICABILITY CONDITIONS, UNLESS THE APPLICABLE SURVEILLANCES HAVE BEEN PERFORMED WITHIN THE REQUIRED SURVEILLANCE PERIOD. THE COMBINED REQUIREMENTS OF THE ABOVE TECH SPEC ITEMS PRESENTS A DIFFICULT SITUATION WHEN ASCENDING MODES FROM MODE 4 AND THE VALVE ALARM FUNCTION IS INOPERABLE. DURING THE PLANT SHUTDOWN ON 3/10/85, THE VALVE ALARM FUNCTION FOR CFV-5 FAILED TO MEET THE ACCEPTANCE CRITERIA AND CORRECTIVE MAINTENANCE WAS PERFORMED DURING THE OUTAGE. ON 8/9/85, CR-3 ENTERED MODE 3 AND THEN RAISED REACTOR COOLANT SYSTEM PRESSURE ABOVE 750 PSIG WITHOUT MEETING THE REQUIRED

OPERABILITY SURVEILLANCES FOR TECH SPEC 3.5.1.A. AS REQUIRED BY THE SURVEILLANCE PROCEDURE, THE NECESSARY TESTING WAS PERFORMED IN MODE 3. THE SYSTEM PRESSURE WAS INCREASED ABOVE 750 PSIG AND CFV-5 WAS CYCLED CLOSED AND OPEN. THE ALARM ANNUNCIATOR FUNCTIONED PROPERLY.

[50] CRYSTAL RIVER 3 DOCKET 50-302 LER 88-002 REV 02
UPDATE ON TECHNICIAN ERROR CAUSES TRIP OF OPERATING FEEDWATER PUMP WHICH RESULTS IN EMERGENCY FEEDWATER ACTUATION AND SUBSEQUENT OVERSPEED OF STEAM DRIVEN EMERGENCY FEEDWATER PUMP.
EVENT DATE: 010788 REPORT DATE: 102689 NSSS: BW TYPE: PWR

(NSIC 215719) ON 1/7/88, CRYSTAL RIVER UNIT 3 WAS IN THE HOT STANDBY MODE (MODE 3). AN EMERGENCY FEEDWATER ACTUATION OCCURRED ON A LOSS OF BOTH MAIN FEEDWATER PUMPS. THE OPERATING FW PUMP TRIPPED WHEN CONTROL POWER WAS LOST TO ITS GOVERNOR. THE MOTOR DRIVEN EF PUMP STARTED AS DESIGNED, BUT THE STEAM DRIVEN EF PUMP STARTED AND THEN TRIPPED ON AN OVERSPEED CONDITION. THE CAUSE OF THE LOSS OF POWER TO THE FW PUMP GOVERNOR WAS AN ERROR BY AN I&C TECHNICIAN WORKING ON A RADIATION MONITOR WHICH IS POWERED FROM THE SAME BREAKER AS THE FW PUMP GOVERNOR. THE MFW PUMP SHOULD NOT HAVE TRIPPED WHEN CONTROL POWER WAS LOST. THE CAUSE OF THE TRIP OF THE STEAM DRIVEN EF PUMP WAS THE IMPROPER POSITIONING OF THE BYPASS VALVE AROUND THE STEAM SUPPLY VALVES TO THE PUMP. THE IDLE FW PUMP WAS STARTED, AND ONCE IT WAS VERIFIED THAT THE MAIN FEEDWATER SYSTEM WAS CONTROLLING LEVEL PROPERLY, THE EF ACTUATION WAS RESET. THE TECHNICIAN INVOLVED HAS BEEN COUNSELLED IN ACCORDANCE WITH APPROVED PLANT POLICIES. A MECHANICAL LOCKING DEVICE HAS BEEN INSTALLED ON THE STEAM SUPPLY BYPASS VALVE, AND A PROCEDURE CHANGE HAS BEEN MADE TO IMPROVE THE MONITORING OF THIS VALVE'S POSITION.

[51] CRYSTAL RIVER 3 DOCKET 50-302 LER 89-016 REV 02
UPDATE ON ADMINISTRATIVE PROBLEMS CAUSED DEFICIENCIES IN THE ENVIRONMENTAL QUALIFICATION PROGRAM RESULTING IN PLANT EQUIPMENT NOT PROPERLY QUALIFIED.
EVENT DATE: 042689 REPORT DATE: 102389 NSSS: BW TYPE: PWR

(NSIC 215727) CRYSTAL RIVER UNIT 3 WAS IN MODE 5 (COLD SHUTDOWN) FROM 2/27/89 TO 6/1/89. DURING THIS OUTAGE, NRC INSPECTORS DISCOVERED DEFICIENCIES RELATED TO ENVIRONMENTAL QUALIFICATION OF PLANT EQUIPMENT. DEFICIENCIES INCLUDED IMPROPER CABLES AND SPLICES, IMPROPER SILICON OIL LEVEL IN INSTRUMENT JUNCTION BOXES, AND PROBLEMS RELATED TO VALVE MOTOR OPERATORS. PROBLEMS WERE THE RESULT OF DEFICIENCIES IN DETAILED DEVELOPMENT AND IMPLEMENTATION OF THE ENVIRONMENTAL QUALIFICATION PROGRAM. UTILITY PERSONNEL HAVE REPAIRED IDENTIFIED ENVIRONMENTAL QUALIFICATION DEFICIENCIES, OR HAVE JUSTIFIED CONTINUED OPERATION WITH THE DEFICIENCIES UNTIL REPAIRS ARE COMPLETED. THE UTILITY HAS EMBARKED ON A MAJOR VOLUNTARY EFFORT TO REVIEW THE EXISTING ENVIRONMENTAL QUALIFICATION PROGRAM, AND TO CORRECT ADDITIONAL ENVIRONMENTAL QUALIFICATION DEFICIENCIES THAT MAY BE DISCOVERED.

[52] CRYSTAL RIVER 3 DOCKET 50-302 LER 89-034
PERSONNEL ERRORS BY ARCHITECT ENGINEER RESULT IN PLANT OPERATION OUTSIDE DESIGN BASIS.
EVENT DATE: 092689 REPORT DATE: 102689 NSSS: BW TYPE: PWR

(NSIC 215728) ON 9/26/89, WHILE IN MODE 5 (COLD SHUTDOWN), TWO RECENTLY IDENTIFIED CONDITIONS WERE DETERMINED TO BE OUTSIDE THE CRYSTAL RIVER UNIT 3 DESIGN BASIS. FIRST; SOLENOID CONTROL VALVES FOR 8 CONTROL COMPLEX HVAC DAMPERS, 1 HVAC CONTROL PANEL AND 6 CONTAINMENT ISOLATION VALVES WERE POWERED FROM A NON-1E DISTRIBUTION PANEL. SECOND; NON-SAFETY RELATED TESTING SOLENOID VALVES SHARED COMMON CIRCUITS WITH SAFETY RELATED ACTUATION SOLENOID VALVES ON EACH OF FOUR MAIN STEAM ISOLATION VALVES, WITHOUT PROPER ISOLATION. ONE OF THESE CONDITIONS HAS EXISTED SINCE INITIAL PLANT OPERATION IN 1977. THE OTHER HAS

EXISTED SINCE 1986. THE CAUSE WAS COGNITIVE PERSONNEL ERROR, THE ARCHITECT ENGINEER AND UTILITY ENGINEERING PERSONNEL FAILED TO RECOGNIZE ALL APPLICABLE DESIGN REQUIREMENTS. AN ENGINEERING EVALUATION WAS PERFORMED AND DETERMINED THAT THE HVAC DAMPERS, THE HVAC CONTROL PANEL, AND CONTAINMENT ISOLATION VALVES REMAINED OPERABLE. THE TEST SOLENOID VALVE CIRCUITS HAVE BEEN PROVIDED WITH ISOLATION FUSES. THE TEST SOLENOID VALVE CIRCUITS HAVE BEEN PROVIDED WITH ISOLATION FUSES. NUCLEAR ENGINEERING PROCEDURES HAVE BEEN UPGRADED TO INSURE THAT ALL APPLICABLE DESIGN REQUIREMENTS ARE SATISFIED DURING DESIGN REVIEW. A MODIFICATION WILL BE INSTALLED TO PROVIDE THE CLASS 1E CIRCUITS WITH CLASS 1E POWER SUPPLIES.

[53] DAVIS-BESSE 1 DOCKET 50-346 LER 89-004 REV 01
UPDATE ON POTENTIAL FOR CIRCULATING WATER LINE BREAK TO CAUSE LOSS OF SERVICE WATER PUMPS.
EVENT DATE: 041189 REPORT DATE: 101689 NSSS: BW TYPE: PWR

(NSIC 215647) ON 4/11/89, AFTER COMPLETING AN EVALUATION FOR ONE OF TWO CONCERNS RAISED BY THE DAVIS BESSE NRC RESIDENT INSPECTORS ON 2/6/89. TOLEDO EDISON DETERMINED THAT A CIRCULATING WATER LINE BREAK WOULD RESULT IN LOSS OF SERVICE WATER, DUE TO FLOODING OF THE SERVICE WATER PUMPS ALLOWED BY A CONSTRUCTION BLOCKOUT THAT HAD EXISTED SINCE ORIGINAL PLANT CONSTRUCTION. THUS THE PLANT HAD OPERATED OUTSIDE ITS DESIGN BASIS SINCE 4/22/77. THE DAVIS BESSE FINAL SAFETY ANALYSIS REPORT (FSAR) DID NOT ADDRESS FLOODING VIA THIS PATHWAY SINCE THIS CONSTRUCTION BLOCKOUT WAS OVERLOOKED. ON FEBRUARY 6, 1989, THE RESIDENT INSPECTORS INFORMED TOLEDO EDISON OF A HOLE IN THE WALL BETWEEN THE CONDENSER PIT AND THE SERVICE WATER (SW) TUNNEL. WHICH WAS CONTRARY TO USAR SECTION 3.6.2.7.2.13, AND THAT THERE WAS NO PROCEDURAL GUIDANCE FOR OPERATORS TO ISOLATE THE SW TUNNEL FROM THE SW PUMP ROOM IN THE EVENT OF FLOODING IN THE SW TUNNEL, WHICH WAS CONTRARY TO USAR SECTION 9.2.1.2. AFTER TOLEDO EDISON'S SUBMITTAL OF THE INITIAL LER ON 5/10/89, THE NRC RESIDENT INSPECTOR POINTED OUT THAT THE EFFECT OF SUBMERGENCE ON THE OPERABILITY OF OTHER EQUIPMENT SHOULD BE ADDRESSED IN THE ANALYSIS OF THE LER. AS A PRECAUTION. UNTIL AN EVALUATION OF THE POTENTIAL CONCERNS COULD BE PERFORMED, A STANDING ORDER WAS ISSUED ON 2/10/89, TO INSTRUCT OPERATORS TO PLUG THE FLOOR DRAINS IN THE SW PUMP ROOM IN EVENT OF SW TUNNEL FLOODING.

[54] DAVIS-BESSE 1 DOCKET 50-346 LER 89-013
HOURLY FIRE WATCH PATROL EXCEEDED ALLOWED INTERVAL BY TWO MINUTES.
EVENT DATE: 091389 REPORT DATE: 101389 NSSS: BW TYPE: PWR

(NSIC 215586) ON SEPTEMBER 13, 1989, AT 0015 HOURS, AN HOURLY FIRE WATCH PATROL EXCEEDED THE ALLOWED INTERVAL. THE PATROL WAS STARTED AT 0017 HOURS AND WAS TWO MINUTES LATE. THIS IS A VIOLATION OF TECHNICAL SPECIFICATION 3 7 10. ACTION "A.2". THE EMPLOYEE ASSIGNED TO THE WATCH FAILED TO INFORM THE SHIFT SUPERVISOR IMMEDIATELY WHEN HIS PROGRESS WAS IMPEDED. HIS PROBLEM WAS QUICKLY RESOLVED, BUT NOT BEFORE THE ALLOWED INTERVAL HAD PASSED. THE INDIVIDUAL WAS COUNSELED. ALL FIRE WATCH PERSONNEL RECEIVED INFORMATION ON THIS INCIDENT AND ON THEIR RESPONSIBILITIES AS ROVING FIRE WATCHES.

[55] DIABLO CANYON 1 DOCKET 50-275 LER 86-007 REV 01
UPDATE ON SPURIOUS CONTAINMENT VENTILATION ISOLATIONS.
EVENT DATE: 071086 REPORT DATE: 100289 NSSS: WE TYPE: PWR

(NSIC 215499) AT 1220 PDT, 7/10/86, 1645, 7/15/86; AND AGAIN AT 0912, 1357 AND 1452, 7/30/86, WITH THE UNIT IN MODE 1 POWER OPERATION, AN AUTOMATIC ISOLATION OF THE CONTAINMENT VENTILATION SYSTEM (CVS) OCCURRED. THE SAMPLE LINE ISOLATION VALVES FOR GASEOUS RADIATION MONITORS (RM) RM11 AND RM12 CLOSED AS DESIGNED. ALL OTHER CVS VALVES THAT RECEIVE ISOLATION SIGNALS WERE ALREADY CLOSED WHEN THE

EVENTS OCCURRED. AS REQUIRED BY 10 CFR 50.72 (B)(2)(II , 4-HOUR SIGNIFICANT EVENT REPORTS WERE MADE AT 1310, 7/10/86; 1837, 7/15/86, AND AT 1017 AND 1515, 7/30/86. THE CVS, ISOLATIONS WERE ATTRIBUTED TO SPURIOUS NOISE SIGNALS ON THE INSTRUMENT AC POWER SUPPLY LINES AS INDICATED BY THE ABSENCE OF A VALID INITIATION SIGNAL AND BY NO INCREASED RADIATION MONITOR INDICATION AND THE SUSCEPTIBILITY OF THE CVS ISOLATION INITIATION CIRCUITRY TO NOISE. THE CVS ISOLATION WAS RESET AT 1231, 7/10/86; 1647, 7/15/86; AND AT 0914, 1402 AND 1456, 7/30/86. THIS EVENT AND THE OTHER SPURIOUS CONTAINMENT VENTILATION ACTUATIONS HAVE BEEN STUDIED BY THE NOISE REDUCTION TASK FORCE IN AN EFFORT TO DETERMINE EFFECTIVE CORRECTIVE ACTIONS TO PREVENT RECURRENCE. THESE CORRECTIVE ACTIONS ARE AS FOLLOWS: TIME DELAY CIRCUITRY HAS BEEN INSTALLED; DCPP PERSONNEL HAVE BEEN TRAINED ON POTENTIAL CAUSES OF CVIS; BYPASS CONTROL SWITCHES HAVE BEEN ADDED ON UNIT 1 AND WILL BE ADDED ON UNIT 2.

[56]   DIABLO CANYON 1   DOCKET 50-275   LER 86-014 REV 01
UPDATE ON ELECTRICAL TRANSIENT RESULTS IN CONTROL ROOM VENTILATION SYSTEM SHIFTING TO THE PRESSURIZATION MODE.
EVENT DATE: 091686   REPORT DATE: 100289   NSSS: WE   TYPE: PWR
OTHER UNITS INVOLVED: DIABLO CANYON 2 (PWR)

(NSIC 215500) ON SEPTEMBER 16, 1986, AT 1548 PDT, WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN AND UNIT 2 IN MODE 1 (POWER OPERATION), THE CONTROL ROOM VENTILATION SYSTEM (CRVS) AUTOMATICALLY SHIFTED FROM THE NORMAL VENTILATION MODE TO THE PRESSURIZATION MODE, ISOLATING THE CONTROL ROOM AND MECHANICAL EQUIPMENT ROOM. THE VENTILATION SYSTEM WAS RESTORED TO ITS NORMAL MODE AFTER ENSURING THAT THIS ESF ACTUATION WAS NOT THE RESULT OF AN ACTUAL EMERGENCY CONDITION. AS REQUIRED BY 10 CFR 50.72 B 2 (II , A 4-HOUR SIGNIFICANT EVENT REPORT WAS MADE AT 1722 PDT, SEPTEMBER 16, 1986. THIS EVENT WAS CAUSED BY AN ELECTRICAL TRANSIENT ON THE UNIT 1 INSTRUMENT AC INVERTER BUS PY-LLA MOMENTARILY PRODUCING AN ALARM CONDITION ON THE CONTROL ROOM VENTILATION RADIATION MONITOR, WHICH AUTOMATICALLY SHIFTED CRVS TO THE PRESSURIZATION MODE. ALARM CIRCUITRY OF THE CRVS RADIATION MONITOR IS HIGHLY SENSITIVE TO ELECTRICAL TRANSIENTS OF THIS TYPE. SUBSEQUENT INVESTIGATIONS COULD NOT DETERMINE THE EXACT CAUSE OF THE ELECTRICAL TRANSIENT. A RADIATION MONITORING SYSTEM UPGRADE PROGRAM HAS BEEN INITIATED TO REPLACE EXISTING RADIATION MONITORS WITH EQUIPMENT THAT IS LESS SENSITIVE TO ELECTRICAL NOISE.

[57]   DIABLO CANYON 1   DOCKET 50-275   LER 86-015 REV 01
UPDATE ON SPURIOUS CONTAINMENT VENTILATION ISOLATIONS.
EVENT DATE: 100186   REPORT DATE: 100289   NSSS: WE   TYPE: PWR

(NSIC 215501) ON OCTOBER 1, 1986, AT 1341 PDT, AND AGAIN ON OCTOBER 8, 1986, AT 1147 PDT, WITH THE UNIT IN MODE 6 (REFUELING), AUTOMATIC ISOLATIONS OF THE CONTAINMENT VENTILATION SYSTEM (CVS) OCCURRED. THE SAMPLE LINE ISOLATION VALVES FOR GASEOUS RADIATION MONITORS (RM) RM11 AND RM12 CLOSED AS DESIGNED. ALL OTHER CVS VALVES THAT RECEIVE ISOLATION SIGNALS WERE ALREADY CLOSED WHEN THE EVENTS OCCURRED. AS REQUIRED BY 10 CFR 50.72 B 2 II), 4-HOUR SIGNIFICANT EVENT REPORTS WERE MADE AT 1430 PDT, OCTOBER 1, 1986, AND AT 1210 PDT, OCTOBER 8, 1986. THE CVS ISOLATIONS WERE ATTRIBUTED TO SPURIOUS NOISE SIGNALS, AS INDICATED BY THE ABSENCE OF VALID INITIATION SIGNALS AND NO INCREASED RADIATION MONITOR INDICATIONS. THE CVS ISOLATIONS WERE RESET AT 1400 PDT, OCTOBER 1, 1986, AND AT 1206 PDT, OCTOB"R 8, 1986. THIS EVENT AND THE OTHER SPURIOUS CONTAINMENT VENTILATION ACTUATIONS HAVE BEEN STUDIED BY THE NOISE REDUCTION TASK FORCE IN AN EFFORT TO DETERMINE EFFECTIVE CORRECTIVE ACTIONS TO PREVENT RECURRENCE. THESE CORRECTIVE ACTIONS ARE AS FOLLOWS: TIME DELAY CIRCUITRY HAS BEEN INSTALLED; DCPP PERSONNEL HAVE BEEN TRAINED ON POTENTIAL CAUSES OF CVIS; BYPASS CONTROL SWITCHES HAVE BEEN ADDED ON UNIT 1 AND WILL BE ADDED ON UNIT 2.

[58]   DIABLO CANYON 1   DOCKET 50-275   LER 87-003 REV 01
UPDATE ON SPURIOUS CONTAINMENT VENTILATION ISOLATION INITIATION.
EVENT DATE: 012687   REPORT DATE: 100289   NSSS: WE   TYPE: PWR

(NSIC 215502) ON JANUARY 26, 1987, AT 0023 PST, WITH THE UNIT IN MODE 1 POWER OPERATION , AN AUTOMATIC INITIATION OF THE CONTAINMENT VENTILATION ISOLATION SYSTEM (CVIS) OCCURRED. THE SAMPLE LINE ISOLATION VALVES FOR GASEOUS RADIATION MONITORS RM RM11 AND RM12 CLOSED AS DESIGNED. ALL OTHER CVIS VALVES THAT RECEIVE ISOLATION SIGNALS WERE ALREADY CLOSED WHEN THE EVENT OCCURRED. AS REQUIRED BY 10 CFR 50.72 B) 2 II), A 4-HOUR SIGNIFICANT EVENT REPORT WAS MADE AT 0123 PST, JANUARY 26, 1987. THE CVIS INITIATION WAS ATTRIBUTED TO SPURIOUS NOISE SIGNALS, AS INDICATED BY THE ABSENCE OF VALID INITIATION SIGNALS AND NO INCREASED RADIATION MONITOR INDICATIONS. THE CVIS WAS RESET AT 0030 PST, JANUARY 26, 1987. THIS EVENT AND THE OTHER SPURIOUS CONTAINMENT VENTILATION ACTUATIONS HAVE BEEN STUDIED BY THE NOISE REDUCTION TASK FORCE IN AN EFFORT TO DETERMINE EFFECTIVE CORRECTIVE ACTIONS TO PREVENT RECURRENCE. THESE CORRECTIVE ACTIONS ARE AS FOLLOWS: TIME DELAY CIRCUITRY HAS BEEN INSTALLED, DCPP PERSONNEL HAVE BEEN TRAINED ON POTENTIAL CAUSES OF CVIS; BYPASS CONTROL SWITCHES HAVE BEEN ADDED ON UNIT 1 AND WILL BE ADDED ON UNIT 2.

[59]   DIABLO CANYON 2   DOCKET 50-323   LER 89-009
MISSED ANALYSIS OF A STEAM GENERATOR BLOWDOWN GRAB SAMPLE DUE TO INADEQUATE CONTROLS.
EVENT DATE: 091089   REPORT DATE: 100989   NSSS: WE   TYPE: PWR

(NSIC 215559) ON SEPTEMBER 10, 1989, A STEAM GENERATOR BLOWDOWN SAMPLE WAS NOT ANALYZED WITHIN 24 HOURS AS REQUIRED BY TECHNICAL SPECIFICATIONS (TS 3.3.3.10). RADIATION MONITOR 2-RE-27 WAS INOPERABLE AND GRAB SAMPLES WERE TO BE TAKEN EVERY 12 HOURS IN ACCORDANCE WITH TS 3.3.3.10. ON SEPTEMBER 9, 1989, A SAMPLE WAS TAKEN AT 0542 PST AND ANALYZED WITHIN 24 HOURS. THE NEXT SAMPLE WAS TAKEN AT 1340 PST BUT WAS INADVERTENTLY DISPOSED OF PRIOR TO ANALYSIS. THIS OCCURRED DUE TO A LACK OF SAMPLE CONTROL AND THE FAILURE OF THE FOLLOWING SHIFT TO RECOGNIZE THE MISSED ANALYSIS. THE MISSED SAMPLE ANALYSIS WAS DISCOVERED BY A CHEMISTRY FOREMAN REVIEWING ANALYSIS AND LOGS IN ACCORDANCE WITH PROCEDURE AP C201-SL. THE FAILURE TO ANALYZE THE SAMPLE TAKEN ON SEPTEMBER 9, 1989, WAS DETERMINED TO BE DUE TO AN ADMINISTRATIVE PROGRAM WHICH LACKED EFFECTIVE CONTROLS FOR PROCESSING SAMPLES THROUGH THE COUNT ROOM. CORRECTIVE ACTIONS INCLUDE: 1 REVISION OF AN ADMINISTRATIVE PROCEDURE TO INCLUDE SAMPLE ANALYSIS COMPLETION AS A SHIFT TURNOVER CHECKLIST ITEM TO ENSURE THAT SAMPLES ARE ANALYZED WITHIN THE SPECIFIED TS PERIOD. 2 COUNSELLING OF PERSONNEL REGARDING THE IMPORTANCE OF EFFECTIVE COMMUNICATION DURING THE WATCH AND AT SHIFT TURNOVER, AND 3 DEVELOPMENT OF A NEW PROCEDURE AND REVISION OF APPLICABLE PROCEDURES TO ENSURE APPROPRIATE POSITIVE CONTROLS FOR SAMPLE PROCESSING.

[60]   DRESDEN 2   DOCKET 50-237   LER 89-025
INADVERTENT AUTOMATIC ISOLATION OF THE HIGH PRESSURE COOLANT INJECTION SYSTEM DUE TO DESIGN DEFICIENCY.
EVENT DATE: 091589   REPORT DATE: 101289   NSSS: GE   TYPE: BWR

(NSIC 215665) ON 9/15/89, AT 2050 HOURS DURING NORMAL UNIT 2 OPERATION AT 99% RATED CORE THERMAL POWER, AN UNPLANNED PRIMARY CONTAINMENT GROUP IV ISOLATION OCCURRED DURING REPLACEMENT OF A HIGH PRESSURE COOLANT INJECTION (HPCI SYSTEM LOW REACTOR PRESSURE ISOLATION MASTER TRIP UNIT (MTU) 2391-01A. THE GROUP IV ISOLATION SIGNAL RESULTED IN AUTOMATIC CLOSURE OF THE HPCI STEAM SUPPLY ISOLATION VALVES. TO PLACE THE HPCI SYSTEM IN A CONSERVATIVE CONDITION DURING REPLACEMENT OF THE MTU, REDUNDANT MTU 2391-01B WAS PREVIOUSLY MANUALLY TRIPPED TO INITIATE A CHANNEL "A" HALF GROUP IV ISOLATION SIGNAL. AS MTU 2391-01A WAS BEING REMOVED FROM ANALOG TRIP SYSTEM (ATS) PANEL 2202-73A, THE UNPLANNED GROUP IV ISOLATION OCCURRED DUE TO INADVERTENT MOVEMENT OF AN ADJACENT MTU. THE ROOT CAUSE OF THIS

EVENT WAS DETERMINED TO BE A DESIGN INDUCED DEFICIENCY WITHIN THE ATS PANEL SUCH THAT REMOVAL OF ADJACENT MTUS RESULTS IN SPURIOUS TRIPS. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS CONSIDERED TO BE MINIMAL SINCE THE GROUP IV ISOLATION SIGNAL WAS RESET WITHIN FOUR MINUTES, AND SINCE THE GROUP IV ISOLATION LOGIC RESPONDED PROPERLY WHEN CHALLENGED. IN ADDITION, ALL OTHER EMERGENCY CORE COOLING SYSTEMS REMAINED OPERABLE THROUGHOUT THIS EVENT. AS CORRECTIVE ACTIONS, PROCEDURES FOR REPLACEMENT OF HPCI ISOLATION MTUS WHICH ARE SENSITIVE TO THIS PROBLEM WILL BE REVISED TO INCLUDE APPROPRIATE PRECAUTIONARY STATEMENTS.

[61] DRESDEN 2 DOCKET 50-237 LER 89-026
STANDBY GAS TREATMENT SYSTEM DUE TO LOOSE REACTOR BUILDING VENTILATION SYSTEM RADIATION MONITOR CONNECTION.
EVENT DATE: 092489 REPORT DATE: 101989 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215723) ON 9/24/89 AT 1715 HRS WITH UNIT 2 AT 78% RATED CORE THERMAL POWER, AND UNIT 3 AT 98% RATED CORE THERMAL POWER, 2B REACTOR BUILDING VENTILATION (RBV) RADIATION MONITOR BEGAN SPIKING ERRATICALLY BETWEEN THE DOWNSCALE TRIP SETPOINT AND THE HI ALARM SETPOINT. WORK REQUEST 87513 WAS INITIATED TO INVESTIGATE THE PROBLEM. AT 2018 HOURS, WHILE THE INSTRUMENT MAINTENANCE DEPARTMENT (IMD) WAS PREPARING WORK PACKAGE TO INVESTIGATE THIS PROBLEM, THE 2B RADIATION MONITOR SPIKED TO HI HI TRIP SETPOINT. THIS CAUSED AN UNPLANNED AUTOMATIC ISOLATION OF RBV AND AUTOMATIC START OF THE STANDBY GAS TREATMENT (SBGT) SYSTEM. CAUSE OF ERRATIC SPIKING OF THE RADIATION MONITOR WAS FOUND TO BE A LOOSE MONITOR SENSOR CONVERTER CABLE CONNECTION. AS AN IMMEDIATE CORRECTIVE ACTION, REDUNDANT 2A RBV RADIATION MONITOR WAS CHECKED TO VERIFY THAT A HIGH RADIATION CONDITION WAS NOT PRESENT. THE CONNECTION WAS RESOLDERED IN ACCORDANCE WITH WORK REQUEST 87513. DRESDEN RADIATION PROCEDURE (DRP) 2000-5, RBV RADIATION MONITOR CALIBRATION, WAS THEN COMPLETED SATISFACTORILY TO ENSURE PROPER CALIBRATION OF 2B MONITOR. THE SBGT SYSTEM WAS THEN SECURED AND RBV SYSTEM WAS RETURNED TO NORMAL. THIS EVENT WAS OF MINIMAL SAFETY SIGNIFICANCE BECAUSE THE REDUNDANT 2A RBV MONITOR WAS OPERABLE, AND THE SBGT SYSTEM STARTED BY DESIGN WHEN CHALLENGED BY THE SPURIOUS TRIP SIGNAL. SIMILAR EVENT: 237/89-018.

[62] FARLEY 1 DOCKET 50-348 LER 89-004
POTENTIAL DESIGN INADEQUACY IN THE SERVICE WATER SYSTEM.
EVENT DATE: 061689 REPORT DATE: 101789 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: FARLEY 2 (PWR)

(NSIC 215648) ON JUNE 16, 1989, DURING A SELF-INITIATED SAFETY SYSTEM ASSESSMENT (SSSA) OF THE SERVICE WATER SYSTEM, IT WAS DETERMINED THAT CALCULATIONS WERE NOT AVAILABLE TO DEMONSTRATE THAT THE SERVICE WATER SYSTEM WAS CAPABLE OF SUPPLYING DESIGN FLOW TO SAFETY RELATED LOADS UNDER THE SCENARIO OF AN LOSP, A SEISMIC EVENT, OR COMBINATION OF THESE TWO EVENTS AND THAT THESE SCENARIOS MAY BE MORE LIMITING THAN THE ASSUMED DESIGN CONDITION FOR THE SERVICE WATER SYSTEM. THE ORIGINAL DESIGN ASSUMED THE LOCA WITH LOSP WOULD BE THE BOUNDING SCENARIO FOR ALL OTHER POSTULATED EVENTS. THE LOCA WITH LOSP EVENT GENERATES AN SI SIGNAL, ISOLATING MAJOR NON-SAFETY RELATED LOADS ON THE SERVICE WATER SYSTEM. THE LOSP, SEISMIC, OR COMBINATION LOSP/SEISMIC EVENTS POSTULATED BY THE SSSA WOULD NOT RESULT IN THE AUTOMATIC ISOLATION OF MAJOR NON-SAFETY RELATED SERVICE WATER LOADS AND THEREFORE MAY REPRESENT A MORE LIMITING DESIGN CONDITION FOR THE SYSTEM.

[63] FARLEY 2 DOCKET 50-364 LER 89-009
INACCURATE FEEDWATER FLOW INDICATION COULD HAVE PREVENTED PROPER OPERATION OF THE HIGH FLUX LEVEL REACTOR TRIP.
EVENT DATE: 052589 REPORT DATE: 101089 NSSS: WE TYPE: PWR
VENDOR: KEROTEST MANUFACTURING CORP.

(NSIC 215587) ON 5/25/89 WITH THERMAL POWER AT APPROXIMATELY 55%, IT WAS DETERMINED THAT THE POWER RANGE NUCLEAR INSTRUMENTATION CHANNELS (NIS) WERE READING APPROXIMATELY 7% LOW. THIS OCCURRED BECAUSE EQUALIZING VALVES WERE LEAKING ON TWO OF THE THREE FEEDWATER FLOW INSTRUMENTS. THIS FEEDWATER FLOW IS USED IN A HEAT BALANCE FOR DETERMINING REACTOR POWER. THIS REPORT IS BEING SUBMITTED VOLUNTARILY TO INFORM THE NRC OF THE FACT THAT FNP HAD A PROBLEM WITH AN INCORRECT FEEDWATER FLOW INDICATION WHICH HAD AN IMPACT ON INDICATED REACTOR THERMAL POWER DURING REDUCED POWER OPERATION. WHEN THIS ERROR WAS IDENTIFIED, THE NIS WERE ADJUSTED TO READ CORRECTLY. THIS EVENT WAS CAUSED BY ERRONEOUS READINGS FROM THE FEEDWATER FLOW INSTRUMENTS. THE EQUALIZING VALVES ON THE SUBJECT FEEDWATER FLOW TRANSMITTERS HAVE BEEN REPAIRED. A SUMMARY OF THIS EVENT HAS BEEN SENT TO ALL SHIFT SUPERVISORS FOR THEIR REVIEW OF THE LESSONS LEARNED. THE RETURN-TO-SERVICE CHECKLIST WHICH IS PERFORMED FOLLOWING EACH MAJOR OUTAGE HAS BEEN REVISED TO VERIFY AGREEMENT BETWEEN VARIOUS INDICATORS OF THERMAL POWER.

[64] FARLEY 2 DOCKET 50-364 LER 89-010
REACTOR TRIP CAUSED BY INADVERTENT OPENING OF THE OVERSPEED TRIP TEST VALVE.
EVENT DATE: 092089 REPORT DATE: 101289 NSSS: WE TYPE: PWR

(NSIC 215672) AT 0722 ON 9/20/89, WITH THE UNIT OPERATING AT APPROXIMATELY 61% POWER, THE REACTOR WAS TRIPPED MANUALLY FOLLOWING THE LOSS OF THE OPERATING STEAM GENERATOR FEED PUMP (SGFP). THE 2A SGFP TURBINE TRIPPED DUE TO A LOW AUTO-STOP OIL PRESSURE SIGNAL. THE SHIFT SUPERVISOR DIRECTED A MANUAL REACTOR TRIP IN ORDER TO PREVENT AN UNNECESSARY CHALLENGE TO THE REACTOR PROTECTION SYSTEM. THE UNIT WAS STABILIZED IN MODE 3 (HOT STANDBY). THE LOW AUTO-STOP OIL SIGNAL WAS CAUSED BY THE INADVERTENT OPENING OF THE OVERSPEED TRIP TEST VALVE. OPENING THIS VALVE LOWERED AUTO-STOP OIL PRESSURE BY DIVERTING AUTO-STOP OIL TO THE OVERSPEED TEST DEVICE. AUTO-STOP OIL PRESSURE DROPPED BELOW THE AS-FOUND TRIP SET POINT OF THE LOW AUTO-STOP OIL PRESSURE SWITCH CAUSING A SGFP TRIP. THE TRIP SETPOINT WAS FOUND TO BE HIGHER THAN IT SHOULD HAVE BEEN. THE UNIT RETURNED TO POWER OPERATION AT 2003 ON 9/26/89.

[65] FARLEY 2 DOCKET 50-364 LER 89-011
SURVEILLANCE NOT PERFORMED ON CONTAINMENT AIR LOCK DUE TO PERSONNEL ERROR.
EVENT DATE: 092589 REPORT DATE: 102589 NSSS: WE TYPE: PWR

(NSIC 215736) AT 0300 ON 9/25/89, IT WAS DISCOVERED THAT THE TEC SPEC 4.6.1.3.A SURVEILLANCE REQUIREMENT TO DEMONSTRATE OPERABILITY OF THE CONTAINMENT AIR LOCK OUTER DOOR HAD NOT BEEN PERFORMED AS REQUIRED FOLLOWING MULTIPLE CONTAINMENT ENTRIES. THE FIRST CONTAINMENT ENTRY OCCURRED AT 1346 ON 9/20/89 AND MULTIPLE CONTAINMENT ENTRIES ENSUED. THIS EVENT WAS CAUSED BY PERSONNEL ERROR IN THAT SHIFT SUPERVISORY PERSONNEL FAILED TO SCHEDULE FNP-2-STP-15.0 (CONTAINMENT AIR LOCK DOOR SEAL OPERABILITY TEST) TO BE PERFORMED WHEN THE CONDITIONS REQUIRING ITS PERFORMANCE OCCURRED. IN ADDITION, THE DAILY SHIFT FOREMEN REVIEW OF THE SURVEILLANCE SCHEDULE FAILED TO IDENTIFY THE PLANT CONDITIONS AS REQUIRING PERFORMANCE OF STP-15.0. INDIVIDUALS INVOLVED HAVE BEEN COUSELED REGARDING THEIR RESPONSIBILITIES AND THE IMPORTANCE OF IDENTIFYING CONDITIONAL SURVEILLANCE REQUIREMENTS. ADMINISTRATIVE CONTROLS WILL BE STRENGTHENED TO ASSIST SUPERVISORY PERSONNEL IN RECOGNIZING THE NEED FOR PERFORMING STP-15.0 WHEN THE CONDITIONS WARRANT. ADDITIONALLY, OPERATIONS ON-SHIFT PERSONNEL WILL BE BRIEFED ON THE LESSONS LEARNED AND THE IMPORTANCE OF THOROUGH REVIEW OF SURVEILLANCE SCHEDULES.

[66] FERMI 2 DOCKET 50-341 LER 89-016 REV 01
UPDATE ON RESIDUAL HEAT REMOVAL SERVICE WATER COOLING TOWER FAN BRAKES INOPERABLE DUE TO LOW NITROGEN PRESSURE.
EVENT DATE: 071189 REPORT DATE: 100689 NSSS: GE TYPE: BWR
VENDOR: MARLEY CO., THE

(NSIC 215566) ON JULY 11, 1989, DIVISION I OF THE RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM (RHRSW) WAS DECLARED INOPERABLE DUE TO LOW NITROGEN PRESSURE FOR ONE OF THE MECHANICAL DRAFT COOLING TOWER FAN BRAKES. THE NITROGEN SUPPLIES THE MOTIVE FORCE FOR THE BRAKES. THE BRAKES WERE INSTALLED TO PROTECT THE FANS FROM OVERSPEEDING IN THE EVENT OF A DESIGN BASIS TORNADO. DURING THE INVESTIGATION AND REPAIR OF THE CONDITION, IT WAS NOTED THAT CONTROL POWER TO THE BRAKES WAS BEING MAINTAINED IN THE AC SUPPLY POSITION RATHER THAN THE DC SUPPLY POSITION AS SHOWN ON THE DRAWING. LACK OF NITROGEN PRESSURE WOULD HAVE PREVENTED THE BRAKE FROM FULFILLING ITS DESIGN FUNCTION IN THE EVENT A TORNADO HAD OCCURRED. A REVIEW OF THE ORIGINAL ANALYSIS AND A MORE REALISTIC EVALUATION OF THE NEED FOR THE MECHANICAL FAN BRAKES HAS BEEN CONDUCTED. THE CONCLUSIONS FROM THIS EVALUATION HAVE REAFFIRMED THE NECESSITY OF THESE BRAKES TO PROTECT THE FAN FROM OVERSPEED IN THE EVENT OF A DESIGN BASIS TORNADO. THE NITROGEN SUPPLY WAS RESTORED AND A LEAKY HOSE REPLACED. THE SYSTEM OPERATING PROCEDURE WAS REVISED AND THE POWER SUPPLY PROPERLY ALIGNED. AS AN ENHANCEMENT, THE DESIGN OF THE BRAKE SYSTEM IS BEING UPGRADED TO QA1 STATUS. THIS MODIFICATION WILL BE COMPLETED DURING THE FIRST REFUELING OUTAGE, WHICH IS CURRENTLY UNDERWAY.

[67] FERMI 2 DOCKET 50-341 LER 89-021
LOCAL LEAK RATE TESTING EXCEEDS TECH SPEC LIMITS.
EVENT DATE: 090689 REPORT DATE: 100689 NSSS: GE TYPE: BWR

(NSIC 215567) PERIODIC LEAKAGE RATE TESTING OF CONTAINMENT ISOLATION VALVES AND PENETRATIONS ARE BEING PERFORMED IN ACCORDANCE WITH THE REQUIREMENTS CONTAINED IN TECHNICAL SPECIFICATION 3.6.1.2 AND 10 CFR 50 APPENDIX J. DURING THE PERFORMANCE OF THIS TESTING, SEVERAL VALVES HAVE EXCEEDED THEIR ADMINISTRATIVE ALLOWABLE LEAKAGE RATE AND THEIR COMBINED LEAKAGE EXCEEDS THE LIMITS AS DEFINED IN THE SUBJECT TECHNICAL SPECIFICATION LIMITING CONDITION FOR OPERATION. ALL CONTAINMENT ISOLATION VALVES THAT HAVE EXCEEDED THEIR ADMINISTRATIVE INDIVIDUAL ALLOWABLE LEAKAGE RATE HAVE HAD WORK REQUESTS GENERATED TO DISASSEMBLE THE SUBJECT VALVES AND REPAIR OR REWORK THE VALVE(S) AS CONDITIONS DICTATE. A FULL REPORT WILL BE MADE IN A SUPPLEMENT TO THIS LER WITHIN 30 DAYS AFTER THE COMPLETION OF THE TYPE A, B AND C TESTING/RETESTING.

[68] FERMI 2 DOCKET 50-341 LER 89-022
REACTOR PROTECTION SYSTEM ACTUATIONS AND REACTOR BUILDING HEATING VENTILATION AND AIR CONDITIONING ISOLATION DUE TO PERSONNEL ERROR.
EVENT DATE: 092389 REPORT DATE: 102389 NSSS: GE TYPE: BWR

(NSIC 215734) ENGINEERING DESIGN PACAKAGE (EDP) 10127 WAS BEING WORKED TO PROVIDE REDUNDANT MANUAL TRIP LOGIC IN THE REACTOR PROTECTION SYSTEM (RPS). DIVISION 1 RPS LOGIC HAD BEEN DE-ENERGIZED TO FACILITATE THE MODIFICATION AND AS A RESULT, THE DIVISION 1 PORTION OF THE RPS LOGIC WAS ALREADY TRIPPED. ON 9/23 AND 24, 1989, RPS LOGIC (ONLY) ACTUATIONS OCCURRED WHEN FUSE C71A-F15B BLEW COMPLETING THE LOGIC FOR THE RPS ACTUATIONS. BOTH EVENTS OCCURRED WHEN ELECTRICIANS ACCIDENTLY GROUNDED/SHORTED LEADS AT PANEL H11-P609. ON 9/28, AT 0303 HOURS, REACTOR BUILDING HEATING VENTILATION AND AIR CONDITIONING (RBHVAC) ISOLATED WHEN A JUMPER WAS LIFTED PER AN INCORRECT WORK PLAN INSTRUCTION FOR THE SAME EDP. CONTROL RODS WERE ALREADY INSERTED FOR ALL THESE EVENTS. FUEL WAS STILL IN THE VESSEL ON THE FIRST EVENT. THE REACTOR WAS DEFUELED ON THE SECOND AND THIRD EVENTS. ALL OF THE EVENTS INVOLVED PERSONNEL ERRORS. THE RPS ACTUATIONS WERE THE RESULT OF POOR WORK PRACTICES. THE RBHVAC ISOLATION WAS THE RESULT OF POOR WORK PLANNING PREPARATION. THE IMMEDIATE CORRECTIVE ACTION WILL BE TO ISSUE THIS LICENSEE EVENT REPORT (LER) AS REQUIRED READING. AS A LONG TERM CORRECTIVE ACTION, PROCEDURES WILL BE REVISED TO PROVIDE IMPROVED GUIDANCE FOR INSTALLATION OF EDPS AND PLANNING OF WORK ACTIVITIES.

[69] FERMI 2 DOCKET 50-341 LER 89-023
LOSS OF POWER TO DIVISION I DUE TO PERSONNEL ERROR BY RELAY DIVISION.
EVENT DATE: 092489 REPORT DATE: 102489 NSSS: GE TYPE: BWR

(NSIC 215735) ON 9/24/89, DETROIT EDISON RELAY PERSONNEL WERE PERFORMING A CHECK OF THE 13.2 KV SWITCHGEAR WHILE THE PLANT WAS DEFUELED. WHILE TESTING A RELAY STRING, PERSONNEL INADVERTENTLY CAUSED THE TRIP OF ESSENTIAL SAFETY SYSTEM BUSES 64B AND 64C. THIS CAUSED VARIOUS ACTUATIONS/ISOLATIONS OF ENGINEERED SAFETY FEATURES, INCLUDING THE START AND LOADING OF EMERGENCY DIESEL GENERATOR 11. IN ADDITION, POWER TO OTHER DIVISION I EQUIPMENT, INCLUDING THE "A" FUEL POOL COOLING PUMP, WAS LOST. OPERATORS TOOK PROMPT ACTIONS TO RESTORE POWER TO THE NECESSARY COMPONENTS AND SYSTEMS. THIS EVENT WAS CAUSED BY THE FAILURE OF THE PERSONNEL INVOLVED TO REVIEW THE APPLICABLE PRINTS PRIOR TO PERFORMING THE TEST. THIS EVENT WAS REVIEWED BY THE INVOLVED PERSONNEL AND THEIR MANAGEMENT. ANY DISCIPLINARY ACTION WARRANTED WILL BE ADMINISTERED IN ACCORDANCE WITH COMPANY POLICY. IN ORDER TO DISSEMINATE THE LESSONS LEARNED DURING THIS EVENT, IT WAS TO BE REVIEWED DURING THE OCTOBER RELAY MONTHLY MEETING.

[70] FITZPATRICK DOCKET 50-333 LER 88-008 REV 01
UPDATE ON EXCESSIVE LEAKAGE OF PRIMARY CONTAINMENT ISOLATION VALVES.
EVENT DATE: 090188 REPORT DATE: 102389 NSSS: GE TYPE: BWR
VENDOR: POWELL, WILLIAM COMPANY, THE

(NSIC 215690) DURING THE 1988 REFUEL OUTAGE, ONE PRIMARY CONTAINMENT (NM) PENETRATION EXCEEDED THE TECHNICAL SPECIFICATION 4.7.A.2.B.(2) LIMIT OF 0.6 LA (3,216 STANDARD CUBIC FEET PER DAY) WHEN SUBJECTED TO LOCAL LEAK RATE TESTING. THE LEAKING PENETRATION VALVES WERE IN THE HIGH PRESSURE COOLING INJECTION (BJ) TURBINE EXHAUST LINE. LEAKAGE WAS ATTRIBUTED TO WEAR. CORRECTIVE ACTION WAS TO REPLACE BOTH LEAKING VALVES. REPLACEMENT WAS PLANNED PRIOR TO THE OUTAGE AS PART OF A COMPREHENSIVE LEAKAGE REDUCTION PROGRAM. SAFETY SIGNIFICANCE AND CONSEQUENCES ARE JUDGED TO BE VERY SMALL BECAUSE THE PATHWAY OF POTENTIAL RELEASES INCLUDES FILTERS, DILUTION, DELAY, AND ELEVATED RELEASE FROM THE PLANT STACK (VL) FOR ACCIDENTS DISCUSSED IN THE FINAL SAFETY ANALYSIS REPORT. LER-87-001, 85-008, 83-002, 81-078, AND 80-050 ARE SIMILAR PREVIOUS EVENTS.

[71] FITZPATRICK DOCKET 50-333 LER 88-014 REV 01
UPDATE ON POTENTIAL COMMON CAUSE FAILURE OF CIRCUIT BREAKERS DUE TO PROCEDURE DEFICIENCY.
EVENT DATE: 120588 REPORT DATE: 100589 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215503) IN (BI) WHILE SHUTDOWN FOR REFUEL, MODIFICATION, AND MAINTENANCE, INSPECTION OF 4160 VAC CIRCUIT BREAKERS REVEALED A POSSIBLE COMMON CAUSE FAILURE AS A RESULT OF A PROCEDURAL DEFICIENCY BY A VENDOR PERFORMING CIRCUIT BREAKER OVERHAUL. THE DEFICIENCY CAUSED FAILURE OF A CIRCUIT BREAKER FOR RESIDUAL HEAT REMOVAL SERVICE WATER (BI) PUMP D ON 11/5/88. ON 12/5/88, INSPECTION REVEALED MISALIGNMENT THAT COULD POSSIBLY HAVE RESULTED IN FUTURE FAILURE OF OTHER CIRCUIT BREAKERS IN BOTH CLASS 1E (EB) AND NON-CLASS 1E (EA) POWER SYSTEMS. THE ACTUAL EFFECT OF THE FAILURE WAS MINIMAL BECAUSE THE PLANT WAS NOT OPERATING. FAILURE DURING OPERATION WAS POSSIBLE BUT NOT LIKELY DUE TO TECHNICAL SPECIFICATIONS LIMITATIONS ON OPERATION OF THE PLANT WITH INOPERABLE COMPONENTS. THE FAILED CIRCUIT BREAKER HAD FAILED TO TRIP ON OPERATOR DEMAND, PROBABLY DUE TO TRIP LINKAGE BINDING, AS A RESULT OF IMPROPER ASSEMBLY OF A PORTION OF THE LINKAGE FOLLOWING MAINTENANCE BY THE MANUFACTURER. ADJUSTMENT OF THE LINKAGE CORRECTED THE DEFICIENCY. PROCEDURE CHANGES WILL REDUCE THE PROBABILITY OF RECURRENCE. THERE HAVE NOT BEEN ANY SIMILAR EVENTS AT THIS FACILITY.

[72] FITZPATRICK DOCKET 50-333 LER 89-015
NINE AIR OPERATED CONTAINMENT ISOLATION VALVES EXHIBIT OPERATIONAL DEFICIENCIES DUE TO PACKING PROBLEMS & IRON BUILD-UP IN REACTOR BUILDING CLOSED LOOP COOLING SYSTEM.
EVENT DATE: 091889 REPORT DATE: 101889 NSSS: GE TYPE: BWR
VENDOR: HAMMEL DAHL

(NSIC 215681) ON SEPTEMBER 18, 1989 DURING A SCHEDULED OUTAGE AND PERFORMANCE OF A SCHEDULED ASME, SECTION XI IN-SERVICE TEST, TWO OF NINE REMOTE MANUALLY OPERATED DIAPHRAGM AIR OPERATED CONTAINMENT ISOLATION VALVES (ISV) ON THE REACTOR BUILDING CLOSED LOOP COOLING WATER SYSTEM (CC) FAILED THE ACCEPTANCE CRITERIA FOR VALVE CLOSING TIME. ONE VALVE WOULD NOT CLOSE EXCEPT BY MANUAL OPERATION. ANOTHER VALVE CLOSING TIME EXCEEDED THE CRITERIA BY 0.4 SECONDS OR 3.6 PERCENT. SUBSEQUENT OUTAGE OPERATIONS REVEALED COMMON PROBLEMS WITH THE SEVEN VALVES WHICH INITIALLY PASSED THE CLOSING TIME TEST. VALVES APPEARED TO BE BINDING DUE PRINCIPALLY TO THE BUILDUP AND HARDENING OF IRON OXIDE SLUDGE IN THE VALVE OPERATING INTERNALS ALTHOUGH THERE IS SOME INDICATION THAT THE ORIGINAL PACKING MAY HAVE CONTRIBUTED TO THE PROBLEM. CORRECTIVE ACTION: DISASSEMBLED AND CLEANED ALL NINE VALVES AND CHANGED PACKING FROM SEVEN RING GRAFOIL TO LIVE LOADED FIVE RING TYPE ON FIVE VALVES. THREE OTHER VALVES HAD PREVIOUSLY BEEN CHANGED TO THIS TYPE OF PACKING IN 1988. LONG-TERM CORRECTIVE ACTION WILL FLUSH SYSTEM TO REDUCE IRON OXIDE BUILDUP AND INVESTIGATE POSSIBLE CHANGES TO INTERNAL VALVE TRIM WHICH IS LESS SUSCEPTIBLE TO CORROSION PRODUCT ACCUMULATION. LERS WITH COMMON ELEMENTS: 88-005, 88-009, AND 86-003.

[73] FITZPATRICK DOCKET 50-333 LER 89-016
REACTOR SCRAM WHILE SHUTDOWN DURING REACTOR PROTECTION SYSTEM POWER SUPPLY TRANSFER DUE TO LACK OF CAUTION STATEMENT IN PROCEDURE.
EVENT DATE: 091889 REPORT DATE: 101889 NSSS: GE TYPE: BWR

(NSIC 215682) ON SEPTEMBER 18, 1989 WITH THE PLANT SHUTDOWN FOR A PLANNED MAINTENANCE OUTAGE, THE POWER SUPPLY FOR THE REACTOR PROTECTION SYSTEM (RPS)(VC) BUS WAS TRANSFERRED FROM NORMAL POWER SUPPLY FROM THE RPS MOTOR GENERATOR SETS (MG)(MG) TO THE ALTERNATE POWER SUPPLY IN ACCORDANCE WITH AN APPROVED PROCEDURE. THE TRANSFER WAS NECESSARY TO PERMIT PLANNED MAINTENANCE WORK ON THE RPS MG SETS. ALTHOUGH THE CONTROL RODS WERE ALREADY FULLY INSERTED, A SCRAM SIGNAL WAS AUTOMATICALLY INITIATED DURING THE BUS TRANSFER. THE PROCEDURE PROVIDES FOR TRANSFER OF THE A RPS POWER WHICH GENERATES A HALF SCRAM DUE TO THE MOMENTARY LOSS OF POWER. THIS IS FOLLOWED BY MANUAL RESET OF THE A SIDE HALF SCRAM AND THEN TRANSFER OF THE B RPS POWER FOLLOWED BY MANUAL RESETTING OF THE RESULTANT B SIDE HALF SCRAM. THE PROCEDURE DID NOT PROVIDE GUIDANCE ON THE TIME INTERVAL TO BE ALLOWED BETWEEN BUS TRANSFER. THE RESET OF THE A SIDE WAS COMPLETED WITHIN FOUR SECONDS OF THE TRANSFER. IN THE NEXT FOUR SECONDS THE B SIDE WAS TRANSFERRED AND A FULL SCRAM RESULTED. THE MAIN STEAM LINE RADIATION MONITOR POWERED FROM THE A RPS BUS WAS NOT YET STABILIZED FOLLOWING THE TRANSFER AND INITIATED A FALSE HALF SCRAM SIGNAL LESS THAN A TENTH OF A SECOND BEFORE THE B SIDE WAS TRANSFERRED. THE COMBINATION OF SCRAM SIGNALS ON BOTH THE A AND B RPS CHANNELS RESULTED IN FULL SCRAM.

[74] FITZPATRICK DOCKET 50-333 LER 89-017
PRIMARY CONTAINMENT ISOLATION OF REACTOR BUILDING VENTILATION AND SHUTDOWN COOLING MODE OF RESIDUAL HEAT REMOVAL DUE TO HUMAN ERROR.
EVENT DATE: 092089 REPORT DATE: 102089 NSSS: GE TYPE: BWR

(NSIC 215683) THE PLANT WAS SHUTDOWN FOR A SCHEDULED MAINTENANCE OUTAGE. IT WAS NECESSARY TO ELECTRICALLY ISOLATE A WIRING TERMINAL STRIP TO REPLACE A TERMINAL BLOCK. THE PROTECTIVE TAGGING REQUEST (PTR) TO ACCOMPLISH THE ISOLATION WAS INDEPENDENTLY REVIEWED BY AN EXPERIENCED INSTRUMENT AND CONTROL TECHNICIAN AND TWO EXPERIENCED SENIOR LICENSED OPERATORS. ON SEPTEMBER 20, 1989 AT 2:05 A.M. A

FUSE WAS PULLED IN ACCORDANCE WITH THE PTR. THE ENGINEERED SAFETY FEATURE ACTUATION SYSTEM (JE) INITIATED A GROUP II PRIMARY CONTAINMENT ISOLATION SIGNAL RESULTING IN ISOLATION OF THE SHUTDOWN COOLING MODE OF THE RESIDUAL HEAT REMOVAL/LOW PRESSURE COOLANT INJECTION SYSTEM (BO) AND THE REACTOR BUILDING VENTILATION SYSTEM (VA). THE FUSE WAS REPLACED AND THE SYSTEMS WERE RESTARTED WITHIN FIVE MINUTES. THE CAUSE WAS COGNITIVE HUMAN ERROR IN NOT FINDING THE ISOLATION SIGNAL CONNECTION WHICH WAS PRESENT ON THE LOGIC DIAGRAMS AND WIRING DRAWING A CONTRIBUTING CAUSE MAY BE THE AWKWARD WIRING AND LOGIC CIRCUIT DRAWING SYSTEM THAT REQUIRES MULTIPLE CROSS-REFERENCES TO MULTIPLE DRAWINGS. THE RESPONSIBLE OPERATORS REVIEWED THE EVENT. THE IMPORTANCE OF THOROUGH PTR REVIEW WAS EMPHASIZED AND ENTERED IN THE NIGHT ORDERS FOR INSTRUCTION OF OTHER OPERATORS. LERS WITH RELATED ELEMENTS:89-013, 87- 016, AND 86-019.

[75] FT. CALHOUN 1 DOCKET 50-285 LER 89-012 REV 01
UPDATE ON FEEDWATER VALVE HCV-1386 INOPERABLE DUE TO MAINTENANCE PROGRAM DEFICIENCY.
EVENT DATE: 050289 REPORT DATE: 101389 NSSS: CE TYPE: PWR
VENDOR: LIMITORQUE CORP.

(NSIC 215640) AT 1610 HOURS ON MAY 2, 1989, PLANT MANAGEMENT OF FORT CALHOUN STATION DETERMINED THAT THE MAIN FEEDWATER ISOLATION VALVE TO THE "A" STEAM GENERATOR WAS INOPERABLE DUE TO AN IMPROPERLY SET TORQUE SWITCH ON THE VALVE'S MOTOR OPERATOR. THE VALVE WOULD NOT HAVE CLOSED COMPLETELY BEFORE BEING TRIPPED BY THE SWITCH DURING CERTAIN ACCIDENT CONDITIONS' AND WAS THEREFORE OUTSIDE THE DESIGN BASIS OF THE PLANT. AT THE TIME OF THE DETERMINATION THE PLANT WAS OPERATING AT 10% POWER AND PREPARING TO GO BACK ON-LINE FOLLOWING A 3 DAY OUTAGE. IN ACCORDANCE WITH 10 CFR 50.72(B)(1)(II)(B), THE NRC OPERATIONS CENTER WAS NOTIFIED AT 1625 HOURS ON THE SAME DAY. THIS LER IS SUBMITTED PURSUANT TO 10 CFR 50.73(A)(2)(II)(B). THE CAUSE OF THIS EVENT IS AN INADEQUATE PROGRAM FOR MAINTENANCE OF MOTOR OPERATED VALVES. THE TORQUE SWITCHES ON THIS VALVE AND ON THREE IDENTICAL VALVE OPERATORS HAVE BEEN RESET TO ENSURE ADEQUATE CLOSING THRUST IS AVAILABLE. AN EXTENSIVELY IMPROVED MOTOR-OPERATED VALVE MAINTENANCE PROGRAM PLAN HAS BEEN APPROVED AND IS BEING IMPLEMENTED.

[76] FT. CALHOUN 1 DOCKET 50-285 LER 89-019
MANUAL UNIT TRIP DUE TO HIGH INDICATED RCP MOTOR BEARING TEMPERATURE.
EVENT DATE: 092489 REPORT DATE: 102489 NSSS: CE TYPE: PWR

(NSIC 215697) ON 9/24/89, FORT CALHOUN STATION UNIT 1 WAS OPERATING AT APPROXIMATELY 70% POWER IN MODE 1. AT 1259 HOURS INDICATION OF HIGH TEMPERATURE FOR REACTOR COOLANT PUMP RC-3A UPPER MOTOR THRUST BEARING WAS RECEIVED IN THE CONTROL ROOM. AFTER MAXIMIZATION OF BOTH THE COOLING WATER FLOW TO THE OIL COOLERS AND THE OIL FLOW TO THE BEARING FAILED TO REDUCE THE INDICATED TEMPERATURE, THE SHIFT SUPERVISOR INITIATED A CONTROLLED PLANT SHUTDOWN AT 1320 HOURS. AT 1518 HOURS, WITH REACTOR POWER BETWEEN 5 AND 6 PERCENT, RC-3A THRUST BEARING TEMPERATURE INDICATION SPIKED TO 267 DEGREES. THE REACTOR WAS IMMEDIATELY MANUALLY TRIPPED AND RC-3A WAS SHUT DOWN. THIS TRIP IS REPORTABLE PURSUANT TO 10 CFR 50.73(A)(2)(IV). INVESTIGATION REVEALED THE CAUSE OF THE INDICATED HIGH TEMPERATURE TO BE DAMAGED CABLE FOR THE BEARING RESISTIVE TEMPERATURE DEVICE (RTD), ALTHOUGH CAUSE OF THE DAMAGE COULD NOT BE DETERMINED. THERE WAS NO EVIDENCE OF ACTUAL EXCESSIVE BEARING TEMPERATURE. THE DAMAGED CABLE AND THE RTD WERE REPLACED. OTHER SIMILAR RTD WIRING WILL BE INSPECTED AND INVESTIGATION INTO THE CAUSE OF THE CABLE DAMAGE WILL BE COMPLETED DURING THE 1990 REFUELING OUTAGE.

[77] GINNA DOCKET 50-244 LER 89-011
SPURIOUS ACTUATION OF CONTAINMENT VENTILATION ISOLATION.
EVENT DATE: 092089 REPORT DATE: 102089 NSSS: WE TYPE: PWR

(NSIC 215678) ON SEPTEMBER 20, 1989 AT 0519 EDST WITH THE REACTOR AT APPROXIMATELY 99% FULL POWER, AN INADVERTENT CONTAINMENT VENTILATION ISOLATION OCCURRED DUE TO A SPURIOUS EVENT. ALL CONTAINMENT VENTILATION ISOLATION VALVES THAT WERE OPEN, CLOSED AS DESIGNED. IMMEDIATE OPERATOR ACTION WAS TO PERFORM THE APPLICABLE ALARM RESPONSE PROCEDURES ACTIONS. THIS INCLUDED VERIFYING AUTOMATIC ACTIONS, ATTEMPTING TO DETERMINE CAUSE OF CONTAINMENT VENTILATION ISOLATION, AND MAKING APPROPRIATE NOTIFICATIONS. THE IMMEDIATE CAUSE OF THE EVENT WAS DETERMINED TO BE SPURIOUS. CORRECTIVE ACTION TAKEN WAS TO RETURN THE CONTAINMENT VENTILATION ISOLATION SYSTEM TO SERVICE FOLLOWED BY A TROUBLESHOOTING EFFORT BY THE INSTRUMENT AND CONTROL DEPARTMENT. FURTHER INVESTIGATION TO DETERMINE THE ROOT CAUSE IS CONTINUING.

[78] GRAND GULF 1 DOCKET 50-416 LER 89-008 REV 02
UPDATE ON FAILURE OF ISOLATION DAMPERS TO CLOSE WITHIN SPECIFIED TIME LIMIT.
EVENT DATE: 052389 REPORT DATE: 103189 NSSS: GE TYPE: BWR
VENDOR: BETTIS CORPORATION
PARKER HANNIFIN CORP.

(NSIC 215780) ON MAY 23, 1989 TWO REDUNDANT SECONDARY CONTAINMENT ISOLATION DAMPERS, Q1T42F019 AND Q1T42F020, FAILED TO CLOSE WITHIN THE 4 SECOND TIME LIMIT OF TECHNICAL SPECIFICATION 3.6.6.2. IT WAS SUSPECTED THAT THE INTERNAL DISK OF EACH EXHAUST VALVE, WHICH SHUTTLES TO VENT OR ADMIT AIR ON THE ACTUATOR CYLINDER FAILED TO PROPERLY LIFT TO VENT AIR PRESSURE DURING INITIAL ATTEMPTS TO CLOSE THE DAMPERS. THE EXHAUST VALVE ASSOCIATED WITH THE ACTUATOR OF EACH DAMPER WAS REPLACED. SERI WILL REMOVE THESE TYPE EXHAUST VALVES ON ALL SAFETY-RELATED BETTIS ACTUATORS AND REPLACE EXISTING SOLENOID VALVES WITH VALVES HAVING LARGER ORIFICES AND AIR TUBING TO ACHIEVE THE SAME REQUIRED STROKE TIME. ON JUNE 16, 1989, THE Q1T42F020 DAMPER AGAIN FAILED TO CLOSE WITHIN THE 4 SECOND TIME LIMIT. THE DAMPER ACTUATOR WAS DISASSEMBLED AND NO OBVIOUS FAILURES OR DEFECTS WERE OBSERVED. THE ACTUATOR CONTAINED MOBILGREASE 28 WHICH WAS PREVIOUSLY IDENTIFIED IN A 10CFR21 REPORT AS HAVING A TENDENCY TO CAUSE SEALS TO SWELL AND PRODUCE SLOWER STROKE TIMES. HOWEVER, THIS MODEL OF ACTUATOR WAS NOT PROJECTED TO DEGRADE IN STROKING TIME. THIS WAS NOT INITIALLY CONSIDERED A POTENTIAL CONTRIBUTOR SINCE THE Q1T42F019 HAD BEEN REFURBISHED AND LUBRICATED WITH THE APPROVED MOLYKOTE 44 DURING THE THIRD REFUELING OUTAGE. GRAND GULF HAS A 5 YEAR PROGRAM TO REFURBISH BETTIS ACTUATORS WHICH ENSURES THAT MOLYKOTE 44 IS USED.

[79] GRAND GULF 1 DOCKET 50-416 LER 89-014
RWCU ISOLATION DURING SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 092989 REPORT DATE: 102789 NSSS: GE TYPE: BWR

(NSIC 215781) ON SEPTEMBER 29, 1989 DURING PERFORMANCE OF AN ANNUAL SURVEILLANCE, THE REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATED ON A SIMULATED STEAM LINE TUNNEL HIGH TEMPERATURE SIGNAL DUE TO PERSONNEL ERROR. THE PERFORMANCE OF THE SURVEILLANCE RESULTS IN ISOLATION SIGNALS TO BOTH THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM ISOLATION VALVES AND THE RWCU SYSTEM ISOLATION VALVES. THE ISOLATIONS ARE PREVENTED BY PLACING THE DIVISIONAL RCIC BYPASS SWITCH AND THE DIVISIONAL RWCU BYPASS SWITCH TO THE BYPASS POSITION. HOWEVER, WHEN THE SURVEILLANCE PROCEDURE DIRECTED THE OPERATOR TO PLACE THE "A" RCIC BYPASS SWITCH TO BYPASS, HE MISTAKENLY PLACED THE "A" RWCU BYPASS SWITCH TO NORMAL. THIS IMMEDIATELY CAUSED AN RWCU ISOLATION. THE TWO SWITCHES ARE ADJACENT ON THE SAME PANEL. AS A RESULT OF THIS EVENT AND TWO OTHER RECENT EVENTS INVOLVING HUMAN PERFORMANCE ERRORS, THE PLANT GENERAL MANAGER HELD BRIEFINGS WITH ALL AVAILABLE OPERATIONS AND INSTRUMENTATION/CONTROL SUPERVISORS AND SUPERINTENDENTS INFORMING THEM OF THE INCIDENTS AND THE KEY ELEMENTS THAT COULD HAVE PREVENTED THE

OCCURRENCES: ATTENTION TO DETAIL, CONCISE VERBAL COMMUNICATIONS, AND SELF-VERIFICATION. THIS WAS ALSO REITERATED TO ALL PLANT PERSONNEL IN A SPECIAL PLANT INFORMATION NEWSLETTER.

[80] HATCH 2 DOCKET 50-366 LER 89-006
PERSONNEL ERROR RESULTS IN AN INADEQUATE PROCEDURE AND MISSED SURVEILLANCE.
EVENT DATE: 092689 REPORT DATE: 102389 NSSS: GE TYPE: BWR

(NSIC 215702) ON 9/26/89 AT APPROX. 1400 CDT, UNIT 2 WAS IN THE REFUEL MODE WITH THE REACTOR PRESSURE VESSEL HEAD REMOVED, THE CAVITY FLOODED, AND ALL FUEL REMOVED FROM THE CORE. AT THAT TIME, NON-LICENSED PERSONNEL DETERMINED PROCEDURE 34SV-SUV-019-2S, "SURVEILLANCE CHECKS," DID NOT FULLY IMPLEMENT THE REQUIREMENTS OF UNIT 2 TECH SPECS TABLE 4.3.2-1, ITEM 1.G, AND TABLE 4.3.6.4-1, ITEM 12. SPECIFICALLY, SINCE 11/11/88, THE CHANNEL CHECK OF DRYWELL HIGH RANGE RADIATION INDICATORS 2D11-K621A AND B WAS BEING PERFORMED ONCE PER SEVEN DAYS INSTEAD OF THE REQUIRED ONCE PER DAY, AND A MONTHLY CHANNEL CHECK WAS NOT BEING PERFORMED FOR THE DRYWELL RADIATION PARAMETER OF RECORDERS 2T48-R601A AND B. THE INDICATORS WERE BEING CALIBRATED AND FUNCTIONALLY TESTED AT THEIR REQUIRED FREQUENCIES. THE ROOT CAUSE OF THIS EVENT IS COGNITIVE PERSONNEL ERROR IN THAT NONLICENSED PERSONNEL FAILED TO ADEQUATELY INCORPORATE TECHNICAL SPECIFICATIONS REQUIREMENTS INTO A MAJOR REVISION OF A PLANT PROCEDURE MADE EFFECTIVE ON 11/11/88 AND FAILED TO IDENTIFY THE CONDITION IN SUBSEQUENT PROCEDURE REVIEWS. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDE COUNSELING INVOLVED PERSONNEL AND REVISING PROCEDURE 34SV-SUV-019-2S PRIOR TO UNIT 2 STARTUP TO ENSURE PERFORMANCE OF THE SURVEILLANCE WHEN THE INSTRUMENTS ARE REQUIRED TO BE OPERABLE.

[81] HATCH 2 DOCKET 50-366 LER 89-007
SAFETY RELIEF VALVES WITH PH13-8MO PILOT VALVE DISCS EXPERIENCE SETPOINT DRIFT.
EVENT DATE: 092689 REPORT DATE: 102389 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 215703) ON 9/26/89, AT APPROXIMATELY 1200 EDT, UNIT 2 WAS IN THE REFUEL MODE AT AN APPROXIMATE POWER LEVEL OF 0 MWT (APPROXIMATELY 0% OF RATED THERMAL POWER . AT THAT TIME PLANT ENGINEERING PERSONNEL RECEIVED WRITTEN NOTIFICATION OF THE RESULTS OF OFF-SITE TESTING OF PRESSURE VESSEL SAFETY RELIEF VALVES (SRVS). OF THE ELEVEN SRVS, FOUR HAD EXHIBITED DRIFT IN THE MECHANICAL LIFT SETPOINTS IN EXCESS OF THE +/- 3% TOLERANCE SPECIFIED BY INSERVICE TESTING (IST) REQUIREMENTS. THIS VOLUNTARY REPORT IS BEING SUBMITTED DUE TO THE POTENTIAL INDUSTRY INTEREST IN THIS EVENT SINCE THREE OF THE FOUR REFERENCED SRVS HAD PILOT VALVE DISCS OF PH13-8MO WHICH IS THE NEW DISC MATERIAL BEING TESTED BY THE BOILING WATER REACTOR OWNERS' GROUP (BWROG) TO REDUCE SETPOINT DRIFT. THE EXPERIENCED SETPOINT DRIFT WAS WELL WITHIN THE ANALYTICAL LIMITS EXISTING FOR REACTOR VESSEL OVERPRESSURE PROTECTION. THE ROOT CAUSE OF THE EVENT IS BEING INVESTIGATED IN COOPERATION WITH THE BWROG EFFORT. THE EXPERIENCED SETPOINT DRIFT IN THIS EVENT IS INCONSISTENT WITH PREVIOUS INDUSTRY DATA SHOWING AVERAGE SETPOINT DRIFT OF ABOUT 50% LESS WITH THIS NEW DISC MATERIAL. CORRECTIVE ACTIONS FOR THIS EVENT THUS FAR INCLUDED REFURBISHING THE VALVES AND CONTINUING THE ROOT CAUSE INVESTIGATION.

[82] INDIAN POINT 3 DOCKET 50-286 LER 89-012 REV 01
UPDATE ON CROSSWIRING OF A HOT LEG AND COLD LEG CHANNEL TEST SWITCH DURING A MODIFICATION.
EVENT DATE: 062489 REPORT DATE: 103189 NSSS: WE TYPE: PWR

(NSIC 215746) ON JUNE 24, 1989 AT 1400 HOURS WHILE COMMENCING LOAD ESCALATION FROM TWENTY-FIVE PERCENT REACTOR POWER, CONTROL ROOM OPERATORS OBSERVED 34 LOOP DELTA TEMPERATURE INDICATION READING LOW AND 34 LOOP AVERAGE REACTOR COOLANT TEMPERATURE READING APPROXIMATELY THREE (3) DEGREES HIGHER THAN NORMAL. THE

INSTRUMENT AND CONTROL DEPARTMENT INSPECTED THE LOOP 34 INSTRUMENTATION PANEL AND FOUND ONE HOT LEG TEMPERATURE CHANNEL TEST SWITCH AND THE COLD LEG TEMPERATURE CHANNEL TEST SWITCH CROSS-WIRED. THE WIRES WERE RECONNECTED TO THE CORRECT SWITCHES AND THE INSTRUMENTS WERE RETURNED TO NORMAL OPERATION. THE ROOT CAUSES FOR THIS EVENT ARE IDENTIFIED AS SEVERAL PERSONNEL ERRORS. THESE ERRORS INCLUDE INADEQUATE INSTALLATION WORKMANSHIP, INADEQUATELY PERFORMED POST INSTALLATION TESTING AND INADEQUATE QUALITY CONTROL. TO IDENTIFY AND CORRECT THE UNDERLYING CAUSES OF THIS EVENT, A STUDY HAS BEEN CONDUCTED COMPARING THE ERRORS IDENTIFIED AS CONTRIBUTING TO THIS EVENT, TO ERRORS FOUND IN SIMILAR EVENTS. THIS STUDY AND ITS RESULTS WERE PRESENTED TO THE MODIFICATION CONTROL MANUAL COMMITTEE FOR INCORPORATION INTO THE MCM PROCEDURES. IN ADDITION, A DEDICATED COMMITTEE HAS BEEN ASSEMBLED TO DEFINE THE WORK PROCESSES THAT LED TO THESE EVENTS AND RECOMMEND SOLUTIONS TO PREVENT RECURRENCE.

[83] LA SALLE 1 DOCKET 50-373 LER 89-023
SPURIOUS CLOSURE OF REACTOR RECIRCULATION PROCESS SAMPLE INBOARD ISOLATION STOP VALVE DUE TO FAILED RELAY.
EVENT DATE: 090889 REPORT DATE: 100589 NSSS: GE TYPE: BWR
VENDOR: AGASTAT RELAY CO.

(NSIC 215523) AT 0956 HOURS ON 9/8, WHILE IN OPERATING CONDITION 1, AT 100% POWER, 1B33-F019, REACTOR RECIRCULATION PROCESS SAMPLE INBOARD ISOLATION STOP VALVE, SPRUIOUSLY WENT CLOSED WITH NO APPARENT ISOLATION SIGNAL PRESENT. (1B33-F020, REACTOR RECIRCULATION PROCESS SAMPLE OUTBOARD ISOLATION STOP VALVE, WAS TAKEN OUT-OF-SERVICE (OOS) CLOSED IN ACCORDANCE WITH TECH SPEC 3.6.3. THE APPARENT CAUSE OF THE 1B33-F019 CLOSURE WAS DUE TO A SHORT IN THE 1B21H-K72 RELAY. THIS RELAY KEEPS THE 1B33-F019'S SOLENOID ENERGIZED AND CONSEQUENTLY, KEEPS THE VALVE OPEN. THROUGH THE EVENT, THE REDUNDANT SAMPLING FLOW PATH THROUGH THE REACTOR WATER CLEAN-UP INLET PROVIDED AN ALTERNATE SOURCE TO SATISFY THE REQUIREMENT OF THE WATER CHEMISTRY TECH SPEC 4.4.4. WORK REQUEST L92433 WAS WRITTEN TO TROUBLESHOOT AND REPAIR THE CAUSE OF THE CLOSURE OF THE 1B33-F019. RELAY 1B21H-K72 AND THE FUSE WERE REPLACED UNDER THIS WORK REQUEST. LASALLE SPECIAL TEST, LST-89-111 WAS WRITTEN TO VERIFY THAT THE NEW 1B21H-K72 RELAY'S RESPONSE TIME WAS WITHIN ACCEPTABLE LIMITS. THIS TASK WAS SUCCESSFULLY COMPLETED ON 9/12/89. ACTION ITEM RECORD 373-200-89-08201 WAS WRITTEN TO TRACK COMPLETION OF POSSIBLE LONG TERM CORRECTIVE ACTION ON AN AGASTAT REPLACEMENT/PREVENTATIVE MAINTENANCE SCHEDULE. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV) DUE TO AN ACTUATION OF AN ENGINEERED SAFETY FEATURE.

[84] LA SALLE 2 DOCKET 50-374 LER 89-012
SHUTDOWN COOLING ISOLATION DURING REACTOR PROTECTION SYSTEM BUS TRANSFER DUE TO RELAY DEENERGIZING.
EVENT DATE: 082989 REPORT DATE: 092289 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215320) ON 8/29/89, AT 1544 HOURS, WITH UNIT 2 IN OPERATIONAL CONDITION 4, WHILE TRANSFERRING THE B REACTOR PROTECTION SYSTEM (RPS) BUS FROM ITS ALTERNATE POWER SUPPLY BACK TO ITS NORMAL POWER SUPPLY IN ACCORDANCE WITH LASALLE OPERATING PROCEDURE LOP-RP-04, "RPS BUS B TRANSFER," A GROUP 6 (SHUTDOWN COOLING) ISOLATION OCCURRED. LOP-RP-04 DIRECTS THE NUCLEAR STATION OPERATOR (NSO) TO MANUALLY MAINTAIN RELAY 2B21H-K35B CLOSED DURING THE RPS BUS B TRANSFER IN ORDER TO MAINTAIN POWER TO THE PRIMARY CONTAINMENT ISOLATION LOGIC FOR VARIOUS PRIMARY CONTAINMENT ISOLATION VALVES. WHEN THE RELAY WAS DEPRESSED IN ACCORDANCE WITH LOP-RP-04, THE CONTACT PAIR ON THE RELAY WHICH MAINTAINS POWER TO THE SHUTDOWN COOLING OUTBOARD ISOLATION LOGIC OPENED, CAUSING THE ISOLATION. THE CAUSE OF THE MOMENTARY LOSS OF POWER TO THE SHUTDOWN COOLING ISOLATION LOGIC WAS DUE TO RELAY 2B21H-K35B BEING DEPRESSED OFF OF ITS CENTER. THIS RELAY IS PIVOTED AT ITS CENTER AND MOMENTARY OPENING OF IT CONTACTS IS POSSIBLE WHEN THE RELAY IS DEPRESSED ON ONE SIDE. BY 1550 HOURS ON 8/29/89, THE SHUTDOWN COOLING ISOLATION

WAS RESET AND SHUTDOWN COOLING WAS PLACED BACK INTO OPERATION. THE SAFETY CONSEQUENCES OF THIS EVENT WERE MINIMAL DUE TO ADHERENCE TO TECH SPEC 3.4.9.2, WHICH REQUIRES THAT SHUTDOWN COOLING BE REESTABLISHED WITHIN 1 HOUR OR AN ALTERNATE METHOD OF REACTOR COOLANT CIRCULATION BE ESTABLISHED.

[85] LA SALLE 2 DOCKET 50-374 LER 89-013
PRIMARY CONTAINMENT ISOLATION DURING INSTRUMENT SURVEILLANCE TESTING OF LIS-MS-401.
EVENT DATE: 090789 REPORT DATE: 100589 NSSS: GE TYPE: BWR

(NSIC 215524) ON 9/7/89 WITH UNIT 2 IN COLD SHUTDOWN (OPERATIONAL CONDITION 4) A GROUP I ISOLATION (CLOSES MAIN STEAM ISOLATION AND STEAM LINE DRAIN VALVES) WAS RECEIVED DURING THE PERFORMANCE OF LASALLE INSTRUMENT SURVEILLANCE LIS-MS-401, "UNIT 2 MAIN STEAM LINE LOW PRESSURE MSIV ISOLATION FUNCTIONAL TEST." THE GROUP I ISOLATION OCCURRED WHEN THE INSTRUMENT MAINTENANCE PERSONNEL DEPRESSURIZED ONE OF THE MAIN STEAM LINE LOW PRESSURE SWITCHES (PS-2B21-N015C). PRESSURE SWITCH 2B21-N015C WAS TESTED WITHOUT RESETTING THE HALF ISOLATION WHICH EXISTED AFTER TESTING ISOLATION LOGIC CHANNEL B1. THE UNIT WAS IN COLD SHUTDOWN WITH ALL MAIN STEAM LINE ISOLATION VALVES CLOSED. NO VALVE MOTOR OR TRANSIENT WAS CAUSED DUE TO THIS EVENT. THE SURVEILLANCE HAD BEEN RECENTLY REVISED TO SPLIT IT INTO TWO SEPARATE PARTS, ONE IF THE UNIT IS IN THE RUN MODE AND THE OTHER IF THE UNIT IS IN THE SHUTDOWN MODE. THE REVIEW FAILED TO IDENTIFY THE REQUIREMENT FOR RESETTING THE HALF ISOLATION WHEN ONE PART OF THE SURVEILLANCE WAS COMPLETED PRIOR TO TESTING THE OTHER HALF OF THE LOGIC. THE GROUP I ISOLATION WAS RESET AND THE SURVEILLANCE WAS COMPLETED WITHOUT FURTHER PROBLEMS. THIS REPORT IS BEING SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV) DUE TO THE ACTUATION OF AN ENGINEERED SAFETY FEATURE SYSTEM.

[86] LIMERICK 1 DOCKET 50-352 LER 88-018 REV 01
UPDATE ON CONTROL ROOM HVAC ISOLATION RESULTING FROM A HIGH CHLORINE CONCENTRATION SIGNAL CAUSED BY RAINWATER CONTACTING AN ANALYZER PROBE.
EVENT DATE: 051188 REPORT DATE: 103188 NSSS: GE TYPE: BWR

(NSIC 215772) ON 5/11/88, AT 1448 HOURS, MAIN CONTROL ROOM VENTILATION SYSTEM ISOLATED DUE TO A 'C' CHANNEL HIGH CHLORINE CONCENTRATION SIGNAL. THE 'A' TRAIN OF CONTROL ROOM EMERGENCY FRESH AIR SUPPLY (CREFAS) SYSTEM, AN ENGINEERED SAFETY FEATURE, INITIATED AS DESIGNED. THE EVENT OCCURRED DURING RAINY AND WINDY WEATHER CONDITIONS AND HIGH CHLORINE CONCENTRATION SIGNAL IS BELIEVED TO HAVE BEEN CAUSED BY RAINWATER COMING IN CONTACT WITH CHLORINE ANALYZER PROBE RESULTING IN A CHEMICAL IMBALANCE IN THE PROBE'S ELECTROLYTE. ANALYZER PROBES ARE LOCATED CLOSE TO OUTSIDE AIR INTAKE PLENUM. AFTER THE 'C' CHANNEL CHLORINE INDICATOR SPIKED, CONTROL ROOM OPERATORS IMPLEMENTED SPECIAL EVENT PROCEDURE SE-2 (TOXIC GAS PROCEDURE). A CHANNEL CHECK OF 'A', 'B' AND 'D' CHLORINE DETECTORS WAS PERFORMED BY OPERATIONS PERSONNEL AND VERIFIED TO BE NORMAL. FOLLOWING THE SPIKE ALL CHLORINE CHANNELS INDICATED NORMAL LEVELS (LESS THAN 0.1 PPM). THE ISOLATION WAS RESET AT 1552 HOURS. THE DURATION OF CONTROL ROOM ISOLATION WAS 1 HOUR 4 MINUTES. THERE WAS NO CHLORINE INTAKE TO THE CONTROL ROOM. THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL TO ENVIRONMENT AS A RESULT OF THIS EVENT. A MODIFICATION TO MOVE THE CHLORINE DETECTOR PROBES WAS IMPLEMENTED BY AUGUST 28, 1988 TO MITIGATE FALSE, ENVIRONMENTALLY RELATED, AUTOMATIC CONTROL ROOM VENTILATION SYSTEM ISOLATIONS.

[87] LIMERICK 1 DOCKET 50-352 LER 88-026 REV 01
UPDATE ON CONTROL ROOM HVAC ISOLATION RESULTING FROM A HIGH CHLORINE ISOLATION SIGNAL CAUSED BY RAIN WATER CONTACTING AN ANALYZER PROBE.
EVENT DATE: 071788 REPORT DATE: 103189 NSSS: GE TYPE: BWR

(NSIC 215773) ON 7/17/88, AT 1730 HOURS, A MAIN CONTROL ROOM VENTILATION SYSTEM

ISOLATION OCCURRED DUE TO A "C" AND "D" CHANNEL HIGH CHLORINE CONCENTRATION SIGNAL. THE "A" AND "B" TRAINS OF THE CONTROL ROOM EMERGENCY FRESH AIR SUPPLY (CREFAS) SYSTEM, AN ENGINEERED SAFETY FEATURE, INITIATED AS DESIGNED. THE EVENT OCCURRED DURING SEVERE LOCAL THUNDERSTORMS. THE HIGH CHLORINE CONCENTRATION SIGNAL WAS CAUSED BY RAINWATER COMING IN CONTACT WITH ANALYZER PROBES RESULTING IN A CHEMICAL IMBALANCE IN PROBES' ELECTROLYTE. THE ANALYZER PROBES ARE LOCATED CLOSE TO THE OUTSIDE AIR INTAKE PLENUM. AFTER THE "C" AND "D" CHANNELS SPIKED, CONTROL ROOM OPERATORS IMPLEMENTED SPECIAL EVENT PROCEDURE SE-2 (TOXIC GAS PROCEDURE) AND MANUALLY TRIPPED THE "A" AND "B" CHLORINE ISOLATION CHANNELS IN ACCORDANCE WITH PROCEDURES. PROPER CONTROL ROOM ISOLATION WAS VERIFIED. CHEMISTRY SAMPLED THE CONTROL ENCLOSURE HVAC INTAKE PLENUM AND NO CHLORINE WAS DETECTED. ALL CHLORINE CHANNELS WERE VERIFIED TO BE WITHIN NORMAL LEVELS (LESS THAN 0.1 PPM). THE ISOLATION WAS RESET AND NORMAL CONTROL ROOM VENTILATION WAS RESTORED AT 2036 HOURS. THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL TO THE ENVIRONMENT AS A RESULT OF THIS EVENT. A MODIFICATION TO RELOCATE THE DETECTORS WAS IMPLEMENTED ON 8/28/88 TO MITIGATE FALSE ENVIRONMENTALLY RELATED CONTROL ROOM VENTILATION SYSTEM ISOLATIONS.

[88] LIMERICK 1 DOCKET 50-352 LER 88-027 REV 01
UPDATE ON CONTROL ROOM HVAC ISOLATION RESULTING FROM A HIGH CHLORINE ISOLATION SIGNAL CAUSED BY RAIN WATER CONTACTING AN ANALYZER PROBE.
EVENT DATE: 072188 REPORT DATE: 103189 NSSS: GE TYPE: BWR

(NSIC 215774) ON JULY 21, 1988 AT 1738 HOURS, THE MAIN CONTROL ROOM VENTILATION SYSTEM ISOLATED ON A "D" CHANNEL HIGH CHLORINE CONCENTRATION SIGNAL. THE "B" TRAIN OF THE CONTROL ROOM EMERGENCY FRESH AIR SUPPLY (CREFAS) SYSTEM, AN ENGINEERED SAFETY FEATURE, INITIATED AS DESIGNED. THE EVENT OCCURRED DURING HEAVY RAIN STORMS ACCOMPANIED BY HIGH WINDS. THE HIGH CHLORINE CONCENTRATION SIGNAL WAS CAUSED BY RAINWATER COMING IN CONTACT WITH THE CHLORINE ANALYZER PROBE RESULTING IN A CHEMICAL IMBALANCE IN THE PROBE'S ELECTROLYTE. THE ANALYZER PROBES ARE LOCATED CLOSE TO THE OUTSIDE AIR INTAKE PLENUM LOUVERS. AFTER THE "D" CHLORINE DETECTOR SPIKED, OPERATORS IMPLEMENTED SPECIAL EVENT PROCEDURE SE-2 (TOXIC GAS PROCEDURE) AND MANUALLY TRIPPED THE "A", "B" AND "C" CHLORINE ISOLATION CHANNELS IN ACCORDANCE WITH PROCEDURES. PROPER CONTROL ROOM ISOLATION WAS VERIFIED. FOLLOWING THE STORMS, INSTRUMENT AND CONTROLS (I&C) INSPECTED THE DETECTOR AND ENSURED THAT THE INTAKE PLENUM AREA WAS DRY. ALL CHLORINE CHANNELS WERE VERIFIED TO BE WITHIN NORMAL LEVELS (LESS THAN 0.1 PPM). THE ISOLATION WAS RESET AT 2350 ON JULY 22, 1988, HOURS AND NORMAL CONTROL ROOM VENTILATION WAS RESTORED. THERE WAS NO CHLORINE INTAKE TO THE MAIN CONTROL ROOM. THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL TO THE ENVIRONMENT AS A RESULT OF THIS EVENT.

[89] LIMERICK 1 DOCKET 50-352 LER 88-028 REV 01
UPDATE CONTROL ROOM HVAC ISOLATION RESULTING FROM A HIGH CHLORINE CONCENTRATION SIGNAL CAUSED BY RAINWATER CONTACTING AN ANALYZER PROBE.
EVENT DATE: 081788 REPORT DATE: 103189 NSSS: GE TYPE: BWR

(NSIC 215775) ON AUGUST 17, 1988 AT 1832 HOURS, THE MAIN CONTROL ROOM VENTILATION SYSTEM ISOLATED ON A "D" CHANNEL HIGH CHLORINE CONCENTRATION SIGNAL. THE "B" TRAIN OF THE CONTROL ROOM EMERGENCY FRESH AIR SUPPLY (CREFAS) SYSTEM, AN ENGINEERED SAFETY FEATURE, INITIATED AS DESIGNED. THE EVENT OCCURRED DURING SEVERE LOCAL THUNDERSTORMS. THE HIGH CHLORINE CONCENTRATION SIGNAL WAS CAUSED BY RAINWATER COMING IN CONTACT WITH THE CHLORINE ANALYZER PROBE RESULTING IN A CHEMICAL IMBALANCE IN THE PROBE'S ELECTROLYTE. THE ANALYZER PROBES ARE LOCATED CLOSE TO THE OUTSIDE AIR INTAKE PLENUM LOUVERS. WHEN THE "D" CHLORINE DETECTOR SPIKED, OPERATORS IMPLEMENTED SPECIAL EVENT PROCEDURE SE-2 (TOXIC GAS PROCEDURE) AND MANUALLY TRIPPED THE "A", "B", AND "C" CHLORINE ISOLATION CHANNELS IN ACCORDANCE WITH PROCEDURES. PROPER CONTROL ROOM ISOLATION WAS VERIFIED. AFTER THE SPIKE, OPERATIONS PERSONNEL VERIFIED THAT ALL CHLORINE CHANNELS WERE WITHIN NORMAL LEVELS (LESS THAN 0.1 PPM). THE ISOLATION WAS RESET AT 1910 HOURS, AND

NORMAL CONTROL ROOM VENTILATION WAS RESTORED. THERE WAS NO CHLORINE INTAKE TO THE MAIN CONTROL ROOM. THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL TO THE ENVIRONMENT. A MODIFICATION TO RELOCATE THE CHLORINE DETECTION PROBES WAS IMPLEMENTED ON AUGUST 29, 1988 TO MITIGATE FALSE, ENVIRONMENTALLY RELATED, AUTOMATIC CONTROL ROOM VENTILATION SYSTEM ISOLATIONS.

[90] LIMERICK 1 DOCKET 50-352 LER 89-052
A CONDITION PROHIBITED BY TECH SPECS FOR PRIMARY COOLANT AND GASEOUS EFFLUENT SAMPLING AND RADIOLOGICAL ANALYSIS DUE TO A PROCEDURAL DEFICIENCY.
EVENT DATE: 091089 REPORT DATE: 101089 NSSS: GE TYPE: BWR

(NSIC 215649) ON 9/10/89, UNIT 1 REACTOR POWER WAS REDUCED TO PERFORM ROUTINE CONTROL ROD PATTERN ADJUSTMENTS AND WAS THEN INCREASED TO 100% RATED THERMAL POWER (RTP) USING NORMAL PLANT OPERATING PROCEDURES. LATER, ON 9/11, PLANT STAFF DISCOVERED THAT TWICE DURING THIS ROUTINE EVOLUTION, REACTOR POWER HAD CHANGED IN EXCESS OF 15% RTP IN ONE HOUR WITHOUT THE TECH SPECS (TS) REQUIREMENTS FOR SAMPLING AND RADIOLOGICAL ANALYSIS BEING PERFORMED. SPECIFICALLY, PRIMARY COOLANT WAS NOT SAMPLED AND ANALYZED FOR SPECIFIC ACTIVITY AS REQUIRED BY ACTION C.1 OF TS 3.4.5. ALSO GASEOUS EFFLUENT SAMPLING AND ANALYSIS WAS NOT INITIATED AS REQUIRED BY SURVEILLANCE REQUIREMENTS OF TS TABLE 4.11.2.1.2-1 UNTIL 1900 ON SEPTEMBER 11, IMMEDIATELY FOLLOWING DISCOVERY OF THE MISSED SAMPLES. THE CONSEQUENCES WERE MINIMAL BECAUSE ANALYSIS RESULTS FOR SAMPLES TAKEN DURING THIS REACTOR POWER CHANGE PERIOD SHOWED LITTLE VARIATION AND WERE WELL WITHIN TS ALLOWABLE LIMITS INDICATING NO APPARENT PROBLEMS OF FUEL CLADDING LEAKAGE. THE EVENT WAS CAUSED BY THE LACK OF DETAILED INSTRUCTIONS FOR THE METHOD OF MONITORING AND NOTIFYING SHIFT CHEMISTRY TECHNICIANS WHEN REACTOR POWER CHANGES EXCEED 15% OF RTP IN ONE HOUR. TO PREVENT RECURRENCE OF THIS EVENT, ADDITIONAL INSTRUCTIONS HAVE BEEN PROVIDED TO OPERATIONS PERSONNEL AND APPLICABLE PROCEDURES WILL BE REVISED.

[91] LIMERICK 2 DOCKET 50-353 LER 89-007
ENTRY INTO STARTUP OPERATING CONDITION WITH RESIDUAL HEAT REMOVAL SYSTEM IN SHUTDOWN COOLING DUE TO PERSONNEL ERROR.
EVENT DATE: 092089 REPORT DATE: 102089 NSSS: GE TYPE: BWR

(NSIC 215650) ON SEPTEMBER 20, 1989, AT 1517 HOURS DURING THE START UP OF LIMERICK GENERATING STATION UNIT 2, THE UNIT WAS PLACED IN THE STARTUP OPERATING CONDITION WITH THE 'B' RESIDUAL HEAT REMOVAL (RHR) SUBSYSTEM IN THE SHUTDOWN COOLING MODE RATHER THAN THE LOW PRESSURE COOLANT INJECTION (LPCI) MODE REQUIRED BY TECHNICAL SPECIFICATIONS (TS). THE 'B' RHR SUBSYSTEM WAS SUBSEQUENTLY ALIGNED IN THE LPCI MODE AT 1823 HOURS. THE CONSEQUENCES OF THIS EVENT WERE MINIMAL BECAUSE THE OTHER THREE RHR SUBSYSTEMS WERE OPERABLE IN THE LPCI MODE. THE CAUSE OF THE EVENT IS PROCEDURE NON-COMPLIANCE DUE TO PERSONNEL ERROR. ALSO, CONTRIBUTING WAS A LACK OF CLARITY IN THE PROCEDURE GOVERNING PLANT STARTUP. THE PERSONNEL INVOLVED WERE COUNSELED AND THE PROCEDURE WAS REVISED TO IMPROVE ITS CLARITY. NO SIMILAR PROBLEMS WERE FOUND IN REVIEW OF OTHER OPERATIONS PROCEDURES THAT HAVE PARALLEL ACTION STEPS.

[92] LIMERICK 2 DOCKET 50-353 LER 89-008
A MISPOSITIONED EMERGENCY SERVICE WATER SYSTEM VALVE COULD HAVE PREVENTED CERTAIN EMERGENCY CORE COOLING SYSTEMS FROM PERFORMING THEIR SAFETY FUNCTIONS.
EVENT DATE: 092289 REPORT DATE: 102089 NSSS: GE TYPE: BWR

(NSIC 215685) ON SEPTEMBER 22, 1989, AT APPROXIMATELY 0700, DURING A ROUTINE EMERGENCY SERVICE WATER (ESW) SYSTEM FLOW BALANCE TEST, A STATION SYSTEM ENGINEER DISCOVERED A NORMALLY OPEN MANUAL ESW ISOLATION VALVE TO BE IN THE CLOSED POSITION. THIS CLOSED VALVE BLOCKED ESW COOLING WATER FLOW TO CERTAIN EMERGENCY CORE COOLING SYSTEMS (ECCS) PUMP ROOM COOLERS, PUMP SEAL COOLERS, AND MOTOR OIL

COOLERS. THE VALVE WAS PROMPTLY OPENED AND THE OTHER ESW SYSTEM VALVES WERE VERIFIED TO BE IN THE CORRECT POSITION. THE VALVE WAS PREVIOUSLY VERIFIED TO BE OPEN ON SEPTEMBER 1, 1989 DURING THE MONTHLY SYSTEM LINEUP VERIFICATION. THE EXACT CAUSE OF THE VALVE MISPOSITIONING WAS NOT DETERMINED, BUT IS BELIEVED TO BE OPERATOR PERSONNEL ERROR BECAUSE A SIMILAR VALVE LOCATED NEAR THE SUBJECT VALVE IS FREQUENTLY CLOSED IN SUPPORT OF ESW LOOP 'B' OPERATION. UNDER POSTULATED ACCIDENT CONDITIONS, WITH NO COOLING WATER TO THE ROOM COOLERS AND EQUIPMENT COOLERS, THE AFFECTED SAFETY RELATED COMPONENTS COULD HAVE BECOME INOPERABLE IF THEIR TEMPERATURE ROSE ABOVE ENVIRONMENTAL QUALIFICATION TEMPERATURE LIMITS. THUS THE HPCI SYSTEM, CORE SPRAY LOOP 'B' SUBSYSTEM AND RHR LOOPS 'B' AND 'D' SUBSYSTEMS COULD HAVE BEEN PREVENTED FROM FULFILLING THEIR SAFETY FUNCTIONS.

[93] MCGUIRE 1 DOCKET 50-369 LER 89-030
THE CONTROL ROOM VENTILATION SYSTEM WAS TECHNICALLY INOPERABLE DUE TO AN OPEN CONDUIT CONNECTION BECAUSE OF UNKNOWN REASONS.
EVENT DATE: 071489 REPORT DATE: 102689 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 215705) ON 7/14/89, BAHNSON PERSONNEL WERE PREPARING TO PERFORM A VISUAL INSPECTION AND A LEAK TEST (PRESSURE TEST) ON THE MODIFIED TRAIN B OUTSIDE AIR PRESSURE FILTER TRAIN (OAPFT) FILTER HOUSING OF THE CONTROL ROOM VENTILATION (VC) SYSTEM. THE VISUAL INSPECTION REVEALED A 3/4 INCH OPEN CONDUIT CONNECTION WHICH WOULD HAVE PREVENTED A SUCCESSFUL LEAK TEST. THE TRAIN B OAPFT FILTER HOUSING WAS SUCCESSFULLY LEAK TESTED AFTER THE CONDUIT CONNECTION WAS REMOVED AND THE HOLE PLUGGED. THE OPEN CONDUIT CONNECTION ALLOWED AN ADDITIONAL INLEAKAGE OF UNFILTERED AIR. THIS LEAKAGE COULD HAVE RESULTED IN THE DOSE TO CONTROL ROOM PERSONNEL TO EXCEED THAT ASSUMED BY THE DESIGN BASIS ANALYSIS. DESIGN ENGINEERING PERFORMED A PAST OPERABILITY DETERMINATION THAT CONCLUDED THAT, ALTHOUGH THE VC SYSTEM WAS TECHNICALLY INOPERABLE, NO DANGER TO THE CONTROL ROOM PERSONNEL EXISTED FOR ANY CREDIBLE ACCIDENT WHICH MAY HAVE OCCURRED. THIS EVENT IS ASSIGNED A CAUSE OF UNKNOWN BECAUSE IT COULD NOT BE DETERMINED WHY THE OPEN CONDUIT CONNECTION WAS NOT REMOVED, PLUGGED, AND SEALED WHEN THE FILTER HOUSING WAS ORIGINALLY MODIFIED. THE BEST ESTIMATE IS THAT THE ORIGINAL MODIFICATION WAS PERFORMED IN THE THIRD QUARTER OF 1980, WHICH WAS PRIOR TO UNIT 1 INITIAL FUEL LOADING. UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER AND UNIT 2 WAS IN MODE 6 (REFUELING) WHEN THIS EVENT WAS DISCOVERED.

[94] MCGUIRE 1 DOCKET 50-369 LER 89-026
BOTH TRAINS OF THE CONTROL AREA VENTILATION SYSTEM WERE INOPERABLE BECAUSE OF A DESIGN DEFICIENCY THAT DELETED A CHECK DAMPER.
EVENT DATE: 072089 REPORT DATE: 101989 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)
VENDOR: RUSKIN MANUFACTURING COMPANY

(NSIC 215686) ON JULY 20, 1989, PERFORMANCE PERSONNEL BEGAN POST MODIFICATION TESTING (PMT) TO FUNCTIONALLY VERIFY THE OPERATION OF THE OUTSIDE AIR PRESSURE FILTER TRAIN (OAPFT) FAN ON TRAIN B OF THE CONTROL AREA VENTILATION (VC) SYSTEM. THE OAPFT WAS ORIGINALLY DESIGNED WITH TWO 50 PERCENT CAPACITY FANS. BECAUSE OF MAINTENANCE AND RELIABILITY PROBLEMS AND SPACE LIMITATIONS, A NUCLEAR STATION MODIFICATION (NSM) WAS WRITTEN TO CHANGE THESE FANS TO ONE 100 PERCENT CAPACITY FAN. WHILE PERFORMING THE PMT, PERFORMANCE PERSONNEL DETERMINED THAT BOTH VC SYSTEM TRAINS A AND B MAY HAVE BEEN INOPERABLE BECAUSE THE NSM DELETED A CHECK DAMPER ON TRAIN B. WITH THIS CHECK DAMPER REMOVED AND THE VC SYSTEM IN CERTAIN OPERATING MODES, A CONDITION COULD EXIST WHERE MORE AIR COULD BE RECIRCULATED THROUGH THE FAILED OAPFT RATHER THAN AIR BEING PULLED IN FROM OUTSIDE. THIS WOULD DECREASE THE ABILITY OF THE OPERATING OAPFT FAN TO PRESSURIZE THE CONTROL ROOM. ON SEPTEMBER 19, 1989, DESIGN ENGINEERING (DE) PERSONNEL CONFIRMED THAT THE VC SYSTEM TRAIN A COULD NOT BE JUSTIFIED OPERABLE WITH THE CHECK DAMPER REMOVED AND ISOLATION DAMPERS OPEN ON TRAIN B. UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT 100

PERCENT POWER AND UNIT 2 WAS IN NO MODE (CORE UNLOADED) AT THE TIME THIS EVENT WAS DISCOVERED. THIS EVENT IS ASSIGNED A CAUSE OF DESIGN DEFICIENCY BECAUSE OF AN UNANTICIPATED INTERACTION OF COMPONENTS.

[95] MCGUIRE 1 DOCKET 50-369 LER 89-015 REV 01
UPDATE ON THE CONTROL ROOM VENTILATION SYSTEM DID NOT MEET THE REQUIRED POSITIVE PRESSURE BECAUSE OF A DESIGN OVERSIGHT.
EVENT DATE: 072289 REPORT DATE: 091889 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 215571) ON JULY 21, 1989, DURING POST MODIFICATION FLOW BALANCING OF OUTSIDE AIR PRESSURE FILTER TRAIN - 2, PERFORMANCE PERSONNEL MEASURED CONTROL ROOM PRESSURE RELATIVE TO OUTSIDE ATMOSPHERE AND DISCOVERED THAT NEUTRAL PRESSURE WAS THE BEST THAT COULD BE ACHIEVED IN SOME REQUIRED SYSTEM CONFIGURATIONS. THE APPLICABLE TECHNICAL SPECIFICATION SPECIFIES THAT THE CONTROL ROOM BE MAINTAINED AT A POSITIVE PRESSURE OF AT LEAST +0.125 INCHES WATER GAUGE RELATIVE TO OUTSIDE ATMOSPHERE. THE CONTROL ROOM PRESSURIZATION HAS BEEN TESTED RELATIVE TO THE PRESSURE IN THE CABLE SPREADING ROOM SINCE INITIAL TESTING AND STARTUP. THIS EVENT IS ASSIGNED A CAUSE OF DESIGN DEFICIENCY BECAUSE OF A DESIGN OVERSIGHT. ON AUGUST 19, 1989, DESIGN ENGINEERING PERSONNEL ISSUED AN OPERABILITY EVALUATION FOR THE CONTROL AREA VENTILATION SYSTEM. THE OPERABILITY EVALUATION STATED THAT THE CONTROL AREA VENTILATION SYSTEM IS CONDITIONALLY OPERABLE IF THE CONTROL ROOM DOORS ARE TAPED AND ALL FOUR OUTSIDE AIR INTAKES REMAIN OPEN AT ALL TIMES. A PERMANENT OUTSIDE AIR REFERENCE POINT WILL BE INSTALLED TO ENSURE THAT FUTURE TESTING IS ACCURATE. UNIT 1 WAS IN MODE 1, POWER OPERATION, AND UNIT 2 WAS IN MODE 6, REFUELING, AT THE TIME THIS EVENT WAS DISCOVERED.

[96] MCGUIRE 1 DOCKET 50-369 LER 89-018
BOTH TRAINS OF THE CONTROL AREA VENTILATION AND CHILLED WATER SYSTEM WERE DECLARED INOPERABLE BECAUSE OF EQUIPMENT FAILURE.
EVENT DATE: 082989 REPORT DATE: 092889 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 215589) ON 8/29/89 AT 0930, OPERATIONS PERSONNEL SECURED TRAIN A OF THE CONTROL AREA VENTILATION (VC) AND CHILLED WATER (YC) SYSTEM BECAUSE THE CURRENT FOR THE TRAIN A VC CHILLER WAS FLUCTUATING DUE TO IMPROPER REFRIGERANT LEVEL. AT 0930, OPERATIONS PERSONNEL STARTED TRAIN B OF THE VC/YC SYSTEM AND THE TRAIN B VC/YC CHILLER TRIPPED ON A LO FLOW ALARM. OPERATIONS ATTEMPTED TO START TRAIN B OF VC/YC A SECOND TIME AND IT WOULD NOT START. AT 1000, AFTER MECHANICAL MAINTENANCE PERSONNEL ADDED REFRIGERANT TO THE TRAIN A VC/YC CHILLER, OPERATIONS PERSONNEL DECLARED TRAIN A OF VC/YC OPERABLE. AT 1004, OPERATIONS PERSONNEL FAILED TO OPEN THE NUCLEAR SERVICE WATER SUPPLY VALVE TO TRAIN B VC/YC CHILLER AND SUCCESSFULLY STARTED TRAIN B OF THE VC/YC SYSTEM. UNIT 1 WAS IN MODE 1, POWER OPERATION AT 48% POWER, AND UNIT 2 WAS IN MODE 5, COLD SHUTDOWN, AT THE TIME THIS EVENT WAS DISCOVERED. THIS EVENT IS ASSIGNED A CAUSE OF EQUIPMENT FAILURE BECAUSE THE NUCLEAR SERVICE WATER SUPPLY VALVE FAILED TO OPEN AS REQUIRED ON A START OF TRAIN B VC/YC SYSTEM CHILLER. OPERATIONS PERSONNEL WILL MAKE APPROFRIATE PROCEDURE CHANGES CONCERNING THE OPERATION OF THE VC/YC CHILLERS.

[97] MCGUIRE 1 DOCKET 50-369 LER 89-023
AN ENGINEERED SAFETY FEATURE ACTUATION OCCURRED WHEN A VALVE WAS REPOSITIONED TO OPEN BECAUSE OF IMPROPERLY FOLLOWING THE CORRECT PROCEDURE.
EVENT DATE: 083189 REPORT DATE: 100289 NSSS: WE TYPE: PWR

(NSIC 215513) ON AUGUST 31, 1989, PERFORMANCE PERSONNEL WERE PREPARING TO PERFORM PERIODIC TEST PT/1/A/4252/01, AUXILIARY FEEDWATER PUMP NUMBER 1 PERFORMANCE TEST. PERFORMANCE PERSONNEL INADVERTENTLY ISOLATED PRESSURE SWITCH ICAPS5042 INSTEAD OF PRESSURE TRANSMITTER ICAPT5050 TO INSTALL A TEST GAUGE. THIS ISOLATION RESULTED

IN AN ENGINEERED SAFETY FEATURE ACTUATION IN THAT, ONE VALVE, ICA-86, NUCLEAR SERVICE WATER (RN) TO AUXILIARY FEEDWATER (CA) ISOLATION VALVE, WAS REPOSITIONED TO OPEN. THIS EVENT IS ASSIGNED A CAUSE OF INAPPROPRIATE ACTION BECAUSE OF IMPROPERLY FOLLOWING THE CORRECT PROCEDURE. THE PUMP TEST WAS SUCCESSFULLY COMPLETED AFTER THE SIGNAL TO VALVE ICA-86 WAS CLEARED. UNIT 1 WAS IN MODE 1, POWER OPERATION, AT 100 PERCENT POWER, AT THE TIME OF THE EVENT. PERFORMANCE TECHNICIANS WILL BE INSTRUCTED CONCERNING THE NEED TO FOLLOW THE SPECIFIC REQUIREMENTS OF WRITTEN PROCEDURES.

[98] MCGUIRE 1 DOCKET 50-369 LER 89-027
THE ANNULUS VENTILATION SYSTEM WAS INOPERABLE BECAUSE OF A DESIGN DEFICIENCY, A MANAGEMENT DEFICIENCY AND INAPPROPRIATE ACTIONS.
EVENT DATE: 091589 REPORT DATE: 101689 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)
VENDOR: SOLON MANUFACTURING COMPANY

(NSIC 215671) DURING PERFORMANCE TESTING OF THE ANNULUS VENTILATION SYSTEM (VE) ON 9/15/89, THE PREHEATERS DID NOT START BECAUSE THE AIR FLOW PERMISSIVE WAS NOT MADE. THE CROSS CONNECT DAMPERS WERE CLOSED AND TAGGED AND THE UNIT 1 VE TRAIN A REMAINED LOGGED INOPERABLE. PERFORMANCE PERSONNEL TESTED THE UNIT 2 VE SYSTEM AND DETERMINED THAT THE SAME PROBLEM EXISTED. THE UNIT 2 VE TRAIN A WAS LOGGED INOPERABLE AND THE CROSS CONNECT DAMPERS WERE CLOSED AND TAGGED. SUBSEQUENTLY, DESIGN ENGINEERING PERSONNEL EVALUATED THE PROBLEM AND DETERMINED THAT THE VE SYSTEM WOULD BE CONSIDERED CONDITIONALLY OPERABLE IF A LEAD WAS LIFTED ON THE DIFFERENTIAL PRESSURE SWITCHES. THIS EVENT IS ASSIGNED CAUSES OF DESIGN DEFICIENCY, INAPPROPRIATE ACTION AND MANAGEMENT DEFICIENCY. UNIT 1 WAS IN MODE 1, POWER OPERATION, AT 100% POWER AND UNIT 2 WAS IN MODE 3, HOT STANDBY, AT THE TIME OF THIS INCIDENT. ON 9/18/89 DURING IMPLEMENTATION OF MCGUIRE EXEMPT VARIATION NOTICE (MEVN) 1078, UNIT 1 ENTERED TECH SPEC 3.0.3 WHEN BOTH TRAINS OF THE VE SYSTEM BECAME INOPERABLE. THE MEVN WAS IMPLEMENTED ON THE UNIT 2 VE TRAIN B, AND TRAIN B WAS RESTORED TO OPERABILITY. THIS EVENT IS ASSIGNED A CAUSE OF INAPPROPRIATE ACTION RESULTING FROM DEFICIENT COMMUNICATION. UNIT 1 WAS IN MODE 1, POWER OPERATION, AT 100% POWER AND UNIT 2 WAS IN MODE 2, STARTUP, AT THE TIME OF ENTRY INTO TECH SPEC 3.0.3.

[99] MCGUIRE 1 DOCKET 50-369 LER 89-029
DIESEL GENERATOR A AUTOMATICALLY STARTED BECAUSE OF AN UNDERVOLTAGE CONDITION CREATED BY GROUNDS FROM UNKNOWN CAUSES.
EVENT DATE: 092289 REPORT DATE: 102389 NSSS: WE TYPE: PWR

(NSIC 215704) ON SEPTEMBER 22, 1989, UNIT 1 WAS IN MODE 1, POWER OPERATION, AT 83 PERCENT POWER. HURRICANE HUGO WAS CAUSING HIGH WINDS AND RAIN AT THIS TIME. AN ENGINEERED SAFEGUARDS FEATURE (ESF) ACTUATION OCCURRED WHEN DIESEL GENERATOR (D/G) 1A STARTED DUE TO A MOMENTARY UNDERVOLTAGE (UV) CONDITION ON THE TRAIN A 4160 VOLT ESSENTIAL SWITCHGEAR (ETA) CAUSED BY THE TRAIN A OFFSITE POWER SOURCE (BL1A) DE-ENERGIZING TO ISOLATE AN APPARENT DIRECTIONAL GROUND FAULT. THE 6900 VOLT SWITCHGEAR AUTOMATICALLY TRANSFERRED TO ITS STANDBY SOURCE, TRAIN B, (BL1B) AS DESIGNED, THUS SUPPLYING POWER TO ETA. THIS PREVENTED THE D/G BREAKER FROM CLOSING BY CLEARING THE UNDERVOLTAGE CONDITION. THE TURBINE RANBACK TO APPROXIMATELY 46 PERCENT POWER WHEN BL1A WAS DE-ENERGIZED. THIS EVENT IS ASSIGNED A CAUSE OF UNKNOWN CAUSE BECAUSE THE EXACT CAUSE OF THE GROUND COULD NOT BE DETERMINED. THE HIGH WINDS FROM THE HURRICANE WERE PROBABLY RESPONSIBLE FOR THE GROUNDS. OPERATIONS (OPS) PERSONNEL VERIFIED PROPER PLANT RESPONSE TO THE RUNBACK AND SECURED D/G 1A. BL1A WAS RESTORED TO NORMAL SERVICE AND THE 6900 VOLT SWITCHGEAR WAS REALIGNED TO ITS NORMAL POWER SOURCE.

[100] MCGUIRE 2 DOCKET 50-370 LER 89-009
MOTOR DRIVEN AUXILIARY FEEDWATER PUMP AUTOMATICALLY STARTED BECAUSE OF AN INADVERTENT ENGINEERED SAFETY FEATURES ACTUATION CAUSED BY INAPPROPRIATE ACTIONS.
EVENT DATE: 090189 REPORT DATE: 100289 NSSS: WE TYPE: PWR

(NSIC 215514) ON SEPTEMBER 1, 1989, INSTRUMENTATION AND ELECTRICAL (IAE) PERSONNEL WERE ASSISTING PERFORMANCE (PRF) PERSONNEL IN THE EXECUTION OF PERIODIC TEST PT/2/A/4200/09A, ENGINEERED SAFETY FEATURES (ESF) ACTUATION. IN THE PERFORMANCE OF THE SUPPORT FOR THIS TEST, IAE PERSONNEL REINSTALLED COIL WIRES FOR RELAYS IN THE SOLID STATE PROTECTION SYSTEM (SSPS) CABINET. THESE WIRES HAD BEEN PREVIOUSLY REMOVED AS DIRECTED BY PROCEDURE OP/2/A/6250/03A, STEAM GENERATOR COLD WET LAY UP RECIRCULATION. AT THE CONCLUSION OF THE TEST, IAE PERSONNEL WERE DIRECTED BY ENGINEERING SUPPORT PERSONNEL TO REMOVE THE COIL WIRES AGAIN PRIOR TO REMOVING JUMPERS ASSOCIATED WITH THE TEST. IN THE PERFORMANCE OF THIS TASK, AN INADVERTENT ESF ACTUATION SIGNAL WAS GENERATED BECAUSE THE IAE PERSONNEL REMOVED THE WRONG COIL WIRES. CONSEQUENTLY, THE MOTOR DRIVEN AUXILIARY FEEDWATER PUMP AUTOMATICALLY STARTED. THIS EVENT IS ASSIGNED A CAUSE OF INAPPROPRIATE ACTION BECAUSE A PROCEDURE WAS NOT USED TO REPLACE OR CONSEQUENTLY REMOVE THE RELAY COIL WIRES. A CONTRIBUTING CAUSE OF DEFECTIVE PROCEDURE IS ALSO ASSIGNED BECAUSE OF INCOMPLETE INFORMATION IN THE PERIODIC TEST PROCEDURE. UNIT 2 WAS IN MODE 5, COLD SHUTDOWN, AT THE TIME OF THE EVENT. APPROPRIATE PROCEDURE CHANGES WILL BE IMPLEMENTED TO PREVENT RECURRENCE OF SIMILAR EVENTS.

[101] MCGUIRE 2 DOCKET 50-370 LER 89-011
TECH SPEC 3.0.3 WAS ENTERED TO PERFORM REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVE LEAK RATE TEST OF TWO EMERGENCY CORE COOLING SYSTEM CHECK VALVES.
EVENT DATE: 091489 REPORT DATE: 101689 NSSS: WE TYPE: PWR

(NSIC 215652) ON SEPTEMBER 14, 1989, AT 1520, OPERATIONS PERSONNEL LOGGED UNIT 2 INTO TECHNICAL SPECIFICATION (TS) 3.0.3 TO ALLOW LEAK RATE TESTING OF TWO EMERGENCY CORE COOLING SYSTEM (ECCS) CHECK VALVES. ENTRY INTO TS 3.0.3 WAS NECESSARY WHEN THE ECCS BECAME TECHNICALLY INOPERABLE BECAUSE ONE ECCS MOTOR OPERATED VALVE WAS REALIGNED. THE LEAK RATE TEST WAS REQUIRED PRIOR TO ENTERING MODE 2, STARTUP. THE TEST CAN ONLY BE PERFORMED IN MODE 3, HOT STANDBY, BECAUSE OF THE REACTOR COOLANT SYSTEM PRESSURE REQUIRED. THE MOTOR OPERATED ECCS VALVE WAS REALIGNED TO MEET TS REQUIREMENTS AFTER COMPLETION OF THE LEAK RATE TEST, RETURNING THE ECCS TO OPERABLE STATUS. OPERATIONS PERSONNEL LOGGED UNIT 2 OUT OF TS 3.0.3 AT 1735. UNIT 2 WAS IN MODE 3 AT THE BEGINNING OF FUEL CYCLE SIX AT THE TIME OF THIS EVENT. NO CAUSE WAS ASSIGNED SINCE THE STATION VOLUNTARILY ENTERED TS 3.0.3 TO PERFORM THE LEAK RATE TESTING. OPERATIONS PERSONNEL WERE AWARE OF THE SYSTEM ALIGNMENT AND IN FULL CONTROL OF THE UNIT DURING THE TEST.

[102] MILLSTONE 2 DOCKET 50-336 LER 87-009 REV 01
UPDATE ON REACTOR TRIP ON LOW #1 STEAM GENERATOR LEVEL.
EVENT DATE: 090287 REPORT DATE: 100289 NSSS: CE TYPE: PWR
VENDOR: COPES-VULCAN, INC.

(NSIC 215517) WHILE OPERATING AT 91% POWER ON 9/2/87 AT 2015 THE UNIT EXPERIENCED AN AUTOMATIC REACTOR TRIP DUE TO LOW STEAM GENERATOR LEVEL IN THE #1 STEAM GENERATOR. THE OPERATION'S STAFF WAS PERFORMING A ROUTINE REDUCTION IN POWER LEVEL FROM 100% TO 90%. DURING THIS EVOLUTION THE SECONDARY PLANT OPERATOR OBSERVED THE FEED FLOW TO THE #1 STEAM GENERATOR LESS THAN THE STEAM FLOW. THE OPERATOR TOOK MANUAL CONTROL OF THE FEEDWATER REGULATING VALVE, 2-FW-51B, AND THE FEEDWATER REGULATING BYPASS VALVE, 2-FW-41B, IN AN ATTEMPT TO RESTORE LEVEL IN THE #1 STEAM GENERATOR. LEVEL CONTINUED TO DECREASE IN THE #1 STEAM GENERATOR AND THE UNIT TRIPPED. OPERATIONS RESPONDED TO THE TRIP BY PERFORMING EOP 2525, "STANDARD POST TRIP ACTIONS" AND EOP 2526 "REACTOR TRIP RECOVERY". NO OTHER SYSTEMS WERE AFFECTED AND THE UNIT WAS PLACED IN A STABLE CONDITION. THE #1 FEEDWATER REGULATING VALVE, 2-FW-51B, WAS DISASSEMBLED AND REPAIRED. DURING THE

DISASSEMBLY IT WAS DISCOVERED THAT THE STEM HAD SEPARATED FROM THE PLUG. THIS EVENT IS BEING REPORTED PURSUANT TO THE REQUIREMENTS OF PARAGRAPH 50.73(A)(2)(IV) DUE TO THE AUTOMATIC REACTOR TRIP ON LOW STEAM GENERATOR LEVEL. SIMILAR LER'S: NONE.

[103] MILLSTONE 2 DOCKET 50-336 LER 88-003 REV 01
UPDATE ON SPURIOUS ENGINEERED SAFEGUARDS ACTUATION.
EVENT DATE: 013088 REPORT DATE: 092989 NSSS: CE TYPE: PWR
VENDOR: CONSOLIDATED CONTROLS CORP.

(NSIC 215518) ON 1/30/88 AT 1527 HOURS WITH THE UNIT IN MODE 6, AN INADVERTENT AUTOMATIC LOSS-OF-NORMAL POWER (LNP) ACTUATION OCCURRED ON FACILITY TWO (Z2) OF THE ENGINEERED SAFEGUARDS ACTUATION SYSTEM (ESAS). ADDITIONALLY, SEVERAL AUXILIARY EXHAUST ACTUATION SIGNALS (AEAS) WERE PROCESSED BY ESAS EARLIER IN THE DAY. AT THE TIME OF THESE ACTUATIONS, THE Z2 FACILITY WAS OUT OF SERVICE FOR MAINTENANCE. THE IMMEDIATE CAUSE OF THE ACTUATIONS WAS NOT APPARENT. IT WAS LATER DETERMINED THAT AN IMPROPERLY IDENTIFIED WIRING PROBLEM DUE TO HUMAN ERROR COUPLED WITH A PROPERLY APPLIED ELECTRICAL JUMPER RESULTED IN 125 VDC BEING APPLIED TO THE COMMON BUS OF THE Z2 ESAS FACILITY. GIVEN THE MODE OF THE UNIT AND THE STATE OF EQUIPMENT ACTUATED BY THE LNP AND AEAS SIGNALS, ALL EQUIPMENT RESPONDED AS EXPECTED. UPON DISCOVERY OF THE ACTUATIONS, OPERATORS TOOK APPROPRIATE ACTIONS TO RESTORE THE AFFECTED EQUIPMENT TO A NORMAL LINEUP. THERE WERE NO SAFETY CONSEQUENCES AS A RESULT OF THESE EVENTS. ONCE THE WIRING ERROR WAS IDENTIFIED, IT WAS CORRECTED AND ALL THE ASSOCIATED ESAS VOLTAGE SENSITIVE EQUIPMENT WAS TESTED AND VERIFIED OPERABLE.

[104] MILLSTONE 2 DOCKET 50-336 LER 88-010 REV 01
UPDATE ON CONTAINMENT ATMOSPHERE GASEOUS RADIATION MONITOR SWITCH ALIGNMENT.
EVENT DATE: 092158 REPORT DATE: 100289 NSSS: CE TYPE: PWR
VENDOR: CONSOLIDATED CONTROLS CORP.
NUCLEAR MEASUREMENTS CORP.

(NSIC 215527) ON 9/21/88 AT 1400 HOURS WITH THE UNIT IN MODE 1 AND AT 100% POWER OPERATION, THE INSTRUMENT AND CONTROLS DEPARTMENT, (I&C), WAS PERFORMING ROUTINE MONTHLY FUNCTIONAL TESTS ON A PROCESS RADIATION MONITOR. WHILE PERFORMING THIS TEST THE I&C TECHNICIAN OBSERVED THAT THE TEST/OPERATE SWITCH FOR EACH OF THE CONTAINMENT ATMOPSHERE GASEOUS RADIATION MONITORS, RM 8123B AND RM 8262B, WAS IN THE TEST POSITION. THIS SWITCH POSITION INTRODUCES AN ARTIFICIAL 3600 COUNT PER MINUTE SIGNAL INTO THE RADIATION MONITOR MAKING EACH OF THE RADIATION MONITORS INOPERABLE. THE UNIT'S TECH SPECS REQUIRE AT LEAST ONE OF THESE RADIATION MONITORS FOR REACTOR COOLANT SYSTEM LEAK DETECTION. THESE TWO MONITORS HAD BEEN FUNCTIONALLY TESTED ON 8/23/88. THE SWITCHES HAD BEEN LEFT IN THE TEST POSITION AT THE CONCLUSION OF THE AUGUST FUNCTIONAL TEST. THE UNIT OPERATED FOR 29 DAYS WITHOUT EITHER OF THE TWO GASEOUS RADIATION MONITORS. UPON DISCOVERY OF THE SWITCH POSITION THE SHIFT SUPERVISOR WAS NOTIFIED AND THE TEST/OPERATE SWITCHES WERE PLACED IN THE OPERATE POSITION.

[105] MILLSTONE 2 DOCKET 50-336 LER 89-004 REV 02
UPDATE ON CRACKING DISCOVERED IN A MECHANICAL PLUG FROM A STEAM GENERATOR TUBE.
EVENT DATE: 032089 REPORT DATE: 100589 NSSS: CE TYPE: PWR
VENDOR: COMBUSTION ENGINEERING, INC.
WESTINGHOUSE ELEC CORP.-NUCLEAR ENERGY SYS

(NSIC 215521) MILLSTONE 2 (MP2) REMOVED (4) MECHANICAL PLUGS FROM STEAM GENERATOR #2 FOLLOWING NOTIFICATION FROM WESTINGHOUSE THAT PLUGS OF SUSPECT HEAT TREATMENT MAY FAIL SIMILAR TO THE INCIDENT AT NORTH ANNA 1. ON 3/20/89 AT 1015 HRS., WITH THE PLANT IN COLD SHUTDOWN (MODE 5) A VISUAL INSPECTION OF A MECHANICAL PLUG WHICH HAD BEEN PULLED REVEALED CIRCUMFERENTIAL CRACKING AT THE TOP OF THE PLUG.

THIS PLUG WAS MARKED WITH ONE OF THE SUSPECT HEATS, NX3513. THERE WAS NO OPERATOR ACTION ASSOCIATED WITH THIS EVENT, SINCE THE STEAM GENERATOR WAS OUT OF SERVICE FOR TESTING AND REPAIR. SUSPECT PLUGS IN BOTH STEAM GENERATORS' HOT LEGS WERE REPAIRED WITH A PLUG LEAK LIMITING DEVICE. SUBSEQUENT TO A RETURN TO SERVICE, WITH THE PLANT OPERATING AT 30% POWER (MODE 1), ON 3/2/89, NORTHEAST UTILITIES WAS NOTIFIED BY WESTINGHOUSE THAT AN ADDITIONAL TUBE PLUG HEAT, NX4523, MAY BE SUSCEPTIBLE TO PRIMARY STRESS CORROSION CRACKING (PWSCC). FIFTY PLUGS LOCATED IN SLEEVED AND STABILIZED HOT LEG TUBES WERE IDENTIFIED BY RECORDS TO BE FROM HEAT NX4523. ANALYSIS AND CALCULATION BY WESTINGHOUSE AND NU CONCLUDED THAT MP2 CAN SAFELY OPERATE UNTIL 12/1/89 WITHOUT REPAIR TO THE PLUGS FROM HEAT NX4523. NORTHEAST UTILITIES INVESTIGATED THE MICROSTRUCTURES AND CORROSION DATA OF HEAT NX5232 AND FOUND THEM TO BE SIMILAR TO THAT EXHIBITED BY HEAT NX4523. THIS HEAT HAS NOT EXPERIENCED CRACKING AT ANY OPERATING PLANT.

[106] MILLSTONE 3 DOCKET 50-423 LER 86-011 REV 01
UPDATE ON CONTROL BUILDING ISOLATION SIGNALS DUE TO NOISE SPIKE.
EVENT DATE: 020586 REPORT DATE: 102589 NSSS: WE TYPE: PWR
VENDOR: STONE & WEBSTER ENGINEERING CORP.

(NSIC 215720) THIS REPORT IS BEING SUBMITTED ON A RECURRING PROBLEM WITH THE CONTROL BUILDING INLET VENTILATION RADIATION MONITORS, HVC*RE16A AND HVC*RE16B. AT RANDOM TIMES, INTERFERENCE SPIKES IN THE INSTRUMENT LOOPS CAUSED SPURIOUS HIGH RADIATION ALARMS. CONTROL BUILDING ISOLATION (CBI) SIGNALS FOR THE RESPECTIVE TRAIN WERE GENERATED AS A RESULT. THESE INADVERTENT ACTUATIONS ARE BEING REPORTED AS A SINGLE EVENT. THERE WERE NO ADVERSE SAFETY IMPLICATIONS ASSOCIATED WITH THIS PROBLEM. BY VIRTURE OF FAIL-SAFE DESIGN, THE INTERFERENCE RESULTED IN SYSTEM ACTUATION TO THE ACCIDENT CONFIGURATION. THE ROOT CAUSE HAS BEEN IDENTIFIED AS BOTH BROADCAST AND CONDUCTED ELECTROMAGNETIC INTERFERENCE. IN SIGNAL PROCESSORS 3HVC*RIY16A AND 3HVC*RIY16B, THE INTERFERENCE SUPERIMPOSED ON THE EXISTING ELECTRICAL SIGNAL REPRESENTING RADIATION COUNTS, PRODUCING SIGNAL LEVELS WHICH MOMENTARILY EXCEEDED THE HIGH RADIATION ALARM SETPOINT. CORRECTIVE ACTION CONSISTED OF INSTALLING A SOFTWARE CHANGE IN BOTH RADIATION MONITORS WHICH PREVENTS ALARM GENERATION FROM SIGNAL SPIKES, YET STILL PROVIDES SAFE, RELIABLE OPERATION ON VALID HIGH RADIATION SIGNALS.

[107] MILLSTONE 3 DOCKET 50-423 LER 88-026 REV 03
UPDATE ON POTENTIAL DAMAGE TO SAFETY RELATED EQUIPMENT DUE TO DESIGN INADEQUACY.
EVENT DATE: 111888 REPORT DATE: 101089 NSSS: WE TYPE: PWR

(NSIC 215528) ON 11/18/88 AT 1630 HOURS, WITH THE PLANT IN MODE 1 AT 100% POWER, ENGINEERING POSTULATED A SCENARIO WHICH COULD, IN THE EXTREME CASE, RESULT IN A LOSS OF REDUNDANT TRAINS OF SAFETY RELATED (VITAL) EQUIPMENT. IT WAS DISCOVERED THAT CERTAIN CIRCUMSTANCES COULD LEAD TO MILLSTONE UNIT 3 BECOMING ISOLATED FROM THE MILLSTONE STATION SWITCHYARD WHILE ON-LINE. THIS COULD LEAD TO AN OUT-OF-PHASE FAS TRANSFER TO THE RESERVE STATION SERVICE TRANSFORMER (RSST) RESULTING IN A POTENTIALLY DAMAGING TRANSIENT ON BOTH TRAINS OF VITAL 4160V BUSSES. ON 12/29/88, WITH THE PLANT IN MODE 1 AT 75% POWER, THREE RELAYS WERE IDENTIFIED THAT, ASSUMING A SINGLE FAIURE, ALSO COULD RESULT IN THE POSTULATED SCENARIO. ROOT CAUSE OF BOTH EVENTS IS INADEQUACY IN PLANT DESIGN. THE OUT-OF-PHASE TRANSFER SCENARIO POSTULATED HAD NOT BEEN PREVIOUSLY IDENTIFIED. AS A RESULT OF THE FIRST EVENT, ADMINISTRATIVE CONTROLS WERE IMPLEMENTED TO REALIGN THE 4160V BUSSES TO THE RSST WHEN EITHER OF THE SWITCHYARD BREAKERS WERE OPEN. AS A RESULT OF THE LATER EVENT, THE POWER SUPPLY TO THE 4160V BUSSES WAS MANUALLY ALIGNED TO THE RSST. WITH THE 4160V BUSSES SUPPLIED BY THE RSST, THE SCENARIO IS NO LONGER FEASIBLE SINCE THE BUSSES ARE IN THE POST FAST TRANSFER STATE. A MODIFICATION WAS COMPLETED ON 6/22/89 TO ELIMINATE THE FAST TRANSFER ON UNDERVOLTAGE.

[108] MILLSTONE 3 DOCKET 50-423 LER 89-011
CONTAINMENT UNFILTERED LEAKAGE IN EXCESS OF LIMITS DUE TO VALVE LEAKAGE.
EVENT DATE: 052789 REPORT DATE: 062689 NSSS: WE TYPE: PWR
VENDOR: FARR CO.
WALWORTH COMPANY

(NSIC 215400) ON 5/27/89 AT APPROXIMATELY 1700 AND ON 6/3/89 AT APPROXIMATELY 1830, WHILE AT 0% POWER IN MODE 6 (REFUELING), ATMOSPHERIC PRESSURE, AND 89 DEGREES, THE "AS FOUND" LOCAL LEAK RATE TESTING (LLRT) CONTAINMENT UNFILTERED LEAKAGE EXCEEDED THE TECH SPEC LIMIT OF 0.01 LA (13,690 SCCM). 3CVS*V20, THE CONTAINMENT VACUUM SYSTEM OUTSIDE CONTAINMENT ISOLATION VALVE AS FOUND LEAKAGE WAS 12,040 SCCM. 3CDS*CTV40B, THE "B" TRAIN CHILLED WATER SYSTEM INSIDE CONTAINMENT ISOLATION VALVE, AS FOUND LEAKAGE WAS 17,980 SCCM. THE ROOT CAUSE OF 3CVS*V20 LEAKAGE WAS A POOR WEDGE-TO-SEAT FIT. THE WEDGE ANGLE WAS CHANGED AND THE VALVE SEATS WERE RESURFACED TO ENSURE THE EXISTENCE OF A PROPER SEAL. THE ROOT CAUSE OF 3CDS*CTV40B LEAKAGE WAS A BURR ON THE DISC. THE BURR WAS REMOVED AND THE ELASTOMER T-RING WAS REPLACED. THE POST-MAINTENANCE LLRTS ON BOTH VALVES WERE SATISFACTORY.

[109] MILLSTONE 3 DOCKET 50-423 LER 89-020
INADVERTENT SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM BREACH DUE TO ADMINISTRATIVE DEFICIENCIES.
EVENT DATE: 083089 REPORT DATE: 092989 NSSS: WE TYPE: PWR

(NSIC 215526) ON 8/30/89 AT APPROXIMATELY 0240 HOURS, IN MODE 1 AT 100% POWER, 2250 PSIA AND 586 DEGREES, A PLANT SECURITY GUARD, WHILE ON ROUTINE ROUNDS, DISCOVERED A POTENTIAL SECURITY BARRIER BREACH VIA AN EIGHT-INCH MAIN STEAM SYSTEM PIPE OPENING. THE CONTROL ROOM WAS CONTACTED AND IT WAS LATER DETERMINED THAT THE OPEN PIPE WAS NOT A SECURITY VIOLATION BUT IT WAS A BREACH OF THE SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM (SLCRS) INTEGRITY. THE SLCRS LIMITED CONDITION FOR OPERATION (LCO) WAS ENTERED AND ACTION WAS TAKEN TO RESTORE SLCRS INTEGRITY. THE TIME ALLOWED BY THE TECH SPECS LCO ACTION STATEMENT WAS NOT VIOLATED. HOWEVER, THE SLCRS BOUNDARY BREACH WAS NOT IDENTIFIED AT THE TIME THE BREACH OCCURRED AND NO COMPENSATORY MEASURES WERE TAKEN. ROOT CAUSE OF THE EVENT WAS INADEQUATE ADMINISTRATIVE CONTROL TO IDENTIFY POTENTIAL BARRIER BREACHES. AS A CONTRIBUTING CAUSE, THE WORK SCOPE WAS NOT IDENTIFIED IN THE AUTOMATED WORK ORDER PACKAGE. IMMEDIATE CORRECTIVE ACTION WAS TO RESTORE SLCRS INTEGRITY. CORRECTIVE ACTION TO PREVENT RECURRENCE WAS TO REINFORCE EXISTING ADMINISTRATIVE CONTROLS AND TO DEVELOP A CHECKLIST FOR THE OPERATIONS DEPARTMENT TO ASSIST IN DETERMINING POSSIBLE BARRIER BREACHES.

[110] MILLSTONE 3 DOCKET 50-423 LER 89-021
MISCALCULATION OF ENGINEERED SAFETY FEATURES RESPONSE TIME DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 092589 REPORT DATE: 102589 NSSS: WE TYPE: PWR

(NSIC 215753) ON 9/25/89, AT 0840 HOURS, WITH THE PLANT OPERATING IN MODE 1 AT 100% POWER, 586F AND 2250 PSIA, AN INADEQUACY WAS DISCOVERED IN THE PROCEDURE USED TO CALCULATE ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIMES. THIS INADEQUACY INVOLVED THE FAILURE TO ACCOUNT FOR THE RESPONSE TIME OF SLAVE RELAYS. IT WAS DISCOVERED DURING A TECHNICAL REVIEW AND REVISION OF THE ESF RESPONSE TIME TEST PROGRAM. THE ROOT CAUSE OF THIS EVENT WAS PROCEDURAL INADEQUACY WHICH EXISTED SINCE INITIAL PROCEDURE DEVELOPMENT. DATA OBTAINED FROM THE SLAVE RELAY TESTING PERFORMED DURING AUGUST AND SEPTEMBER, 1989 HAS BEEN REVIEWED TO DETERMINE THE EFFECT OF SLAVE RELAY RESPONSE TIME ON THE "AS CALCULATED" CHANNEL RESPONSE TIME CALCULATED DURING THE PAST REFUELING OUTAGE. FOR THOSE SLAVE RELAYS THAT ARE "BLOCK" TESTED (I.E. LOGIC TESTED WITHOUT EQUIPMENT ACTUATION), A SPECIAL TEST WAS PERFORMED TO MEASURE THE RESPONSE TIMES. THIS REVIEW CONCLUDED THAT ACCEPTANCE CRITERIA WAS MET WHEN SLAVE RELAY RESPONSE TIMES WERE INCLUDED.

SLAVE RELAY SURVEILLANCES WILL BE REVISED PRIOR TO 4/1/90, TO INCLUDE DETERMINATION OF SLAVE RELAY RESPONSE TIMES AT LEAST ONCE PER 18 MONTHS.

[111] MILLSTONE 3 DOCKET 50-423 LER 89-022
VALVE STROKE TIME TESTING IN THE WRONG DIRECTION DUE TO TRANSCRIPTION ERROR.
EVENT DATE: 092589 REPORT DATE: 102589 NSSS: WE TYPE: PWR

(NSIC 215738) ON 9/25/89 AT 1215, IN MODE 1 AT 100% POWER, 586F AND 2250 PSIA, IT WAS DISCOVERED THAT AN INSERVICE TEST (IST) SURVEILLANCE, REQUIRED STROKE TIME TESTING OF THE HIGH PRESSURE SAFETY INJECTION (SIH) PUMP INLET ISOLATION VALVES IN THE WRONG DIRECTION (CLOSE TO OPEN VICE OPEN TO CLOSE). THE DISCOVERY WAS MADE DURING A ROUTINE REVIEW OF SURVEILLANCE DATA. IMMEDIATE ACTION WAS TO PERFORM THE STROKE TIME TEST IN THE PROPER DIRECTION (I.E., OPEN TO CLOSE) WHICH WAS SATISFACTORILY COMPLETED. THE ROOT CAUSE OF THE EVENT WAS A TRANSCRIPTION ERROR WHEN WRITING THE SURVEILLANCE PROCEDURE. THE STROKE DIRECTION WHICH WAS PROPERLY IDENTIFIED IN THE IST MANUAL, WAS INCORRECTLY STATED IN THE PROCEDURE AND WAS NOT IDENTIFIED DURING THE PROCEDURE REVIEW PROCESS. THE ASSOCIATED SURVEILLANCE PROCEDURE WAS CHANGED TO REFLECT THE CORRECT STROKE DIRECTION FOR THE VALVES. A REVIEW OF ALL IST STROKE TIME TEST DIRECTION REQUIREMENTS VERSUS THE STROKE TIME TEST PROCEDURES HAS BEEN COMPLETED. NO OTHER REPORTABLE DEFICIENCIES WERE IDENTIFIED. ONE NON-REPORTABLE DISCREPANCY WAS IDENTIFIED. THE IST PROGRAM WAS UPDATED TO REFLECT THE PROPER STROKE DIRECTION IDENTIFIED IN THE SURVEILLANCE PROCEDURE.

[112] MONTICELLO DOCKET 50-263 LER 89-023
FABRICATION FLAWS DISCOVERED IN HPCI LINE WELDS.
EVENT DATE: 090789 REPORT DATE: 101089 NSSS: GE TYPE: BWR
VENDOR: BIF

(NSIC 215539) REJECTABLE FLAWS WERE DISCOVERED IN DISSIMILAR METAL WELDS JOINING A TYPE 304 FLOW VENTURI IN THE STEAM SUPPLY TO THE HIGH PRESSURE COOLANT INJECTION (HPCI) TURBINE. THESE WERE THE ONLY WELDS REMAINING IN THE SYSTEM WHICH REQUIRED AUGMENTED INSERVICE INSPECTION (ISI) BY GENERIC LETTER 88.01 AND NUREG 0313 REV 2 AFTER REPLACEMENT PROJECTS COMPLETED IN 1984 (RECIRCULATION PIPING) AND 1986 (CORE SPRAY PIPING). THE FLAWS WERE NOT PREVIOUSLY DISCOVERED BY ISI BECAUSE IRREGULAR I.D. JOINT GEOMETRY MASKED ULTRASONIC TESTING (UT) INDICATIONS. FOR THIS REASON, RADIOGRAPHY WAS USED TO SUPPLEMENT UT DURING THE CURRENT REFUELING OUTAGE. THE FLAWS WERE DISCLOSED ON 9/7/89. THE FLAWS APPEAR TO BE FROM THE ORIGINAL FABRICATION OF THE SPOOL PIECE WELDS JOINING CARBON STEEL PUP PIECES TO EACH END OF STAINLESS STEEL VENTURI. RADIOGRAPHY WAS NOT PERFORMED DURING FABRICATION, NOR WAS IT REQUIRED AT THE TIME BY THE DESIGN AND FABRICATION CODE B31.1.1967. THE REPLACEMENT VENTURI SPOOL PIECE HAS A VENTURI OF NUCLEAR GRADE STAINLESS STEEL AND WELD JOINT GEOMETRY DESIGNED TO FACILITATE UT INSPECTABILITY.

[113] MONTICELLO DOCKET 50-263 LER 89-018
FAILURE OF STARTER COIL CAUSES SECONDARY CONTAINMENT ISOLATION.
EVENT DATE: 091189 REPORT DATE: 101189 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215538) THE MOTOR STARTER MAGNET COIL FOR THE "B" REACTOR PROTECTION SYSTEM- MOTOR GENERATOR SET SHORTED. THIS RESULTED IN A CONTROL VOLTAGE FUSE OPENING AND SHUTDOWN OF THE MOTOR. LOSS OF POWER TO THE BUS "B" LOADS RESULTED IN SECONDARY CONTAINMENT ISOLATION, AND STARTUP OF BOTH STANDBY GAS TREATMENT SYSTEMS. POWER WAS RESTORED TO BUS "B" FROM THE ALTERNATE SOURCE, ISOLATIONS WERE RESET, AND THE STANDBY GAS TREATMENT SYSTEMS WERE SHUTDOWN. THE COIL AND FUSE WERE REPLACED AND THE MG SET WAS RETURNED TO SERVICE. FUTURE CORRECTIVE

ACTIONS INCLUDE PERIODIC REPLACEMENT OF NORMALLY ENERGIZED COMPONENTS AND INVESTIGATION OF THE FEASIBILITY OF REDUCING THE AREA TEMPERATURES.

[114] MONTICELLO DOCKET 50-263 LER 89-021
CRACK ON JET PUMP RISER BRACE DUE TO FATIGUE.
EVENT DATE: 091689 REPORT DATE: 101689 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215635) A CRACK WAS DISCOVERED ON THE UPPER SUPPORT BRACE OF ONE JET PUMP RISER BY USE OF AN UNDERWATER CAMERA. THE CRACK LOCATION IS ON THE LOWER LEAF ON THE RIGHT SIDE (WHILE FACING OUT FROM VESSEL CENTERLINE) OF THE UPPER SUPPORT BRACE ON THE RISER FOR JET PUMPS 7 AND 8. THIS EVENT IS BEING REPORTED BECAUSE OF ITS POTENTIAL SAFETY SIGNIFICANCE AND GENERIC NATURE. THE CRACK IS BELIEVED TO BE THE RESULT OF HIGH CYCLE FATIGUE DUE TO RESONANCE OF THE UPPER BRACE LEAF CAUSED BY RECIRCULATION PUMP VANE PASSING FREQUENCY AT 95% PUMP SPEED. THERE ARE NO OTHER CRACK INDICATIONS ON THE OTHER LEAVES OF THIS RISER SUPPORT BRACKET (BRACE) OR ANY OTHER RISER SUPPORT BRACKET (BRACE) IN THE VESSEL. PRELIMINARY EVALUATION CONCLUDES THAT THE LOWER SUPPORT BRACE IS NOT SUSCEPTIBLE TO THIS RESONANCE CONDITION AND IS ADEQUATE TO SUPPORT THE JET PUMP RISER ALONE. EVALUATION IS CONTINUING TO VERIFY PRELIMINARY FINDINGS.

[115] MONTICELLO DOCKET 50-263 LER 89-022
FAILURE OF EMERGENCY SERVICE WATER PUMPS TO AUTO START ON TRANSFER TO 1AR TRANSFORMER DUE TO DESIGN ERROR.
EVENT DATE: 092089 REPORT DATE: 102089 NSSS: GE TYPE: BWR

(NSIC 215636) ON SEPTEMBER 20, 1989, WITH THE PLANT IN COLD SHUTDOWN, ENGINEERS DISCOVERED THAT THE #13 AND #14 EMERGENCY SERVICE WATER (ESW) PUMPS DO NOT AUTOMATICALLY START UPON ESSENTIAL BUS TRANSFER TO THE EMERGENCY OFFSITE POWER TRANSFORMER (1AR). IF THIS EVENT OCCURRED COINCIDENT WITH AN ECCS INITIATION, POWER TO THE NORMAL SERVICE WATER PUMPS WOULD HAVE BEEN LOST RESULTING IN NO COOLING WATER FOR THE RHR AND HPCI ROOM COOLERS, THE RHR AND CORE SPRAY PUMP MOTOR COOLERS AND THE CONTROL ROOM VENTILATION SYSTEM. THE CONTROL LOGIC WAS CHANGED TO PROVIDE A PUMP AUTO START FEATURE UPON TRANSFER TO 1AR. A REVIEW OF 1AR LOADS WILL BE COMPLETED PRIOR TO START-UP TO ASSURE ALL EMERGENCY LOADS REQUIRED TO BE POWERED BY 1AR HAVE APPROPRIATE AUTO INITIATION FEATURES IN THEIR CONTROL LOGIC. THIS EVENT IS BEING REPORTED AS A VOLUNTARY LER TO NOTIFY THE NRC OF THE EVENT.

[116] NINE MILE POINT 1 DOCKET 50-220 LER 89-009
IMPROPER INSTALLATION OF PENETRATION PLUG ASSEMBLY DUE TO POOR WRITTEN COMMUNICATIONS.
EVENT DATE: 080389 REPORT DATE: 090689 NSSS: GE TYPE: BWR

(NSIC 215204) AT 1615 HOURS ON 8/3/89, WITH THE MODE SWITCH IN "SHUTDOWN" AND NINE MILE POINT UNIT 1 (NMP1) IN AN EXTENDED REFUELING OUTAGE WITH THE CORE OFF-LOADED, THE CABLE CONNECTING PLUG TO ENVIRONMENTALLY QUALIFIED (EQ) PENETRATION X-E200U WAS FOUND TO BE INCORRECTLY INSTALLED. THE OBSERVED CONDITION OF THIS PENETRATION PLUG DID NOT MEET THE EQ REQUIREMENTS ESTABLISHED FOR PENETRATION PLUGS, SO THE EQUIPMENT CONNECTED TO THIS PENETRATION, TEMPERATURE ELEMENT 36-29A(TE36-29A), WAS DETERMINED TO BE IN A CONDITION WHICH WAS FUNCTIONAL BUT NOT ENVIRONMENTALLY QUALIFIED. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE POOR WRITTEN COMMUNICATIONS. SPECIFICALLY, THERE WAS A LACK OF ADEQUATE PROCEDURAL CONTROL OF THE WORK TO BE PERFORMED. IMMEDIATE CORRECTIVE ACTION WAS TO INITIATE WORK REQUEST 167052 TO BRING VENDOR PERSONNEL ON SITE TO ASSIST IN THE PERFORMANCE OF CORRECTIVE MAINTENANCE. AN ADDITIONAL CORRECTIVE ACTION IS TO HAVE VENDOR PERSONNEL TRAIN NMPC PERSONNEL ON PENETRATION PLUG ASSEMBLY.

[117] NINE MILE POINT 1 DOCKET 50-220 LER 89-012
REACTOR SCRAM AND REACTOR BUILDING EMERGENCY VENTILATION INITIATION DUE TO LOSS OF REACTOR PROTECTION SYSTEM BUS 11.
EVENT DATE: 091789 REPORT DATE: 101789 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215677) BETWEEN SEPTEMBER 17 AND SEPTEMBER 28, 1989, WITH NINE MILE POINT UNIT ONE IN AN EXTENDED REFUELING OUTAGE WITH THE CORE OFF-LOADED, FOUR REACTOR SCRAMS AND ASSOCIATED REACTOR BUILDING EMERGENCY VENTILATION SYSTEM (RBEVS) INITIATIONS WERE RECEIVED DUE TO A LOSS OF POWER TO REACTOR PROTECTION SYSTEM (RPS) BUS 11. LOSS OF POWER TO RPS BUS 11 WILL GENERATE A FULL SCRAM SIGNAL IN BOTH RPS TRIP CHANNELS DUE TO A DESIGN NON-COINCIDENT LOGIC RELAY CONFIGURATION WHEN THE MODE SWITCH IS IN SHUTDOWN, REFUEL OR STARTUP AND REACTOR PRESSURE IS LESS THAN 600 PSIG. IN ADDITION, NON-COINCIDENT LOGIC WILL INITIATE RBEVS DURING ALL OPERATING MODES ON THIS POWER LOSS. THE 10CFR50.72(A)(2)(IV) NOTIFICATION WAS MADE FOR EACH EVENT. THE MOST PROBABLE CAUSE FOR THIS SEQUENCE OF EVENTS IS A MALFUNCTIONING VOLTAGE REGULATOR/SPEED CONTROLLER ON MOTOR GENERATOR SET 162 (MG 162), WHICH PROVIDES CONTINUOUS POWER TO RPS BUS 11. MG 162 HAS BEEN REMOVED FROM SERVICE AND ADMINISTRATIVE CONTROLS IMPLEMENTED TO PREVENT A RECURRENCE OF THESE EVENTS. A ROOT CAUSE INVESTIGATION HAS BEEN INITIATED AND A SUPPLEMENT TO THIS LER WILL BE ISSUED OUTLINING ITS RESULTS AND FINAL CORRECTIVE ACTIONS TAKEN.

[118] NINE MILE POINT 2 DOCKET 50-410 LER 88-022 REV 01
UPDATE ON DESIGN RATED REACTOR CORE FLOW EXCEEDED DUE TO POOR ELECTRICAL CONNECTION RESULTS IN PLANT OPERATION IN AN UNANALYZED CONDITION.
EVENT DATE: 041988 REPORT DATE: 100589 NSSS: GE TYPE: BWR
VENDOR: BAILEY CONTROLS CO.

(NSIC 215504) ON APRIL 19, 1988 AT 0930 HOURS WITH THE REACTOR AT APPROXIMATELY 99% OF RATED THERMAL POWER, THE NINE MILE POINT UNIT 2 (NMP2) REACTOR WAS INADVERTENTLY OPERATED WITH GREATER THAN 100% OF RATED REACTOR CORE FLOW. OPERATIONS PERSONNEL, WHILE RAISING REACTOR POWER BY ADJUSTING REACTOR RECIRCULATION FLOW, OBSERVED THAT THE REACTOR CORE RECIRCULATION SYSTEM HAD EXCEEDED ITS FULL POWER DESIGN RATED FLOW OF 108.5 MILLION POUNDS/HOUR (MLB/HR). ACTUAL REACTOR RECIRCULATION FLOW WAS 114.933 MLBS/HR OR 105.9% OF RATED. THIS PLACED THE UNIT IN A CONDITION NOT SPECIFICALLY ADDRESSED IN THE NMP2 FINAL SAFETY ANALYSIS REPORT (FSAR). THE ROOT CAUSE OF THIS EVENT WAS A POOR ELECTRICAL CONTACT BETWEEN THE SUMMER CARD AND ITS MATING CONNECTOR, IN THE REACTOR RECIRCULATION SYSTEM. THE REACTOR TOTAL CORE FLOW SIGNAL PASSES THROUGH THIS CARD. THIS CONDITION RESULTED IN AN ERRONEOUS FLOW INDICATION. IMMEDIATE CORRECTIVE ACTION TAKEN BY NIAGARA MOHAWK LICENSED OPERATORS WAS TO REDUCE "INDICATED" CORE FLOW TO LESS THAN 100% OF RATED. (IT WAS LATER DETERMINED THAT AN "INDICATED" CORE FLOW OF 100% OF RATED WAS AN ACTUAL CORE FLOW OF 104.5% OF RATED.)

[119] NINE MILE POINT 2 DOCKET 50-410 LER 89-023
ENGINEERED SAFETY FEATURE ACTUATION AS A RESULT OF A SPURIOUS HIGH SIGNAL FROM A REACTOR BUILDING VENTILATION RADIATION MONITOR.
EVENT DATE: 090689 REPORT DATE: 100589 NSSS: GE TYPE: BWR

(NSIC 215525) ON 9/6/89, AT 2335 HOURS WITH THE REACTOR AT 96% POWER AND THE MODE SWITCH IN "RUN", NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED AN ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF). THE EVENT CONSISTED OF AN AUTOMATIC INITIATION OF THE STANDBY GAS TREATMENT SYSTEM-TRAIN B (SBGT-B) AND EMERGENCY RECIRCULATION UNIT COOLER 2HVR*UC413B. THE ESF ACTUATION OCCURRED AS A RESULT OF A SPURIOUS HIGH SIGNAL FROM THE REACTOR BUILDING VENTILATION EXHAUST RADIATION MONITOR 2HVR*CAB14B. DURING TROUBLESHOOTING, ATTEMPTS WERE MADE TO DUPLICATE THE EVENT, HOWEVER, THESE EFFORTS PROVED UNSUCCESSFUL. CORRECTIVE ACTIONS INCLUDED NMP2

LICENSED OPERATORS IDENTIFYING THE CAUSE OF THE ESF INITIATION AND VERIFYING THE PLANT STATUS AS NORMAL.

[120] NINE MILE POINT 2 DOCKET 50-410 LER 89-024
MANUAL REACTOR SCRAM DUE TO EQUIPMENT FAILURE AND ENTRY INTO RESTRICTED ZONE.
EVENT DATE: 090889 REPORT DATE: 100589 NSSS: GE TYPE: BWR
VENDOR: LAMBDA ELECTRONICS

(NSIC 215577) AT 18:00:45 HOURS ON SEPTEMBER 8, 1989, WITH THE REACTOR MODE SWITCH IN "RUN" AND THE REACTOR AT 88% RATED THERMAL POWER (930 MWE), NINE MILE POINT #2 EXPERIENCED A DOWNSHIFT OF REACTOR RECIRCULATION PUMPS TO SLOW SPEED. THIS PLACED THE UNIT ABOVE THE 100% ROD LINE WITH CORE FLOW <45% (RESTRICTED AREA OF OPERATION). THE CONTROL ROOM ANNUNCIATORS NOTIFIED THE OPERATIONS PERSONNEL OF THE DOWNSHIFT. OPERATIONS PERSONNEL IMMEDIATELY PERFORMED THE REQUIRED ACTIONS PER N2-OP-29, REACTOR RECIRCULATION SYSTEM, BY PLACING THE REACTOR MODE SWITCH TO "SHUTDOWN", INITIATING A REACTOR SCRAM. OPERATIONS PERSONNEL PROCEEDED WITH SCRAM RECOVERY PER N2-OP-101C, PLANT SHUTDOWN. 10CFR50.72 NOTIFICATION WAS MADE ON SEPTEMBER 8, 1989, AT 2013. THE ROOT CAUSE FOR RECIRCULATION PUMP TRIP IS A FAILURE OF A 24 VOLT DC POWER SUPPLY (C33A-K613) WITHIN THE FEEDWATER CONTROL SYSTEM. CORRECTIVE ACTION WAS TO REPLACE THE FAULTY POWER SUPPLY.

[121] NINE MILE POINT 2 DOCKET 50-410 LER 89-025
TECH SPEC SURVEILLANCE REQUIREMENT NOT PERFORMED DUE TO PERSONNEL ERROR.
EVENT DATE: 090889 REPORT DATE: 100989 NSSS: GE TYPE: BWR

(NSIC 215578) ON SEPTEMBER 9, 1989, AT 0640 HOURS, IT WAS DISCOVERED THAT AN AREA UNIT COOLER WHICH IS REQUIRED FOR OPERABILITY OF DIVISION 3 DIESEL GENERATOR (DG) HAD NOT BEEN RETURNED TO SERVICE. THE DG HAD BEEN DECLARED OPERABLE THE PREVIOUS DAY, SEPTEMBER 8, 1989, AT 1850 HOURS FOLLOWING ELECTRICAL PREVENTIVE MAINTENANCE ON THE AREA UNIT COOLER. AS A RESULT, A REQUIRED TECHNICAL SPECIFICATION (TS) SURVEILLANCE REQUIREMENT WAS NOT PERFORMED WHEN THE DIESEL GENERATOR WAS DETERMINED TO BE INOPERABLE. AT THE TIME OF THE TS VIOLATION, NINE MILE POINT UNIT 2 (NMP2) WAS IN OPERATIONAL MODE 3 "HOT SHUTDOWN". AT THE TIME OF DISCOVERY NMP2 WAS IN OPERATIONAL MODE 4 "COLD SHUTDOWN". THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR, SPECIFICALLY, AN EXTRANEOUS ACT WAS PERFORMED BY AN ELECTRICIAN WHICH WAS NOT INCLUDED IN THE PROCEDURE. IMMEDIATE CORRECTIVE ACTIONS WERE TO DECLARE THE DIVISION 3 DG INOPERABLE AND CLOSE THE LOCAL DISCONNECT FOR THE AREA UNIT COOLER, PLACING THE UNIT COOLER IN A STANDBY CONFIGURATION. ADDITIONAL CORRECTIVE ACTIONS INCLUDED REPRIMANDING INDIVIDUALS INVOLVED IN THE EVENT, A CHANGE TO THE ELECTRICAL PREVENTIVE MAINTENANCE PROCEDURE AND THE ISSUANCE OF A "LESSONS LEARNED" TRANSMITTAL.

[122] NINE MILE POINT 2 DOCKET 50-410 LER 89-027
MISSED SURVEILLANCE DUE TO INADEQUATE TASK SCHEDULING AND ASSIGNMENT RESULTS IN TECH SPEC VIOLATION.
EVENT DATE: 091389 REPORT DATE: 101389 NSSS: GE TYPE: BWR

(NSIC 215654) ON SEPTEMBER 13, 1989, AT 1158 HOURS WITH THE REACTOR IN "COLD SHUTDOWN" (OPERATIONAL CONDITION 4), AN ACTION STATEMENT REQUIRED BY A LIMITING CONDITION FOR OPERATION WAS NOT COMPLETED AS REQUIRED. THE SERVICE WATER LINE "A" RADIOACTIVITY MONITOR WAS INOPERABLE. A SERVICE WATER LINE "A" GRAB SAMPLE WAS COLLECTED AND ANALYZED AT 2058 ON SEPTEMBER 12, 1989. THE NEXT SAMPLE WAS NOT COLLECTED AND ANALYZED UNTIL 1331 ON SEPTEMBER 13, 1989. TECHNICAL SPECIFICATION (TS) REQUIRES GRAB SAMPLES BE COLLECTED AND ANALYZED AT LEAST EVERY 12 HOURS WHEN THE MONITOR IS INOPERABLE. THEREFORE, NMP2 WAS NOT IN COMPLIANCE WITH TS SECTION 3.3.7.9.B AND ACTION STATEMENT 130. THE ROOT CAUSE FOR THIS EVENT WAS INADEQUATE SCHEDULING OF THE SERVICE WATER GRAB SAMPLE. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED DELEGATION OF IMMEDIATE ACTION TASKS TO BE

COMPLETED TO A SPECIFIC TECHNICIAN AND THE REVISION OF THE WORK ASSIGNMENT FORM USED TO ASSIGN TASKS. ADDITIONAL CORRECTIVE ACTIONS INCLUDE THE DEVELOPMENT OF A UNIT 2 CHEMISTRY DEPARTMENT INSTRUCTION ON SHIFT TURNOVER WORK ASSIGNMENT PROCESSES AND THE DEVELOPMENT AND TRAINING ON A LESSONS LEARNED TRANSMITTAL FOR THE APPROPRIATE DEPARTMENTS.

[123] NINE MILE POINT 2 DOCKET 50-410 LER 89-029
INCORRECT SETTING OF SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS DUE TO ENGINEERING CHANGE.
EVENT DATE: 092089 REPORT DATE: 102389 NSSS: GE TYPE: BWR

(NSIC 215712) ON SEPTEMBER 20, 1989, AT 0445 HOURS IT WAS DISCOVERED THAT NINE MILE POINT UNIT 2 HAD BEEN IN A CONDITION NOT ALLOWED BY TECHNICAL SPECIFICATIONS (TS). SPECIFICALLY, THERE WAS A FAILURE TO COMPLY WITH THE REQUIREMENT THAT THE SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS HAVE AN OPENING SETPOINT TO BE LESS THAN OR EQUAL TO 0.25 POUNDS/SQUARE INCH DELTA (PSID). AT THE TIME OF THE DETERMINATION THE REACTOR WAS IN COLD SHUTDOWN (OPERATIONAL CONDITION 4) AT AMBIENT PRESSURE WITH THE REACTOR COOLANT WATER TEMPERATURE ABOUT 115 DEGREES FAHRENHEIT. THE IMMEDIATE CAUSE OF THE CONDITION WAS AN ENGINEERING CHANGE ISSUED IN AUGUST 1986. THIS CHANGE ALLOWED THE MAINTENANCE SETPOINT FOR THE SUPPRESSION POOL/DRYWELL VACUUM BREAKERS TO BE SET NON-CONSERVATIVE TO THE TS REQUIREMENT. A ROOT CAUSE ANALYSIS WILL BE PERFORMED AND WILL BE PRESENTED IN A SUPPLEMENTAL REPORT TO THIS LER. CORRECTIVE ACTION WAS TO CANCEL THE ENGINEERING AND DESIGN CHANGE AND CORRECT THE MAINTENANCE PROCEDURE. ALL SUPPRESSION POOL/DRYWELL VACUUM BREAKERS WERE IMMEDIATELY CHECKED AND ADJUSTED AS REQUIRED PER THE MAINTENANCE PROCEDURE TO WITHIN THE TS LIMIT.

[124] NINE MILE POINT 2 DOCKET 50-410 LER 89-030
FAILURE TO SUBMIT A SPECIAL REPORT DUE TO INADEQUATE MANAGERIAL METHODS.
EVENT DATE: 092089 REPORT DATE: 102089 NSSS: GE TYPE: BWR

(NSIC 215688) ON SEPTEMBER 20, 1989, DURING A REVIEW OF DIESEL GENERATOR OCCURRENCE REPORTS (OR) BY THE INDEPENDENT SAFETY ENGINEERING GROUP (ISEG) IT WAS DISCOVERED THAT NINE MILE POINT UNIT 2 (NMP2) WAS NOT IN COMPLIANCE WITH TECHNICAL SPECIFICATION (TS) SECTION 4.8.1.1.3. SPECIFICALLY, A SPECIAL REPORT REQUIRED BY TS SECTION 4.8.1.1.3 WAS NOT SUBMITTED IN THE DESIGNATED TIME PERIOD FOR A VALID DIESEL GENERATOR (D/G) ENGINE FAILURE. THE DIVISION I DIESEL ENGINE (2EGS*EG1) FAILURE OCCURRED ON MARCH 12, 1989, AND THE DUE DATE FOR THIS SPECIAL REPORT WAS APRIL 11, 1989. THE ROOT CAUSE FOR THIS EVENT WAS INADEQUATE MANAGERIAL METHODS. THE CORRECTIVE ACTIONS FOR THIS EVENT ARE THAT THE TS SPECIAL REPORT REPORTABILITY REQUIREMENTS FOR THE MARCH 12, 1989, EVENT HAVE BEEN INCLUDED IN THIS REPORT. THE REQUIREMENT FOR AN ADDITIONAL REVIEW OF ORS BY THE NUCLEAR REGULATORY COMPLIANCE GROUP (NRCG) AFTER THE SIGNATURE OF THE STATION SUPERINTENDENT WAS ADDED TO PROCEDURE S-NRCP-1, OR TRACKING. ADDITIONALLY, REPORTABLE CONDITIONS IDENTIFIED BY TS WERE ADDED TO AP-10.2.2, REPORTABLE OCCURRENCES.

[125] NINE MILE POINT 2 DOCKET 50-410 LER 89-033
ENGINEERED SAFETY FEATURE ACTUATION DUE TO PERSONNEL ERROR.
EVENT DATE: 092589 REPORT DATE: 102589 NSSS: GE TYPE: BWR

(NSIC 215737) ON 9/25/89, AT 1529 HOURS, A TRANSIENT OCCURRED ON THE REACTOR WATER CLEANUP (RWCU) SYSTEM. THIS EVENT CAUSED A PRIMARY CONTAINMENT ISOLATION VALVE 2WCS*MOV-112 TO CLOSE, DUE TO HIGH TEMPERATURE (140F) ON THE OUTLET OF THE RWCU NON-REGENERATIVE HEAT EXCHANGER. THE ISOLATION, ALTHOUGH NOT A TECH SPEC REQUIREMENT, UTILIZES THE NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NS4) LOGIC TO SIGNAL THE CLOSURE OF 2WCS*MOV-112. THE NS4 LOGIC SYSTEM IS AN ENGINEERED SAFETY SYSTEM. AT THE TIME OF THE EVENT, THE PLANT WAS OPERATING AT 31% OF RATED

THERMAL POWER WITH THE REACTOR MODE SWITCH IN THE "RUN" POSITION. THIS EVENT OCCURRED WHILE OPERATION DEPARTMENT PERSONNEL WERE PLACING "D" RWCU FILTER DEMINERALIZER ON THE LINE. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE PERSONNEL ERROR. THE IMMEDIATE CORRECTIVE ACTIONS WERE AS FOLLOWS: 1. OPERATORS VERIFIED THE PRIMARY CONTAINMENT ISOLATION FOR THE RWCU SYSTEM OCCURRED. (RWCU VERIFIED ISOLATED). 2. OPERATORS VERIFIED RELATED SYSTEMS WERE OPERATIONAL TO ENSURE PLANT AND PERSONNEL SAFETY.

[126] NINE MILE POINT 2 DOCKET 50-410 LER 89-026
ENGINEERED SAFETY FEATURE INITIATION DUE TO SPURIOUS TRIP SIGNALS CAUSED BY HIGH FREQUENCY WELDING.
EVENT DATE: 092689 REPORT DATE: 102589 NSSS: GE TYPE: BWR
VENDOR: KAMAN SCIENCES CORP.

(NSIC 215751) ON SEPTEMBER 26, 1989, AT 1320 HOURS AND AT 1435 HOURS WITH THE REACTOR AT APPROXIMATELY 35% POWER AND THE MODE SWITCH IN "RUN", NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED TWO ENGINEERED SAFETY FEATURE (ESF) ACTUATIONS. THE FIRST EVENT CONSISTED OF A SECONDARY CONTAINMENT ISOLATION AND THE AUTOMATIC INITIATION OF THE STANDBY GAS TREATMENT SYSTEM (TRAINS A&B), EMERGENCY RECIRCULATION UNIT COOLER AND REACTOR BUILDING UNIT COOLERS. THE SECOND EVENT CONSISTED OF AN AUTOMATIC INITIATION OF TRAIN A OF THE STANDBY GAS TREATMENT SYSTEM. THE ESF ACTUATIONS OCCURRED ON A SPURIOUS HIGH RADIATION SIGNAL. THE IMMEDIATE CAUSE OF THE SPURIOUS SIGNAL HAS BEEN DETERMINED TO BE ELECTRICAL NOISE RESULTING FROM AC (HIGH FREQUENCY) TUNGSTEN INERT GAS (TIG) WELDING. THE ROOT CAUSE HAS BEEN DETERMINED TO BE A DESIGN DEFICIENCY. CORRECTIVE ACTIONS CONSIST OF THE FOLLOWING: 1. WR 163948 WAS WRITTEN AND COMPLETED TO INSPECT AND TIGHTEN GROUND CONNECTIONS ON AFFECTED RADIATION MONITORS. 2. A PROBLEM REPORT (PR 08867) HAS BEEN ISSUED TO REVIEW THE INSTRUMENT SYSTEM GROUND (ISG) DESIGN. 3. A MODIFICATION REQUEST HAS BEEN ISSUED TO EVALUATE THE NEED TO MODIFY THE MONITORING SYSTEM GROUNDS IN ORDER TO ISOLATE IT FROM POTENTIAL GROUND NOISE.

[127] NINE MILE POINT 2 DOCKET 50-410 LER 89-028
OPERATIONAL SURVEILLANCE PROCEDURE DEFICIENCY DUE TO INADEQUATE DEVELOPMENT OF ASME SECTION XI REQUIREMENTS.
EVENT DATE: 092789 REPORT DATE: 102789 NSSS: GE TYPE: BWR

(NSIC 215752) ON SEPTEMBER 27, 1989, AS THE RESULT OF AN AUDIT, IT WAS DETERMINED THAT OPERATIONS SURVEILLANCE PROCEDURE N2-OSP-ICS-Q002 (RCIC PUMP AND VALVE OPERABILITY TEST AND SYSTEM INTEGRITY TEST), REVISION 0 THROUGH REVISION 2, DID NOT PROVIDE STEPS TO ADJUST THE REACTOR CORE ISOLATION COOLING PUMP 2ICS*P1 SPEED TO A REFERENCE (PREDETERMINED) SPEED PRIOR TO MEASURING FLOW AND DIFFERENTIAL PRESSURE AS REQUIRED BY ASME SECTION XI, SUBSECTION IWP, SUBARTICLE IWP-3100. THIS CONDITION EXISTED FROM 12/86 UNTIL 3/89 AT WHICH TIME THE ERROR WAS DETECTED AND CORRECTED. NINE MILE POINT UNIT 2 (NMP2) WAS OPERATING AT 33 PERCENT POWER WITH REACTOR TEMPERATURE AT 518 DEGREES FAHRENHEIT AND REACTOR PRESSURE AT 942 POUNDS PER SQUARE INCH GAUGE AT THE TIME OF DISCOVERY OF THIS EVENT. THE ROOT CAUSE FOR THIS EVENT WAS A PROCEDURAL DEFICIENCY DUE TO AN INADEQUATE PROCEDURE REVIEW. INITIAL CORRECTIVE ACTIONS INCLUDED PREPARING A REVISION TO OPERATIONS SURVEILLANCE PROCEDURE N2-OSP-ICS-Q002 TO ADJUST PUMP SPEED TO A REFERENCE SPEED. UTILIZING THE NEW TEST METHOD, TESTING WAS COMPLETED ON APRIL 3, 1989, WITH SATISFACTORY RESULTS.

[128] NORTH ANNA 2 DOCKET 50-339 LER 89-008 REV 01
UPDATE ON INADVERTENT ESF ACTUATION DURING SSPS TESTING UPDATED TO REFLECT MOV CONFIGURATION DURING FUTURE TESTING AND CLARIFY ENTRY INTO TECH SPEC 3.0.3.
EVENT DATE: 080689 REPORT DATE: 100489 NSSS: WE TYPE: PWR

(NSIC 215597) AT 1202 HOURS ON 8/6/89, WITH UNIT 2 AT 100% POWER (MODE 1),

ENGINEERED SAFETY FEATURES VALVE 2-RS-MOV-201B WAS INADVERTENTLY CLOSED DURING THE INITIAL ON-LINE PERFORMANCE OF 2-PT-36.5.3A, "SOLID STATE PROTECTION SYSTEM OUTPUT SLAVE RELAY TEST (TRAIN A),". THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV) DUE TO THE INADVERTENT CLOSURE OF ENGINEERED SAFETY FEATURES VALVE 2-RS-MOV-201B. THE INADVERTENT CLOSURE OF 2-RS-MOV-201B RESULTED BECAUSE AN INTERLOCK FOR 2-RS-MOV-201B, ENERGIZED BY TRAIN A SLAVE RELAY K645, WAS NOT CORRECTLY IDENTIFIED DURING THE DEVELOPMENT OF 2-PT-36.5.3A. 2-RS-MOV-201B WAS OPENED FOLLOWING TESTING OF SLAVE RELAY K645. 2-PT-36.5.3A/B HAVE BEEN REVISED TO INCLUDE DEENERGIZING 2-RS-MOV-201B AND 2-RS-MOV-201A IN THE OPEN POSITION DURING FUTURE ON-LINE TESTING. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE 2-RS-MOV-201B RECEIVES AN "ASSURE" OPEN SIGNAL UPON INITIATION OF A CONTAINMENT DEPRESSURIZATION SIGNAL AND WOULD HAVE ALLOWED CASING COOLING FLOW TO 2-RS-P-2B.

[129] OCONEE 1 DOCKET 50-269 LER 89-014
OVERHEAD EMERGENCY POWER PATH RENDERED INOPERABLE DUE TO MANAGEMENT DEFICIENCY WHEN CERTAIN PCBS WERE REMOVED FROM SERVICE.
EVENT DATE: 092189 REPORT DATE: 102089 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 215724) ON 9/21/89, AT 1030 HOURS, IT WAS DISCOVERED THAT WHEN CERTAIN 230 KV SWITCHYARD POWER CIRCUIT BREAKERS (PCBS) ARE REMOVED FROM SERVICE, THE "SWITCHYARD ISOLATE COMPLETE" LOGIC OF THE EXTERNAL GRID TROUBLE PROTECTIVE SYSTEM WOULD NOT COMPLETE ITS CIRCUITRY. THIS PREVENTED THE AIR CIRCUIT BREAKERS (ACB 1 OR 2) OF THE KEOWEE HYDRO UNIT CONNECTED TO THE OVERHEAD EMERGENCY POWER PATH FROM CLOSING. THERE HAVE BEEN SOME INSTANCES IN THE PAST IN WHICH A LIMITING CONDITION OF OPERATION (LCO) IN TECH SPEC 3.7 WAS NOT ENTERED DUE TO LACK OF KNOWLEDGE OF HOW THE EXTERNAL GRID TROUBLE PROTECTIVE SYSTEM AFFECTED THE OVERHEAD EMERGENCY POWER PATH. THIS CONDITION WAS DISCOVERED WHEN LICENSEE TRAINEES QUESTIONED THE OPERABILITY OF KEOWEE TRANSFORMERS WHILE CERTAIN PCBS WERE REMOVED FROM SERVICE. THE ROOT CAUSE OF THIS EVENT IS DETERMINED TO BE MANAGEMENT DEFICIENCY. IMMEDIATE CORRECTIVE ACTIONS INCLUDED PLACING A STATEMENT ON THE SHIFT TURNOVER SHEET ADDRESSING ACTIONS TO BE PERFORMED PRIOR TO ALLOWING MAINTENANCE OR INSPECTION ON CERTAIN 230 KV SWITCHYARD PCBS.

[130] OCONEE 1 DOCKET 50-269 LER 89-015
INOPERABLE CONTAINMENT ISOLATION VALVES FOLLOWING FAILURE TO TEST AFTER MAINTENANCE/MODIFICATION RESULTING FROM INAPPROPRIATE ACTION AND MANAGEMENT DEFICIENCY.
EVENT DATE: 092189 REPORT DATE: 102189 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 215725) ON 9/21/89, WITH UNIT 1 AT 75% FULL POWER (FP) AND UNITS 2 AND 3 AT 100% FP, DURING PROCEDURE REVIEW FOR AN UPCOMING TEST, AN ENGINEER DISCOVERED THAT PENETRATION 53 HAD NOT BEEN PROPERLY VENTED AND DRAINED DURING PREVIOUS 10CFR50, APPENDIX J, TYPE A LEAK RATE TESTS ON ALL THREE UNITS. FURTHER INVESTIGATION SHOWED THAT ON UNITS 1 AND 2 PENETRATIONS 39 AND 53 HAD NOT BEEN PROPERLY TESTED SINCE A MODIFICATION HAD INSTALLED NEW BUILDING ISOLATION VALVES IN 1982. CONTINUED INVESTIGATION FOUND THAT ON UNIT 2 THEY HAD NOT BEEN RETESTED FOLLOWING VALVE REPLACEMENT IN JUNE, 1989. SEVERAL OTHER CONTAINMENT ISOLATION VALVES WERE ALSO FOUND TO HAVE NOT BEEN TESTED FOLLOWING MAINTENANCE. BY TECH SPECS TYPE C TESTING IS NOT REQUIRED FOR THESE PENETRATIONS, BUT TYPE A TESTING IS REQUIRED. ROOT CAUSES INCLUDE INAPPROPRIATE ACTION, AND MANAGEMENT DEFICIENCY. CORRECTIVE ACTIONS INCLUDE CLOSURE OF BACKUP VALVES, MODIFICATION OF PENETRATION 53 ON UNIT 1 TO PERMIT TYPE C TESTS, REQUESTS FOR TECH SPEC CHANGE AND A WAIVER OF COMPLIANCE, AND COMMITMENT FOR FUTURE MODIFICATIONS TO ENABLE TYPE C TESTING.

[131] OYSTER CREEK DOCKET 50-219 LER 89-019
EMERGENCY DIESEL GENERATOR I INOPERABLE.
EVENT DATE: 091189 REPORT DATE: 101189 NSSS: GE TYPE: BWR
VENDOR: ELECTRO-MOTIVE DIV

(NSIC 215663) A REACTOR PLANT SHUTDOWN AND COOLDOWN WAS COMPLETED AS REQUIRED BY TECH SPECS DUE TO THE INOPERABILITY OF EMERGENCY DIESEL GENERATOR 1 (EDG 1) FOR GREATER THAN SEVEN DAYS. DURING A BI-WEEKLY SURVEILLANCE TEST, EDG 1 EXHIBITED ERRATIC LOAD SWINGS DURING ITS UNLOADING SEQUENCE. ONCE HOUR AFTER IT WAS SHUT DOWN, AN ATTEMPT TO START EDG 1 FAILED, AND IT WAS DECLARED INOPERABLE. THE EDG-1 INOPERABILITY WAS CAUSED BY THE FOLLOWING: (1) THE UNLOADING PROBLEM WAS CAUSED BY DIRT ON THE WIPER OF A MOTORIZED POTENTIOMETER IN THE PEAKING LOAD CONTROL CIRCUIT (2) THE FAILURE TO START WAS ATTRIBUTED TO LATENT HEAT EXPANSION OF THE ENGINE OCCURRING APPROXIMATELY ONE HOUR AFTER SHUTDOWN. ADDED ENGINE FRICTION CAUSED STARTERS TO STALL IN THE REDUCED VOLTAGE SLOW ROLL MODE. (3) THE LOAD SWINGS WERE CAUSED BY TWO LOOSE ELECTRICAL CONNECTIONS ON A "DROOP" INPUT CONTACT TO THE ELECTRIC GOVERNOR CONTROL. FUEL INJECTORS WERE REPLACED WITH NEW UNITS BASED ON THE SUSPICION THAT 50KW OF THE SWINGS MAY HAVE BEEN DUE TO INJECTOR CHECK VALVES LEAKING INTERNALLY. THE SAFETY SIGNIFICANCE OF THIS EVENT IS CONSIDERED MINIMAL SINCE DURING THIS PERIOD EDG 2 WAS AVAILABLE FOR OPERATION, AND ITS OPERABILITY WAS DEMONSTRATED DAILY IN ACCORDANCE WITH TECH SPECS. IT SHOULD BE NOTED THAT ONLY THE LEAKY FUEL INJECTORS WOULD HAVE AFFECTED PERFORMANCE UNDER EMERGENCY START AND LOADING CONDITIONS.

[132] OYSTER CREEK DOCKET 50-219 LER 89-020
DC CONTROL POWER SELECTOR SWITCH FOR AC SWITCHGEAR SELECTED TO NON-SAFETY RELATED DC SUBSYSTEM DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 091689 REPORT DATE: 101389 NSSS: GE TYPE: BWR

(NSIC 215664) ON 9/15/89, AN OPERATOR TRAINEE IDENTIFIED THAT THE KNIFE SWITCH USED TO SELECT THE DC CONTROL POWER SOURCE FOR 480V AC UNIT SUBSTATION (USS) 1B2 WAS SELECTED TO THE NON-SAFETY RELATED SOURCE. INVESTIGATIONS HAVE REVEALED THAT THE DC CONTROL POWER SELECTOR SWITCH FOR USS 1B2 HAD BEEN IN THE WRONG POSITION SINCE NOVEMBER OF 1986. THE CAUSE OF THIS OCCURRENCE IS ATTRIBUTED TO PROCEDURAL INADEQUACY. THE SYSTEM WAS RETURNED TO NORMAL AFTER PLANT MODIFICATIONS USING THE COMPONENT LINEUP SHEET IN THE SYSTEM OPERATING PROCEDURE. THIS LINEUP SHEET DID NOT INCLUDE THE DC CONTROL POWER SELECTOR SWITCHES FOR THE 4160V AC OR THE 480V AC BUSES AND IT WAS NOT RECOGNIZED THAT THE SWITCH FOR USS 1B2 HAD BEEN LEFT IN THE WRONG POSITION. CHANGES HAVE BEEN MADE TO THE COMPONENT LINEUP SHEETS IN THE SYSTEM OPERATING PROCEDURE TO INCLUDE THE CONTROL POWER SELECTOR SWITCHES FOR ALL OF THE 4160V AND 480V AC SWITCHGEAR UNITS. THIS EVENT REPORT WILL BE MADE REQUIRED READING FOR ALL LICENSED AND EQUIPMENT OPERATORS.

[133] OYSTER CREEK DOCKET 50-219 LER 89-021
REACTOR SCRAM DUE TO TURBINE TRIP AS A RESULT OF PERSONNEL ERROR DURING SURVEILLANCE TESTING.
EVENT DATE: 092289 REPORT DATE: 102389 NSSS: GE TYPE: BWR

(NSIC 215694) ON 9/22/89, AT APPROX. 1418 HOURS, MECHANICAL TEST EQUIPMENT WAS INADVERTENTLY LEFT CONNECTED TO ONE OF THE REACTOR PRESSURE VESSEL (RPV) WATER LEVEL INSTRUMENTS AFTER TESTING WAS COMPLETE. WHILE BEING PLACED BACK IN SERVICE, THE REFERENCE LEG OF THE INSTRUMENT WAS VENTED TO THE TEST EQUIPMENT CAUSING A FALSE HIGH RPV WATER LEVEL SIGNAL TO BE GENERATED IN ALL FIVE LEVEL INSTRUMENTS ATTACHED TO THAT COMMON REFERENCE LEG. THE FALSE HIGH RPV WATER LEVEL CAUSED A TRIP OF THE TURBINE GENERATOR WHICH RESULTED IN A REACTOR SCRAM. THE TECHNICIAN PERFORMING THE VALVE MANIPULATION RECOGNIZED THE PROBLEM AND CLOSED THE ROOT VALVE. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR SINCE THE TEST EQUIPMENT WAS NOT REMOVED AS REQUIRED BY THE SURVEILLANCE PROCEDURE. THIS EVENT HAD MINIMAL SAFETY SIGNIFICANCE BECAUSE THE REACTOR PROTECTION SYSTEM IS

DESIGNED TO PROTECT THE REACTOR FROM ANY TURBINE TRIP CONDITION, AND THE EXCESS FLOW CHECK VALVES IN THE INSTRUMENT LINE WOULD HAVE PREVENTED ANY SIGNIFICANT LOSS OF COOLANT. ALL ENGINEERED SAFETY FEATURES WOULD HAVE FUNCTIONED NORMALLY DUE TO REDUNDANT RPV LEVEL INSTRUMENTATION. ALL INSTRUMENTS INVOLVED IN THIS EVENT WERE CALIBRATION CHECKED TO ENSURE NO PROBLEMS HAD RESULTED FROM THE MOMENTARY DEPRESSURIZATION. THE TECHNICIANS INVOLVED IN THIS EVENT WERE COUNSELED.

[134] PALISADES DOCKET 50-255 LER 89-021
SINGLE FAILURE POTENTIAL WITHIN CR HVAC CIRCUITRY.
EVENT DATE: 091889 REPORT DATE: 101889 NSSS: CE TYPE: PWR

(NSIC 215633) DURING A SENIOR REACTOR OPERATOR (SRO) CLASS WALKDOWN OF THE CONTROL ROOM HEATING, VENTILATION AND AIR CONDITIONING (CR HVAC) SYSTEM (VI) ON SEPTEMBER 18, 1989 ONE OF THE SRO CANDIDATES IDENTIFIED THE POSSIBILITY FOR A SINGLE RELAY (JM;RLY) FAILURE LEAVING IN CR HVAC TRAIN IN THE NORMAL MODE WHEN THE EMERGENCY MODE WOULD BE DESIRED. THE CR HVAC SYSTEM WAS SWITCHED INTO THE EMERGENCY MODE AS A PRECAUTIONARY MEASURE WHILE THE DESIGN BASIS FOR THE MODIFIED CR HVAC SYSTEM WAS REVIEWED. THE REACTOR WAS CRITICAL WITH THE PLANT OPERATING AT 80 PERCENT OF RATED POWER WHEN THIS POTENTIAL CONDITION WAS IDENTIFIED. THE AUTOMATIC SWITCHOVER CIRCUITRY FROM NORMAL TO EMERGENCY MODE IS PROVIDED BY RECEIPT OF A CHP AND/OR CHR SIGNAL TO EITHER OR BOTH OF THESE INSERVICE RELAYS. FOR THE SINGLE FAILURE TO OCCUR, AN EVENT PROVIDING ONLY A CHP OR CHR MUST OCCUR CONCURRENT WITH A SINGLE RELAY FAILURE FOR THE RELAY UPON WHICH THE SIGNAL WAS RECEIVED. DURING EVALUATION OF THE CR HVAC DESIGN BASIS, AN ADDITIONAL CONCERN REGARDING AN INHERENT TIME DELAY WAS IDENTIFIED. THIS DELAY MAY BE EITHER TWO MINUTES OR 55 SECONDS DEPENDING ON OFFSITE POWER AVAILABILITY. THE TWO MINUTE DELAY PRECLUDES ACTUATION OF AIR HANDLING UNITS AND THE OPENING OF REMOTE INTAKE DAMPERS. THESE FINDINGS AND THE RESULTANT CONSEQUENCES ARE BEING EVALUATED BY THE CR HVAC SYSTEM DESIGNER.

[135] PALO VERDE 1 DOCKET 50-528 LER 89-006 REV 01
UPDATE ON INADVERTENT ENGINEERED SAFETY FEATURE ACTUATION.
EVENT DATE: 073189 REPORT DATE: 103189 NSSS: CE TYPE: PWR

(NSIC 215787) ON JULY 31, 1989, PALO VERDE UNIT 1 WAS IN A REFUELING OUTAGE WITH THE CORE OFF-LOADED TO THE SPENT FUEL POOL. ON THE NIGHT SHIFT OF JULY 31, 1989, UNIT 1 PERSONNEL WERE MAKING PREPARATIONS FOR AN OUTAGE OF ALL THE TRAIN "B" CLASS 1E ELECTRICAL SWITCHGEAR. AUXILIARY OPERATORS WERE STRIPPING TRAIN "B" LOADS WHEN AT APPROXIMATELY 0115 MST ON JULY 31, 1989, THERE WAS A LOSS OF POWER TO PANEL 1E-PNB-D26 WHICH CAUSED A LOSS OF POWER TO THE REMOTE INDICATING AND CONTROL (RIC) UNIT FOR RADIATION MONITOR RU-38 THUS INITIATING A TRAIN "B" CONTAINMENT PURGE ISOLATION ACTUATION SIGNAL (CPIAS). THE CPIAS CROSS-TRIPPED TRAIN "B" CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNAL (CREFAS) WHICH IN TURN CROSS-TRIPPED CREFAS "A", ALL IN ACCORDANCE WITH DESIGN. DUE TO THE PLANNED ELECTRICAL OUTAGE, ALL SAFETY EQUIPMENT FOR CPIAS AND CREFAS WAS IN ITS ACTUATED CONDITION PRIOR TO THE EVENT WITH THE EXCEPTION OF THE TRAIN "B" CONTROL ROOM ESSENTIAL AIR HANDLING UNIT WHICH STARTED AS DESIGNED. APPROXIMATELY ONE (1) MINUTE AFTER THE EVENT INITIATION, POWER WAS RESTORED TO 1E-PNB-D26 AND RU-38. RU-38 WAS PLACED BACK ON LINE AND AT APPROXIMATELY 0221 MST ON JULY 31, 1989, THE CPIAS AND CREFAS WERE RESET. AN INVESTIGATION OF THE EVENT HAS BEEN COMPLETED. ATTEMPTS TO RECREATE THIS EVENT, INTERVIEWS WITH ALL INVOLVED PERSONNEL, AND TROUBLESHOOTING OF THE ELECTRICAL EQUIPMENT COULD NOT IDENTIFY THE ROOT CAUSE OF THE EVENT.

[136] PALO VERDE 1 DOCKET 50-528 LER 89-016
ENGINEERED SAFETY FEATURE CAUSED BY FAILED INTEGRATED CIRCUIT CHIP.
EVENT DATE: 090289 REPORT DATE: 100289 NSSS: CE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 215515) ON SEPTEMBER 2, 1989 AT APPROXIMATELY 0237 MST, THE PALO VERDE UNIT 1 BALANCE OF PLANT ENGINEERED SAFETY FEATURE ACTUATION SYSTEM (BOP ESFAS) LOAD SEQUENCER MALFUNCTIONED CAUSING THE INITIATION OF CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SYSTEM, FUEL BUILDING ESSENTIAL VENTILATION ACTUATION SYSTEM, AND DIESEL GENERATOR "A" START. AT APPROXIMATELY 0402 MST THE BOP ESFAS CABINET WAS DOWNPOWERED AND NORMAL PLANT ELECTRICAL ALIGNMENT WAS ESTABLISHED BY APPROXIMATELY 0430 MST. THE CAUSE OF THIS EVENT WAS THE FAILURE OF AN INTEGRATED CIRCUIT CHIP IN THE BOP ESFAS LOAD SEQUENCER. THE LOAD SEQUENCER WAS REPLACED AND BOP ESFAS RETURNED TO SERVICE ON SEPTEMBER 3, 1989 AT APPROXIMATELY 1824 MST.

[137] PALO VERDE 1 DOCKET 50-528 LER 89-019
ENGINEERED SAFETY FEATURE ACTUATION CAUSED BY LOOSE CONNECTION.
EVENT DATE: 092989 REPORT DATE: 102589 NSSS: CE TYPE: PWR

(NSIC 215756) ON SEPTEMBER 29, 1989 AT APPROXIMATELY 0720 MST, UNIT 1 WAS IN A REFUELING OUTAGE WITH THE CORE OFF-LOADED WHEN THE "B" TRAIN OF FUEL BUILDING ESSENTIAL VENTILATION ACTUATION SYSTEM (FBEVAS) SPURIOUSLY ACTUATED. THE "B" TRAIN FBEVAS LOGIC CROSS TRIPPED THE "A" TRAIN FBEVAS AND CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SYSTEM (CREFAS), TRAINS "A" AND "B" AS DESIGNED. AT APPROXIMATELY 0744 MST ON SEPTEMBER 29, 1989 THE FUEL BUILDING VENT LOW AND HIGH RANGE GAS CHANNELS WERE DECLARED INOPERABLE AND THE "B" FBEVAS WAS RESET AND PLACED IN BYPASS. TECHNICAL SPECIFICATION (TS) 3.3.3.8 ACTION 37, AND ACTION 41 REQUIREMENTS WERE INITIATED AS REQUIRED. THE CAUSE OF THIS EVENT WAS A LOOSE CONNECTION BETWEEN THE RADIATION MONITOR'S REMOTE INDICATION AND CONTROL UNIT INSTRUMENT DRAWER AND INSTRUMENT CABINET. AS IMMEDIATE CORRECTIVE ACTION, THE CONNECTION WAS FULLY ENGAGED BY FIRMLY INSTALLING THE INSTRUMENT DRAWER. AN ENGINEERING EVALUATION WILL BE PERFORMED TO DETERMINE THE CORRECTIVE ACTION TO PREVENT RECURRENCE. ON OCTOBER 2, 1989 AT APPROXIMATELY 0744 MST, THE HIGH RANGE RADIATION MONITOR GAS CHANNEL HAD BEEN INOPERABLE FOR 72 HOURS. THEREFORE, THIS REPORT IS ALSO BEING SUBMITTED IN ACCORDANCE WITH TS 3.3.3.8 ACTION 42(B) AND 6.9.2.

[138] PALO VERDE 2 DOCKET 50-529 LER 88-015 REV 01
UPDATE ON ACTION STATEMENT NOT MET FOR INOPERABLE RADIATION MONITOR.
EVENT DATE: 120388 REPORT DATE: 100389 NSSS: CE TYPE: PWR
VENDOR: KAMAN SCIENCES CORP.

(NSIC 215530) ON DECEMBER 7, 1988 AT APPROXIMATELY 0942 MST, A UNIT 2 CHEMISTRY TECHNICIAN 1 (CONTRACTOR, NON-LICENSED) DISCOVERED THE NEW FUEL AREA RADIATION MONITOR RU-19 WAS INOPERABLE. RU-19 INDICATED A CONSTANT 0.00E-0 MILLIREM PER HOUR RADIATION LEVEL INSTEAD OF THE ACTUAL LEVEL. RU-19 MEASURES AREA RADIATION 1 ADJACENT TO THE NEW FUEL STORAGE RACKS. A REVIEW OF PREVIOUS READINGS DETERMINED THAT THE LAST ACCURATE READING OCCURRED ON DECEMBER 3, 1988 AT APPROXIMATELY 0516 MST. ON DECEMBER 4, 1988 AT APPROXIMATELY 0516 MST, AREA SURVEYS WERE NOT PERFORMED WITHIN 24 HOURS AS REQUIRED BY TECHNICAL SPECIFICATIONS 3.3.3.1 ACTION 22. THE CAUSE OF THE INOPERABLE MONITOR IS BELIEVED TO BE A MALFUNCTION OF A CLOCK IN THE COMPUTER INTERNAL TO THE MONITOR. A ROOT CAUSE OF FAILURE WAS UNABLE TO CONFIRM THE CAUSE. THE CAUSE OF THE MISSED ACTION STATEMENT REQUIREMENTS IS A COGNITIVE PERSONNEL ERROR CONTRARY TO AN APPROVED PROCEDURE. AS IMMEDIATE CORRECTIVE ACTION, ON DECEMBER 7, 1988 AT APPROXIMATELY 1030 MST, THE AREA MONITOR RU-19 WAS RESET, TESTED, AND DECLARED OPERABLE.

[139] PALO VERDE 2 DOCKET 50-529 LER 89-008 REV 01
UPDATE ON IMPROPER PERFORMANCE OF SURVEILLANCE TEST FOR STARTUP CHANNEL 2.
EVENT DATE: 050989 REPORT DATE: 100689 NSSS: CE TYPE: PWR

(NSIC 215585) AT APPROXIMATELY 0745 MST ON MAY 9, 1989, PALO VERDE UNIT 2 WAS IN MODE 5 (COLD SHUTDOWN) WHEN A WORK GROUP SUPERVISOR IDENTIFIED A CALCULATIONAL ERROR IN A SURVEILLANCE TEST THAT WOULD HAVE RESULTED IN BORON DILUTION ALARM CHANNEL 2 BEING INOPERABLE. AT APPROXIMATELY 1712 MST ON MAY 8, 1989, STARTUP CHANNEL 2 WAS DECLARED INOPERABLE TO PERFORM SURVEILLANCE TEST 36ST-9SE05, "BORON DILUTION FUNCTIONAL ALARM CHECK." AT APPROXIMATELY 2030 MST ON MAY 8, 1989, AN INSTRUMENT AND CONTROL TECHNICIAN SUCCESSFULLY COMPLETED 36ST-9SE05 BASED ON A CALCULATIONAL ERROR WHICH RESULTED IN THE SURVEILLANCE TEST MEETING THE ACCEPTANCE CRITERIA. STARTUP CHANNEL 2 WAS RETURNED TO SERVICE AT APPROXIMATELY 2154 MST ON MAY 8, 1989 AND TECHNICAL SPECIFICATION 3.1.2.7 ACTION B.1 WAS EXITED. AT APPROXIMATELY 0745 MST ON MAY 9, 1989 THE WORK GROUP SUPERVISOR IDENTIFIED THE CALCULATIONAL ERROR. THE CAUSE OF THE EVENT WAS A PERSONNEL ERROR ON THE PART OF THE INSTRUMENT AND CONTROL TECHNICIAN. A CONTRIBUTORY CAUSE WAS A PROCEDURE DEFICIENCY. STARTUP CHANNEL 2 WAS THEN DECLARED INOPERABLE AT APPROXIMATELY 0840 MST ON MAY 9, 1989 AND TECHNICAL SPECIFICATION 3.1.2.7 ACTION B.1 WAS ENTERED. 36ST-9SE05 WAS REPERFORMED SATISFACTORILY AND STARTUP CHANNEL 1 WAS DECLARED OPERABLE AT APPROXIMATELY 1132 MST ON MAY 9, 1989. TO PREVENT RECURRENCE, THE TECHNICIANS WERE COUNSELED AND 36ST-9SE05 HAS BEEN REVISED.

[140] PALO VERDE 3 DOCKET 50-530 LER 89-005 REV 01
UPDATE ON PLANT VENT LOW RANGE EFFLUENT MONITOR ALARM NOT PROPERLY INVESTIGATED.
EVENT DATE: 062889 REPORT DATE: 092889 NSSS: CE TYPE: PWR

(NSIC 215516) AT APPROXIMATELY 1115 MST ON JUNE 28, 1989, PALO VERDE UNIT 3 WAS IN A REFUELING OUTAGE WITH THE CORE OFF-LOADED TO THE SPENT FUEL POOL WHEN A UNIT 3 TECHNICIAN DISCOVERED THAT THE SAMPLE FLOW RATE FOR THE PLANT VENT LOW RANGE RADIOACTIVE EFFLUENT MONITOR (RU-143) WAS BELOW THE LOW FLOW ALARM SETPOINT RENDERING THE MONITOR INOPERABLE. INVESTIGATION DETERMINED THAT THE LOW FLOW ALARM HAD OCCURRED AT APPROXIMATELY 0531 MST ON JUNE 28, 1989; HOWEVER, THE ALARM WAS NOT PROPERLY INVESTIGATED. THIS RESULTED IN NOT MEETING ACTION REQUIREMENTS 36 AND 40 OF TECHNICAL SPECIFICATION (T.S.) 3.3.3.8. THE CAUSE OF THE LOW FLOW CONDITION ON RU-143 WAS A LOOSE SET SCREW ON THE COUPLING BETWEEN THE MONITOR'S SAMPLE PUMP AND ITS DRIVE MOTOR. THE CAUSE OF THE IMPROPER FOLLOW-UP ACTION FOR THE LOW FLOW ALARM WAS INADEQUATE COMMUNICATION BETWEEN CONTROL ROOM PERSONNEL AND PERSONNEL RESPONSIBLE FOR INVESTIGATING THE CAUSE OF THE ALARM CONDITION. AS CORRECTIVE ACTION, THE PRE-PLANNED ALTERNATE SAMPLING PROGRAM WAS IMPLEMENTED BY 1150 MST ON JUNE 28, 1989, FULFILLING T.S. 3.3.3.8 ACTION REQUIREMENTS. THE LOOSE SET SCREW WAS TIGHTENED AND RU-143 WAS RETURNED TO SERVICE AT APPROXIMATELY 1558 MST ON JUNE 24, 1989. A PREVIOUS SIMILAR EVENT WAS REPORTED IN UNIT 1 LER 528/85-067.

[141] PALO VERDE 3 DOCKET 50-530 LER 89-010
SURVEILLANCE INTERVAL EXCEEDED FOR RADIOACTIVE EFFLUENT MONITORING SYSTEM.
EVENT DATE: 092389 REPORT DATE: 101789 NSSS: CE TYPE: PWR

(NSIC 215661) AT APPROXIMATELY 1535 MST ON 9/25/89 PALO VERDE UNIT 3 WAS IN MODE 5 (COLD SHUTDOWN) WHEN A PVNGS TECHNICIAN (UTILITY, NON-LICENSED) DISCOVERED THAT THE SURVEILLANCE TESTING INTERVAL FOR THE RADIOACTIVE EFFLUENT MONITORING SYSTEM HAD BEEN EXCEEDED. THE SURVEILLANCE TESTING IS REQUIRED TO BE PERFORMED AT LEAST ONCE PER 24 HOURS PURSUANT TO SPECIFICATION 4.3.3.8 AND WAS REQUIRED TO HAVE BEEN PERFORMED NO LATER THAN 1500 MST ON 9/25/89 (THIS INCLUDES THE 25% MAXIMUM ALLOWABLE EXTENSION PURSUANT TO SPECIFICATION 4.0.2). THE EFFLUENT MONITORING SYSTEM SURVEILLANCE TEST WAS SATISFACTORILY COMPLETED AT APPROXIMATELY 1550 MST ON 9/23/89. THE EFFLUENT MONITORING SYSTEM WAS ADMINISTRATIVELY INOPERABLE FOR APPROXIMATELY 50 MINUTES. THERE WERE NO SAFETY SYSTEM RESPONSES AND NONE WERE

NECECESSARY. THE ROOT CAUSE OF THIS EVENT WAS A COGNITIVE PERSONNEL ERROR ON THE PART OF THE PVNGS TECHNICIAN RESPONSIBLE FOR THE PERFORMANCE OF THE SURVEILLANCE TESTING. THE TECHNICIAN NEGLECTED TO PERFORM THE REQUIRED DAILY SURVEILLANCE TESTING IN A TIMELY MANNER. A HUMAN PERFORMANCE EVALUATION SYSTEM ANALYSIS OF THE PERSONNEL ERROR IS BEING CONDUCTED AND APPROPRIATE CORRECTIVE ACTIONS WILL BE TAKEN. PREVIOUS SIMILAR EVENTS OCCURRED AS DISCUSSED IN UNIT 2 LER 2-88-010 AND UNIT 3 LER 3-88-006.

[142] PALO VERDE 3 DOCKET 50-530 LER 89-013
INADVERTENT CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNAL ESF ACTUATION.
EVENT DATE: 092689 REPORT DATE: 102589 NSSS: CE TYPE: PWR

(NSIC 215757) AT APPROXIMATELY 1439 MST ON SEPTEMBER 26, 1989, PALO VERDE UNIT 3 WAS IN MODE 5 DURING A REFUELING OUTAGE WHEN A CONTROL ROOM OPERATOR INADVERTENTLY TURNED THE WRONG HANDSWITCH DURING POST MAINTENANCE TESTING WHICH RESULTED IN A CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNAL (CREFAS) BALANCE OF PLANT ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (BOP ESFAS) ACTUATION. THE ESF ACTUATION WAS VERIFIED NOT TO BE THE RESULT OF A VALID SIGNAL, ALL ACTUATED COMPONENTS WERE RETURNED TO THEIR NORMAL CONFIGURATION, AND TESTING WAS CONTINUED. SUBSEQUENTLY DURING TEST RESTORATION, AN INADVERTENT CONTROL ROOM VENTILATION ISOLATION ACTUATION SIGNAL (CRVIAS) BOP ESFAS ACTUATION OCCURRED DURING REMOVAL OF INSTRUMENTATION TEST LEADS. ALL CREFAS AND CRVIAS COMPONENTS RESPONDED PROPERLY. THE CAUSE OF THE CREFAS WAS A COGNITIVE PERSONNEL ERROR. THE CAUSE OF THE CRVIAS WAS THE INADVERTENT GROUNDING OF A HANDSWITCH CONTACT VIA THE TEST LEADS AS THEY WERE BEING REMOVED. AS CORRECTIVE ACTION FOR THE CREFAS, APPROPRIATE DISCIPLINARY MEASURES HAVE BEEN TAKEN. AS CORRECTIVE ACTION FOR THE CRVIAS THE WORK AUTHORIZATION DOCUMENTS BEING USED FOR BOP ESFAS HANDSWITCH TESTING WERE REVISED. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED PURSUANT TO 10CFR50.73.

[143] PEACH BOTTOM 2 DOCKET 50-277 LER 89-020
ENVIRONMENTAL QUALIFICATION NON-COMPLIANCE RESULTING IN INOPERABLE RESIDUAL HEAT REMOVAL PUMP MOTORS DUE TO INCOMPLETE PROCEDURAL GUIDANCE DURING INITIAL INSTALLATION.
EVENT DATE: 091589 REPORT DATE: 101689 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 215637) ON SEPTEMBER 15, 1989 AN EVALUATION OF A NON-CONFORMING ENVIRONMENTAL QUALIFICATION (EQ) CONDITION ASSOCIATED WITH THE 4 KILOVOLT (KV) MOTOR LEAD TO FIELD CABLE SPLICE INSULATION METHOD WAS COMPLETED. THE AS FOUND CONFIGURATION CONSISTED OF A MOLDED INSULATING BOOT HELD IN PLACE BY EITHER ELECTRICAL TAPE OR CABLE TIE WRAPS. AN INSPECTION WAS INITIATED TO DETERMINE THE SCOPE OF THE NON-CONFORMANCE WITH RESPECT TO OTHER 4 KV EQ MOTORS (RHR AND CORE SPRAY PUMP MOTORS). THREE UNIT 3 RHR PUMP MOTORS, FOUR UNIT 3 CORE SPRAY PUMP MOTORS, ONE UNIT 2 RHR PUMP MOTOR, AND ONE UNIT 2 CORE SPRAY PUMP MOTOR HAD THE NON-CONFORMING BOOT INSULATION CONFIGURATION. THE NONCONFORMING SPLICE CONFIGURATIONS WERE RESTORED TO THE CORRECT CONFIGURATIONS. THE ROOT CAUSE OF THIS EVENT WAS A LESS THAN ADEQUATE OR INCOMPLETE PROCEDURE(S) USED DURING INITIAL PLANT CONSTRUCTION, AND EACH SUBSEQUENT DE-TERMINATION/RE-TERMINATION OF THE RHR PUMP MOTORS. APPROPRIATE MAINTENANCE PROCEDURES ASSOCIATED WITH 4 KV MOTORS WILL BE REVIEWED AND REVISED AS NECESSARY TO INCLUDE THE DETAILS FOR DETERMINATION OF THE MOTOR LEADS. THERE WERE NO PREVIOUS SIMILAR EVENTS.

[144] PEACH BOTTOM 2 DOCKET 50-277 LER 89-021
CONTROL ROOM EMERGENCY VENTILATION SYSTEM ACTUATION DUE TO SPURIOUS HIGH RADIATION SIGNAL DURING TROUBLESHOOTING.
EVENT DATE: 092089 REPORT DATE: 101989 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 215638) ON SEPTEMBER 20, 1989 AT 1730, A CONTROL ROOM EMERGENCY VENTILATION ACTUATION OCCURRED. THE ACTUATION OCCURRED DUE TO A MOMENTARY FALSE HIGH RADIATION SIGNAL FROM THE CONTROL ROOM VENTILATION "A" RADIATION MONITOR. THE ACTUATION OCCURRED WHILE A TECHNICIAN WAS PERFORMING TROUBLESHOOTING ON THE RADIATION MONITOR SAMPLE PUMP FLOW SWITCH. A JUMPER WAS PLACED ACROSS THE FLOW SWITCH CONTACTS. THE FLOW SWITCH TERMINAL POINTS ARE ON THE SAME TERMINAL STRIP AS TERMINAL POINTS FOR THE RADIATION MONITOR CHECK SOURCE. ALTHOUGH THE CAUSE IS UNKNOWN, THE TROUBLESHOOTING ACTIVITY IS BELIEVED TO HAVE CREATED AN ELECTRICAL DISTURBANCE RESULTING IN THE FALSE HIGH RADIATION SIGNAL ON THE "A" MONITOR. THE "B" MONITOR INDICATED ONLY BACKGROUND RADIATION LEVELS. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. ATTEMPTS TO RECREATE THE DISTURBANCE DID NOT REPRODUCE THE ACTUATION. WIRE CONNECTIONS TO THE TERMINAL STRIP WERE VERIFIED TO BE TIGHT. THERE WERE NO PREVIOUS SIMILAR LERS.

[145] PEACH BOTTOM 3 DOCKET 50-278 LER 89-003
CONTAINMENT ATMOSPHERIC CONTROL PIPING SEISMIC SUPPORTS NOT IN ACCORDANCE WITH APPLICABLE DESIGN CRITERIA DUE TO DESIGN AND INSTALLATION ERRORS DURING ORIGINAL CONSTRUCTION.
EVENT DATE: 080889 REPORT DATE: 100689 NSSS: GE TYPE: BWR

(NSIC 215542) ON 6/16/89, A DESIGN CONTRACTOR PERFORMING CALCULATIONS FOR A PNEUMATIC TUBING MODIFICATION TO THE CONTAINMENT ATMOSPHERIC CONTROL (CAC) SYSTEM IDENTIFIED THAT OPERATORS FOR TWO VALVES WOULD HAVE EXCESSIVE DISPLACEMENTS DURING A POSTULATED SEISMIC EVENT. THE OPERATORS FOR VALVES AO-3509 AND AO-3510 WERE LACKING SEISMIC SUPPORTS. ALSO, 3 PIPING SUPPORTS WERE NOT IN ACCORDANCE WITH DESIGN FIELD SKETCHES. SUBSEQUENT SYSTEM WALKDOWN AND ANALYSIS REVEALED ON 8/8/89, THAT THE AS-FOUND CONFIGURATION, INCLUDING SEVEN ADDITIONAL DEFICIENT EXISTING PIPING SUPPORTS, WOULD RESULT IN PIPING STRESSES ADJACENT TO THE VALVES EXCEEDING CODE LIMITS. EVALUATION CONCLUDED THAT SYSTEM OPERABILITY COULD NOT BE ASSURED DURING AN OPERATING BASIS EARTHQUAKE SINCE LOADING CONDITIONS WOULD EXCEED YIELD STRESS OF THE PIPING MATERIAL. THE DEFICIENCIES HAVE EXISTED SINCE INITIAL PLANT CONSTRUCTION, AND ARE DUE TO DESIGN AND INSTALLATION ERRORS. NO ADVERSE SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS CONDITION. FAILURE OF 1 INCH CAC SYSTEM PIPING DURING A DESIGN BASIS EARTHQUAKE COULD HAVE RESULTED IN A LOSS OF PRIMARY CONTAINMENT INTEGRITY. THE DEFICIENCIES WILL BE CORRECTED PRIOR TO UNIT 3 RESTART. A SMALL BORE PIPING (3 INCH NOMINAL OR LESS) SURVEY WILL BE CONDUCTED FOR UNIT 2 AND UNIT 3 TO ASSURE INTEGRITY OF SMALL BORE PIPING.

[146] PEACH BOTTOM 3 DOCKET 50-278 LER 89-004
PRIMARY CONTAINMENT ISOLATION OF THE REACTOR WATER CLEANUP SYSTEM DUE TO FAILURE TO FOLLOW PROCEDURES WHILE INVESTIGATING A MALFUNCTIONING DIFFERENTIAL PRESSURE INDICATOR.
EVENT DATE: 092689 REPORT DATE: 102589 NSSS: GE TYPE: BWR

(NSIC 215696) AT 0902 AM, ON 9/26/89, WITH UNIT 3 IN COLD SHUTDOWN, A GROUP IIA PRIMARY CONTAINMENT ISOLATION ACTUATED, RESULTING IN AUTOMATIC CLOSURE OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM (INBOARD) ISOLATION VALVE AND TRIPPING OF THE "3B" RWCU PUMP. THE ROOT CAUSE OF THE EVENT WAS IMPROPER ACTION RESULTING FROM A PERSONNEL ERROR. A NON-LICENSED UTILITY MAINTENANCE PLANNER OPENED A RWCU LOW PRESSURE SIDE INSTRUMENT DRAIN VALVE. OPENING THE INSTRUMENT DRAIN VALVE SIMULATED A HIGH FLOW CONDITION IN THE RWCU SUCTION PIPING AND THE ISOLATION OCCURRED AS DESIGNED. AT 0932 AM THE ISOLATION LOGIC WAS RESET AND THE RWCU SYSTEM WAS RETURNED TO SERVICE. NO SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. HAD THIS EVENT OCCURRED AT POWER, THE TEMPORARY ISOLATION OF THE RWCU SYSTEM WOULD HAVE NO SIGNIFICANT IMPACT ON CONTINUED POWER OPERATIONS OR REACTOR WATER CHEMISTRY. THE INDIVIDUAL INVOLVED IN THE EVENT WAS COUNSELLED. THIS EVENT AND ITS CONSEQUENCES WERE DISCUSSED WITH APPROPRIATE MAINTENANCE, INSTRUMENT AND CONTROL, AND PLANT SUPERVISORY PERSONNEL. THERE WERE NO PREVIOUS SIMILAR EVENTS.

[147] PERRY 1 DOCKET 50-440 LER 88-025 REV 01
UPDATE ON OVERTRAVEL OF THE REACTOR PROTECTION SYSTEM POWER TRANSFER SWITCH RESULTS IN A LOSS OF POWER TO BOTH BUSSES AND A FULL RPS ACTUATION.
EVENT DATE: 061888 REPORT DATE: 101389 NSSS: GE TYPE: BWR

(NSIC 215529) ON 6/18/88, AT 0923, THE B REACTOR PROTECTION SYSTEM (RPS) BUS WAS INADVERTENTLY ALSO DEENERGIZED WHEN DEENERGIZING THE A RPS BUS, RESULTING IN A FULL RPS ACTUATION AND THE CONCURRENT NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) DIVISION II BALANCE OF PLANT (BOP) ISOLATION. NO CONTROL ROD MOVEMENT OCCURRED SINCE ALL RODS WERE ALREADY FULLY INSERTED. THIS EVENT OCCURRED WHEN A CONTROL ROOM OPERATOR WAS TRANSFERRING THE RPS BUS A POWER SUPPLY FROM THE A ALTERNATE FEED TRANSFORMER TO THE NORMAL A MOTOR-GENERATOR (MG) SET. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR WITH A CONTRIBUTING FACTOR OF A POOR HUMAN FACTORS DESIGN. THE OPERATOR INADVERTENTLY OVERSHOT NORMAL POSITION OF THE RPS POWER TRANSFER SWITCH WHEN TRANSFERRING THE SWITCH FROM ALTERNATE A TO NORMAL. EXTENSIVE INVESTIGATION INTO THE OPERATION OF THIS SWITCH IDENTIFIED THAT THE B RPS BUS POWER BREAK OCCURS WHEN THE POWER TRANSFER SWITCH TRAVELS APPROXIMATELY 15 DEGREES FROM NORMAL POSITION. AN INFORMATION TAG WAS IMMEDIATELY INSTALLED ON THE POWER TRANSFER SWITCH CAUTIONING THE OPERATORS THAT ONLY A SLIGHT AMOUNT OF OVERTRAVEL OF THIS SWITCH IS REQUIRED TO DEENERGIZE THE REDUNDANT RPS BUS. THIS INFORMATION TAG WAS LATER REPLACED BY PERMANENT LABELS FIXED BOTH ABOVE AND BELOW THE SWITCH. THE SYSTEM OPERATING INSTRUCTION WAS ALSO REVISED TO ADD THIS CAUTION.

[148] PERRY 1 DOCKET 50-440 LER 89-024 REV 01
UPDATE ON PERSONNEL ERROR DURING VALVE LINE-UP AND INSTRUMENTATION DEFICIENCIES CAUSE TECH SPEC VIOLATION OF SUPPRESSION POOL MAKE-UP SYSTEM.
EVENT DATE: 072589 REPORT DATE: 102789 NSSS: GE TYPE: BWR

(NSIC 215739) ON 7/23/89 AT 0413, ENTRY INTO OPERATIONAL CONDITION 2 WAS COMPLETED WITH THE SUPPRESSION POOL MAKEUP (SPMU) SYSTEM INOPERABLE, IN VIOLATION OF TECH SPEC 3.0.4. ON 7/25/89 A VENT VALVE ON THE REFERENCE LEG OF A SP LEVEL INSTRUMENT WAS FOUND OPEN AND UNCAPPED. ON 8/2/89 THE UPPER CONTAINMENT POOL (UCP) WAS FOUND TO BE BELOW THE WATER LEVEL ALLOWABLE LIMIT. THE INSTRUMENTS AND UCP LEVEL WERE RESTORED TO AN OPERABLE CONDITION. THE CAUSES OF THESE EVENTS ARE PERSONNEL ERROR AND INADEQUATE INSTRUMENT CALIBRATION INSTRUCTIONS. DURING SPMU SYSTEM INSTRUMENT FILL AND VENT ON 7/18/89 TECHNICIANS APPARENTLY FAILED TO PROPERLY RESTORE THE SYSTEM. ADDITIONALLY, FOLLOWING COMPLETION OF OUTAGE ACTIVITIES, UCP SKIMMER PLATES WERE NOT RETURNED TO THEIR CORRECT POSITION. THE UCP LEVEL INSTRUMENTS HAD BEEN REPLACED DURING THE REFUEL OUTAGE WITH A MORE RELIABLE DESIGN HOWEVER, VENDOR MANUAL CALIBRATION PROCEDURES DID NOT PROVIDE ADEQUATE INSTRUCTIONS FOR THE PERFORMANCE OF THE INSTRUMENT CALIBRATION. TO PREVENT RECURRENCE, THE TECHNICIANS INVOLVED WITH THE FILL AND VENT ACTIVITY HAVE BEEN COUNSELED, WHILE ALL OTHER INSTRUMENT AND CONTROL FIELD PERSONNEL HAVE BEEN INSTRUCTED ON THE LESSONS LEARNED FROM THIS EVENT. ALSO, THE ASSOCIATED INSTRUMENT MAINTENANCE INSTRUCTION WILL BE REVISED TO INCLUDE STEP-BY-STEP SIGNOFFS FOR SYSTEM VERIFICATION.

[149] PERRY 1 DOCKET 50-440 LER 89-026
RECIRCULATION SYSTEM FLOW TRANSIENT DUE TO SOLENOID VALVE FAILURE RESULTS IN THERMAL POWER LEVEL EXCEEDING THE OPERATING LICENSE REQUIREMENTS.
EVENT DATE: 091389 REPORT DATE: 101389 NSSS: GE TYPE: BWR
VENDOR: SPERRY VICKERS (SPERRY RAND CORP)

(NSIC 215496) ON 9/13/89 AT 1636 REACTOR THERMAL POWER LEVEL EXCEEDED THAT SPECIFIED IN THE OPERATING LICENSE DUE TO AN UNEXPECTED RECIRCULATION SYSTEM FLOW CONTROL TRANSIENT. PRIOR TO THE EVENT THE PLANT WAS IN OPERATIONAL CONDITION 1 (POWER OPERATION) AT 100% OF RATED POWER. OPERATORS RECOVERED FROM THE TRANSIENT BY INSERTING CONTROL RODS AND ADJUSTING RECIRCULATION LOOP FLOW IN ACCORDANCE

WITH APPROVED PLANT INSTRUCTIONS. THE TOTAL DURATION POWER EXCEEDED 100% WAS APPROXIMATELY 2 MINUTES AND 10 SECONDS. THE CAUSE OF THIS EVENT WAS A COMPONENT FAILURE. FAILURE OF A SOLENOID VALVE IN THE HYDRAULIC POWER UNIT CAUSED THE FLOW CONTROL VALVE TO STROKE FORM 56% TO A FINAL POSITION OF 89% OPEN. SUBSEQUENT TROUBLESHOOTING FOUND A VARNISH-TYPE MATERIAL ON THE PLUNGER INTERNAL TO THE ISOLATION SOLENOID VALVE. THIS MATERIAL IS BELIEVED TO HAVE CAUSED THE VALVE TO FAIL IN MIDPOSITION ALLOWING OIL TO PASS TO THE FLOW CONTROL VALVE ACTUATOR. TO PREVENT RECURRENCE THE FAILED ISOLATION SOLENOID VALVE HAS BEEN REPLACED. ANOTHER ISOLATION SOLENOID WILL BE DISASSEMBLED AND INSPECTED. BASED ON THE INSPECTION RESULTS, THE NEED TO REPLACE THE REMAINING SOLENOID VALVES AND THE FREQUENCY AND NECESSITY OF A PERIODIC REPLACEMENT OF THESE VALVES WILL BE EVALUATED.

[150] PERRY 1 DOCKET 50-440 LER 89-027
LOSS OF EMERGENCY SERVICE WATER A (DIVISION I) WHILE THE DIVISION III DIESEL GENERATOR WAS OUT-OF-SERVICE FOR MAINTENANCE PLACED UNIT INTO TECH SPEC 3.0.3.
EVENT DATE: 092589 REPORT DATE: 102089 NSSS: GE TYPE: BWR

(NSIC 215655) ON SEPTEMBER 25, 1989 AT 1846 A LOSS OF CONTROL POWER TO THE A EMERGENCY SERVICE WATER (ESW) SYSTEM PUMP DISCHARGE VALVE RENDERED THE ESW SYSTEM AND ITS DEPENDENT SYSTEMS INCLUDING THE DIVISION I EMERGENCY CORE COOLING SYSTEMS AND THE DIESEL GENERATOR (DG) INOPERABLE. DURING THE SAME TIME PERIOD THE DIVISION III ECCS (HIGH PRESSURE CORE SPRAY) AND DG WERE OUT-OF-SERVICE FOR MAINTENANCE PLACING THE PLANT INTO TECHNICAL SPECIFICATION (TS) 3.0.3. WORK WAS STOPPED ON DIVISION III AND RESTORATION BEGAN SO AS TO RETURN IT TO OPERATION. DIVISION III WAS DECLARED OPERABLE AT 2030 AND THE OPERATORS EXITED FROM TS 3.0.3. DIVISION I WAS DECLARED OPERABLE AT 0430 ON SEPTEMBER 26 AFTER REPLACEMENT AND RETEST OF THE ESW CONTROL POWER TRANSFORMER. THE CAUSE OF THIS EVENT IS PERSONNEL ERROR. WHILE THE INSTRUMENT AND CONTROL SECTION (ICS) TECHNICIANS WERE PERFORMING A SURVEILLANCE INSTRUCTION, THEY UNKNOWINGLY PLACED A JUMPER ON THE TERMINALS WITH THE CORRECT NUMBER BUT IN THE WRONG PANEL AND CLOSED THE CONNECTING TOGGLE SWITCH THEREBY BURNING OUT THE CONTROL POWER TRANSFORMER TO THE ESW PUMP DISCHARGE VALVE. INDEPENDENT VERIFICATION BY THE SECOND TECHNICIAN FAILED TO IDENTIFY THE ERROR. TO PREVENT RECURRENCE, THE ICS TECHNICIANS HAVE BEEN COUNSELED ON THE IMPORTANCE OF INDEPENDENT VERIFICATION AND ATTENTION TO DETAIL.

[151] PERRY 1 DOCKET 50-440 LER 89-028
OPERATION OF FUEL POOL COOLING AND CLEANUP SYSTEM TO THE UPPER CONTAINMENT POOLS CAUSES AIR EVACUATION AND CONTAINMENT VACUUM BREAKER ACTUATIONS.
EVENT DATE: 100889 REPORT DATE: 110389 NSSS: GE TYPE: BWR

(NSIC 215782) ON 10/8/89 IT WAS CONCLUDED THAT OPERATION OF THE FUEL POOL COOLING AND CLEANUP SYSTEM (FPCC) TO THE UPPER CONTAINMENT POOLS (UCP) CAUSED CONTAINMENT VACUUM BREAKER ACTUATIONS THAT OCCURRED BETWEEN 10/19 AND 10/8/89. FLOW FROM THE UCP TO THE FPCC SURGE TANK WAS SECURED FOR ADDITIONAL TESTING. CONTAINMENT PRESSURE INCREASED AND THE CONTAINMENT VACUUM BREAKERS STOPPED CYCLING. THE CAUSE OF CONTAINMENT VACUUM BREAKER ACTUATION WAS A PREVIOUSLY UNRECOGNIZED SYSTEM INTERACTION. THE FPCC WAS BEING OPERATED IN A MANNER WHICH ENTRAINED AIR IN THE RETURN LINE FROM THE UPPER CONTAINMENT POOLS TO THE FPCC SURGE TANK, LOCATED OUTSIDE CONTAINMENT. THIS REMOVAL OF AIR FROM THE CONTAINMENT RESULTED IN LOW PRESSURE IN CONTAINMENT AND SUBSEQUENT VACUUM BREAKER ACTUATION. IN ORDER TO PREVENT RECURRENCE, THE FPCC SYSTEM OPERATING INSTRUCTION WAS REVISED TO ELIMINATE AIR ENTRAINMENT. THE CONTAINMENT VACUUM RELIEF SYSTEM OPERATING INSTRUCTION WAS REVISED TO PROVIDE A CAUTION THAT FPCC UPPER CONTAINMENT POOLS OPERATION WITH A LOW SURGE TANK LEVEL CAN CAUSE CONTAINMENT VACUUM BREAKER ACTUATION. ADDITIONALLY, ACTIONS WERE INITIATED TO REVISE FPCC SURGE TANK LOW LEVEL ALARM SETPOINTS. AS PART OF THE ESTABLISHED REQUALIFICATION TRAINING

PROGRAM ALL PLANT LICENSED OPERATORS WILL BE INSTRUCTED ON THE LESSONS LEARNED FROM THIS EVENT.

[152] PILGRIM 1 DOCKET 50-293 LER 89-027
UNPLANNED AUTOMATIC START OF DIESEL GENERATOR 'A' AND ACTUATION OF THE RESIDUAL HEAT REMOVAL SYSTEM (RHRS)/LOW PRESSURE COOLANT INJECTION (LPCI) CIRCUITRY DURING LOGIC RELAY TESTING.
EVENT DATE: 090589 REPORT DATE: 100589 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215508) ON SEPTEMBER 5, 1989 AT 1805 HOURS, AN UNPLANNED ACTUATION OF A PORTION OF RHRS/LPCI LOOP SELECTION LOGIC CIRCUITRY OCCURRED DURING LOGIC RELAY TESTING. THE ACTUATION RESULTED IN AN AUTOMATIC START OF THE 'A' DIESEL GENERATOR; AUTOMATIC CLOSING OF THE RECIRCULATION SYSTEM LOOP 'B' PUMP DISCHARGE VALVE, AUTOMATIC OPENING OF THE RHRS/LPCI LOOP 'B' INJECTION VALVE; AUTOMATIC CLOSING OF THE RHRS/LPCI LOOP 'A' INJECTION VALVE AND THE AUTOMATIC OPENING OF THE RHRS LOOP 'A' HEAT EXCHANGER BYPASS VALVE. NO INJECTION FLOW OCCURRED AS A RESULT OF THE ACTUATION. THE CAUSE FOR THE ACTUATION WAS UTILITY NON-LICENSED INSTRUMENTATION AND CONTROL TECHNICIAN ERROR. THE TECHNICIAN JUMPERED THE CONTACTS OF THE INCORRECT LOGIC RELAY FOR THE TESTING THAT WAS BEING PERFORMED IN ACCORDANCE WITH APPROVED PROCEDURE. THE TESTING WAS SUSPENDED AND THE AFFECTED COMPONENTS WERE RESTORED TO NORMAL. THE PROCEDURE HAS BEEN REVISED TO INCLUDE VERIFICATION WHEN A JUMPER IS TO BE INSTALLED FOR THE TEST. THIS EVENT OCCURRED WHILE SHUTDOWN WITH THE REACTOR MODE SELECTOR SWITCH IN THE SHUTDOWN POSITION. THE CONTROL RODS WERE IN THE INSERTED POSITION. REACTOR VESSEL (RV) WATER TEMPERATURE WAS 160 DEGREES F AND RV PRESSURE WAS ZERO PSIG. REACTOR POWER LEVEL WAS ZERO%.

[153] PILGRIM 1 DOCKET 50-293 LER 89-028
HIGH PRESSURE COOLANT INJECTION SYSTEM INOPERABLE DUE TO FAILURE OF SPEED CONTROL RAMP GENERATOR SIGNAL CONVERTER MODULE.
EVENT DATE: 090789 REPORT DATE: 101089 NSSS: GE TYPE: BWR
VENDOR: WOODWARD GOVERNOR COMPANY

(NSIC 215552) ON 9/7/89 AT 1855 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) BEGAN AT THAT TIME. THE SYSTEM WAS DECLARED INOPERABLE BECAUSE A MECHANICAL OVERSPEED TRIP OCCURRED DURING A SURVEILLANCE TEST. THE CAUSE FOR THE OVERSPEED TRIP WAS FAILURE OF THE RAMP GENERATOR SIGNAL CONVERTER RGSC . MODULE THAT IS PART OF THE SYSTEM'S TURBINE SPEED CONTROL SYSTEM. THE RGSC MODULE WAS MANUFACTURED BY WOODWARD GOVERNOR CO. PART NUMBER 8271-083 . THE FAILED MODULE WAS REMOVED AND WILL BE SENT TO THE MANUFACTURER FOR EXAMINATION. A NEW MODULE WAS INSTALLED AND CALIBRATED AND TURBINE SPEED CONTROL SYSTEM WAS SUBSEQUENTLY CALIBRATED. THE HPCI SYSTEM WAS TESTED FOR OPERABILITY WITH SATISFACTORY RESULTS AND THE SEVEN DAY LCO WAS TERMINATED ON 9/9/89 AT 2023 HOURS. WHILE THE HPCI SYSTEM WAS INOPERABLE, TESTING OF APPLICABLE SYSTEMS, INCLUDING THE REACTOR CORE ISOLATION COOLING SYSTEM, WAS CONDUCTED WITH SATISFACTORY RESULTS. THIS EVENT OCCURRED DURING POWER OPERATION WITH THE REACTOR MODE SELECTOR SWITCH IN THE RUN POSITION. THE REACTOR POWER LEVEL WAS APPROXIMATELY 25 PERCENT. THE REACTOR VESSEL (RV) PRESSURE WAS 952 PSIG WITH THE RV WATER TEMPERATURE AT 538F. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(V)(D).

[154] PILGRIM 1 DOCKET 50-293 LER 89-029
LOCKED HIGH RADIATION AREA DOOR TO THE CONDENSER BAY FOUND UNSECURED.
EVENT DATE: 091489 REPORT DATE: 101689 NSSS: GE TYPE: BWR

(NSIC 215641) ON 9/14/89, A LOCKED HIGH RADIATION AREA (LHRA) ACCESS DOOR WAS

FOUND TO HAVE BEEN UNSECURED FROM ABOUT 1120 HRS TO 1235 HRS. THE DOOR LATCHING MECHANISM WAS LOCKED BUT THE LATCH DID NOT EXTEND INTO THE DOORJAMB. THE UNSECURED DOOR PROVIDES 1 OF 3 ACCESS POINTS TO THE CONDENSER BAY (CB). WHILE UNSECURED, THE DOOR WAS ACCESSED BY TWO INDIVIDUALS WHO HAD NOT BEEN AUTHORIZED ENTRY. THE DOSE RECEIVED WAS BELOW ALLOWABLE LIMITS. POCKET DOSIMETER READINGS FOR THE ENTRIES WERE LOGGED AT 15 MILLI-REM AND 5 MILLI-REM. THE EVENT WAS CONTRARY TO THE REQUIREMENTS OF TECH SPEC 6.13.2 WHICH REQUIRED DOOR TO BE LOCKED. ALSO, ENTRY WAS MADE WITHOUT THE REQUIRED DOSE RATE INDICATING DEVICE OR DOSE INTEGRATING DEVICE WITH PRESET ALARM, AND THE TWO INDIVIDUALS WERE NOT ACCOMPANIED BY AN INDIVIDUAL QUALIFIED IN RADIATION PROTECTION PROCEDURES. INITIAL ACTIONS TAKEN VERIFIED THAT ONLY TWO INDIVIDUALS HAD GAINED UNAUTHORIZED ACCESS, AND THAT LHRA DOORS WERE SECURE AND LHRA DOOR KEYS WERE ACCOUNTED FOR. CORRECTIVE ACTIONS INCLUDE SPECIALIZED TRAINING ON ACCESS REQUIREMENTS TO HIGH RADIATION AREAS, AND DEVELOPMENT OF A JOB AID TO ASSURE THAT ACCESSIBLE LHRA DOORS ARE PROPERLY CHECKED UPON EXITING THE APPLICABLE AREA. THE PLANT WAS AT 75% POWER AT THE TIME OF EVENT. THE REACTOR MODE SELECTOR SWITCH WAS IN RUN POSITION, REACTOR VESSEL PRESSURE WAS 995 PSIG, AND REACTOR COOLANT TEMPERATURE WAS APPROX. 540F.

[155] PILGRIM 1 DOCKET 50-293 LER 89-030
CONTROL ROOM HIGH EFFICIENCY AIR FILTRATION SYSTEM FLOWRATE NON-CONSERVATIVE DUE TO PROCEDURE ERROR.
EVENT DATE: 092689 REPORT DATE: 102189 NSSS: GE TYPE: BWR

(NSIC 215698) ON 9/26/89 AT APPROX. 1530 HRS, A PREVIOUS CONDITION CONTRARY TO TECH SPECS 3.7.B.2.D AND 4.7.B.2.A REGARDING THE CONTROL ROOM HIGH EFFICIENCY AIR FILTRATION (CRHEAF) SYSTEM WAS IDENTIFIED. THE PREVIOUS CONDITION INVOLVED THE FLOWRATE OF AIR FOR EACH OF THE TWO CRHEAF SYSTEM TRAINS ('A' AND 'B'). SPECIFICALLY, THE CORRECTED FLOWRATE, CALCULATED AT 862 CUBIC FEET PER MINUTE (CFM), WAS LESS THAN THE MINIMUM SPECIFIED VALUE OF 1000 CFM +/- 10 PERCENT. THE CAUSE HAS BEEN ATTRIBUTED TO A TRANSCRIPTION ERROR FROM A DRAWING THAT OCCURRED DURING THE PROCESS OF REVISING A CRHEAF SYSTEM SURVEILLANCE TEST PROCEDURE. THE ERROR INVOLVED USING THE INCORRECT FLOW AREA WHEN CALCULATING THE CRHEAF SYSTEM FLOWRATE IN ACCORDANCE WITH THE PROCEDURE. CORRECTIVE ACTION TAKEN INCLUDED REVISING THE PROCEDURE AND SUBSEQUENTLY REPERFORMING THE TEST USING THE CORRECTED PROCEDURE. THE CONDITION WAS DISCOVERED DURING POWER OPERATION WITH THE REACTOR MODE SELECTOR SWITCH IN THE RUN POSITION. THE REACTOR POWER LEVEL WAS APPROXIMATELY 75 PERCENT. THE REACTOR VESSEL (RV) PRESSURE WAS APPROXIMATELY 990 PSIG WITH THE RV WATER TEMPERATURE AT 542 DEGREES. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73 SUBPARTS (A)(2)(I)(B) AND (A)(2)(VII)(D), AND THIS CONDITION POSED NO THREAT TO THE HEALTH AND SAFETY OF THE PUBLIC.

[156] POINT BEACH 1 DOCKET 50-266 LER 89-008
ATWS MITIGATING ACTUATION CIRCUITRY NOT ENABLED AS REQUIRED BY TECHNICAL SPECIFICATIONS.
EVENT DATE: 091089 REPORT DATE: 101989 NSSS: WE TYPE: PWR

(NSIC 215695) ON 9/10/89, POINT BEACH NUCLEAR PLANT UNIT 1 WAS DECREASING POWER FROM GREATER THAN 40% TO LESS THAN 40%. AT SOMEWHAT LESS THAN 40% POWER, THE ATWS MITIGATING ACTUATION CIRCUITRY (AMSAC) IS DESIGNED TO BE BYPASSED AUTOMATICALLY BASED UPON FIRST-STAGE TURBINE PRESSURE. POWER AS INDICATED BY FIRST-STAGE PRESSURE AT THIS POWER LEVEL DURING POWER DECREASES IS LESS THAN REACTOR POWER. IN FACT, THE AMSAC WAS AUTOMATICALLY BYPASSED AT ABOUT 42% REACTOR POWER. TECHNICAL SPECIFICATIONS REQUIRE THE AMSAC SYSTEM TO BE ENABLED AT GREATER THAN 40% REACTOR POWER. POWER WAS REDUCED TO 38% AND HELD UNTIL THE BISTABLE COULD BE RESET. THE ENABLE/DISABLE BISTABLE SETPOINT WAS RESET TO 30% TO ENSURE THE OPERATION OF THE SYSTEM AT LESS THAN 40% REACTOR POWER. A REVISION TO THE TECHNICAL SPECIFICATIONS WILL BE SUBMITTED TO THE NRC TO CLARIFY THAT TURBINE POWER OF GREATER THAN 40% IS THE POWER LEVEL AT WHICH AMSAC IS REQUIRED.

[157] PRAIRIE ISLAND 1 DOCKET 50-282 LER 89-014
SURVEILLANCE TEST PERFORMED LATE DUE TO PERSONNEL ERROR.
EVENT DATE: 090589 REPORT DATE: 100589 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)

(NSIC 215549) TECHNICAL SPECIFICATION TABLE TS.4.1.1, ITEMS #19 AND 39 REQUIRE THAT AUTOMATIC LOAD SEQUENCING AND LOSS OF VOLTAGE ON THE 4KV SAFEGUARDS BUSES BE TESTED MONTHLY. THIS IS NORMALLY ACCOMPLISHED DURING SURVEILLANCE TESTING OF THE SAFEGUARDS DIESEL GENERATORS. DL DIESEL GENERATOR CAN SUPPLY BUS 15 FOR UNIT 1 AND BUS 26 FOR UNIT 2. SEPARATE PROCEDURES, SP1093.1 (DL DIESEL GENERATOR TO BUS 15) AND SPI093.2 (DI DIESEL GENERATOR TO BUS 26), ARE EACH PERFORMED ONCE A MONTH. SPI093.2 WAS SCHEDULED TO BE PERFORMED ON AUGUST 22, 1989. WHEN PREPARING TO PERFORM THE PROCEDURE, SPI093.1 WAS APPARENTLY SELECTED FROM THE FILE DRAWER BY MISTAKE, AND WAS PERFORMED SATISFACTORILY. ON SEPTEMBER 5, 1989, WHEN SPI093.1 WAS SCHEDULED TO BE PERFORMED, OPERATIONS PERSONNEL RECALLED THAT SPI093.1 HAD BEEN PERFORMED ONLY TWO WEEKS PREVIOUSLY. A REVIEW OF THE REACTOR LOGS SHOWED THAT SPI093.1 HAD BEEN PERFORMED ON AUGUST 22, 1989, INSTEAD OF THE SCHEDULED SP1093.2. PROPER NOTIFICATIONS WERE MADE, AND SPI093 2 WAS PERFORMED IMMEDIATELY AND THE TEST WAS SATISFACTORY. CAUSE OF THE EVENT WAS PERSONNEL ERROR IN SELECTING THE INCORRECT PROCEDURE FROM THE CONTROLLED PROCEDURE FILE.

[158] PRAIRIE ISLAND 1 DOCKET 50-282 LER 89-015
AUTOMATIC CONTROL ROOM ISOLATION AND START OF CONTROL ROOM CLEANUP FAN DUE TO FAILURE OF A CHLORINE GAS MONITOR.
EVENT DATE: 090589 REPORT DATE: 100589 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)
VENDOR: M D A SCIENTIFIC, INC.

(NSIC 215550) THIS EVENT CONSISTS OF TWO SIMILAR OCCURRENCES. ON SEPTEMBER 5, 1989, THE AUTOMATIC ACTUATION OF 122 CONTROL ROOM SPECIAL VENTILATION SYSTEM, WAS CAUSED BY A SPURIOUS HIGH CHLORINE SIGNAL. ON SEPTEMBER 23, 1989 THE AUTOMATIC ACTUATION OF 121 CONTROL ROOM SPECIAL VENTILATION SYSTEM WAS CAUSED BY A SPURIOUS HIGH CHLORINE SIGNAL. TWO DIFFERENT CHLORINE DETECTORS WERE INVOLVED. THE CAUSE OF THE FIRST OCCURRENCE WAS DETERMINED TO BE A FALSE HIGH CHLORINE SIGNAL GENERATED WHEN ITS OPTIC BLOCK MALFUNCTIONED. THE OPTICS BLOCK WAS REPLACED AND THE MONITOR WAS RETURNED TO SERVICE. THE CAUSE OF THE SECOND OCCURRENCE WAS DETERMINED TO BE FROM A BROKEN CHLORINE SENSITIVE PAPER TAPE. THE TAPE WAS REPLACED WITH A NEW ONE AND THE MONITOR WAS RETURNED TO SERVICE. CORRECTIVE ACTIONS TO BE TAKEN INCLUDE THE INSTALLATION OF ADDITIONAL MONITORS AND A MODIFICATION OF THE ACTUATION LOGIC WHICH INITIATES THE AUTOMATIC ACTUATION. THE CONTROL ROOM VENTILATION SYSTEM ENTERED ITS SAFEGUARDS MODE OF OPERATION AS DESIGNED.

[159] PRAIRIE ISLAND 1 DOCKET 50-282 LER 89-016
AUTO-START OF TRAIN B OF THE AUXILIARY BUILDING SPECIAL VENTILATION SYSTEM AS A RESULT OF A RADIATION MONITOR SPIKE.
EVENT DATE: 090889 REPORT DATE: 101089 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)
VENDOR: L N D, INC.
NUCLEAR MEASUREMENTS CORP.

(NSIC 215551) ON SEPTEMBER 8, 1989. BOTH UNITS WERE OPERATING AT 100% POWER. AT 1555 HOURS, THE CONTROL ROOM RECEIVED A TRAIN B HIGH RADIATION ALARM, WHICH INITIATED AN AUTOMATIC START OF THE AUXILIARY BUILDING SPECIAL VENTILATION SYSTEM (ABSVS). THIS WAS A NON-ENGINEERED SAFEGUARDS FEATURE ACTUATION OF AN ESF SYSTEM. RADIATION MONITOR 2R-30, WHICH ACTUATES THE ABSVS, WAS FOUND TO BE IN ALARM. 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 ABSVS TO THE NORMAL STANDBY CONDITION AND RETURNED THE AUXILIARY BUILDING

NORMAL VENTILATION SYSTEM TO SERVICE. THE RADIATION MONITOR DETECTOR TUBE AND ONE ELECTRONICS CARD WERE REPLACED AND THE MONITOR TESTED SATISFACTORILY. A RADIATION MONITOR MODULE UPGRADE IS PLANNED.

[160] PRAIRIE ISLAND 1 DOCKET 50-282 LER 89-017
DISCOVERY THAT THE PRESENT POSITION OF THE TRANSFER SWITCH FOR POWER SUPPLYING CONTROL AND PROTECTION RELAYS FOR D2 DIESEL GENERATOR DOES NOT MEET THE REQUIREMENTS OF APP. R.
EVENT DATE: 091489 REPORT DATE: 101689 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)

(NSIC 215669) AN INTERNAL TECHNICAL REVIEW OF THE PLANT 10 CFR 50 APPENDIX R DESIGN AGAINST THE AS-BUILT CONFIGURATION IS BEING CONDUCTED. ON 9/14/89, THIS REVIEW REVEALED THAT THE NORMAL CONTROL POWER TO D2 EMERGENCY DIESEL GENERATOR WAS SUPPLIED FROM 22 BATTERY INSTEAD OF 12 BATTERY AS DESCRIBED IN THE SAFE SHUTDOWN ANALYSIS. ONE FIRE AREA EXISTS CONTAINING NORMAL CONTROL POWER CABLES FOR BOTH D1 AND D2 DIESEL GENERATORS, THUS CREATING A CONDITION THAT WAS OUT OF COMPLIANCE WITH THE REQUIREMENTS OF 10 CFR 50 APPENDIX R. THE CONTROL POWER FOR D2 DIESEL GENERATOR WAS TRANSFERRED TO 12 BATTERY (THE ALTERNATE OR STANDBY SUPPLY), THEREBY PLACING THE PLANT BACK WITHIN ITS 10 CFR 50 APPENDIX R DESIGN BASIS. ADMINISTRATIVE CONTROLS OF THE MODIFICATION PROCESS IN 1983 HAD NOT YET IMPLEMENTED THE MEANS TO ENSURE INCORPORATION OF 10 CFR 50 APPENDIX R CRITERIA INTO THE DESIGN. THIS SITUATION WAS RECTIFIED WITH THE ADMINISTRATIVE CONTROL PROCEDURE REVISIONS ISSUED IN 1984. THE INTERNAL TECHNICAL REVIEW IS CONTINUING AND WILL BE COMPLETED TO ENSURE THAT THE AS-BUILT CONFIGURATION IS IN COMPLIANCE WITH 10 CFR 50 APPENDIX R.

[161] QUAD CITIES 1 DOCKET 50-254 LER 87-017 REV 91
UPDATE ON HIGH PRESSURE COOLANT INJECTION SYSTEM INOPERABLE DUE TO INVALID SYSTEM ISOLATION FROM FAILED DIFFERENTIAL PRESSURE TRANSMITTER.
EVENT DATE: 080587 REPORT DATE: 102489 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 215769) AUGUST 5, 1987, UNIT ONE WAS OPERATING IN THE RUN MODE AT 100 PERCENT OF RATED CORE THERMAL POWER. AT 1212 HOURS DURING THE PERFORMANCE OF THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM MONTHLY OPERABILITY TEST, A GROUP IV ISOLATION WAS RECEIVED WHICH RESULTED IN CLOSURE OF THE HPCI STEAM SUPPLY VALVES. THE HPCI SYSTEM WAS DECLARED INOPERABLE AND TECHNICAL SPECIFICATION REQUIRED SURVEILLANCES WERE INITIATED. THE ISOLATION WAS THE RESULT OF A FAILED HPCI STEAMLINE DIFFERENTIAL PRESSURE TRANSMITTER THAT DETECTS EXCESSIVE FLOW IN THE STEAMLINE. THE EXACT CAUSE OF THE TRANSMITTER FAILURE WAS A LOSS OF OIL IN THE TRANSMITTER'S SENSING CELL. THE TRANSMITTER WAS REPLACED WITH A LIKE-FOR-LIKE REPLACEMENT. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 1FR50.73(A)(2)(IV) AND (A)(2 (V).

[162] QUAD CITIES 1 DOCKET 50-254 LER 89-014
EXCEEDING TECH SPEC LEAKAGE LIMITS FOR CONTAINMENT ISOLATION VALVES AND MAIN STEAM ISOLATION VALVES DUE TO UNKNOWN CAUSES.
EVENT DATE: 091089 REPORT DATE: 100289 NSSS: GE TYPE: BWR

(NSIC 215535) ON SEPTEMBER 10, 1989, QUAD CITIES UNIT ONE WAS SHUT DOWN FOR THE END OF CYCLE 10 REFUELING AND MAINTENANCE OUTAGE. AT 0955 HOURS WHILE LOCAL LEAK RATE TESTING (LLRT) THE DRYWELL/TORUS PURGE VOLUME (VB) BOUNDED BY AO-1-1601-23, 24, 60, 61, 62, AND 63 VALVES (NH)(ISV), IT WAS DETERMINED THAT THE MEASURED COMBINED LEAKAGE RATE OF 522.0 STANDARD CUBIC FEET PER HOUR (SCFH) FROM ALL PENETRATIONS (PEN) AND VALVES (V) EXCLUDING THE MAIN STEAM (SB) ISOLATION VALVES (MSIV)(ISV) HAD EXCEEDED THE TECHNICAL SPECIFICATION 3.7.A.2.A.2 LIMIT OF 293.75 SCFH (0.6 LA). ON SEPTEMBER 11, 1989, AT 2035 HOURS WHILE PERFORMING LLRT ON THE

UNIT ONE MAIN STEAM ISOLATION VALVES, AO 1-203-2A AND AO 1-203-2D WERE FOUND TO LEAK IN EXCESS OF THE TECHNICAL SPECIFICATION (3.7.A.2.A.3) LIMIT OF 11.5 SCFH. AO 1-203-2A LEAKED AT A RATE OF 27.65 SCFH AND AO 1-203-2D LEAKED AT 24.19 SCFH. THE ROOT CAUSE OF THE EXCESSIVE LEAKAGES WILL NOT BE KNOWN UNTIL REPAIRS HAVE BEEN COMPLETED AND THE VALVES HAVE BEEN RETESTED. NO CORRECTIVE ACTION HAS BEEN TAKEN AT THIS TIME. A SUPPLEMENTAL REPORT WILL BE ISSUED TO DOCUMENT THE CAUSES AND CORRECTIVE ACTIONS TAKEN TO BRING THE COMBINED LEAKAGE AND THE MSIV LEAKAGE BELOW THE REQUIRED LIMITS. THIS REPORT IS BEING SUBMITTED TO COMPLY WITH 10CFR50.73(A)(2)(I)(B .

[163] QUAD CITIES 1 DOCKET 50-254 LER 89-015
OFF GAS ISOLATION AFTER REACTOR PROTECTION SYSTEM BUS POWER SUPPLY CHANGE DUE TO INADEQUATE PROCEDURES.
EVENT DATE: 091689 REPORT DATE: 101689 NSSS: GE TYPE: BWR

(NSIC 215632) ON SEPTEMBER 16, 1989. UNIT ONE WAS IN THE REFUEL MODE AT 0 PERCENT POWER. BOTH OFF GAS (WF) RADIATION MONITORS (MON) WERE DOWNSCALE, WHICH IS THE NORMAL CONDITION WHEN THE REACTOR (RCT) IS IN THE REFUEL MODE. THE OPERATIONS DEPARTMENT COMPLETED A POWER SUPPLY CHANGEOVER (JX) OF THE REACTOR PROTECTION BUS FROM THE MOTOR GENERATOR SETS (MG) TO BUS (BU) 15-2. DURING THE TRANSFER, THE LOSS OF POWER TO THE "A" OFF GAS MONITOR CAUSED ITS CONTACTS TO OPEN GIVING AN UPSCALE RADIATION SIGNAL, THUS STARTING THE OFF GAS TIMER (TMR). THE TIMER COMPLETED ITS 15-MINUTE CYCLE AND AS DESIGNED ISOLATED THE OFF GAS SYSTEM. THE PROPER PROCEDURES WERE USED TO COMPLETE THIS CHANGEOVER; HOWEVER, THEY DO NOT ALERT THE OPERATORS TO THE POTENTIAL ENGINEERED SAFETY FEATURE (ESF) ACTUATION. THE PROCEDURES WILL BE CHANGED TO REFLECT THE POTENTIAL ESF ACTUATION. THE OFF GAS SYSTEM WAS RESET AND NO OTHER UNEXPECTED ACTIONS OCCURRED. THIS REPORT IS SUBMITTED TO COMPLY WITH THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV).

[164] QUAD CITIES 2 DOCKET 50-265 LER 87-012 REV 01
UPDATE ON SUPPRESSION CHAMBER TO DRYWELL VACUUM BREAKER FAILURE TO CLOSE DUE TO TEST CYLINDER BINDING.
EVENT DATE: 091887 REPORT DATE: 103089 NSSS: GE TYPE: BWR
VENDOR: ATWOOD & MORRILL CO., INC.

(NSIC 215770) ON SEPTEMBER 18, 1987, QUAD CITIES UNIT TWO WAS IN THE RUN MODE AT 90% OF CORE THERMAL POWER. AT 1300 HOURS, WHILE PERFORMING QOS 1600-1 (SUPPRESSION CHAMBER TO DRYWELL VACUUM BREAKERS MONTHLY EXERCISE), THE 2-1601-33E VACUUM BREAKER REMAINED OPEN AFTER TESTING. EFFORTS WERE DIRECTED AT RECLOSING THE VACUUM BREAKER. AT 1350 HOURS, THESE EFFORTS WERE SUCCESSFUL. BECAUSE OF THE STUCK OPEN VACUUM BREAKER, NRC NOTIFICATION HAD BEEN COMPLETED AT 1346 HOURS. POSITION INDICATION PROBLEMS WERE ALSO IDENTIFIED FOR VACUUM BREAKERS 2-1601-32A AND 33A. SUPPRESSION CHAMBER TO DRYWELL DIFFERENTIAL PRESSURE WAS REESTABLISHED AT 1725 HOURS. THE CAUSE OF THE 2-1601-33E FAILURE WAS DETERMINED TO BE DUE TO BINDING IN THE TEST CYLINDER PORTION OF THE VACUUM BREAKER. THIS IS CONSIDERED AN ISOLATED EVENT AND DOES NOT AFFECT THE OPERATION NECESSARY TO MITIGATE THE CONSEQUENCES OF A LOSS OF COOLANT ACCIDENT. THE POSITION INDICATION PROBLEMS ON 2-1601-32A AND 33A SHOULD BE CORRECTED UPON COMPLETION OF MODIFICATION M-4-1(2)-88-009 WHICH IS THE RESULT OF THE INSPECTION AND EVALUATION PERFORMED ON UNIT ONE'S VACUUM BREAKERS. THIS REPORT IS SUBMITTED PER 10 CFR 50.73(A)(2)(II).

[165] QUAD CITIES 2 DOCKET 50-265 LER 88-003 REV 01
UPDATE ON REACTOR CORE ISOLATION COOLING INOPERABLE DUE TO FAILED GOVERNOR ACTUATOR.
EVENT DATE: 030188 REPORT DATE: 102489 NSSS: GE TYPE: BWR
VENDOR: TERRY STEAM TURBINE COMPANY
WOODWARD GOVERNOR COMPANY

(NSIC 215771) ON 3/1/88, UNIT TWO WAS IN THE RUN MODE AT APPROXIMATELY 96 PERCENT POWER. AT 1005 HOURS WHILE PERFORMING QOS 1300-1, "REACTOR CORE ISOLATION COOLING (RCIC) MONTHLY TEST," IT WAS FOUND THAT THE RCIC PUMP (P) COULD ONLY ACHIEVE 400 GPM (RATED) AGAINST A DISCHARGE PRESSURE OF 500 PSIG WITH A TURBINE SPEED OF 3,000 RPM. NORMAL PRESSURE AND SPEED IS 1,250 PSIG AT 4,500 RPM. RCIC WAS DECLARED INOPERABLE AND NRC NOTIFICATION OF THIS EVENT WAS COMPLETED AT 1155 HOURS PER 10CFR50.72. WORK REQUEST Q64717 WAS WRITTEN TO INVESTIGATE THE PROBLEM AND TO MAKE REPAIRS. IT WAS DETERMINED THAT THE CAUSE OF THIS EVENT WAS DUE TO A FAILED HYDRAULIC ACTUATOR (15) ON THE TURBINE (TRB) GOVERNOR (65) VALVE (V). THE ACTUATOR WAS REPLACED ON 3/2/88, AND RCIC WAS SUCCESSFULLY TESTED AT 1600 HOURS. THE FAILED ACTUATOR WAS SENT TO THE MANUFACTURER WHO DETERMINED THAT THE PROBLEM WAS AN ELECTRONIC FAILURE. THIS REPORT IS PROVIDED PER 10CFR50.73(A)(2)(V)(B).

[166] RIVERBEND 1 DOCKET 50-458 LER 89-033
ESF ACTUATIONS RESULT FROM POWER LINE CONDITIONING TRANSFORMER FAILURE DUE TO CAUSE UNKNOWN.
EVENT DATE: 091689 REPORT DATE: 101689 NSSS: GE TYPE: BWR
VENDOR: ELGAR, CORP.

(NSIC 215658) AT APPROXIMATELY 0900 ON 9/16/89 WITH THE UNIT IN OPERATIONAL CONDITION 1 AT 100% POWER, ELECTRICAL POWER WAS LOST TO DIVISION II 120VAC DISTRIBUTION PANEL 1SCM*PNL01B. LOSS OF POWER TO THIS PANEL RESULTED IN THE AUTO-START OF THE DIVISION II STANDBY GAS TREATMENT AND ANNULUS MIXING SYSTEMS AND THE DIVISION II FUEL BUILDING FILTER TRAIN. ADDITIONALLY, THE REACTOR BUILDING FLOOR AND EQUIPMENT DRAIN ISOLATION VALVES ISOLATED, REACTOR WATER SAMPLING SYSTEM ISOLATION VALVE CLOSED, AUXILIARY BUILDING EQUIPMENT DRAINS ISOLATED AND INSTRUMENT AIR TO THE CONTAINMENT AIR LOCKS ISOLATED. A HALF ISOLATION SIGNAL OF THE MAIN STEAM ISOLATION VALVES OCCURRED AND SEVERAL DIVISION II SYSTEMS WERE DECLARED INOPERABLE FOR VARIOUS REASONS. SEVERAL DIVISION II STATUS LIGHTS AND ANNUNCIATORS WERE ALSO INOPERABLE. THE CAUSE OF THE POWER LOSS WAS DETERMINED TO BE THE FAILURE OF AN ELGAR POWER LINE CONDITIONING TRANSFORMER 1SCM*XRC14B1. THE ROOT CAUSE OF THE FAILURE IS INDETERMINATE AT THIS TIME, HOWEVER, GSU'S INVESTIGATION IS CONTINUING. UPON COMPLETION OF THIS INVESTIGATION, A SUPPLEMENTAL REPORT WILL BE PROVIDED. SINCE ALL PLANT SYSTEMS WERE VERIFIED TO HAVE PERFORMED AS DESIGNED, THERE WAS NO IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT.

[167] RIVERBEND 1 DOCKET 50-458 LER 89-034
FAILURE TO PERFORM TECH SPEC SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 093089 REPORT DATE: 102789 NSSS: GE TYPE: BWR

(NSIC 215789) AT 1535 ON 9/30/89 WITH THE UNIT IN OPERATIONAL CONDITION 3 (HOT SHUTDOWN), IT WAS DETERMINED THAT THE 10 HOUR RUN OF THE "B" STANDBY GAS TREATMENT SYSTEM (SGTS) TRAIN REQUIRED BY TECH SPEC (TS) 4.6.5.4.A HAD NOT BEEN PERFORMED WITHIN THE ALLOWABLE SURVEILLANCE TOLERANCE WHICH HAD ENDED 9/4/89. THIS CAUSED THE "B" SGTS TRAIN TO BE INOPERABLE (PER TS) FOR A PERIOD OF 26 DAYS, WHICH EXCEEDED THE 7 DAY LIMIT OF THE ACTION STATEMENT. ALSO DURING THIS TIME PERIOD, THE "A" SGTS TRAIN WAS INOPERABLE DURING SHORT PERIODS. THE IMMEDIATE ACTION TAKEN BY THE SHIFT SUPERVISOR WAS TO COMPLETE THE 10 HOUR RUN OF THE "B" SGTS TRAIN, RESTORING THE "B" SGTS TO OPERABLE STATUS. THE ROOT CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE HUMAN ERROR IN THE SCHEDULING OF SURVEILLANCE TEST PROCEDURE (STP)-257-0201, "STANDBY GAS TREATMENT SYSTEM OPERABILITY TEST." CORRECTIVE ACTION WILL CONSIST OF: 1) SPLITTING THE STP TO PROVIDE BETTER TRACKING FOR INDIVIDUAL TRAINS, 2) TRAINING TO BE GIVEN TO OPERATIONS PERSONNEL, 3) TRAINING OF STP SCHEDULING GROUP, AND 4) AN ADDITIONAL WEEKLY REVIEW WILL BE PERFORMED ON THE PREVIOUS WEEKS PERFORMANCES. EVALUATION OF THIS EVENT HAS CONCLUDED THAT THE FILTER TRAIN COULD HAVE PERFORMED ITS DESIGNED SAFETY FUNCTION IN THE EVENT OF AN ACCIDENT.

[168] RIVERBEND 1 DOCKET 50-458 LER 89-035
REACTOR AUTOMATICALLY SCRAMMED DURING SURVEILLANCE TEST DUE TO A DEFECTIVE TEST SWITCH.
EVENT DATE: 093089 REPORT DATE: 102789 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215783) AT 0340 ON 9/30/89 WITH THE UNIT AT 78% POWER(OPERATIONAL 1), THE REACTOR AUTOMATICALLY SCRAMMED DURING THE PERFORMANCE OF A ROUTINE REACTOR PROTECTION SYSTEM (RPS) - MAIN STEAM LINE ISOLATION VALVE CLOSURE MONTHLY SURVEILLANCE TEST. THE CAUSE OF THE SCRAM WAS DETERMINED TO BE A DEFECTIVE TEST SWITCH IN THE REACTOR PROTECTION SYSTEM. THE DEFECTIVE GENERAL ELECTRIC SWITCH (MODEL CR 2940) HAD INADVERTENTLY PLACED THE SYSTEM IN A CONSERVATIVE STATE ALLOWING A SCRAM TO OCCUR AFTER A HALF SCRAM SIGNAL WAS INITIATED AS REQUIRED BY THE SURVEILLANCE TEST. THE TEST SWITCH WAS REPLACED VIA A PROMPT MAINTENANCE WORK ORDER. PRIOR TO SUCCESSFULLY COMPLETING THE SURVEILLANCE TEST, THE PROCEDURE WAS REVISED TO REQUIRE VERIFICATION OF PROPER SWITCH POSITION PRIOR TO PERFORMING THE SURVEILLANCE AND UPON RESTORATION. THE RPS SYSTEM WAS RETURNED TO SERVICE. THE RPS SYSTEM ACTUATED PER DESIGN IN RESPONSE TO THE DEFECTIVE TEST SWITCH CONDITION AND THE REACTOR SCRAM PLACED THE UNIT IN A SAFE SHUTDOWN CONDITION. THERE WAS NO ADVERSE IMPACT ON THE SAFE OPERATION OF THE PLANT NOR TO THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT.

[169] SALEM 2 DOCKET 50-311 LER 89-015
TECH SPEC SURV. 4.3.3.7 NON-COMPLIANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 072688 REPORT DATE: 101889 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 1 (PWR)

(NSIC 215729) ON 9/18/89, IT WAS DISCOVERED THAT THE CHANNEL CALIBRATION FOR THE TWO (2) CONTAINMENT WIDE RANGE PRESSURE POST ACCIDENT CHANNELS HAD NOT BEEN COMPLETED WITHIN THE REQUIRED TIME FRAME AS REQUIRED BY TECH SPEC SURVEILLANCE 4.3.3.7. THE SURVEILLANCE HAD BEEN REQUIRED TO BE COMPLETED NO LATER THAN 7/26/88. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO PERSONNEL ERROR. THIS WAS A RESULT OF INATTENTION TO DETAIL UPON IMPLEMENTATION REVIEW OF THE APRIL 1987 TECH SPEC AMENDMENT (BOTH UNITS) WHICH ADDED THE WIDE RANGE PRESSURE INDICATION REQUIREMENTS. THE PRESSURE CHANNELS WERE SURVEILLED PRIOR TO THIS AMENDMENT BUT AS A PREVENTIVE MAINTENANCE (PM) REQUIREMENT. WHEN THE TECH SPEC BECAME EFFECTIVE, TWO OF THE FOUR NARROW RANGE PRESSURE TRANSMITTERS WERE INCORRECTLY IDENTIFIED AS THE WIDE RANGE TRANSMITTERS. THE ACTUAL WIDE RANGE TRANSMITTERS PM SPECIFICATION WAS THEREFORE NOT CHANGED TO A SURVEILLANCE TASK (ST) SPECIFICATION. CONTRIBUTING TO THIS EVENT WAS A RELATIVELY WEAK TECH SPEC AMENDMENT IMPLEMENTATION PROCESS IN 1987. THIS PROCESS HAS SINCE BEEN UPGRADED, REFERENCE UNIT 1 LER 272/89-028-00. THE UNIT 2 WIDE RANGE CHANNELS WERE SURVEILLED AS PER THE TECH SPECS AND ON 9/19/89 THE ACTION STATEMENT WAS EXITED. THE MMIS DATA BASE HAS BEEN REVISED TO CORRECTLY IDENTIFY THE NARROW AND WIDE RANGE CONTAINMENT PRESSURE TRANSMITTERS.

[170] SAN ONOFRE 1 DOCKET 50-206 LER 89-001 REV 01
UPDATE ON REACTOR VESSEL THERMAL SHIELD SUPPORT BLOCK BOLTS OUT OF TOLERANCE.
EVENT DATE: 010889 REPORT DATE: 101889 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215675) ON 1/8/89, WITH UNIT 1 IN MODE 6 FOR THE CYCLE 10 REFUELING OUTAGE, AFTER PERFORMING A REMOTE VIDEO CAMERA INSPECTION OF THE THERMAL SHIELD SUPPORT SYSTEM, IT WAS OBSERVED THAT THREE THERMAL SHIELD SUPPORT BLOCK BOLTS WERE OUT OF TOLERANCE. THREE BOLTS, TWO ON ONE SUPPORT BLOCK AND ONE ON ANOTHER SUPPORT BLOCK, WERE PROTRUDING FROM THE INNER SURFACE OF THE CORE BARREL BY AN AMOUNT THAT EXCEEDS NORMAL, ORIGINAL ASSEMBLY, TOLERANCE. THE REMAINING FOUR BLOCKS AND THEIR BOLTS WERE ACCEPTABLE. THE THREE BOLTS PROTRUDING FROM THE INNER SURFACE OF THE CORE BARREL ARE POSTULATED TO HAVE FAILED. THE CAUSE OF THE FAILURE IS

BELIEVED TO BE HIGH-CYCLE FATIGUE DUE TO FLOW INDUCED VIBRATION. AS CORRECTIVE ACTION ALL ACCESSIBLE SUPPORT FEATURES HAVE BEEN INSPECTED BY REMOTE VIDEO CAMERA. THE BOTTOM OF THE REACTOR VESSEL HAS BEEN INSPECTED AND NO BROKEN BOLTS WERE FOUND. ENGINEERING ANALYSIS HAS BEEN PERFORMED TO VERIFY THAT CONTINUED OPERATION THROUGH FUEL CYCLE 10 WOULD NOT DEGRADE THE THERMAL SHIELD SUPPORT SYSTEM TO AN EXTENT WHICH WOULD LEAD TO SIGNIFICANT DAMAGE OF CORE INTERNALS. ALSO, ANALYSIS HAS SHOWN THAT SIGNIFICANT LOOSE PARTS, LEADING TO FLOW BLOCKAGE, WOULD NOT BE GENERATED; AND THAT INCREASED VIBRATION DUE TO LOOSE PARTS, OR FURTHER BOLT FAILURE COULD BE MONITORED. LICENSE AMENDMENT NUMBER 127 PROVIDES FOR A THERMAL SHIELD MONITORING PROGRAM WHICH HAS BEEN IMPLEMENTED.

[171] SAN ONOFRE 1 DOCKET 50-206 LER 89-005 REV 01
UPDATE ON DIESEL GENERATOR NO. 2 AUTOMATIC START DUE TO SPURIOUS LOSS OF BUS SIGNAL.
EVENT DATE: 022089 REPORT DATE: 103189 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215743) ON 02/20/89 AT 1704, AND AGAIN ON 03/17/89 AT 1756, WITH THE UNIT IN COLD SHUTDOWN, EMERGENCY DIESEL GENERATOR (DG) NO. 2 WAS AUTOMATICALLY STARTED BY SEQUENCER NO. 2 ON A LOSS OF BUS (LOB) SIGNAL. ON BOTH OCCASIONS, THE DG AND ALL SUPPORT SYSTEMS WERE CONFIRMED TO BE OPERATING IN ACCORDANCE WITH DESIGN AND THE LOB SIGNALS WERE VERIFIED TO BE SPURIOUS. BASED UPON THE ALARMS OBSERVED DURING EACH EVENT, THE LOB SIGNAL MAY HAVE BEEN CAUSED BY PARTIAL CYCLING OF A RELAY DESIGNED TO ACTUATE DURING AN UNDER-VOLTAGE (UV) CONDITION ON 4KV BUS 2C. COINCIDENT WITH BOTH DG STARTS, AN OUTER DOOR TO THE 4KV BREAKER CUBICLE WHICH HOUSES UV RELAY 127-12X WAS BEING OPENED (OR CLOSED). SOME OF THE WIRING INSIDE THE CUBICLE FLEXES AS THE OUTER CUBICLE DOOR IS OPERATED. AFTER THE FIRST ACTUATION, EXTENSIVE TESTING AND INSPECTION OF THE UV RELAYS AND ASSOCIATED CIRCUITS DID NOT DETECT ANY ANOMALIES. FOLLOWING THE SECOND DG START , ALL COMPONENTS AND WIRING IN THE 4KV CUBICLE THAT COULD HAVE CAUSED THE LOB SIGNAL WERE REPLACED AND THE CIRCUITS WERE TESTED SATISFACTORILY. ALL COMPONENTS AND WIRING REMOVED WERE INSPECTED AND TESTED ONSITE AND AT AN OFFSITE LABORATORY. TESTING OF THE TWO RELAYS, THEIR ASSOCIATED WIRING, AND THE RELAY CASE ASSEMBLY WAS UNABLE TO DETERMINE A ROOT CAUSE FOR THE TWO SPURIOUS LOB SIGNALS.

[172] SAN ONOFRE 1 DOCKET 50-206 LER 89-009 REV 01
UPDATE ON DELINQUENT EFFLUENT DOSE DETERMINATIONS DUE TO PROCEDURE DEFICIENCY.
EVENT DATE: 030889 REPORT DATE: 103089 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 2 (PWR)
SAN ONOFRE 3 (PWR)

(NSIC 215744) ON 3/8/89, WITH UNIT 1 IN COLD SHUTDOWN AND UNITS 2 AND 3 AT FULL POWER, DURING QUALITY ASSURANCE AUDIT OF TECH SPEC (TS) REQUIRED EFFLUENT SURVEILLANCES, IT WAS DETERMINED THAT SEVERAL EFFLUENT MONTHLY REPORTS HAD EXCEEDED THE 31-DAY SURVEILLANCE INTERVAL, INCLUDING THE 25% EXTENSION PERMITTED BY TS 4-0-2. THE EMRS DOCUMENT THE COMPLETION OF THE 31- DAY SURVEILLANCES FOR CUMULATIVE AND PROJECTED DOSES FROM LIQUID AND GASEOUS EFFLUENT REQUIRED BY SEVERAL UNIT 1, 2 AND 3 TSS. THE AUDIT IDENTIFIED FIVE INSTANCES (6/87, 7/87, 11/87, 9/88 AND 12/88) IN WHICH THE INTERVAL BETWEEN EMR DATES HAD EXCEEDED REQUIREMENTS OF TS 4.0.2 BY 1 TO 8 DAYS. THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE THE DELINQUENT 31-DAY CUMULATIVE AND PROJECTED DOSE SURVEILLANCES WERE ALL SATISFACTORILY PERFORMED AND EFFLUENT RELEASES WERE WELL WITHIN TS LIMITS. DOSE DETERMINATIONS AND ISSUANCE OF EMRS ARE ACCOMPLISHED MONTHLY IN ACCORDANCE WITH CHEMISTRY PROCEDURE, "LIQUID AND GASEOUS EFFLUENT DOSE DETERMINATIONS". SINCE THIS PROCEDURE DOES NOT ADDRESS THE REQUIREMENTS OF TS 4.0.2, THE EFFLUENT ENGINEERS RESPONSIBLE FOR THE DOSE DETERMINATIONS AND ISSUANCE OF THE EMRS DID NOT ENSURE THAT THE SURVEILLANCES ARE ACCOMPLISHED AND DOCUMENTED WITHIN THE TS 4.0.2 REQUIRED TIME INTERVAL. THE CHEMISTRY PROCEDURE

HAS BEEN REVISED TO REQUIRE THAT THE SURVEILLANCES BE PERFORMED AND DOCUMENTED WITHIN TIME INTERVAL.

[173] SAN ONOFRE 1 DOCKET 50-206 LER 89-022
NON-CONSERVATIVE TECH SPEC FOR OVERPRESSURE MITIGATION SYSTEM.
EVENT DATE: 091489 REPORT DATE: 101689 NSSS: WE TYPE: PWR

(NSIC 215630) ON 9/14/89, AT 0845, WITH UNIT 1 AT 91% POWER, AN ENGINEERING REVIEW OF THE REACTOR COOLANT SYSTEM (RCS) OVERPRESSURE MITIGATION SYSTEM (OMS) DETERMINED THAT TECH SPEC (TS) 3.20, "OVERPRESSURE PROTECTION SYSTEMS" AND ITS ADMINISTRATIVE CONTROLS, WHICH PERMIT OPERATION WITH OMS OUT OF SERVICE WITH A PRESSURIZER LEVEL < 50% AND RCS PRESSURES < 50% AND RCS PRESSURES < 400 PSIG, ARE NON-CONSERVATIVE. AT LOW RCS PRESSURES, THE FLOW RATE OF ONE CHARGING PUMP (320 GPM) COULD EXCEED THAT ASSUMED BY A 1978 ANALYSIS (110 GPM) SUCH THAT < 10 MINUTES IS AVAILABLE FOR OPERATOR ACTION TO TERMINATE THE EVENT, AS DESCRIBED IN TS 3.20 BASIS, PRIOR TO EXCEEDING 10CFR50, APP. G, RCS PRESSURE LIMITS AT LOW TEMPERATURE. AS A RESULT, OMS PROTECTION IS REQUIRED WITH A PRESSURIZER LEVEL < 50%. CONSEQUENTLY, UNIT 1 HAS BEEN OPERATED WITH OMS OUT OF SERVICE WHEN OMS PROTECTION WAS, IN FACT, REQUIRED. ON 9/18/89, AS A CONSEQUENCE OF THIS REVIEW, IT WAS ALSO DETERMINED THAT TS 3.2, "CHEMICAL AND VOLUME CONTROL SYSTEM" WAS NON-CONSERVATIVE SINCE IT PERMITS OPERATION OF BOTH CHARGING PUMPS WHEN PRESSURIZER LEVEL IS < 50%. CAUSE OF THIS EVENT IS ATTRIBUTED TO WEAKNESSES IN SCE'S ENGINEERING AND TECHNICAL SUPPORT TO SAN ONOFRE, WHICH IS DESCRIBED IN DETAIL IN 10/3/88 SUBMITTAL TO THE NRC ADDRESSING THIS SUBJECT. THE CORRECTIVE ACTIONS IDENTIFIED IN THAT SUBMITTAL ARE ALSO APPLICABLE TO CAUSES OF THIS CONDITION.

[174] SAN ONOFRE 1 DOCKET 50-206 LER 89-023
MANUAL REACTOR TRIP FOLLOWING DROPPED SHUTDOWN BANK RODS DUE TO FAILED CONTACTOR COIL AND BLOWN FUSE.
EVENT DATE: 091889 REPORT DATE: 101889 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215692) AT 1921 ON 9/18/89, WITH UNIT 1 IN MODE 1 AT 91% POWER, 4 CONTROL RODS (CRS) FROM SHUTDOWN BANK 2 DROPPED INTO THE CORE. APPROX. 23 SEC. LATER, THE REMAINING 12 SHUTDOWN BANK CRS (4 FROM SHUTDOWN BANK 2 AND 8 FROM SHUTDOWN BANK 1) DROPPED INTO THE CORE. OPERATIONS PERSONNEL RESPONDED BY MANUALLY TRIPPING REACTOR, VERIFYING THAT ALL SYSTEMS RESPONDED NORMALLY TO THE TRIP, AND COMPLETING REQUIREMENTS OF THE REACTOR TRIP RESPONSE PROCEDURE. AT 1942, THE PLANT WAS STABILIZED IN MODE 3 (HOT STANDBY) AND THE REACTOR TRIP RESPONSE PROCEDURE WAS EXITED. INVESTIGATION INTO THE CAUSE OF THE DROPPED CRS REVEALED THAT: 1) SHUTDOWN BANK 2 CONTACTOR COIL 2MS1 FAILED, RESULTING IN THE DE-ENERGIZATION OF 4 MOVEABLE GRIPPER COILS AND SUBSEQUENT DROP OF INITIAL 4 CRS; AND 2) AS THE FAULT FROM 2MS1 PROGRESSED, CURRENT PASSING THROUGH THE (+) 125 VDC FUSE TO BOTH SHUTDOWN BANK 1 AND 2 CONTACTOR COILS INCREASED TO THE POINT AT WHICH THE FUSE BLEW, DE-ENERGIZING MOVEABLE GRIPPER COILS FOR THE REMAINING 12 CRS AND CAUSING THE CRS TO DROP INTO THE CORE. THE FAILED CONTACTOR COIL AND FUSE WERE REPLACED WITH IN-KIND PARTS. INSULATION RESISTANCE OF ALL SHUTDOWN BANK AND CONTROL BANK CONTACTOR COILS WAS CHECKED AND DETERMINED TO BE SATISFACTORY. A THERMOGRAPHIC INSPECTION WAS PERFORMED AND NO ANOMALOUS CONDITIONS WERE IDENTIFIED. THE ROOT CAUSE OF THE CONTACTOR COIL 2MS1 FAILURE IS UNKNOWN.

[175] SAN ONOFRE 2 DOCKET 50-361 LER 87-024 REV 01
UPDATE ON FUEL HANDLING AND CONTAINMENT PURGE ISOLATION SPURIOUS ACTUATION DURING VITAL BUS TRANSFER.
EVENT DATE: 110987 REPORT DATE: 103089 NSSS: CE TYPE: PWR

(NSIC 215741) AT 1813 ON 11/9/87, A SPURIOUS ACTUATION OF TRAIN "A" OF BOTH THE

FUEL HANDLING ISOLATION SYSTEM (FHIS) AND THE CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) OCCURRED. OPERATORS VERIFIED THAT RADIATION LEVELS WERE NORMAL AND RESET FHIS AND CPIS AT 1817. ALL TRAIN "A" FHIS AND CPIS COMPONENTS FUNCTIONED AS DESIGNED. AT THE TIME OF THE ACTUATIONS, THE TRAIN "A" 1E 120 VAC BUS, WHICH PROVIDES POWER TO TRAIN "A" OF BOTH CPIS AND FHIS, WAS BEING TRANSFERRED FROM ITS NORMAL POWER SOURCE TO ITS ALTERNATE POWER SOURCE; HOWEVER, THIS EVOLUTION COULD NOT BE DEMONSTRATED TO HAVE CAUSED THE FHIS AND CPIS ACTUATIONS. THERE HAVE BEEN NO RECURRENCES OF RADIATION MONITOR ACTUATIONS DURING A VITAL BUS TRANSFER. REEVALUATION OF AVAILABLE INFORMATION LED TO THE CONCLUSION THAT A MORE LIKELY IMMEDIATE CAUSE WAS THE RESET OF THE TRAIN "A" FHIS MONITOR WHICH WAS LOGGED AS OCCURRING 1 MINUTE BEFORE THE FHIS/CPIS ACTUATION, BUT WHICH MAY HAVE ACTUALLY OCCURRED AT THE TIME OF THE FHIS/CPIS ACTUATION. LER 88-012 (50-362) PROVIDES A COMPLETE DISCUSSION OF THE ROOT CAUSE AND CORRECTIVE ACTIONS WHICH HAVE SINCE BEEN COMPLETED, AND TO DATE, HAVE PRECLUDED RECURRENCE. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE ALL TRAIN "A" FHIS AND CPIS COMPONENTS FUNCTIONED AS DESIGNED.

[176] SAN ONOFRE 2 DOCKET 50-361 LER 88-011 REV 01
UPDATE ON FUEL HANDLING ISOLATION SYSTEM TRAIN "A" ACTUATION DUE TO FAILURE OF RADIATION MONITOR 2RT-7822 POWER SUPPLY.
EVENT DATE: 051288 REPORT DATE: 103089 NSSS: CE TYPE: PWR
VENDOR: INTERNATIONAL RECTIFIER
MOTOROLA
NUCLEAR MEASUREMENTS CORP.

(NSIC 215742) ON MAY 12, 1988, AT 1100, WITH UNIT 2 AT 100% REACTOR POWER, THE TRAIN "A" FUEL HANDLING ISOLATION SYSTEM WAS ACTUATED BY RADIATION MONITOR GAS CHANNEL 2RI-7822B1. AFTER THE AIRBORNE ACTIVITY LEVELS IN THE FHB WERE DETERMINED TO BE NORMAL, FHIS TRAIN "A" WAS SECURED. THE MONITOR WAS PLACED IN BYPASS AND THE FHB VENTILATION SYSTEM WAS RETURNED TO NORMAL. ALL FHIS TRAIN "A" COMPONENTS FUNCTIONED AS DESIGNED. THE REDUNDANT FHIS TRAIN "B" REMAINED OPERABLE THROUGHOUT THE EVENT. THE ACTUATION WAS DUE TO LOSS OF POWER IN THE RADIATION MONITOR MODULE RESULTING FROM A FAILURE OF A -15 VDC POWER SUPPLY (COMPONENTS AFFECTED INCLUDED A CAPACITOR, DIODES AND A VOLTAGE REGULATOR). A REPLACEMENT MODULE WAS VERIFIED TO CONFORM TO THE CURRENT DESIGN AND WAS INSTALLED IN PLACE OF THE FAILED MODULE. IN ADDITION, THE 2RI-7822B1 MODULE INTERFACING CIRCUITS/COMPONENTS WERE TESTED AND DETERMINED TO BE OPERATING SATISFACTORILY. FAILURE ANALYSIS HAS DETERMINED THAT THE CAUSE OF THE -15 VDC POWER SUPPLY FAILURE WAS A BURR ON A METAL FLAT PLATE HEAT SINK WHICH RESULTED IN A SHORT CIRCUIT FROM THE -15 VDC VOLTAGE REGULATOR TO THE HEAT SINK ON WHICH THE REGULATOR WAS MOUNTED; THE SHORT CIRCUIT CAUSED THE FAILURE OF TWO DIODES IN THE -33 VDC SUPPLY TO THE VOLTAGE REGULATOR.

[177] SAN ONOFRE 2 DOCKET 50-361 LER 88-037
AUXILIARY FEEDWATER VALVE INOPERABLE FOR MAIN STEAM ISOLATION RESULTING IN TECH SPEC 3.0.3 ENTRY.
EVENT DATE: 101488 REPORT DATE: 101889 NSSS: CE TYPE: PWR
VENDOR: LIMITORQUE CORP.

(NSIC 215691) AT 0615 ON 10/14/88, WITH UNIT 2 AT FULL POWER, FOLLOWING THE COMPLETION OF MAINTENANCE ACTIVITIES ON THE MOTOR OPERATOR TO AUX. FEEDWATER VALVE 2HV-4706, STROKE TESTING OF THE VALVE WAS PERFORMED. PER TECH SPEC (TS) 3.3.2, THIS VALVE HAS A MAIN STEAM ISOLATION SIGNAL (MSIS) RESPONSE TIME REQUIREMENT TO CLOSE. WHILE PERFORMING THE STROKE TEST, THE VALVE WOULD NOT CLOSE UPON DEMAND FROM THE CONTROL ROOM HANDSWITCH, AND AS A RESULT, WAS NOT CAPABLE OF AUTOMATIC CLOSURE BY A MSIS SIGNAL WITHIN THE MINIMUM RESPONSE TIME REQUIRED BY TS 3.3.2. SINCE THERE ARE NO TS ACTION STATEMENTS WHICH ADDRESS THE CONDITION WHERE AN AFW VALVE CAN NOT CLOSE ON A MSIS SIGNAL, TS 3.0.3 WAS INVOKED. AT APPROX. 0645, TS 3.0.3 WAS EXITED WHEN OPERATORS MANUALLY CLOSED THE

VALVE. DURING MAINTENANCE WORK ON THE ACTUATOR TO 2HV-4706, WHICH INVOLVED THE REPLACEMENT OF A MOTOR HEATER AND THE APPLICATION OF LUBRICATION TO THE MOTOR BEARINGS, A SMALL PIECE OF DEBRIS SETTLED ONTO THE "CLOSE" TORQUE SWITCH CONTACTS, PREVENTING THE "CLOSE" MOTOR WINDINGS FROM BEING ENERGIZED. THE ACTUATOR TORQUE "CLOSE" CONTACTS WERE CLEANED, AND 2HV-4706 WAS SUCCESSFULLY STROKE TESTED. THE ROOT CAUSE OF THE EVENT IS THAT EXISTING TSS DO NOT INCLUDE A LIMITING CONDITION FOR OPERATION AND ACCOMPANYING ACTION STATEMENT SPECIFICALLY FOR THESE AFW COMPONENTS.

[178] SAN ONOFRE 2 DOCKET 50-361 LER 89-019
MANUAL REACTOR TRIP DURING PLANNED SHUTDOWN DUE TO APPROACH TO CORE PROTECTION CALCULATOR AXIAL SHAPE INDEX AUXILIARY TRIP SETPOINT.
EVENT DATE: 090289 REPORT DATE: 100289 NSSS: CE TYPE: PWR

(NSIC 215512) AT 0534 ON 9/2/89, 1989, DURING A PLANNED SHUTDOWN OF UNIT 2 AT THE END OF CYCLE 4, A MANUAL TRIP WAS INITIATED DUE TO THE APPROACH OF AXIAL SHAPE INDEX (ASI) TO THE CORE PROTECTION CALCULATOR (CPC) AUXILIARY TRIP SETPOINT. ASI DESCRIBES THE AXIAL POWER DISTRIBUTION OF THE REACTOR CORE. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE THE REACTOR PROTECTION SYSTEM FUNCTIONED IN ACCORDANCE WITH DESIGN. AT THE END OF A FUEL CYCLE, THE EFFECT OF A DECREASE IN PLANT POWER ON ASI IS GREATER THAN AT ANY OTHER TIME IN THE CYCLE. AS A RESULT, STRICT CONTROLS MUST BE EMPLOYED TO MAINTAIN ASI WITHIN LIMITS AND PREVENT A TRIP. ALTHOUGH ACTION WAS TAKEN TO CONTROL ASI, IT WAS NOT SUFFICIENT TO MAINTAIN ASI WITHIN ITS LIMITS. THE GUIDANCE IN THE OPERATING PROCEDURE GOVERNING PLANT SHUTDOWN WAS NOT SUFFICIENTLY SPECIFIC TO PROVIDE ASSURANCE THAT THE OPERATORS COULD SUCCESSFULLY CONTROL ASI DURING A PLANT SHUTDOWN AT END-OF-CYCLE. AN EVALUATION OF THIS EVENT WAS PERFORMED BY THE CORE ANALYSIS ENGINEERING GROUP. AS A RESULT OF THIS EVALUATION, THE ABOVE CAUSES WERE DETERMINED AND A PROPER STRATEGY FOR END-OF-CYCLE PLANT SHUTDOWNS WAS DEVELOPED, WHICH WILL BE INCORPORATED INTO THE PLANT SHUTDOWN PROCEDURE. THIS EVENT WILL BE DISCUSSED WITH APPROPRIATE OPERATIONS PERSONNEL AND ADDITIONAL ASI TRAINING WILL BE INCLUDED.

[179] SAN ONOFRE 2 DOCKET 50-361 LER 89-020
CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) AND FUEL HANDLING BUILDING ISOLATION SYSTEM (FHIS) TRAIN "A" UNPLANNED ACTUATIONS DURING MAINTENANCE ACTIVITIES DUE TO PERSONNEL ERROR.
EVENT DATE: 090389 REPORT DATE: 100389 NSSS: CE TYPE: PWR

(NSIC 215569) ON 9/3/89, AT 1658, UNANTICIPATED TRAIN "A" CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) AND FUEL HANDLING ISOLATION SYSTEM (FHIS) ACTUATIONS OCCURRED AS A RESULT OF OPENING THE POWER SUPPLY BREAKER TO RADIATION MONITORS 2RT-2804 (TRAIN "A" CPIS) AND 2RT-7822 (TRAIN "A" FHIS) TO SUPPORT MAINTENANCE ACTIVITIES ON THE MONITORS. ALL CPIS AND FHIS TRAIN "A" COMPONENTS FUNCTIONED AS REQUIRED. AT 1705, TRAIN "A" CPIS AND FHIS WERE RESET AND ALIGNMENT OF TRAIN "A" COMPONENTS RETURNED TO NORMAL. PROCEDURAL PRECAUTIONS CORRECTLY STATE THAT SELECTING "BYPASS" ON THE MONITORS WILL NOT PREVENT ACTUATION DUE TO LOSS OF POWER. THE EQUIPMENT CONTROL (EC) EVALUATOR WHO HAD PREPARED WORK AUTHORIZATION RECORD (WAR), CONCLUDED THAT THE PRECAUTION WAS NOT APPLICABLE TO OPENING OF THE POWER SUPPLY BREAKER. INSTEAD, THE EC EVALUATOR INCORRECTLY INDICATED IN THE WAR THAT BYPASSING THE MONITORS WOULD PREVENT CPIS AND FHIS ACTUATIONS. CONTROL ROOM OPERATORS WHO APPROVED AND IMPLEMENTED THE WAR ALSO FAILED TO RECOGNIZE THIS INCORRECT STATEMENT REGARDING THE BYPASSING OF THE MONITORS. OPERATORS HAD NOT ANTICIPATED THE CPIS AND FHIS ACTUATIONS SINCE THE MONITORS WERE BYPASSED BEFORE THE BREAKER WAS OPENED PER THE WAR. CAUSE OF THESE PERSONNEL ERRORS IS INSUFFICIENT TRAINING OF OPERATIONS PERSONNEL ON RADIATION MONITOR BYPASS SWITCHES.

[180] SAN ONOFRE 2 DOCKET 50-361 LER 89-022
CORE ALTERATIONS PERFORMED WITHOUT COMPLETE CONTAINMENT CLOSURE DUE TO DEFICIENT ADMINISTRATIVE CONTROLS.
EVENT DATE: 091789 REPORT DATE: 102089 NSSS: CE TYPE: PWR

(NSIC 215701) AT 0247 ON 9/20/89, WITH UNIT 2 IN A REFUELING OUTAGE, DURING CORE ALTERATIONS TO REMOVE THE INCORE NUCLEAR INSTRUMENTS, IT WAS DISCOVERED THAT A PATH PERMITTING FREE FLOW OF AIR FROM INSIDE TO OUTSIDE CONTAINMENT EXISTED THROUGH OPEN 3/4 INCH VENT/DRAIN VALVES ASSOCIATED WITH A HOT LEG INJECTION LINE. THIS CONDITION IS PROHIBITED BY TECH SPEC (TS) 3.9.4, WHICH REQUIRES THAT DURING CORE ALTERATIONS, EACH PENETRATION PROVIDING DIRECT ACCESS FROM CONTAINMENT ATMOSPHERE TO OUTSIDE ATMOSPHERE SHALL BE EITHER CLOSED MANUALLY OR CAPABLE OF BEING CLOSED AUTOMATICALLY. CORE ALTERATIONS WERE IMMEDIATELY SUSPENDED; AT 0308, OUTSIDE-CONTAINMENT DRAIN VALVE WAS CLOSED, REESTABLISHING THE REQUIREMENTS OF TS 3.9.4. THE OPEN PATHWAY WAS PRESENT WHEN CORE ALTERATIONS WERE INITIALLY COMMENCED AT 2155 ON 9/17. RESPONSIBLE PERSONNEL FAILED TO EVALUATE WORK ON THE HOT LEG INJECTION COMPONENTS FOR ITS IMPACT ON CORE ALTERATIONS. ALSO, ADMINISTRATIVE CONTROLS WHICH PROVIDED ASSURANCE OF CONTAINMENT CLOSURE PRIOR TO AUTHORIZING INITIATION OF CORE ALTERATIONS WERE DETERMINED TO BE DEFICIENT. A COMPREHENSIVE CHECK OF CONTAINMENT PENETRATION STATUS WAS PERFORMED; NO OTHER DISCREPANCIES WERE IDENTIFIED. THIS EVENT HAS BEEN REVIEWED WITH APPROPRIATE PERSONNEL, EMPHASIZING ADMINISTRATIVE CONTROLS UTILIZED TO VERIFY CONTAINMENT CLOSURE REQUIREMENTS PRIOR TO AUTHORIZING CORE ALTERATIONS.

[181] SAN ONOFRE 2 DOCKET 50-361 LER 89-017
CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) ACTUATION ON INSTRUMENT FAILURE DUE TO INADEQUATE ADMINISTRATIVE CONTROLS.
EVENT DATE: 092789 REPORT DATE: 102789 NSSS: CE TYPE: PWR
VENDOR: NUCLEAR MEASUREMENTS CORP.

(NSIC 215779) AT 1653 ON 9/27/89, DURING REFUELING OPERATION, CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) TRAIN "B" ACTUATED ON AN INSTRUMENT FAILURE SIGNAL FROM CONTAINMENT AIRBORN MONITOR 2RT-7807. AT 1702, AFTER DETERMINATION THAT CONTAINMENT RADIATION LEVELS WERE NORMAL AND THAT THE CPIS HAD ACTUATED ON AN INSTRUMENT FAILURE SIGNAL, THE MONITOR WAS PLACED IN BYPASS, CPIS TRAIN "B" WAS RESET, AND CONTAINMENT VENTILATION WAS RETURNED TO NORMAL. THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE RADIATION LEVELS REMAINED NORMAL AND ALL CPIS TRAIN "B" COMPONENTS FUNCTIONED AS DESIGNED. THE INSTRUMENT FAILURE WAS CAUSED BY AN INTRUSION OF WATER INTO TWO OF THE RADIATION MONITOR'S DETECTION CHAMBERS WHICH RESULTED IN A HIGH VOLTAGE BREAKDOWN OF THE PREAMPLIFIER PRINTED CIRCUIT ASSEMBLY. THE SOURCE OF THE WATER WAS FROM A TEMPORARY DRAIN HOSE WHICH WAS ERRONEOUSLY CONNECTED TO THE MONITOR'S SAMPLE LINE INLET DURING PREPARATION FOR CONTAINMENT PENETRATION LOCAL LEAK RATE TESTING (LLRT). THE ROOT CAUSE OF THIS EVENT IS INADEQUATE ADMINISTRATIVE CONTROLS. THE LLRT PROCEDURE DOES NOT PROVIDE INSTRUCTIONS FOR THE PREPARATION OF PENETRATIONS. ADDITIONALLY, THE WORK AUTHORIZATION FAILED TO IDENTIFY THE NEED TO REMOVE 2RT-7807 FROM SERVICE PRIOR TO PERFORMING THE WORK. THE WATER WAS DRAINED, THE MONITOR WAS DRIED, AND THE PREAMPLIFIER ASSEMBLY WAS REPLACED.

[182] SAN ONOFRE 3 DOCKET 50-362 LER 89-009
VOLUNTARY ENTRY INTO TECH SPEC 3.0.3 TO TRANSFER POWER SUPPLY TO EMERGENCY CHILLER E-336.
EVENT DATE: 090789 REPORT DATE: 100689 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 2 (PWR)

(NSIC 215570) AT 0852 ON 9/7/89, WITH UNIT 2 IN MODE 5 FOR REFUELING OUTAGE AND UNIT 3 AT 100% POWER, AND WITH THE TRAIN B EMERGENCY CHILLER E-335 REMOVED FROM SERVICE FOR MAINTENANCE, A VOLUNTARY ENTRY OF UNIT 3 INTO TECH SPEC (TS) 3.0.3 WAS MADE TO TRANSFER THE TRAIN A EMERGENCY CHILLER E-336 POWER SUPPLY FROM THE

UNIT 2 TO THE UNIT 3 CLASS IE ELECTRICAL BUS. THE TRANSFER OF THE E-336 POWER SUPPLY WAS NECESSARY TO MAINTAIN AN EMERGENCY BACKUP POWER SOURCE AVAILABLE TO THE CHILLER PRIOR TO REMOVING FROM SERVICE THE UNIT 2 TRAIN A EMERGENCY DIESEL GENERATOR (EDG) 2G002 FOR SCHEDULED MAINTENANCE. OPERATIONS MANAGER APPROVAL TO VOLUNTARILY ENTER TS 3.0.3 WAS OBTAINED AT 0745. UPON COMPLETION OF THE TRANSFER OPERATION AT 0908, TS 3.0.3 WAS EXITED. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT, SINCE THE SECOND TRAIN OF EMERGENCY CHILLERS WAS INOPERABLE FOR ONLY 16 MINUTES, AND NORMAL COOLING WAS MAINTAINED AS NECESSARY TO ALL ASSOCIATED ROOMS. AN EARLY START OF THE UNIT 2 REFUELING OUTAGE CREATED A CONFLICT BETWEEN THE TWO MAINTENANCE PLANS FOR E-335 AND 2G002. PERSONNEL RESPONSIBLE FOR REVIEWING MAINTENANCE SCHEDULES AND PROVIDING DIRECTION FOR CONTROLLING PLANT EQUIPMENT STATUS FAILED TO RECOGNIZE THIS CONFLICT, AND AS A RESULT, E-335 WAS REMOVED FROM SERVICE FOR MAINTENANCE EVEN THOUGH THE 2G002 WORK WAS SCHEDULED TO COMMENCE PRIOR TO THE EXPECTED COMPLETION DATE OF THE E-335 WORK.

[183] SEABROOK 1 DOCKET 50-443 LER 89-011
UNSEALED PENETRATIONS IN THE CST ENCLOSURE.
EVENT DATE: 090589 REPORT DATE: 100589 NSSS: WE TYPE: PWR

(NSIC 215592) CONTRARY TO TECH SPEC 3.7.1.3 IT HAS BEEN DETERMINED THAT THERE ARE 3 UNSEALED PIPING PENETRATIONS IN THE CONDENSATE STORAGE TANK (CST) ENCLOSURE. ON 9/5/89, WHILE IN MODE 5, THE QUESTION WAS RAISED AS TO WHETHER THESE PENETRATIONS WERE SEALED. AFTER INVESTIGATING FURTHER, IT WAS DETERMINED THEY ARE NOT SEALED AND THUS THE CST ENCLOSURE IS INOPERABLE. THE CST ENCLOSURE IS REQUIRED BY TECH SPECS TO BE OPERABLE IN MODES 1, 2, AND 3. SEABROOK STATION HAS ENTERED MODE 3 TWICE AND MODE 2 ONCE WITHOUT THE CST ENCLOSURE BEING OPERABLE. PRIOR TO INITIAL CRITICALITY, AS WAS THE ENTIRETY OF THE FIRST OF THE TWO ENTRIES INTO MODE 3, THERE WAS NO SAFETY SIGNIFICANCE. THE SECOND ENTRY INTO MODE 3 WAS DURING LOW POWER TESTING, WHEN MODE 2 WAS ALSO ENTERED. AT THAT TIME THE SAFETY SIGNIFICANCE WAS MINIMAL. THE ROOT CAUSE OF THIS INCIDENT IS CURRENTLY BEING INVESTIGATED. IT SHALL BE SUPPLIED IN A SUPPLEMENTAL REPORT ON OR BEFORE NOVEMBER 16, 1989. CORRECTIVE ACTION FOR THIS INCIDENT INCLUDES SEALING THE UNSEALED PENETRATIONS AND ALSO REVISING THE TECH SPECS SURVEILLANCE LOG TO PROVIDE CLARIFICATION OF THE REQUIREMENT FOR CST ENCLOSURE INTEGRITY. THIS IS THE FIRST EVENT OF THIS TYPE AT SEABROOK STATION.

[184] SEQUOYAH 2 DOCKET 50-328 LER 89-011 REV 01
UPDATE ON RE-90-119 RADIATION MONITORS INOPERABLE BECAUSE OF INADEQUATE SOURCE CHECK PERFORMANCE.
EVENT DATE: 082389 REPORT DATE: 102389 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 1 (PWR)

(NSIC 215732) ON 8/23/89, WITH UNITS 1 AND 2 IN MODE 1 AT 100% POWER, 2,235 PSIG, 578F, IT WAS DISCOVERED THAT A TECH SPEC (TS) SURVEILLANCE REQUIREMENT (SR) TO SOURCE CHECK THE RADIOACTIVE GASEOUS EFFLUENT MONITORS ON THE CONDENSER VACUUM PUMP EXHAUST WAS NOT BEING FULLY MET. A SOURCE CHECK IS DEFINED IN THE SQN TS AS A QUALITATIVE ASSESSMENT OF CHANNEL RESPONSE WHEN THE CHANNEL SENSOR IS EXPOSED TO A RADIOACTIVE SOURCE. THE SUBJECT MONITORS USE A LIGHT-EMITTING DIODE (LED) LIGHT SOURCE TO SOURCE CHECK ALL COMPONENTS EXCEPT THE SCINTILLATION CRYSTAL. ADDITIONALLY, THE SOURCE CHECK METHOD USED FOR OTHER GASEOUS EFFLUENT RADIATION MONITORS THAT EXPOSE A SECOND, NONPROCESS SCINTILLATION CRYSTAL TO A RADIOACTIVE SOURCE DURING SOURCE CHECKING HAS BEEN INVESTIGATED AND DETERMINED TO BE ADEQUATE. ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO A PREVIOUS LACK OF EMPHASIS ON RECOGNITION OF THE LITERAL REQUIREMENTS OF THE TECH SPEC DEFINITION OF SOURCE CHECK. AS INTERIM CORRECTIVE ACTION, THE TWO MONITORS WITH LEDS WERE SOURCE CHECKED WITH A RADIOACTIVE SOURCE TO DEMONSTRATE THEIR OPERABILITY. THE SURVEILLANCE INSTRUCTION HAS BEEN REVISED TO REQUIRE A RADIOACTIVE SOURCE TO BE USED FOR SOURCE CHECKING THESE TWO MONITORS. THE TECH SPEC CHANGES SPECIFIED IN

GENERIC LETTER 89-01 WILL REMOVE THE EFFLUENT SPECIFICATIONS FROM THE TECH SPEC AND PLACE THEM IN THE OFFSITE DOSE CALCULATION MANUAL.

[185] SEQUOYAH 2 DOCKET 50-328 LER 89-012
ONE TRAIN OF THE REACTOR VESSEL LEVEL INSTRUMENTATION SYSTEM LEVEL INDICATION INOPERABLE BECAUSE AN ISOLATION VALVE WAS INADVERTENTLY LEFT MISPOSITIONED DURING PREVENTIVE MAINTENANCE.
EVENT DATE: 091289 REPORT DATE: 101289 NSSS: WE TYPE: PWR
VENDOR: AUTOCLAVE ENGINEERS, INC.

(NSIC 215598) ON 9/12/89, WITH UNITS 1 AND 2 AT 100%, 2,235 POUNDS PER SQUARE INCH GAUGE, 578F, A UNIT 2 REACTOR VESSEL LEVEL INSTRUMENTATION SYSTEM LEVEL INDICATOR FAILED A MONTHLY CHANNEL CHECK AND WAS DECLARED INOPERABLE. A MAGNETICALLY OPERATED ISOLATION VALVE FOR THE LEVEL INDICATOR WAS SUBSEQUENTLY FOUND CLOSED AND WAS DETERMINED TO HAVE BEEN CLOSED SINCE BEING INADVERTENTLY MISPOSITIONED WHILE BEING EXERCISED DURING PREVENTIVE MAINTENANCE (PM) ON 8/15/89. AFTER THE ISOLATION VALVE WAS OPENED AND OTHER ISOLATION VALVES WERE VERIFIED TO BE FULLY OPEN, THE LEVEL INDICATOR RETURNED TO A NORMAL INDICATION AND WAS DECLARED OPERABLE. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INADEQUATE TRAINING FOR THE TWO CRAFT PERSONNEL WHO PERFORMED THE PM IN AUGUST. ADDITIONALLY, THE PM PROCEDURE SHOULD HAVE INCLUDED MORE DETAILED WORK INSTRUCTIONS. AS CORRECTIVE ACTION, THE APPROPRIATE CRAFT PERSONNEL HAVE BEEN TRAINED, AND THE PM PROCEDURE HAS BEEN REVISED TO PROVIDE MORE DETAILED GUIDANCE.

[186] SHEARON HARRIS 1 DOCKET 50-400 LER 89-016
1B-SB EMERGENCY SERVICE WATER PUMP START DUE TO A FAILED RELAY IN THE EMERGENCY SEQUENCER TEST CIRCUIT.
EVENT DATE: 091189 REPORT DATE: 101189 NSSS: WE TYPE: PWR
VENDOR: POTTER & BRUMFIELD

(NSIC 215576) THE PLANT WAS OPERATING IN MODE 1, POWER OPERATION, AT 100 PERCENT REACTOR POWER ON SEPTEMBER 11, 1989. THE EMERGENCY LOAD SEQUENCER WAS BEING RUN IN THE TEST MODE TO PERFORM ENGINEERING PERIODIC TEST (EPT)- 033, EMERGENCY SAFEGUARDS SEQUENCER SYSTEM TEST. AT 0930 HOURS WHEN THE TEST STOP PUSHBUTTON WAS PRESSED THE SEQUENCER DID NOT PROPERLY RESET AND GENERATED A START SIGNAL TO THE 1B-SB EMERGENCY SERVICE WATER (ESW) PUMP. THE INADVERTENT START OF THE 1B-SB EWS PUMP WAS OBSERVED BY CONTROL ROOM OPERATORS WHO IMMEDIATELY VERIFIED PLANT CONDITIONS AND SECURED THE PUMP. THE CAUSE OF THE EVENT WAS THE FAILURE OF A TEST CIRCUIT RELAY TO RESET AT THE PROPER TIME. THE RELAY FAILURE CAUSED POWER TO CONTINUE TO BE SUPPLIED TO THE EQUIPMENT ACTUATION RELAYS LONGER THAN DESIGNED RESULTING IN THE INADVERTENT START OF THE 1B-SB ESW PUMP. THE ROOT CAUSE HAS BEEN DETERMINED TO BE THE OVERLOADING OF THE TEST RELAY CONTACTS. THE CORRECTIVE ACTIONS INCLUDE AN EVALUATION TO ELIMINATE THE CONTACT OVERLOADING, AND AN EVALUATION TO DETERMINE IF OTHER SEQUENCER CIRCUITS HAVE THE SAME PROBLEM. THERE WERE NO SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT AS THE DAMAGED TESTING CIRCUITRY DOES NOT AFFECT THE SEQUENCERS ABILITY TO RESPOND TO A VALID ACCIDENT SIGNAL. THIS EVENT IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV) AS AN ENGINEERED SAFETY FEATURE ACTUATION.

[187] SOUTH TEXAS 1 DOCKET 50-498 LER 89-017
INADVERTENT CONTAMINATION OF NON-RADIOACTIVE SYSTEMS DUE TO INADEQUATE CONTROLS AND PERSONNEL ERROR.
EVENT DATE: 081489 REPORT DATE: 102489 NSSS: WE TYPE: PWR

(NSIC 215715) ON AUGUST 14, 1989, UNIT 1 WAS IN MODE 5 FOR A REFUELING OUTAGE. AT 1400 HOURS, HEALTH PHYSICS PERSONNEL DETECTED RADIATION LEVELS OF .3 TO .8 MR/HR IN THE AUXILIARY STEAM LINE TO THE LIQUID WASTE PROCESSING SYSTEM (LWPS) WASTE EVAPORATOR. FURTHER SURVEYS INDICATED THE PRESENCE OF CONTAMINATION IN THE

LWPS CONDENSATE TANK, THE AUXILIARY BOILER AND THE INORGANICS BASIN. THE SYSTEMS WERE IMMEDIATELY ISOLATED AND THE AREAS POSTED. THE SOURCES OF THE CONTAMINATION WERE TWO VALVES WHICH WERE LEFT OPEN DURING SHUTDOWN OF THE LWPS WASTE EVAPORATOR WHICH ALLOWED LIQUID WASTE BEING TRANSFERRED TO A WASTE MONITOR TANK TO BACKUP THROUGH THE WASTE EVAPORATOR GAS STRIPPER INTO THE AUXILIARY STEAM SYSTEM. THE CAUSES OF THIS EVENT WERE INADEQUATE CONTROLS OVER INTERFACES BETWEEN RADIOACTIVE AND NON-RADIOACTIVE SYSTEMS, FAILURE OF CHEMICAL OPERATIONS PERSONNEL TO FOLLOW PROCEDURES, AND THE LACK OF RADIATION MONITORS IN THE AUXILIARY STEAM SYSTEM. CORRECTIVE ACTIONS INCLUDE A REVIEW OF RADIOACTIVE TO NON-RADIOACTIVE SYSTEM INTERFACES, PROCEDURE REVISIONS AND A REVIEW OF THE IMPLEMENTATION OF THE STATION POLICY REGARDING PROCEDURE COMPLIANCE. NO RELEASE OCCURRED FROM THE INORGANICS BASIN. ANALYSIS RESULTS INDICATE THAT HAD A RELEASE OCCURRED TO UNCONTROLLED AREAS, IT WOULD HAVE BEEN A SMALL FRACTION OF TECHNICAL SPECIFICATION LIMITS.

[188] SOUTH TEXAS 1 DOCKET 50-498 LER 89-018
IMPROPER INSTALLATION OF REGULATORY GUIDE 1.97 CATEGORY 2 INSTRUMENTATION DUE TO AN ERROR IN THE INSTALLATION DETAILS.
EVENT DATE: 092289 REPORT DATE: 102689 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SOUTH TEXAS 2 (PWR)

(NSIC 215755) ON SEPTEMBER 22, 1989 UNIT 1 WAS IN MODE 6 IN A REFUELING OUTAGE AND UNIT 2 WAS IN MODE 3, IT WAS DETERMINED THAT THE CONSTRUCTION INSTALLATION PROCEDURE FOR REGULATORY GUIDE 1.97 CATEGORY 2 INSTRUMENTATION DID NOT INCLUDE INSTALLATION DETAILS THAT WOULD ASSURE THE DEVICES ARE QUALIFIED TO THE REQUIREMENTS OF 10CFR50.49 AND IEEE 323-1974. THIS WAS DISCOVERED AS A RESULT OF THE FOLLOWUP INVESTIGATION INTO CONCERNS RAISED DURING THE NRC ENVIRONMENTAL QUALIFICATION AUDIT CONDUCTED ON SITE FROM SEPTEMBER 18, 1989 TO SEPTEMBER 22, 1989. POTENTIALLY DEFICIENT EQUIPMENT WAS IDENTIFIED, INSPECTED AND RESTORED TO A QUALIFIED INSTALLATION PRIOR TO THE AFFECTED UNIT RETURNING TO POWER. NO EQUIPMENT FAILED AS A RESULT OF A LOSS OF FUNCTION DUE TO INCORRECTLY INSTALLED SPLICES AND SUBSEQUENT INVESTIGATION DETERMINED THAT THERE WERE NO SAFETY CONCERNS. ALTHOUGH SOME OF THESE POST-ACCIDENT MONITORING INSTRUMENTS REQUIRED REWORK OF CABLE SPLICES, PLANT SAFETY SYSTEMS WOULD HAVE FUNCTIONED PROPERLY UNDER ALL REQUIRED CONDITIONS AND WOULD NOT HAVE JEOPARDIZED RECOVERY FROM AN ACCIDENT. THERE WERE NO ADVERSE SAFETY OR RADIOLOGICAL CONSEQUENCES AS A RESULT OF THIS EVENT. THE EVENT DID NOT RESULT IN ADDITIONAL RISK TO THE PUBLIC. THIS CONDITION IS NOT REPORTABLE PURSUANT TO 10CFR50-73. AS SUCH, THIS IS A VOLUNTARY LER.

[189] SOUTH TEXAS 1 DOCKET 50-498 LER 89-019
FAILURE TO OBTAIN REACTOR CONTAINMENT BUILDING ATMOSPHERE GRAB SAMPLES WITHIN 24 HOURS AS REQUIRED BY TECH SPECS.
EVENT DATE: 092889 REPORT DATE: 103089 NSSS: WE TYPE: PWR

(NSIC 215785) AT APPROXIMATELY 1440, ON SEPTEMBER 25, 1989, WITH UNIT 1 IN MODE 5 AFTER REFUELING, THE REACTOR CONTAINMENT BUILDING ATMOSPHERE RADIATION MONITOR WAS REMOVED FROM SERVICE FOR CALIBRATION. THIS PLACED UNIT 1 IN A LIMITING CONDITION FOR OPERATION AS DEFINED IN TECHNICAL SPECIFICATION 3.3.3 REQUIRING THAT A GRAB SAMPLE BE OBTAINED AND ANALYZED AT LEAST ONCE PER 24 HOURS. TWICE THIS ACTION STATEMENT WAS EXCEEDED, ONCE ON SEPTEMBER 27, 1989 AND ONCE ON SEPTEMBER 28, 1989. BOTH TIMES THE REQUIREMENT WAS NOT EXCEEDED BY MORE THAN AN HOUR. THE CAUSE OF THIS EVENT WAS THAT NO PROGRAMMATIC CONTROLS WERE IN PLACE TO CONTROL THE COLLECTION AND ANALYSIS OF GRAB SAMPLES REQUIRED BY THE TECHNICAL SPECIFICATION ACTION STATEMENT. CORRECTIVE ACTIONS INCLUDE ISSUANCE OF NIGHT ORDERS TO ENSURE GRAB SAMPLE ARE OBTAINED AND ANALYZED, DEVELOPING AND IMPLEMENTING A HEALTH PHYSICS PROCEDURE TO CONTROL COLLECTION OF GRAB SAMPLES AND A REVIEW OF INSTRUMENT TECHNICAL SPECIFICATIONS TO ENSURE APPROPRIATE CONTROLS ARE IN PLACE.

[190] SOUTH TEXAS 2 DOCKET 50-499 LER 89-024
FAILURE TO RESTORE AN ESSENTIAL CHILLER TO SERVICE WITHIN TECH SPEC LIMITS.
EVENT DATE: 052889 REPORT DATE: 103089 NSSS: WE TYPE: PWR

(NSIC 215786) ON MAY 25, 1989, UNIT 2 WAS IN MODE 1 AT 100 PERCENT POWER. AT 0630 THE TRAIN C ESSENTIAL COOLING WATER (ECW) SYSTEM WAS REMOVED FROM SERVICE FOR MAINTENANCE. PRIOR TO RESTORING THE ECW SYSTEM, THE TRAIN C CHILLED WATER SYSTEM WHICH IS SUPPORTED BY THE ECW SYSTEM WAS REMOVED FROM SERVICE FOR PREVENTIVE AND CORRECTIVE MAINTENANCE. THE CHILLED WATER SYSTEM WAS NOT FULLY RESTORED TO SERVICE UNTIL 0855 HOURS ON MAY 28, 1989. THIS RESULTED IN A VIOLATION OF THE TECHNICAL SPECIFICATION REQUIREMENT FOR RESTORATION OF THE SYSTEM IN 72 HOURS. THE CAUSE OF THIS EVENT WAS AN INCORRECT DECISION BY MANAGEMENT PERSONNEL TO COUNT THE INOPERABLE TIME OF THE CHILLED WATER SYSTEM FROM THE FIRST TIME IT WAS WORKED ON RATHER THAN FROM THE TIME THE ECW SYSTEM WAS REMOVED FROM SERVICE. THIS DECISION WAS BASED ON A MISCOMMUNICATION OF SYSTEM STATUS TO MANAGEMENT, AND INADEQUATE LOGGING. CORRECTIVE ACTIONS INCLUDE BRIEFINGS OF OPERATIONS PERSONNEL REGARDING THIS EVENT, PROCEDURE REVISIONS, AND A DISCUSSION OF THIS EVENT IN OPERATOR REQUALIFICATION TRAINING.

[191] SOUTH TEXAS 2 DOCKET 50-499 LER 89-021
REACTOR TRIP DUE TO A DEFECTIVE FEEDWATER PUMP SPEED CONTROLLER CARD EDGE CONNECTOR.
EVENT DATE: 090589 REPORT DATE: 100589 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215583) ON SEPTEMBER 5, 1989, UNIT 2 WAS IN MODE 1 AT 100 PERCENT POWER. AT APPROXIMATELY 1607 HOURS, CONTROL ROOM OPERATORS OBSERVED SPEED OSCILLATIONS ON THE TURBINE DRIVEN STEAM GENERATOR FEEDWATER PUMP (SGFP) 21. THE OPERATORS ATTEMPTED TO REGAIN SPEED CONTROL; HOWEVER, THE PUMP DID NOT RESPOND AND SUBSEQUENTLY TRIPPED ON OVERSPEED. THE RESULTANT LOSS OF STEAM GENERATOR LEVEL CAUSED A REACTOR TRIP AND AUXILIARY FEEDWATER SYSTEM ACTUATION. NO SAFETY INJECTION ACTUATION OCCURRED. THE PLANT WAS STABILIZED IN MODE 3. THE CAUSE OF THIS EVENT WAS A DEFECTIVE SGFP 21 SPEED CONTROLLER CARD EDGE CONNECTOR WHICH WAS DISTURBED BY A MAINTENANCE TECHNICIAN DURING TROUBLESHOOTING OF A CARD ASSOCIATED WITH SGFP 22 IN THE SAME CARD FRAME. THE DEFECTIVE CONNECTOR WAS REPAIRED, THE CARD FRAME ALIGNMENT CHECKED, THE REMAINING PRINTED CIRCUIT CARDS AND EDGE CONNECTORS WERE INSPECTED AND THE CONTACT SURFACES WERE CLEANED. THE PRINTED CIRCUIT CARDS IN THE SPEED CONTROLLER CIRCUITS ON UNIT 1 WERE ALSO INSPECTED AND CLEANED.

[192] SOUTH TEXAS 2 DOCKET 50-499 LER 89-022
REACTOR TRIP DUE TO ACTUATION OF THE OVERTEMPERATURE DELTA TEMPERATURE TURBINE RUNBACK CIRCUIT.
EVENT DATE: 091989 REPORT DATE: 101989 NSSS: WE TYPE: PWR

(NSIC 215716) ON SEPTEMBER 19, 1989, UNIT 2 WAS IN MODE 1 AT 100 PERCENT POWER. AT 1237 HOURS DURING THE CROSS CALIBRATION OF THE INCORE AND EXCORE NUCLEAR INSTRUMENTATION, AN OVERTEMPERATURE DELTA TEMPERATURE (OTDT) TURBINE RUNBACK OCCURRED RESULTING IN AN OTDT REACTOR TRIP. THE TURBINE TRIPPED ON THE REACTOR TRIP AND AN AUXILIARY FEEDWATER ACTUATION OCCURRED ON LOW STEAM GENERATOR LEVEL. THE MAIN STEAM ISOLATION VALVES WERE CLOSED TO PREVENT EXCESSIVE REACTOR COOLANT SYSTEM COOLDOWN. THE ENGINEERED SAFETY FEATURES FUNCTIONED AS DESIGNED. THE CAUSE OF THIS EVENT WAS THAT THE TURBINE RUNBACK SETPOINT DESIGN DID NOT PROVIDE SUFFICIENT OPERATING MARGIN TO ALLOW FOR SMALL DEVIATIONS IN RCS AVERAGE TEMPERATURE. THE OTDT RUNBACK FEATURE HAS BEEN DISABLED AND ANALYSIS IS BEING PERFORMED TO DETERMINE IF IT CAN BE RESTORED WITHOUT CAUSING UNWANTED PLANT TRANSIENTS.

[193] SOUTH TEXAS 2 DOCKET 50-499 LER 89-023
REACTOR TRIP DUE TO A TURBINE TRIP CAUSED BY AN INVERTER FAILURE.
EVENT DATE: 092289 REPORT DATE: 102389 NSSS: WE TYPE: PWR
VENDOR: ELGAR, CORP.

(NSIC 215717) ON 9/22/89, UNIT 2 WAS IN MODE 1 AT 94% POWER. AT 0201 HOURS A TURBINE TRIP OCCURRED ON LOSS OF POWER TO THE FOUR MAIN TURBINE AUTO STOP SOLENOIDS. THE REACTOR TRIPPED ON THE TURBINE TRIP. A MAIN FEEDWATER ISOLATION AND AUXILIARY FEEDWATER ACTUATION OCCURRED ON LOW AVERAGE REACTOR COOLANT SYSTEM TEMPERATURE AS EXPECTED. THE UNIT WAS STABILIZED IN MODE 3 WITH NO UNEXPECTED POST-TRIP TRANSIENTS. THE CAUSE OF THIS EVENT WAS FAILURE OF A NON-SAFETY-RELATED INVERTER WHICH INTERRUPTED POWER TO THE MAIN TURBINE AUTO STOP SOLENOIDS. A CONTRIBUTING CAUSE WAS THAT THE DESIGN DID NOT PROVIDE FOR THE SINGLE FAILURE OF A POWER FEED. THE INVERTER HAS BEEN REPAIRED AND RETURNED TO SERVICE. THE POWER FEED DESIGN HAS BEEN CHANGED TO ADD AN ADDITIONAL POWER SOURCE.

[194] ST. LUCIE 1 DOCKET 50-335 LER 89-005
REACTOR TRIP DUE TO INADEQUATE PRE-MAINTENANCE REVIEW.
EVENT DATE: 091389 REPORT DATE: 101389 NSSS: CE TYPE: PWR

(NSIC 215565) ON SEPTEMBER 13, 1989, AT 1409, WHILE IN MODE 1 AT 98% POWER, UNIT 1 TRIPPED ON LOSS OF LOAD. PRIOR TO THE TRIP, THERE WERE TWO NUCLEAR PLANT WORK ORDERS (NPWO) BEING WORKED CONCURRENTLY ON THE REACTOR PLANT PROTECTION SYSTEM (RPS). ONE OF THE NPWO'S INVOLVED REMOVING TCB-1 FOR MAINTENANCE WHILE THE OTHER NPWO WAS FOR REPLACING A "C" CHANNEL POWER SUPPLY IN THE RPS CABINET. IN ORDER TO REPLACE THE POWER SUPPLY, BREAKER CB-3 INSIDE THE RPS CABINET WAS OPENED. WHEN THIS WAS DONE, TCB-7 AND TCB-3 OPENED AND WITH TCB-1 ALREADY OPEN, A TURBINE AND REACTOR TRIP OCCURRED. THE ROOT CAUSE OF THE REACTOR TRIP WAS DETERMINED TO BE AN INADEQUATE NPWO WORK DESCRIPTION, INADEQUATE COMMUNICATIONS TO OPERATIONS BY I&C AND THE PROCEDURE FOR UNIT RELIABILITY-SENSITIVE SYSTEMS DID NOT CLEARLY SHOW THAT ITS USE WAS REQUIRED. CORRECTIVE ACTIONS REVIEW/REVISE SENSITIVE SYSTEMS PROCEDURES, WRITE STANDARD WORK DESCRIPTIONS TO BE ATTACHED TO NPWO'S FOR RPS POWER SUPPLY REPLACEMENTS, ADD CAUTION STATEMENTS TO I&C PROCEDURES TO CHECK THE TCB'S POSITION AND EXPEDITE PLANT CHANGES TO REPLACE RPS POWER SUPPLIES.

[195] ST. LUCIE 2 DOCKET 50-389 LER 89-007
MANUAL REACTOR TRIP RESULTING FROM MULTIPLE DROPPED RODS DUE TO UNRELATED EQUIPMENT FAILURES.
EVENT DATE: 092389 REPORT DATE: 102389 NSSS: CE TYPE: PWR
VENDOR: HEINEMANN ELECTRIC CO.
INTERNATIONAL RECTIFIER
LIMITORQUE CORP.

(NSIC 215708) AT 0558 ON SEPTEMBER 23, 1989, WITH ST. LUCIE UNIT 2 IN MODE 1 AT 100% POWER, THE UNIT EXPERIENCED A DROPPED CONTROL ELEMENT ASSEMBLY (CEA) IN REGULATING GROUP 1. THE REACTOR CONTROL OPERATOR (RCO) MANUALLY REDUCED MAIN TURBINE LOAD TO MATCH REACTOR POWER. AT 0613, AS THE CONTROL RODS WERE BEING INSERTED, FOUR CEAS IN REGULATING GROUP 5, THE LEAD BANK, DROPPED IN THE CORE. THE RCO IMMEDIATELY TRIPPED THE REACTOR FROM 96% POWER DUE TO THE MULTIPLE DROP CEAS. THE STANDARD POST TRIP ACTIONS WERE COMPLETED AND THE UNIT WAS QUICKLY STABILIZED IN HOT STANDBY, MODE 3. THE MOST PROBABLE CAUSE OF THE EVENT WAS A BLOWN FUSE CAUSING THE INITIAL DROPPED CEA. THE APPARENT CAUSE FOR THE FOUR CEAS TO DROP WAS DUE TO THE TRIPPING OF THE CEA SUBGROUP BREAKER. THE ROOT CAUSE FOR THE SUBGROUP BREAKER TRIP HAS NOT BEEN CONCLUSIVELY IDENTIFIED; HOWEVER, TESTING THE SUBGROUP BREAKER REVEALED THAT THE BREAKER TRIPPED AT A CURRENT LESS THAN DESIGNED. THE FOLLOWING CORRECTIVE ACTIONS HAVE BEEN IMPLEMENTED: REPLACED THE BLOWN FUSE AND REPLACED THE CEA SUBGROUP BREAKER.

[196] ST. LUCIE 2 DOCKET 50-389 LER 89-008
TECH SPEC SURVEILLANCES PERFORMED IMPROPERLY DUE TO PERSONNEL ERROR.
EVENT DATE: 092489 REPORT DATE: 102489 NSSS: CE TYPE: PWR

(NSIC 215709) ON SEPTEMBER 23, 1989, ST. LUCIE UNIT 2 WAS IN MODE 3 FOLLOWING A MANUAL REACTOR TRIP. AT 1100 HOURS, OPERATORS FILLED TWO SAFETY INJECTION TANKS (SIT). THIS REQUIRES THAT SPECIFIC CHECK VALVES IN THE SAFETY INJECTION HEADERS BE TESTED TO VERIFY THEIR CLOSURE WITHIN 24 HOURS AND THEIR LEAKAGE MEASURED WITHIN 31 DAYS. ON SEPTEMBER 24, 1989, AT 0930, A PLANT ENGINEER FAILED TO FOLLOW APPROVED PLANT PROCEDURES AND PERFORMED AN IMPROPER SURVEILLANCE ON THE SAFETY INJECTION HEADER CHECK VALVES TO VERIFY THEIR CLOSURE. ON SEPTEMBER 25, 1989, A SUPERVISOR DETERMINED THAT THE METHOD COULD NOT PROVIDE VERIFICATION OF CLOSURE OF THESE VALVES. THESE SAFETY INJECTION HEADER CHECK VALVES SUBSEQUENTLY PASSED THEIR LEAK TESTS AT 1245 ON SEPTEMBER 25, 1989. FURTHER REVIEW INDICATED THAT ON MAY 7, 1989, WHILE ST. LUCIE UNIT 2 WAS AT 85% POWER, THE FILLING OF A SIT REQUIRED THE TWO CHECK VALVES TO BE TESTED WITHIN 24 HOURS. HOWEVER, DUE TO A MISCOMMUNICATION, A UTILITY ENGINEER ONLY TESTED ONE OF THE TWO CHECK VALVES WITHIN THE REQUIRED PERIOD FOR VALVE CLOSURE VERIFICATION. THE VALVE WAS LATER TESTED SATISFACTORILY ON JUNE 2, 1989. THE CAUSE OF BOTH EVENTS WAS PERSONNEL ERROR BY UTILITY ENGINEERS. CORRECTIVE ACTIONS WERE TO PROPERLY TEST THE CHECK VALVES INVOLVED AND TO COUNSEL PERSONNEL INVOLVED ON THE IMPORTANCE OF FOLLOWING APPROVED PROCEDURES.

[197] SUMMER 1 DOCKET 50-395 LER 89-013 REV 01
UPDATE ON SERVICE WATER PUMP HOUSE FLOODING DUE TO NON-SAFETY RELATED COOLING COIL FAILURE.
EVENT DATE: 080889 REPORT DATE: 092789 NSSS: WE TYPE: PWR

(NSIC 215379) ON 8/8/89, A DESIGN DEFICIENCY IN THE SERVICE WATER PUMP HOUSE (SWPH) VENTILATION SYSTEM WAS CONFIRMED. IN 1984, NON-SAFETY RELATED, NON-SEISMIC COOLING COILS WITH ASSOCIATED SERVICE WATER (SW) SUPPLY PIPING WERE ADDED TO THE VENTILATION SYSTEM TO PRECLUDE HIGH TEMPERATURES IN THE SWPH DURING WARM WEATHER. THE INITIAL DESIGN OF THE COOLING COILS WHICH WAS PERFORMED BY THE ARCHITECT ENGINEER FAILED TO ADEQUATELY ASSESS SWPH FLOODING DUE TO A FAILURE IN THE COOLING COILS. DURING A SUBSEQUENT SOUTH CAROLINA ELECTRIC & GAS COMPANY (SCE&G) INVESTIGATION, A PATH WAS IDENTIFIED WHERE FAILURE OF THE COILS OR ASSOCIATED SUPPLY PIPING IN THE AIR INTAKE CHASE COULD CAUSE FLOODING IN THE DUCTWORK AND EVENTUALLY THE LOWER ELEVATION SWITCHGEAR ROOMS. THIS FLOODING COULD CAUSE FAILURE OF SAFETY RELATED ELECTRICAL POWER SUPPLIES FOR SW COMPONENTS. THE IMMEDIATE CORRECTIVE ACTION WAS TO VERIFY THAT THE NON-SAFETY RELATED SUPPLY PIPING TO THE COILS WAS ISOLATED. A DESIGN CHANGE HAS BEEN GENERATED TO CUT HOLES IN THE COOLING COIL SUPPORT PLATE TO ALLOW DRAINAGE. ALSO, BAFFLES WILL BE INSTALLED IN THE VENTILATION PLENUMS TO PREVENT DIRECT SPRAY INTO THE DUCTWORK. THE ARCHITECT ENGINEER FILED A REPORT WITH THE NRC ON 8/11/89, PURSUANT TO 10 CFR 21.

[198] SUMMER 1 DOCKET 50-395 LER 89-015 REV 01
UPDATE ON MANUAL REACTOR TRIP DUE TO PRESSURIZER SAFETY VALVE FAILURE.
EVENT DATE: 082589 REPORT DATE: 102489 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE & GAGE CO.

(NSIC 215710) AT 1000 HOURS ON AUGUST 25, 1989, THE "A" PRESSURIZER SAFETY VALVE BODY INLET TEMPERATURE INCREASED TO GREATER THAN 450F AND A PLANT SHUTDOWN WAS INITIATED. SHORTLY AFTER THE LOAD REDUCTION WAS STARTED, THE "A" PRESSURIZER SAFETY VALVE OPENED AT A SYSTEM PRESSURE OF APPROXIMATELY 2260 PSIG. AT 1003 HOURS, THE ACOUSTIC LEAK MONITOR ALARM, THE REACTOR COOLANT SYSTEM (RCS) BEGAN TO RAPIDLY DEPRESSURIZE AND AT APPROXIMATELY 1004 HOURS THE SHIFT SUPERVISOR DIRECTED A MANUAL REACTOR TRIP. THE PRESSURIZER SAFETY VALVE RESEATED PRIOR TO REACHING THE SAFETY INJECTION SETPOINT OF 1850 PSIG. ALL PLANT PARAMETERS

RECOVERED TO THEIR EXPECTED POST TRIP VALUES EXCEPT RCS PRESSURE WHICH WAS CONTROLLED AROUND 2000 PSIG TO AVOID LIFTING THE SAFETY VALVE AGAIN. THE PLANT WAS TAKEN TO COLD SHUTDOWN, THE "A" PRESSURIZER SAFETY VALVE REPLACED, AND THE REACTOR WAS RESTARTED AT 0635 HOURS ON SEPTEMBER 1, 1989. LER 89-011, DATED JUNE 27, 1989, DOCUMENTS A SIMILAR EVENT INVOLVING "C" PRESSURIZER SAFETY VALVE. IT WAS IDENTIFIED IN THE REPORT THAT THE LICENSEE IS STILL INVESTIGATING THE EVENT AND THE FINDINGS WILL BE DISCUSSED IN A SUPPLEMENT REPORT. AS OF THIS DATE, THE LICENSEE CONTINUES TO INVESTIGATE THESE EVENTS AND THE SUPPLEMENT REPORT WILL BE SUBMITTED DETAILING THE FINDINGS. NOTE: EXPECTED SUBMISSION DATE DECEMBER 20, 1989 AS IDENTIFIED IN LER 89-011.

[199] SUMMER 1 DOCKET 50-395 LER 89-016
SHUTDOWN DUE TO INOPERABLE MSIV.
EVENT DATE: 090889 REPORT DATE: 100789 NSSS: WE TYPE: PWR

(NSIC 215573) AT 0410 HOURS ON SEPTEMBER 8, 1989, A CONTROLLED REACTOR SHUTDOWN WAS COMMENCED AS REQUIRED BY TECHNICAL SPECIFICATION 3.7.1.5, "MAIN STEAM LINE ISOLATION VALVES" (MSIV). "A" MSIV HAD BEEN DECLARED INOPERABLE AT 0001 HOURS ON SEPTEMBER 8, 1989 WHEN IT WAS FOUND THAT A NORMALLY EXTINGUISHED INDICATING LIGHT IN THE ASSOCIATED TEST CIRCUIT WAS ILLUMINATED. WITH THIS INDICATION, IT WAS PERCEIVED THAT THE VALVE WOULD NOT CLOSE IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS. THE PLANT ENTERED MODE 2 AT 0550 HOURS ON SEPTEMBER 8, 1989. AT 1600 HOURS ON SEPTEMBER 8, 1989, THE CAUSE OF THE ILLUMINATED LIGHT IN THE TEST CIRCUIT WAS IDENTIFIED TO BE A CRUSHED CONDUIT THAT RESULTED IN A SHORT CIRCUIT OF THE TEST CIRCUIT WIRING. THE CRUSHED CONDUIT WAS THE RESULT OF MOVEMENT OF A FOUR INCH FEEDWATER FORWARD FLUSH LINE DUE TO THE SYSTEM FLASHING TO STEAM DURING OPERATION AND THE TRANSIENT THAT RESULTED WHEN THE SYSTEM REFILLED WITH WATER. THE SHORT CIRCUIT CAUSED BY THE CRUSHED CONDUIT DID NOT AFFECT THE OPERATION OF THE VALVE. THE VALVE WOULD HAVE FUNCTIONED AS DESIGNED DUE TO THE CAPABILITY TO BREAK THE SEAL-IN CIRCUIT EITHER BY CONTROL SWITCH OR A MAIN STEAM ISOLATION ACTUATION SIGNAL. CORRECTIVE ACTION CONSISTED OF REPAIRING THE CONDUIT/TEST CIRCUIT WIRING. IN ADDITION, THE METHOD OF OPERATING THIS NON-SAFETY SYSTEM IS BEING REVIEWED WITH INPUT FROM ENGINEERING TO MINIMIZE THE PROBABILITY OF FUTURE SYSTEM TRANSIENTS.

[200] SUMMER 1 DOCKET 50-395 LER 89-017
MISSED SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 091889 REPORT DATE: 102489 NSSS: WE TYPE: PWR

(NSIC 215711) SURVEILLANCE TEST PROCEDURE (STP) 209.001, "INCORE VERSUS EXCORE AXIAL OFFSET EVALUATION," IS REQUIRED TO BE PERFORMED MONTHLY (31 DAYS) WITH THE PLANT IN MODE 1 AND GREATER THAN 15% POWER. THE TEST WAS ORIGINALLY SCHEDULED FOR 9/8/89, BUT DUE TO AN UNSCHEDULED SHUTDOWN, THE TEST WAS UNABLE TO BE PERFORMED. AFTER RETURNING TO POWER ON 9/17, THE ENGINEER REVIEWED THE PLANT CONDITIONS ON 9/18 AND AGAIN DID NOT FEEL THE INITIAL CONDITIONS WERE MET FOR PERFORMING THE TEST. FOLLOWING THE "LOSS OF CONDENSER VACUUM/TURBINE TRIP" ON 9/19, THE ENGINEER CONTINUED TO REVIEW PLANT PARAMETERS. ON 9/21, THE ENGINEER QUESTIONED THE OPERATORS ON WHEN THEY ANTICIPATED REACHING 100% POWER AND WAS TOLD IT WOULD PROBABLY BE 9/22 OR 9/23. BASED ON THE CONVERSATION WITH OPERATIONS AND THE XENON EQUILIBRIUM REQUIREMENTS ADDRESSED IN THE STP, THE STP WAS RESCHEDULED AND SUCCESSFULLY COMPLETED ON 9/25. AT 0930 HRS ON 9/25, AN OFF-NORMAL OCCURRENCE (ONO) REPORT WAS INITIATED AFTER DETERMINING THAT THE TECH SPEC SURVEILLANCE INTERVAL HAD NOT BEEN MET. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO A COMBINATION OF PERSONNEL ERROR AND PROCEDURAL INADEQUACY RESULTING FROM THE STP NOT ACCURATELY ADDRESSING TECH SPEC REQUIREMENTS FOR THE PERFORMANCE OF THE SURVEILLANCE. THE STP IS TO BE REVISED TO ACCURATELY ADDRESS TECH SPEC REQUIREMENTS FOR THE PERFORMANCE OF SURVEILLANCE.

[201] SURRY 1 DOCKET 50-280 LER 89-038
UNPLANNED ENGINEERED SAFETY FEATURES COMPONENT ACTUATION, AUTO START OF #3 EDG WHILE ATTEMPTING TO SHUT DOWN DUE TO EXISTING HI HI CLS SIGNAL.
EVENT DATE: 090389 REPORT DATE: 100289 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 215543) ON SEPTEMBER 3, 1989 AT 2048 HOURS, WITH UNIT 1 AT 100% POWER AND UNIT 2 AT COLD SHUTDOWN, WHILE ATTEMPTING TO SHUT DOWN THE #3 EMERGENCY DIESEL GENERATOR (EDG) FOLLOWING ITS OPERATION IN SUPPORT OF A SPECIAL TEST, THE EDG AUTOMATICALLY STARTED WHEN ITS CONTROL SWITCH WAS PLACED FROM THE "EXERCISE" TO THE "AUTO" POSITION. THE AUTOMATIC START WAS DUE TO AN EXISTING HI HI CONSEQUENCE LIMITING SAFEGUARDS (CLS) SIGNAL THAT HAD BEEN INITIATED FOR THE SPECIAL TEST. DURING THE RECENT OUTAGES, SEVERAL EVENTS HAVE OCCURRED DURING SPECIAL TESTING. THESE EVENTS WILL BE COMPILED AND ANALYZED, AND THE ADMINISTRATIVE PROCEDURE GOVERNING THE PREPARATION AND CONDUCT OF SPECIAL TESTS WILL BE REVISED TO INCORPORATE THE LESSONS LEARNED. THIS EVENT WAS REPORTED TO THE NUCLEAR REGULATORY COMMISSION AS AN UNPLANNED ENGINEERED SAFETY FEATURES COMPONENT ACTUATION PER 10CFR50.72(B)(2)(II).

[202] SURRY 2 DOCKET 50-281 LER 89-005 REV 01
UPDATE ON UNPLANNED ESF COMPONENT ACTUATION CLOSURE OF "A" AND "C" CONDENSER WATERBOX CIRCULATING WATER INLET VALVES.
EVENT DATE: 090889 REPORT DATE: 101689 NSSS: WE TYPE: PWR

(NSIC 215639) ON SEPTEMBER 8, 1989 AT 2142 HOURS, WITH UNIT 2 IN COLD SHUTDOWN, DURING PERFORMANCE OF A PERIODIC TEST ON THE TURBINE BUILDING FLOOD CONTROL CIRCUITRY, TWO OF THE FOUR CONDENSER WATERBOX CIRCULATING WATER (CW) INLET ISOLATION VALVES CLOSED UNEXPECTEDLY. THE CONDENSER INLET VALVES ARE DESIGNED TO CLOSE UPON THE INITIATION OF A HI HI CONSEQUENCE LIMITING SAFEGUARDS (CLS) SIGNAL IN COINCIDENCE WITH A LOSS OF OFF SITE POWER; HOWEVER, NO ACTUAL HI HI CLS SIGNAL WAS PRESENT. THE EVENT IS BEING REPORTED AS AN UNPLANNED ENGINEERED SAFETY FEATURES (ESF) COMPONENT ACTUATION. A FOUR HOUR NON-EMERGENCY REPORT WAS MADE TO THE NUCLEAR REGULATORY COMMISSION PER 10CFR50.72. THE EVENT WAS CAUSED BY A RELAY IN THE FLOOD CONTROL CIRCUIT THAT DID NOT DROP OUT AS REQUIRED DURING TESTING WHICH RESULTED IN ACTUATION OF THE VALVES. IT WAS DETERMINED THAT THE ORIGINAL RELAY HAD BEEN REPLACED WITH A NEW RELAY WHICH REQUIRED 1 LESS HOLD-IN CURRENT CAUSING THIS EVENT TO OCCUR. AN INVESTIGATION WILL BE CONDUCTED TO DETERMINE HOW THE NEW RELAYS WERE INSTALLED.

[203] SURRY 2 DOCKET 50-281 LER 89-006
UNPLANNED ESF COMPONENT ACTUATION, CLOSURE OF CONTAINMENT ISOLATION VALVE DUE TO CONTAINMENT PARTICULATE RADIATION MONITOR ALARM.
EVENT DATE: 091189 REPORT DATE: 101189 NSSS: WE TYPE: PWR

(NSIC 215545) ON SEPTEMBER 11, 1989 AT 1845 HOURS, WITH UNIT 2 AT INTERMEDIATE SHUTDOWN, THE CONTAINMENT PARTICULATE RADIATION MONITOR INDICATION INCREASED ABOVE THE ALARM SETPOINT. THE ALARM RESULTED IN THE CLOSURE OF THE CONTAINMENT INSTRUMENT AIR (IA) COMPRESSORS' NORMAL SUCTION VALVES FROM CONTAINMENT AND THE OPENING OF THE ALTERNATE SUCTION VALVE OUTSIDE CONTAINMENT. THE CONTAINMENT VALVES ALSO CLOSE UPON AN ENGINEERED SAFETY FEATURES (ESF) SIGNAL. THIS EVENT IS BEING REPORTED AS AN UNPLANNED ESF COMPONENT ACTUATION. A FOUR HOUR NON-EMERGENCY REPORT WAS MADE TO THE NUCLEAR REGULATORY COMMISSION PER 10CFR50.72. THE CAUSE OF THE EVENT WAS HIGH DETECTOR READINGS DUE TO A REDUCTION IN CONTAINMENT PRESSURE THAT RELEASED CONTAMINATED PARTICLES AND A FAILED DETECTOR HEAT TRACE CIRCUIT. THE ALARM WAS RESET, AND THE IA COMPRESSOR SUCTION VALVES WERE REALIGNED TO THEIR NORMAL POSITION. THE BASIS FOR THE ALERT AND ALARM SETPOINTS OF THE CONTAINMENT PARTICULATE AND GASEOUS MONITOR ARE BEING EVALUATED.

[204] SURRY 2 DOCKET 50-281 LER 89-007
MANUAL REACTOR TRIP INITIATED TO RESET CONTROL RODS AFTER IMPROPER BANK OVERLAP NOTED DURING REACTOR STARTUP.
EVENT DATE: 091689 REPORT DATE: 101389 NSSS: WE TYPE: PWR

(NSIC 215546) ON SEPTEMBER 16, 1989 AT 1228 HOURS WITH UNIT 2 SUBCRITICAL, DURING A REACTOR STARTUP, A MANUAL REACTOR TRIP WAS INITIATED WHEN IT WAS DETERMINED THAT IMPROPER BANK OVERLAP EXISTED BETWEEN THE "A" AND "B" CONTROL ROD BANKS. THE REACTOR TRIP WAS INITIATED TO INSERT ALL CONTROL RODS AND TO RESET THE CONTROL ROD STEP COUNTERS TO ZERO. A FOUR HOUR NON-EMERGENCY REPORT WAS MADE TO THE NUCLEAR REGULATORY COMMISSION PER 10CFR50.72. TROUBLESHOOTING DID NOT REVEAL THE CAUSE OF THE IMPROPER BANK OVERLAP. DURING THE SUBSEQUENT REACTOR STARTUP, NO PROBLEMS WERE ENCOUNTERED WITH CONTROL ROD BANK OVERLAP.

[205] SURRY 2 DOCKET 50-281 LER 89-009
TURBINE TRIP/REACTOR TRIP DUE TO 86 BU TRIP CAUSED BY SPURIOUS ACTUATION OF KD-41 RELAY.
EVENT DATE: 091889 REPORT DATE: 101389 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215547) ON SEPTEMBER 18, 1989, AT 1042 HOURS, WITH UNIT 2 REACTOR AT 14% POWER AND THE TURBINE AT 1800 RPM UNDER NO LOAD CONDITIONS, A REACTOR TRIP SIGNAL WAS GENERATED. A GENERATOR BACKUP DIFFERENTIAL LOCKOUT RELAY 86 BU TRIPPED THE TURBINE, AND SINCE REACTOR POWER WAS GREATER THAN 10%, THE TURBINE TRIP INITIATED A REACTOR TRIP. OPERATORS PERFORMED THE APPROPRIATE PLANT PROCEDURES AND QUICKLY STABILIZED THE PLANT FOLLOWING THE TRIP. THE 86 BU GENERATOR BACKUP LOCKOUT RELAY TRIP WAS CAUSED BY THE SPURIOUS ACTUATION OF THE GENERATOR BACKUP IMPEDANCE RELAY (KD- 41). THE EXACT CAUSE OF THE SPURIOUS ACTUATION OF THE RELAY COULD NOT BE DETERMINED, HOWEVER FAULTS WERE DISCOVERED IN THE RELAY. THE FAULTED KD-41 RELAY WAS REPLACED AND APPROPRIATE TESTING WAS PERFORMED. THE GENERATOR STARTUP PROCEDURE HAS BEEN REVISED TO ENSURE THAT REACTOR POWER IS LESS THAN 10% PRIOR TO CLOSING THE EXCITER FIELD BREAKER. A FOUR HOUR NON-EMERGENCY REPORT WAS MADE TO THE NUCLEAR REGULATORY COMMISSION IN ACCORDANCE WITH 10CFR50.72.

[206] SURRY 2 DOCKET 50-281 LER 89-010
REACTOR TRIP DUE TO LOW STEAM GENERATOR LEVEL FOLLOWING A HIGHER THAN EXPECTED LOAD INCREASE DURING UNIT STARTUP.
EVENT DATE: 091989 REPORT DATE: 101389 NSSS: WE TYPE: PWR

(NSIC 215548) ON 9/19/89 WITH UNIT 2 REACTOR POWER AT 25%, SHORTLY AFTER PLACING UNIT 2 ON LINE, AN AUTOMATIC REACTOR TRIP WAS GENERATED AT 0051 HOURS AS A RESULT OF A LO LO STEAM GENERATOR LEVEL. A FOUR HOUR NON-EMERGENCY REPORT WAS MADE TO THE NRC AT 0120 HOURS PER 10CFR50.72. A RAPID TURBINE LOAD INCREASE OCCURRED DURING STARTUP WHICH LED TO OSCILLATIONS IN THE STEAM GENERATOR (S/G) WATER LEVELS, EVENTUALLY LEADING TO THE LO LO S/G WATER LEVEL TRIP. THE CAUSE OF THE RAPID TURBINE LOAD INCREASE WAS THE OPERATOR RAISING THE GOVERNOR VALVE POSITION LIMITER MORE RAPIDLY THAN DESIRED. THE OPERATORS PERFORMED THE APPROPRIATE EMERGENCY PROCEDURES, AND QUICKLY STABILIZED THE UNIT FOLLOWING THE REACTOR TRIP. THE LESSONS LEARNED FROM THIS EVENT AND SUBSEQUENT UNIT STARTUP HAVE BEEN DISSEMINATED TO ALL OPERATIONS PERSONNEL.

[207] SURRY 2 DOCKET 50-281 LER 89-011
CONTAINMENT INTEGRITY VIOLATED DUE TO CONTAINMENT PENETRATION BEING IN AN UNANALYZED CONDITION.
EVENT DATE: 091989 REPORT DATE: 101889 NSSS: WE TYPE: PWR

(NSIC 215679) ON SEPTEMBER 19, 1989 WITH UNIT 2 CRITICAL AT LESS THAN 2% POWER, A REVIEW OF AN EXISTING PIPE SUPPORT, RC 19.LA, FOUND THAT THE SUPPORT WAS RIGIDLY

ATTACHED TO BOTH THE AUXILIARY BUILDING STRUCTURE AND THE UNIT 2 CONTAINMENT. THIS CONDITION IS IN VIOLATION OF THE DESIGN REQUIREMENTS FOR "RATTLE SPACE" BETWEEN THE TWO STRUCTURES TO ACCOMMODATE DIFFERENTIAL SEISMIC DISPLACEMENTS, AND THE SUPPORT WAS CONSIDERED INOPERABLE. BECAUSE THE SUPPORT FORMED THE SEISMIC BOUNDARY OF THE STRESS ANALYSIS PROBLEM FROM THE CONTAINMENT PENETRATION, CONTAINMENT INTEGRITY COULD NOT BE ASSURED. A SIX HOUR ACTION STATEMENT TO HOT SHUTDOWN WAS ENTERED DUE TO THE PLANT BEING IN AN UNANALYZED CONDITION. A ONE HOUR NON-EMERGENCY REPORT WAS MADE TO THE NUCLEAR REGULATORY COMMISSION IN ACCORDANCE WITH 10CFR50.72.B.1.II. THE SUPPORT WAS MODIFIED TO SATISFY INTERIM OPERABILITY WITHIN THE SIX HOUR ACTION STATEMENT. IT WAS SUBSEQUENTLY MODIFIED TO SATISFY LONG TERM OPERABILITY BY THE ADDITION OF A BRACE ATTACHED TO THE AUXILIARY BUILDING. SUPPORTS ON THE CORRESPONDING UNIT 1 LINE WERE CHECKED TO ENSURE THAT UNIT 1 DID NOT HAVE A SIMILAR PROBLEM. IN ADDITION, A REVIEW OF SUPPORT DRAWINGS DESIGNED BY THE SAME ENGINEERING CONSULTING FIRM RESPONSIBLE FOR THE UNACCEPTABLE SUPPORT DESIGN HAS BEEN COMPLETED. NO OTHER DISCREPANCIES WERE NOTED.

[208] SURRY 2 DOCKET 50-281 LER 89-012
INDIVIDUAL ROD POSITION INDICATORS OUT OF SPECIFICATION FOR GREATER THAN 60 MINUTES IN A 24 HOUR PERIOD.
EVENT DATE: 091989 REPORT DATE: 101889 NSSS: WE TYPE: PWR
VENDOR: MAGNETICS DIV SPANG INDUSTRIES, INC.

(NSIC 215680) ON SEPTEMBER 19, 1989 AT 0815 HOURS DURING A UNIT STARTUP WITH THE REACTOR CRITICAL AND LESS THAN 2% POWER, IT WAS DISCOVERED THAT TWO INDIVIDUAL ROD POSITION INDICATORS (IRPI) IN CONTROL BANK "A", GROUP TWO, HAD DIFFERED IN INDICATED POSITION FROM THE BANK'S STEP DEMAND COUNTER BY MORE THAN 12 STEPS FOR 63 AND 64 MINUTES WITHIN THE LAST 24 HOURS. THIS OCCURRENCE IS CONTRARY TO TECHNICAL SPECIFICATION (T.S.) 3.12.E, WHICH ALLOWS THIS CONDITION TO EXIST FOR A MAXIMUM OF 60 MINUTES IN ANY 24 HOUR PERIOD. THIS EVENT WAS DUE TO AN INADEQUATE METHOD USED TO LOG ACCUMULATED TIMES FOR IRPI DIFFERENCES IN EXCESS OF 12 STEPS. THE PROCEDURE USED TO LOG THESE ACCUMULATED TIMES WILL BE MODIFIED AND THE PLANT COMPUTER WILL BE REPROGRAMMED TO MONITOR ACCUMULATED TIMES AND ALARM TO ALERT THE OPERATOR PRIOR TO EXCEEDING THE 60 MINUTE LIMIT.

[209] SUSQUEHANNA 1 DOCKET 50-387 LER 89-023
PLANT SHUTDOWN COMPLETED WHEN VACUUM RELIEF VALVES DECLARED INOPERABLE.
EVENT DATE: 090889 REPORT DATE: 101089 NSSS: GE TYPE: BWR

(NSIC 215590) AT 1620 HOURS ON 9/8/89, WITH UNIT 1 AT 100% POWER, IT WAS DETERMINED THAT A POTENTIALLY INOPERABLE CONDITION EXISTED ON THE SUPPRESSION CHAMBER - DRYWELL VACUUM RELIEF VALVES DUE TO A COMPONENT CHANGE (MALE ELBOW FITTING ON THE VALVE ACTUATING CYLINDER) WHICH HAD TAKEN PLACE DURING THE UNIT 1 4TH REFUELING OUTAGE. THE ORIGINAL FITTING CONTAINED AN ORIFICE TO LIMIT THE CLOSING VELOCITY OF THE VACUUM RELIEF VALVES DURING CYCLING IN THE EVENT OF A LOCA; THE NEW FITTING DID NOT. PRELIMINARY ANALYSIS OF THE EFFECT OF THE NEW FITTING ON RELIEF VALVE CYCLING VELOCITY WAS INCONCLUSIVE AT THE TIME, SO THE CONSERVATIVE MEASURE OF DECLARING THE VALVES INOPERABLE WAS TAKEN IN THE ABSENCE OF A COMPLETE ANALYSIS. DECLARING THE VACUUM RELIEF VALVES INOPERABLE REQUIRES ENTRY INTO TECH SPEC 3.0.3 AND INITIATION OF A PLANT SHUTDOWN. THIS INCIDENT WAS ATTRIBUTED TO THE FOLLOWING: CONTRARY TO ADMINISTRATIVE PROCEDURES, A FIELD DRAWING CHANGE MECHANISM WAS USED TO EXPAND THE SCOPE OF A MODIFICATION PACKAGE. HUMAN ERROR RESULTED IN A COMPONENT CHANGE REQUEST THAT WAS NOT EQUIVALENT TO THE ORIGINAL COMPONENT. ENGINEERING REVIEWS FAILED TO IDENTIFY THE ABOVE ERRORS. DOCUMENTATION AVAILABLE DID NOT CLEARLY DESCRIBE THE INVOLVED COMPONENT NOR ITS FUNCTION. THE ORIFICES WERE REINSTALLED AND THE VALVES WERE SATISFACTORILY RETESTED.

[210] SUSQUEHANNA 2 DOCKET 50-388 LER 89-008
DRYWELL PURGE AIR SUPPLY OUTBOARD ISOLATION VALVE FAILS CLOSED DURING DRYWELL DEINERTING ACTIVITIES.
EVENT DATE: 091089 REPORT DATE: 100689 NSSS: GE TYPE: BWR
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)
CIRCLE SEAL

(NSIC 215572) AT 1500 ON 9/10/89 AND 0600 ON 9/11/89, DURING A SCHEDULED SHUTDOWN FOR THE UNIT 2 THIRD REFUELING OUTAGE, THE DRYWELL PURGE AIR SUPPLY OUTBOARD ISOLATION VALVE, HV-25723, AUTOMATICALLY CLOSED. THE CLOSURES WERE CAUSED BY A BLOWN FUSE IN THE VALVE'S CONTROL CIRCUIT AND AN OPEN COIL IN THE VALVE'S AIR ACTUATOR SOLENOID CONTROL VALVE, RESPECTIVELY. THE SOLENOID VALVE COIL FAILED ONCE DURING VALVE FUNCTIONAL TESTING AFTER THE BLOWN FUSE WAS REPLACED AND AGAIN AFTER THE SOLENOID VALVE WAS REPLACED. THE CAUSE OF THESE FAILURES HAS NOT BEEN DETERMINED. FURTHER INVESTIGATIONS WILL BE CONDUCTED. FOR BOTH OCCURRENCES, THE AFFECTED PRIMARY CONTAINMENT PENETRATION WAS ISOLATED BY DEACTIVATING THE REDUNDANT (INBOARD) ISOLATION VALVE IN THE CLOSED POSITION. ADMINISTRATIVE CONTROLS ARE IN PLACE WHICH WILL ENSURE THAT HV-25723 IS REWORKED, AND DEMONSTRATED TO BE OPERABLE PRIOR TO STARTUP. THIS EVENT WAS DETERMINED TO BE REPORTABLE UNDER 10CFR50.73(A)(2)(IV) IN THAT THE UNANTICIPATED CLOSURES OF HV-25723 (A PRIMARY CONTAINMENT ISOLATION VALVE) CONSTITUTED UNPLANNED ACTUATIONS OF AN ENGINEERED SAFETY FEATURE. SINCE, ON BOTH OCCURRENCES, HV-25723 ACTUATED, PER DESIGN, TO THE SAFE, CLOSED POSITION, THERE WERE NO SAFETY CONSEQUENCES OR COMPROMISES TO THE HEALTH OR SAFETY OF THE PUBLIC.

[211] SUSQUEHANNA 2 DOCKET 50-388 LER 89-009
CONTROL POWER LOST TO SEVERAL CONTAINMENT ISOLATION VALVES DURING APPLICATION OF PROTECTIVE BLOCKING.
EVENT DATE: 091789 REPORT DATE: 101789 NSSS: GE TYPE: BWR

(NSIC 215673) ON 9/17/89 AT APPROXIMATELY 0430 HOURS WITH UNIT 2 IN THE REFUELING CONDITION, AN UNPLANNED ENGINEERED SAFETY FEATURE ACTUATION OCCURRED WHEN CONTROL POWER WAS LOST TO SEVERAL CONTAINMENT ISOLATION VALVES. CONTAINMENT ATMOSPHERE CONTROL AND CONTAINMENT INSTRUMENT GAS CONTAINMENT ISOLATION VALVES WERE AFFECTED BY THE EVENT. THE EVENT OCCURRED WHEN PERSONNEL PROTECTIVE BLOCKING WAS BEING APPLIED IN SUPPORT OF PLANNED OUTAGE WORK. SPECIFICALLY, A MODIFICATION WAS BEING PERFORMED TO THE DRYWELL SUMP LOGIC. WHEN THE SLIDING LINKS WERE OPENED TO DE-ENERGIZE THE DRYWELL SUMP LOGIC, THE CONTROL LOGIC FOR SEVERAL CONTAINMENT ISOLATION VALVES WAS DE-ENERGIZED AS WELL. THE ELECTRICAL SCHEME DRAWING THE WORK GROUP USED FOR THE BLOCKING POINT SELECTION DID NOT REFLECT THE OTHER CONTROL SCHEMES CONNECTED DOWNSTREAM OF THE SELECTED BLOCKING POINTS. FURTHER REVIEW OF THE PANEL WIRING DIAGRAMS AND CONNECTION LISTS WOULD HAVE IDENTIFIED MULTIPLE SCHEMES POWERED FROM THE SLIDING LINKS. IN ADDITION, SUBSEQUENT REQUIRED REVIEWS OF THE BLOCKING REQUEST FAILED TO IDENTIFY THE DEFICIENCY. THE AFFECTED VALVES ARE DESIGNED TO CLOSE UPON LOSS OF CONTROL POWER AS SUCH NO SAFETY CONSEQUENCES OR COMPROMISE TO PUBLIC HEALTH RESULTED. SPECIFIC GUIDANCE WHICH ENHANCES AND REFINES THE METHOD FOR DETERMINING THE CORRECT PROTECTIVE BLOCKING POINTS HAS BEEN ISSUED AND REVIEWED BY THE WORK GROUP.

[212] SUSQUEHANNA 2 DOCKET 50-388 LER 89-010
MAIN STEAM LINE PENETRATIONS EXCEED MAXIMUM ALLOWABLE LEAK RATE DURING REGULARLY SCHEDULED LOCAL LEAK RATE TESTING.
EVENT DATE: 092089 REPORT DATE: 102089 NSSS: GE TYPE: BWR
VENDOR: ATWOOD & MORRILL CO., INC.

(NSIC 215707) AT 0915 ON 9/20/89, WITH UNIT 2 IN ITS THIRD REFUELING AND INSPECTION OUTAGE, IT WAS DETERMINED THAT THE MAIN STEAM LINES (MSL) HAD EXHIBITED A LEAK RATE IN EXCESS OF THE MAXIMUM ALLOWABLE SPECIFIED IN TECHNICAL SPECIFICATION 3.6.1.2.C DURING REGULARLY SCHEDULED LOCAL LEAK RATE TESTING (LLRT)

OF THE MSL PENETRATIONS. THE LLRT ACTIVITIES HAD COMMENCED ON 9/16/89. THE LEAK RATE WAS DETERMINED TO BE ABOVE THE TECHNICAL SPECIFICATION LIMIT OF 46 STANDARD CUBIC FEET PER HOUR (SCFH) DURING THE EVALUATION OF THE LLRT DATA. THE "A" AND "B" MSL PENETRATIONS HAD INDICATED LEAKAGE IN EXCESS OF 53 SCFH AND 318 SCFH RESPECTIVELY. ADDITIONAL TESTING IS BEING CONDUCTED TO MORE ACCURATELY QUANTIFY THE LEAK RATES. THE APPROPRIATE MAIN STEAM ISOLATION VALVES WILL BE REWORKED AND POST MAINTENANCE LLRTS WILL BE PERFORMED DURING THE CURRENT REFUELING OUTAGE. THIS EVENT WAS DETERMINED TO BE REPORTABLE UNDER 10CFR50.73(A)(2)(II) IN THAT CONTAINMENT PENETRATION LEAKAGE IN EXCESS OF TECHNICAL SPECIFICATION LIMITS HAS THE POTENTIAL TO RESULT IN A PRINCIPAL SAFETY BARRIER BEING IN A DEGRADED CONDITION. THE RESULTS OF THESE ACTIVITIES AND A SAFETY ASSESSMENT OF THE EVENT INCLUDING CONFIRMATION OF TEST RESULTS WILL BE PROVIDED IN A SUPPLEMENTAL REPORT. A VERBAL REPORT WAS MADE TO THE COMMISSION AT 1115 ON 9/20/89 VIA THE EMERGENCY NOTIFICATION SYSTEM.

[213] TROJAN DOCKET 50-344 LER 88-032 REV 01
UPDATE ON CONTROL ROOM EMERGENCY VENTILATION SYSTEM INOPERABLE DUE TO INABILITY TO MAINTAIN SUFFICIENT POSITIVE PRESSURE IN THE CONTROL ROOM.
EVENT DATE: 092488 REPORT DATE: 101389 NSSS: WE TYPE: PWR

(NSIC 215629) ON SEPTEMBER 24, 1988, DURING PERIODIC OPERATING TEST (POT) 20.1, "CONTROL ROOM EMERGENCY VENTILATION PERFORMANCE", A POSITIVE PRESSURE OF GREATER THAN OR EQUAL TO 0.125 INCHES WATER GAGE (W.G) COULD NOT BE OBTAINED IN THE CONTROL ROOM IN RELATION TO OUTSIDE AIR AND THE TURBINE BUILDING. TRAIN A OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CB.1A) WAS BEING TESTED. INITIAL TROUBLESHOOTING LED TO THE CONCLUSION THAT OPERATION OF THE AUXILIARY BUILDING EXHAUST SYSTEM (AB.3), WITH AUXILIARY BUILDING ACCESS DOOR 30 OPEN TO THE CONTROL BUILDING, WAS INFLUENCING THE PRESSURIZATION CAPABILITY OF CB.1. THIS WAS CONFIRMED WHEN, WITH DOOR 30 CLOSED, CB.1A CREATED A POSITIVE PRESSURE OF > 0.125 INCHES W.G. CB.1B WAS NOT TESTED AT THIS TIME AND WAS THOUGHT TO BE AFFECTED IN THE SAME WAY. THE REAL CAUSE, HOWEVER, WAS DETERMINED TO BE A MISPOSITIONED BALANCING DAMPER, BD.104, IN THE CB.1A RETURN AIR DUCT. THIS WAS IDENTIFIED DURING THE PERFORMANCE OF POT 20.1 ON OCTOBER 22, 1988 WHEN CB.1A FAILED AND CB.1B PASSED WITH A TEST CONFIGURATION SIMILAR TO THAT ON SEPTEMBER 24. BASED ON THE OCTOBER TEST IT IS PRESUMED THAT CB.1B WAS OPERABLE IN SEPTEMBER. THE CAUSE OF THIS EVENT WAS BD.104 BEING OUT OF POSITION AND INADEQUATE CONTROLS BEING IN PLACE FOR MAINTAINING ITS POSITION. DAMPER BD.104 WAS RETURNED TO ITS PROPER POSITION.

[214] TROJAN DOCKET 50-344 LER 89-007 REV 01
UPDATE ON CONTROL ROOM ISOLATION DAMPER CLOSURE TIME EXCEEDS REQUIRED MAXIMUM.
EVENT DATE: 040689 REPORT DATE: 101389 NSSS: WE TYPE: PWR

(NSIC 215644) AT 1700 HOURS ON APRIL 6, 1989, WITH THE PLANT IN MODE 3 (HOT STANDBY), IT WAS DISCOVERED THAT THE ISOLATION DAMPERS IN THE CONTROL ROOM VENTILATION SYSTEM (CB.2) DID NOT CLOSE IN THE TIME REQUIRED BY THE TROJAN TECHNICAL SPECIFICATIONS. THE TIME REQUIRED FOR THE CLOSURE OF THE DAMPERS IS NO MORE THAN THREE SECONDS MEASURED TIMES FOR THE FOUR DAMPERS RANGED FROM 3.77 TO 4.30 SECONDS. THE CAUSE OF THE EVENT COULD NOT BE FIRMLY ESTABLISHED. IT IS BELIEVED TO BE PRESSURE SWITCHES THAT WERE INCORRECTLY SET DURING SYSTEM INSTALLATION. THE CONDITION WAS MASKED BY A POSSIBLE WIRING ERROR AND POOR TESTING PROCEDURES. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO PLACE THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CB.1) IN OPERATION AND TO CLOSE THE CONTROL ROOM ISOLATION DAMPERS IN THE CB.2 SYSTEM. THE SETPOINTS OF THE PRESSURE SWITCHES CONTROLLING THE VENTILATION FANS WERE CHANGED TO INSURE THE DAMPERS SHUT MORE QUICKLY. THE TEST PROCEDURE WAS CHANGED AND THE DAMPERS RETESTED IN THE REQUIRED CLOSING TIME. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[215] TROJAN DOCKET 50-344 LER 89-018
BOTH TRAINS OF RESIDUAL HEAT REMOVAL INOPERABLE DUE TO COGNITIVE ERROR.
EVENT DATE: 090889 REPORT DATE: 100989 NSSS: WE TYPE: PWR

(NSIC 215568) ON 9/8/89, THE PLANT WAS IN MODE 1 AT 99% POWER WITH REACTOR COOLANT SYSTEM (RCS) CONDITIONS OF 585F AND 2242 PSIG. AT APPROXIMATELY 0945, WORK WAS INITIATED TO CHANGE SETPOINTS FOR THE 'B' TRAIN RESIDUAL HEAT REMOVAL (RHR) PUMP S FLOW INDICATING SWITCH. OPERATIONS PERSONNEL REVIEWED WORK TO BE DONE, BUT CONSIDERED THE 'B' TRAIN RHR PUMP OPERABLE AS IT WOULD AUTOMATICALLY START AND THE RECIRCULATION FLOW CONTROL VALVE COULD BE OPENED FROM THE CONTROL ROOM WHILE WORK WAS IN PROGRESS. THE SETPOINTS HAD BEEN ADJUSTED, BUT FUNCTIONAL TESTING WAS DELAYED WHILE CHANGES TO TESTING METHODS, DESIRED BY OPERATIONS PERSONNEL, WERE REVIEWED. AT 1250, THE 'A' TRAIN COMPONENT COOLING WATER (CCW) SYSTEM WAS DECLARED INOPERABLE AND CROSS-CONNECTED TO THE 'B' TRAIN OF CCW PER THE CONTROLLING PROCEDURE FOR A SERVICE WATER SYSTEM (SWS) BIOCIDE TREATMENT. SYSTEMS COOLED BY 'A' TRAIN CCW, SPECIFICALLY THE 'A' TRAIN RHR PUMP, WERE THEREFORE ALSO INOPERABLE. THE ON-COMING SWING SHIFT WAS BRIEFED ON BOTH EVOLUTIONS BUT THE TRAIN OF RHR BEING WORKED WAS NOT ADEQUATELY COMMUNICATED. WHEN THE REVISED FUNCTIONAL TEST WAS PRESENTED TO THE SHIFT SUPERVISOR AT 1520 IT WAS REALIZED THAT BOTH TRAINS OF RHR WERE INOPERABLE. IMMEDIATE ACTION WAS TO RESTORE THE 'A' TRAIN OF CCW (AND RHR) TO SERVICE. THIS WAS ACCOMPLISHED AT 1550. 'B' TRAIN RHR WAS RESTORED AT 2037.

[216] TROJAN DOCKET 50-344 LER 89-020
INSTRUMENT LOOP INACCURACIES NOT ACCOUNTED FOR IN PERFORMING SURVEILLANCE ON REACTOR COOLANT SYSTEM AVERAGE TEMPERATURE COULD ALLOW OPERATION OUTSIDE ANALYSIS.
EVENT DATE: 091489 REPORT DATE: 101689 NSSS: WE TYPE: PWR

(NSIC 215643) ON 9/14/89, FOLLOWING A DISCUSSION WITH THE NUCLEAR STEAM SUPPLY SYSTEM VENDOR IT WAS IDENTIFIED THAT THE VALUES FOR REACTOR COOLANT SYSTEM (RCS) AVERAGE TEMPERATURE (T-AVG) AND PRESSURIZER PRESSURE IN TECH SPEC 3 2.5, "DNB PARAMETERS", WERE THE ACTUAL VALUES USED IN THE SAFETY ANALYSES RATHER THAN INDICATED VALUES (ACTUAL VALUE WITH INSTRUMENT UNCERTAINTIES INCLUDED). THE PLANT PROCEDURE FOR PERFORMING THE SURVEILLANCE TO COMPLY WITH THIS TECH SPEC COMPARED THE INDICATED VALUE FOR T-AVG TO THE ACTUAL VALUE WITHOUT APPLYING AN INSTRUMENT UNCERTAINTY TO EITHER NUMBER. THE CAUSE OF THIS ERROR WAS A FAILURE (IN 1976) TO UNDERSTAND THE BASIS FOR THESE VALUES IN THE TECH SPECS. CORRECTIVE ACTION WAS TO DEVIATE THE SURVEILLANCE PROCEDURE TO ACCOUNT FOR INSTRUMENT UNCERTAINTIES. ADDITIONALLY. PORTLAND GENERAL ELECTRIC COMPANY WILL COMPLETE A REVIEW OF ALL OTHER TECHNICAL SPECIFICATION VALUES TO DETERMINE IF INSTRUMENT UNCERTAINTIES ARE BEING PROPERLY APPLIED. THIS REVIEW WILL BE COMPLETED BY MARCH 1992 (CTL #21066). THE INITIAL CONDITIONS FOR THE SAFETY ANALYSES MIGHT HAVE BEEN EXCEEDED IF INDICATED T-AVG WAS ABOVE 585 DEGREES F. HOWEVER, CONSERVATIVE MARGINS IN THE DEPARTURE FROM NUCLEATE BOILING PARAMETERS WOULD COMPENSATE FOR THIS. THIS EVENT DID NOT CAUSE A SIGNIFICANT EFFECT ON HEALTH AND SAFETY.

[217] TROJAN DOCKET 50-344 LER 89-010
FAILURE TO FOLLOW PROCEDURE ACTUATES AUXILIARY FEEDWATER SYSTEM .
EVENT DATE: 091689 REPORT DATE: 101689 NSSS: WE TYPE: PWR

(NSIC 215670) ON 9/16/89, THE PLANT WAS AT 1% POWER (MODE 2) AND SHUTTING DOWN TO ENTER AN OUTAGE. ONE MAIN FEEDWATER PUMP AND THE ELECTRIC (NON-ENGINEERED SAFETY FEATURE - ESF) AUXILIARY FEEDWATER PUMP WERE IN USE TO MAINTAIN STEAM GENERATOR WATER LEVEL. PLANT PROCEDURE GENERAL OPERATING INSTRUCTION (GOI)-3, "PLANT SHUTDOWN FROM POWER OPERATION TO HOT STANDBY" WAS BEING USED BY THE CONTROL OPERATOR TO DIRECT ACTIVITIES TO SHUT DOWN THE PLANT. THE CONTROL OPERATOR DIRECTED AN OPERATOR TO STOP THE OPERATING MAIN FEEDWATER PUMP. WHEN THE OPERATING MAIN FEEDWATER PUMP WAS STOPPED THE (ESF) AUXILIARY FEEDWATER PUMPS (AFPS) AUTOMATICALLY STARTED. AFTER CONFIRMING THAT THE CAUSE OF THE AUTOMATIC

AFP START WAS FROM A "LOSS OF BOTH MAIN FEEDWATER PUMPS" SIGNAL, THE SIGNAL WAS BLOCKED. BOTH ESF AUXILIARY FEEDWATER PUMPS WERE THEN SECURED AND THEIR AUTOMATIC START SIGNALS RESET. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR IN NOT FOLLOWING THE GOI-3 PROCEDURE. THE PROCEDURE HAS A SPECIFIC STEP TO BLOCK THE AFP'S START SIGNAL PRIOR TO STOPPING THE OPERATING MAIN FEEDWATER PUMP. CORRECTIVE ACTION WAS TO HOLD THE RESPONSIBLE INDIVIDUAL ACCOUNTABLE BY IMPLEMENTING APPROPRIATE DISCIPLINARY ACTION. ADDITIONALLY, THIS EVENT WAS DISCUSSED IN A LESSONS LEARNED BULLETIN. STEAM GENERATOR (SG) WATER LEVELS WERE BEING MAINTAINED BY THE NON-ESF AUXILIARY FEEDWATER PUMP.

[218] TROJAN DOCKET 50-344 LER 89-019
PRESSURIZER SAFETY VALVE SETPOINT FOUND OUT-OF-TOLERANCE DURING TESTING.
EVENT DATE: 091689 REPORT DATE: 101689 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE & GAGE CO.

(NSIC 215645) DURING PERFORMANCE OF PRESSURIZER SAFETY VALVE IN-PLACE LIFT SETPOINT TESTING ON SEPTEMBER 16, 1989. PSV.8010B LIFTED AT 2552 PSIG. THIS WAS OUTSIDE THE TROJAN TECHNICAL SPECIFICATION (TTS) ALLOWED TOLERANCE OF 2485 2 PERCENT (I.E., 2435 TO 2535 PSIG). THE VALVE LIFT SETPOINT WAS ADJUSTED AND THE VALVE RETESTED SATISFACTORILY. THE CAUSE OF THIS EVENT APPEARS TO BE A CHANGE IN THE TEST METHOD. THE PREVIOUS TEST METHOD TESTED THE PRESSURIZER SAFETY VALVES WITH THE LOOP SEAL DRAINED. RECENT INDUSTRY EXPERIENCE INDICATED THAT THIS METHOD COULD CAUSE LIFT SETPOINT ERRORS. NO POSITIVE CORRELATION HAS BEEN ESTABLISHED AT THIS TIME. BUT THE SITUATION IS UNDER INVESTIGATION. THE PRESSURIZER SAFETY VALVE TEST PROCEDURE HAS BEEN REVISED TO TEST WITH THE LOOP SEAL INTACT. THE OTHER TWO PRESSURIZER SAFETY VALVES HAVE BEEN TESTED WITH THE LOOP SEAL INTACT AND ARE WITHIN SPECIFICATION (8010A DURING THE SHUTDOWN AND THE NEW 8010C PRIOR TO INSTALLATION TO CORRECT EXCESSIVE LEAKAGE). THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH THE SPECIAL REPORTING REQUIREMENTS OF TTS 3.4.3.1, "SAFETY AND RELIEF VALVES-OPERATING." IF THE INVESTIGATION REVEALS ADDITIONAL SIGNIFICANT INFORMATION THIS LICENSEE EVENT REPORT WILL BE REVISED.

[219] TROJAN DOCKET 50-344 LER 89-023
CONTAINMENT INTEGRITY VIOLATED DURING LOCAL LEAK RATE TESTING.
EVENT DATE: 091889 REPORT DATE: 102389 NSSS: WE TYPE: PWR
VENDOR: AMPHENOL

(NSIC 215699) DURING AN EVENT EVALUATION IT WAS DETERMINED THAT DURING THE PERFORMANCE OF PERIODIC ENGINEERING TEST (PET) 5.2 "CONTAINMENT LOCAL LEAK RATE TESTING (LLRT)" ON THE ELECTRICAL PENETRATION ASSEMBLIES (EPA), TROJAN TECHNICAL SPECIFICATION (TTS) 3.6.1.1 "CONTAINMENT INTEGRITY" WAS VIOLATED WHEN IN MODES 1-4. EACH EPA HAS TWO SELF ENERGIZED SEALS. THE INBOARD SEAL PROVIDES A SEAL FOR LEAK TESTING PURPOSES, BUT IS NOT CONSIDERED A CONTAINMENT PRESSURE BOUNDARY SEAL. THE OUTBOARD SEAL PERFORMS THE CONTAINMENT PRESSURE BOUNDARY FUNCTION. WITH THE TEST CART HOOKED UP TO THE PENETRATION A FLOW PATH EXISTS FROM CONTAINMENT PAST THE INBOARD SEAL TO THE OUTSIDE ATMOSPHERE. THE ROOT CAUSE OF THIS EVENT HAS NOT BEEN FIRMLY ESTABLISHED, AND AN EVALUATION CONTINUES. IT APPEARS TO BE A COMBINATION OF PROCEDURAL INADEQUACIES IN PET 5-2, A LACK OF UNDERSTANDING OF ELECTRICAL PENETRATION DESIGN, AND SEVERAL MISSED OPPORTUNITIES TO IDENTIFY THE PROBLEM. THE IMMEDIATE CORRECTIVE ACTION WAS TO STOP LLRTS OF EPAS DURING MODES 1-4. THE PERMANENT CORRECTIVE ACTIONS WILL INCLUDE REVISING PET 5-2, EVALUATING ALTERNATE TESTING METHODS. ADDITIONAL DETAILS AND CORRECTIVE ACTIONS WILL FOLLOW WITHIN THE NEXT 60 DAYS AS A REVISION TO THIS LICENSEE EVENT REPORT. THIS EVENT HAD NO EFFECT ON THE HEALTH AND SAFETY OF THE PUBLIC.

[220] TROJAN DOCKET 50-344 LER 89-022
100% POWER REFERENCE TEMPERATURE USED IN ROD CONTROL SYSTEM PROGRAM DIFFERENT THAN THE VALUE IN THE SAFETY ANALYSIS.
EVENT DATE: 091989 REPORT DATE: 101989 NSSS: WE TYPE: PWR

(NSIC 215646) ON 9/19/89 AN INVESTIGATION WAS CONTINUING INTO A REACTOR COOLANT SYSTEM (RCS) LOOP AVERAGE TEMPERATURE IMBALANCE AS A CONTRIBUTING CAUSE OF SPURIOUS OVERTEMPERATURE DELTA TEMPERATURE ALARMS. DURING A DISCUSSION WITH THE NUCLEAR STEAM SUPPLY SYSTEM (NSSS) VENDOR IT WAS DETERMINED THAT THE CURRENT MAXIMUM SETPOINT TEMPERATURE (T-REF) FOR THE ROD CONTROL SYSTEM WAS NON-CONSERVATIVE WITH RESPECT TO THE T-REF USED IN THE TROJAN SAFETY ANALYSES. THIS NON-CONSERVATISM WAS DUE TO A MISCOMMUNICATION BETWEEN PORTLAND GENERAL ELECTRIC AND THE NSSS VENDOR IN 1976. THE TEMPERATURE IMBALANCE, DISCOVERED DURING INITIAL STARTUP TESTING, CAUSED RCS BULK AVERAGE TEMPERATURE (T-AVG) TO BE LOWER THAN THE DESIGN VALUE OF 584.7 DEGREES F, WHICH CAUSED THE SECONDARY STEAM PRESSURE TO ALSO BE LOWER THAN DESIGN. TO ACHIEVE DESIGN T-AVG (FOR THE BULK RCS) AND SECONDARY STEAM PRESSURE, T-REF WAS RAISED 1.7 DEGREES F. THE NSSS VENDOR, WHO WAS AWARE OF THE LOOP TEMPERATURE IMBALANCE, WAS INFORMED BY LETTER IN SEPTEMBER 1976 THAT T-AVG WAS TO BE INCREASED BUT SPECIFICS ON HOW THIS WAS TO BE ACCOMPLISHED WERE NOT PROVIDED IN THE LETTER. NO NEGATIVE RESPONSE WAS RECEIVED FROM THE NSSS VENDOR ON THIS CHANGE, NOR DID THEY IDENTIFY A NEED FOR A NEW ANALYSIS. CORRECTIVE ACTIONS INCLUDE CHANGING THE 100 PERCENT POWER T- REF TO THE VALUE USED IN THE SAFETY ANALYSES.

[221] TROJAN DOCKET 50-344 LER 89-024
PERSONNEL ERROR IN CONNECTING A PROCESS EFFLUENT RADIATION MONITOR COULD HAVE PREVENTED AN AUTOMATIC TERMINATION OF RELEASE.
EVENT DATE: 092589 REPORT DATE: 102589 NSSS: WE TYPE: PWR

(NSIC 215700) ON SEPTEMBER 25, 1989, WITH THE PLANT IN MODE 5 (COLD SHUTDOWN) IT WAS DETERMINED THAT THE TRAIN OF THE HYDROGEN VENT SYSTEM (CS-9) DESIGNATED FOR USE TO VENT CONTAINMENT WAS NOT MONITORED BY A PROCESS EFFLUENT RADIATION MONITOR (PERM-1). IF A RADIOACTIVE RELEASE HAD OCCURRED WHICH EXCEEDED EFFLUENT RELEASE LIMITS, AUTOMATIC TERMINATION OF THE RELEASE COULD NOT BE ASSURED, UNLESS A SAFETY INJECTION SIGNAL OCCURRED IN CONJUNCTION WITH THE RELEASE. THE WRONG TRAIN OF CS-9 WAS MONITORED BY PERM-1 DUE TO THE CRAFTSMAN SIGNING OFF FOR INSTALLING A TEMPORARY MODIFICATION ON 'A' TRAIN WHEN IT WAS INSTALLED IN 'B' TRAIN AS A RESULT OF NOT BEING SURE OF THE TRAIN ON WHICH WORK WAS PERFORMED, AND USE OF VERBAL INFORMATION TO RESOLVE A QUESTION AS TO WHICH TRAIN OF CS-9 WAS MONITORED BY PERM-1. CORRECTIVE ACTION WAS TO PREVENT USE OF CS-9 FOR CONTAINMENT PRESSURE CONTROL UNTIL IT COULD BE CONFIRMED THAT PERM-1 WAS MONITORING 'B TRAIN. THE ERRORS INVOLVED IN THIS EVENT WERE DISCUSSED WITH THE INDIVIDUALS INVOLVED BY THEIR SUPERVISOR. THIS EVENT DID NOT HAVE ANY EFFECT ON PUBLIC HEALTH AND SAFETY AS THE CONDITION WAS DISCOVERED THE DAY AFTER THE MAINTENANCE WORK WAS COMPLETED AND NO ACTUAL RELEASE THAT EXCEEDED EFFLUENT LIMITS OCCURRED.

[222] TURKEY POINT 3 DOCKET 50-250 LER 89-014
COMPONENT COOLING WATER FLOW RATE TO THE EMERGENCY CONTAINMENT COOLERS BELOW DESIGN BASIS ACCIDENT REQUIREMENT DUE TO INADEQUATE ADMINISTRATIVE CONTROLS.
EVENT DATE: 091289 REPORT DATE: 101289 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)

(NSIC 215532) ON 9/12/89, WITH UNITS 3 AND 4 IN MODE 1 AT 100%, THE COMPONENT COOLING WATER (CCW) SYSTEM ENGINEER (NON-LICENSED UTILITY PERSONNEL) DISCOVERED THAT THE MECHANICAL STOPS FOR THE DISCHARGE VALVES FROM THE EMERGENCY CONTAINMENT COOLERS (ECC) TO THE CCW SYSTEM WERE IN POSITIONS THAT MAY NOT ENSURE THAT EACH ECC WOULD RECEIVE THE MINIMUM DESIGN CCW FLOWRATE DURING POST-ACCIDENT CONDITIONS. AN ANALYSIS IS BEING PERFORMED TO DETERMINE THE IMPACT THIS

CONDITION MAY HAVE (IF ANY) ON THE CONTAINMENT TEMPERATURE AND PRESSURE PROFILES. A CAUSE OF THE EVENT WAS INADEQUATE ADMINISTRATIVE CONTROLS. THE MECHANICAL STOPS OF THE SUBJECT VALVES WERE RE-ADJUSTED TO THE SETTINGS PREVIOUSLY DETERMINED BY THE APPROPRIATE SPECIAL TESTS. INFORMATION TAGS WILL BE PLACED ON THE SUBJECT VALVES IDENTIFYING THE BASIS FOR THROTTLING THE VALVES AND THE CORRECT SETTINGS FOR THE MECHANICAL STOPS.

[223] TURKEY POINT 4 DOCKET 50-251 LER 85-002 REV 01
UPDATE ON LOSS OF EMERGENCY DIESEL GENERATOR DUE TO A SHORT IN DIESEL LOCAL PANEL LIGHT SOCKET.
EVENT DATE: 012985 REPORT DATE: 101989 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 3 (PWR)
VENDOR: FAIRBANKS CO, THE
SCHOONMAKER, A. G.

(NSIC 215674) ON JANUARY 29, 1985, WHILE UNIT 3 WAS AT 100% POWER AND UNIT 4 WAS COOLING DOWN FROM HOT SHUTDOWN, A MALFUNCTION CAUSED EMERGENCY DIESEL GENERATOR A (EDG A) TO BE OUT OF SERVICE (OOS) SO VITAL BUSSES 3A AND 4A DID NOT HAVE ACCESS TO ON-SITE EMERGENCY POWER. IN ADDITION, A PREVIOUS MALFUNCTION ON JANUARY 16, 1985, CAUSED VITAL BUS 4B TO NOT HAVE ACCESS TO EMERGENCY POWER FROM EDG B. THERE ARE A TOTAL OF FOUR VITAL BUSSES FOR BOTH UNITS 3 AND 4 BUT ONLY VITAL BUS 3B HAD ON-SITE EMERGENCY POWER AVAILABLE. SIMILAR OCCURRENCES: LER 250-84-036. DURING A ROUTINE CHECK, IT WAS NOTED AND REPORTED THAT LAMP INDICATORS ON THE LOCAL EDG A CONTROL PANEL WERE DARK. A SHORT IN A DIESEL LOCAL PANEL LIGHT SOCKET CAUSED THE LOSS OF CONTROL POWER TO EDG A AND IT WAS DECLARED OUT OF SERVICE (OOS) TO BOTH UNITS. THE LOSS OF LOCAL CONTROL POWER WOULD PREVENT AUTOMATIC DIESEL START ON DEMAND BUT THE DIESEL COULD HAVE BEEN STARTED LOCALLY FROM THE CONTROL PANEL THROUGHOUT THE EVENT. ALSO THE PREVIOUS AND INDEPENDENT MALFUNCTION OF BREAKER 4AB21 HAD PUT IT OOS TO UNIT 4. ON LOSS OF OFF-SITE POWER, BREAKER 4AB21 CONNECTS VITAL BUS 4B TO EMERGENCY POWER FROM EDG B.

[224] TURKEY POINT 4 DOCKET 50-251 LER 89-003 REV 01
UPDATE ON REACTOR TRIP WHILE PERFORMING STEAM GENERATOR PROTECTION SET III CHANNEL TEST DUE TO INADEQUATE ADMINISTRATIVE CONTROLS.
EVENT DATE: 050589 REPORT DATE: 101989 NSSS: WE TYPE: PWR

(NSIC 215631) ON 5/5/89, AT 0152, WITH UNIT 4 IN HOT STANDBY, AND DURING ROD DROP TESTING, AN RPS ACTUATION OCCURRED WHILE PERFORMING PROCEDURE 4-SMI-071.4. THE REACTOR TRIPPED WHEN I&C PERSONNEL (NON-LICENSED UTILITY PERSONNEL) PLACED BISTABLE BS-4-446-1 IN THE TEST POSITION IN A ACCORDANCE WITH PROCEDURE 4-SMI-071.4. THIS SIMULATED A REACTOR POWER GREATER THAN 10%, ENABLING THE LOW POWER PERMISSIVE'S REACTOR TRIPS. AN INVESTIGATION DETERMINED THAT REACTOR TRIP LOGIC WAS COMPLETED BY A TURBINE TRIP SIGNAL GENERATED BY INDICATION OF CLOSED TURBINE STOP VALVES. ALTHOUGH THE TURBINE STOP VALVES (TSV) WERE PHYSICALLY VERIFIED TO BE IN OPEN POSITION, THE RPS INDICATED THEY WERE CLOSED DUE TO PRESENCE OF LIFTED LEADS IN THE TSVS POSITION SENSING CIRCUITRY. THE EVENT RESPONSE TEAM HAS IDENTIFIED THE PHYSICAL ROOT CAUSE AS THE PRESENCE OF LIFTED LEADS IN THE TSV'S. POSITION SENSING CIRCUITRY DUE TO INADEQUATE ADMINISTRATIVE CONTROLS. A CONTRIBUTING FACTOR WAS DETERMINED TO BE THAT THE SEQUENCE OF EVENTS GENERAL ALARM SUMMARY DID NOT IDENTIFY THE TSVS IN THE "ALARM CONDITION." THE SUBJECT LEADS WERE LANDED. A NEW ADMINISTRATIVE SITE PROCEDURE WAS DEVELOPED TO IMPROVE CONTROL OF PROCESS SHEETS AND INSTALLATION LISTS. THE GENERAL ALARM SUMMARY SOFTWARE HAS BEEN MODIFIED TO PREVENT LOSS OF ALARM STATUS.

[225] TURKEY POINT 4 DOCKET 50-251 LER 89-010
TURBINE RUNBACK DUE TO AN INADEQUATE PROCEDURE RESULTING IN A FALSE NUCLEAR INSTRUMENTATION SYSTEM ROD DROP SIGNAL.
EVENT DATE: 090689 REPORT DATE: 100689 NSSS: WE TYPE: PWR

(NSIC 215533) ON 9/6/89, AT APPROXIMATELY 0846, WITH UNIT 4 IN MODE 1, A NUCLEAR INSTRUMENTATION SYSTEM (NIS) ROD DROP SIGNAL WAS RECEIVED RESULTING IN A TURBINE RUNBACK. THE NIS SIGNAL WAS RECEIVED WHILE PERFORMING A FAST LOAD REDUCTION FROM 100 PERCENT POWER IN ACCORDANCE WITH PROCEDURE 4-ONOP-100, "FAST LOAD REDUCTION." THIS LOAD REDUCTION WAS BEING PERFORMED FOLLOWING RECEIPT OF HIGH CONDUCTIVITY READINGS INDICATING THAT AN UNKNOWN NUMBER OF CONDENSER TUBES WERE LEAKING. THE NIS ROD DROP SIGNAL WAS IN RESPONSE TO THE RATE OF REDUCTION IN REACTOR POWER REACHING THE NIS ROD DROP SETPOINT. THERE WAS NO ACTUAL DROPPED ROD EVENT. THE ROD POSITION INDICATION (RPI) ROD BOTTOM SIGNAL NORMALLY PROVIDES THE TURBINE RUNBACK SIGNAL BUT IN THIS CASE NIS WAS SELECTED SINCE AN RPI HAD PREVIOUSLY BEEN DECLARED INOPERABLE. THE ROOT CAUSE OF THIS EVENT IS THAT PROCEDURE 4-ONOP-100 DID NOT PROVIDE GUIDANCE REGARDING THE RATE OF REDUCTION IN REACTOR POWER TO PREVENT THE TURBINE RUNBACK. PROCEDURES 3/4-ONOP-100 WILL BE REVISED TO PROVIDE ADDITIONAL GUIDANCE TO MINIMIZE THE PROBABILITY OF A TURBINE RUNBACK BY 11/6/89. THE LEAKING CONDENSER TUBES WERE IDENTIFIED AND PLUGGED. THE INOPERABLE RPI, G-3, WAS LATER REPAIRED, FOLLOWING A UNIT SHUTDOWN ON 9/15 WHICH RESULTED IN THE UNIT BEING PLACED IN COLD SHUTDOWN.

[226]   TURKEY POINT 4   DOCKET 50-251   LER 89-011
TURBINE STOP VALVE CLOSURE DUE TO AUTO STOP OIL LINE LEAK RESULTED IN A MANUAL REACTOR TRIP AND A MANUAL SAFETY INJECTION.
EVENT DATE: 091589   REPORT DATE: 101289   NSSS: WE   TYPE: PWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)
        HAGAN CONTROLS
        UNITED ELECTRIC CONTROLS COMPANY

(NSIC 215534) AT 0426, ON 9/15/89, WITH UNIT 4 OPERATING AT 100% POWER, A MANUAL REACTOR TRIP WAS INITIATED. A HIGH PRESSURE (HP) TURBINE STOP VALVE AUTO STOP OIL LINE WELD LEAK IDENTIFIED AT 0045 LED TO CLOSURE OF THE STOP VALVE. WHEN THE CONTROL RODS FAILED TO INSERT IN AUTOMATIC OR MANUAL IN RESPONSE TO THE SUDDEN TURBINE POWER DECREASE, AS CALLED FOR BY A T-AVE/T-REF MISMATCH SIGNAL, A MANUAL REACTOR TRIP WAS INITIATED. A FAILURE OF THE 4C STEAM GENERATOR FEEDWATER CONTROL VALVE TO CLOSE DURING A SUBSEQUENT FEEDWATER ISOLATION (SLOW CLOSURE) SIGNAL RESULTED IN OVERFEED OF THE STEAM GENERATOR AND "SHRINK" OF THE REACTOR COOLANT SYSTEM INVENTORY. A MANUAL SAFETY INJECTION SIGNAL WAS INITIATED BY PROCEDURE BECAUSE PRESSURIZER LEVEL DROPPED BELOW 12%. THE AUTO STOP OIL LINE WELD FAILURE WAS DUE TO AN INADEQUACY IN THE REFUELING PREVENTIVE MAINTENANCE PROGRAM. THE AUTOMATIC ROD CONTROL SPEED SIGNAL OUTPUT SUMMATOR WAS OUT OF CALIBRATION. THE FEEDWATER CONTROL VALVE FAILURE TO CLOSE IS DUE TO AN INADEQUATE PROCEDURE USED DURING A RECENT MODIFICATION. THE THERMAL TRANSIENT EXPERIENCED DID NOT AFFECT THE STRUCTURAL INTEGRITY OF REACTOR COOLANT SYSTEM COMPONENTS. VARIOUS CORRECTIVE ACTIONS HAVE BEEN/WILL BE PERFORMED.

[227]   TURKEY POINT 4   DOCKET 50-251   LER 89-012
"A" LOOP OF WIDE RANGE CONTAINMENT WATER LEVEL INDICATION DE-ENERGIZED FOR A TIME PERIOD WHICH EXCEEDED ACTION STATEMENT 1 OF TECH SPEC TABLE 3.5-5, ITEM 9.
EVENT DATE: 092789   REPORT DATE: 102789   NSSS: WE   TYPE: PWR

(NSIC 215745) ON 9/27/89, 0930, AN INDIVIDUAL IN THE OPERATIONS SUPPORT GROUP (NON-LICENSED CONTRACTOR PERSONNEL) REPORTED FINDING ONE OF THE TWO WIDE RANGE CONTAINMENT WATER LEVEL INDICATORS (LI-4-6309A) DE-ENERGIZED. THIS CONDITION WAS REPORTED TO THE PLANT SUPERVISOR NUCLEAR (LICENSED UTILITY PERSONNEL) AND LI-4-6309A WAS RE-ENERGIZED. AT 1050, IT WAS NOTED THAT LI-4-6309A DID NOT DISPLAY PROPER EMERGENCY RESPONSE DATA ACQUISITION AND DISPLAY SYSTEM (ERDADS) RESPONSE FOLLOWING RE-ENERGIZATION. A PLANT WORK ORDER (PWO) WAS INITIATED TO CORRECT THIS CONDITION. THE "A" LOOP OF WIDE RANGE CONTAINMENT WATER LEVEL INDICATION WAS ESTABLISHED AS DE-ENERGIZED ON 7/12/89. THE LAST TIME THE SUBJECT LOOP WAS ESTABLISHED AS FUNCTIONAL WAS ON 10/5/88. THIS CONDITION PLACED UNIT IN A CONDITION OUTSIDE OF TECHNICAL SPECIFICATION 3.5, TABLE 3.5-5, ITEM 9, ACTION

STATEMENT 1. FPL BELIEVES THAT DE-ENERGIZATION OF LI-4-6309A AND DAMAGE TO THE CABLE CONNECTION TO LY-4-6309A WERE BOTH DUE TO NON-COGNITIVE ERROR BY UTILITY/CONTRACTOR PERSONNEL. BUMP COVERS WILL BE PLACED ON THE CONTAINMENT LEVEL RECEIVERS IN THE CONTAINMENT PENETRATION ROOMS TO PREVENT THE RECEIVERS FROM BEING INADVERTANTLY DE-ENERGIZED. A PERIODIC CHECK TO ENSURE THAT THE CONTAINMENT WATER LEVEL LOOPS ARE ENERGIZED HAS BEEN ADDED TO THE APPROPRIATE SURVEILLANCE PROCEDURE.

[228] VERMONT YANKEE DOCKET 50-271 LER 89-022
REACTOR BUILDING CLOSED COOLING WATER RETURN MOTOR OPERATED VALVE 70-117 NOT POWERED FROM EMERGENCY BUS AS REQUIRED BY FSAR.
EVENT DATE: 083189 REPORT DATE: 092889 NSSS: GE TYPE: BWR

(NSIC 215507) DURING A REVIEW OF THE REACTOR BUILDING CLOSED COOLING WATER (RBCCW) SYSTEM (EIIS=BI) AUGUST 31, 1989 (WITH THE REACTOR OPERATING AT 100% POWER), IT WAS IDENTIFIED THAT THE OUTBOARD ISOLATION VALVE FOR THE COOLING WATER RETURN LINE (V70-117) WAS FED FROM MOTOR CONTROL CENTER 7A (MCC 7A) (EIIS=EC) A NON EMERGENCY POWER BUS. THE PLANT FINAL SAFETY ANALYSIS REPORT REQUIRES THAT THIS VALVE BE FED FROM ONE OF THE AC EMERGENCY POWER BUSES (EIIS=EK). A JUSTIFICATION FOR CONTINUED OPERATION (JCO) WAS PREPARED AND PRESENTED TO THE PLANT OPERATIONS REVIEW COMMITTEE (PORC) ON AUGUST 31,1989, WHICH FOUND THAT ON A TEMPORARY BASIS THE PRESENT CONFIGURATION WAS ACCEPTABLE. A TEMPORARY MODIFICATION WAS IMPLEMENTED ON SEPTEMBER 15, 1989 TO PLACE THE POWER FEED FOR MOV 70-117 ON AN AC EMERGENCY BUS (MCC 8B)(EIIS=ED). THE EXACT ROOT CAUSE OF THIS EVENT IS UNKNOWN. THE SUSPECTED ROOT CAUSE APPEARS TO HAVE BEEN IN THE INTERPRETATION OF REQUIREMENTS IN TWO SEPARATE DESIGN SPECIFICATIONS.

[229] VERMONT YANKEE DOCKET 50-271 LER 89-023
FAILURE TO PERFORM DAILY INSTRUMENT CHECKS ON THE LOW PRESSURE COOLANT INJECTION SYSTEM CROSSTIE MONITOR DUE TO INTERPRETATION OF TECH SPEC REQUIREMENTS.
EVENT DATE: 091189 REPORT DATE: 101189 NSSS: GE TYPE: BWR

(NSIC 215540) VERMONT YANKEE TECH SPEC 4.2.A, TABLE 4.2.A, REQUIRES THAT AN INSTRUMENT CHECK OF THE INDICATION FOR THE RESIDUAL HEAT REMOVAL (RHR) (EIIS = BO) SYSTEM CROSSTIE VALVE, RHR-20, BE COMPLETED ONCE PER DAY. CONTRARY TO THIS REQUIREMENT, IT WAS DISCOVERED, ON 9/11/89, THAT THE REMOTE INDICATION TO THE VALVE HAD NOT BEEN AVAILABLE FROM 3/20/89, WHEN THE POWER SUPPLY BREAKER TO THE INDICATION WAS REMOVED, TO 9/12/89, WHEN THE BREAKER WAS REPLACED. THE ROOT CAUSE OF THIS EVENT IS THAT OPERATIONS PERSONNEL DID NOT RECOGNIZE THAT THE RHR-20 BREAKER WAS NEEDED TO MAINTAIN CONTINUITY FOR THE LPCI CROSSTIE MONITORING CIRCUIT WHICH IS REQUIRED BY TECH SPECS 4.2.A. ON 9/12/89 AT 1455, THE BREAKER WAS REPLACED AND INDICATION WAS RESTORED TO THE VALVE (RHR-20). A CHANGE WAS MADE TO OP 0150 "RESPONSIBILITIES AND AUTHORITIES OF OPERATIONS DEPARTMENT PERSONNEL" TO INCLUDE A SPECIFIC REFERENCE TO THE RHR-20 VALVE INDICATION. LONG TERM CORRECTIVE ACTIONS ARE CURRENTLY BEING PROCESSED.

[230] VERMONT YANKEE DOCKET 50-271 LER 89-024
MISSED RESIDUAL HEAT REMOVAL SYSTEM VALVE LEAKAGE SURVEILLANCE DUE TO INCOMPLETE PROCEDURE REVIEW.
EVENT DATE: 091389 REPORT DATE: 101389 NSSS: GE TYPE: BWR

(NSIC 215541) IN ORDER TO IMPLEMENT THE RECENTLY DEVELOPED REVISION 10 OF THE VERMONT YANKEE INSERVICE TESTING (IST) PROGRAM, VARIOUS SYSTEM SURVEILLANCE PROCEDURES WERE BEING REVISED TO INCORPORATE PROGRAM CHANGES. ON 9/13/89, WHILE REVIEWING THE RESIDUAL HEAT REMOVAL (RHR) (EIIS=BO) SYSTEM SURVEILLANCE PROCEDURE, IT WAS NOTED THAT A REQUIREMENT FROM PREVIOUS IST PROGRAM REVISION 9 CONCERNING LEAK TESTING OF RHR VALVE V10-18 HAD NOT BEEN INCORPORATED AND IMPLEMENTED. THE ROOT CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE INCOMPLETE

REVIEW AND OMISSION OF THE V10-18 LEAK TEST REQUIREMENT DURING RHR SYSTEM IMPLEMENTING PROCEDURE DEVELOPMENT. A REVIEW OF THE PRIMARY CONTAINMENT PENETRATION CORRESPONDING TO VALVE V10-18 WAS PERFORMED. IT IS CONCLUDED THAT THE INTEGRITY OF THIS PENETRATION IS ASSURED VIA CLOSURE OF DOWNSTREAM RHR VALVE V10-17 AND CONTINUOUS PENETRATION PRESSURE MONITORING. THE REQUIREMENT FOR LEAK TESTING VALVE V10-18 WILL BE INCORPORATED INTO THE CORRESPONDING IMPLEMENTING PROCEDURE PRIOR TO THE NEXT REFUELING OUTAGE. TO PRECLUDE RECURRENCE OF A SIMILAR EVENT, FUTURE CHANGES TO IMPLEMENTING PROCEDURES INVOLVING IST PROGRAM TESTING WILL BE ROUTED TO THE IST PROGRAM COORDINATOR FOR A COMPLETE REVIEW. PREVIOUS SIMILAR EVENTS HAVE BEEN REPORTED TO THE COMMISSION, IN THE LAST FIVE YEARS, AS LER 87-04 AND LER 89-20.

[231] VOGTLE 1 DOCKET 50-424 LER 87-082 REV 01
UPDATE ON FAILURE TO PERFORM RESPONSE TIME TEST RESULTS IN TECH SPEC VIOLATION.
EVENT DATE: 101787 REPORT DATE: 101389 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215628) ON 6-20-89, PLANT PERSONNEL WERE REVIEWING THE MAINTENANCE HISTORY ASSOCIATED WITH THE UNIT 1 REACTOR TRIP BREAKERS (RTB'S) WHEN IT WAS DISCOVERED THAT A RTB HAD BEEN SWAPPED-OUT WITHOUT PERFORMING A RESPONSE TIME TEST FOR THE BREAKER BEING INSTALLED. THIS SWAP-OUT OCCURRED ON 10-17-87 AND INVOLVED THE REPLACEMENT OF BREAKER SERIAL NO. 02YN072B-4, WHICH WAS INSTALLED IN THE MAIN RTB "B" CUBICLE, WITH BREAKER SERIAL NO. 860.759-1. THEREFORE, FROM 10-17-87 UNTIL THE REPLACEMENT OF BREAKER 860.759-1 ON 3-6-88, THE MINIMUM CHANNELS OPERABLE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.3.1 WERE NOT MET AS FAR AS RESPONSE TIME TESTING IS CONCERNED. THE ROOT CAUSE OF THIS EVENT IS CONSIDERED TO BE PROCEDURAL INADEQUACY IN THAT PROCEDURE 27765-C, "WESTINGHOUSE TYPE DS-416 CIRCUIT BREAKER MAINTENANCE", DID NOT CONTAIN INSTRUCTIONS FOR PERFORMING A RESPONSE TIME BENCH TEST. PROCEDURE 27765-C, IMPLEMENTS THE RTB PM PROGRAM, AND THIS PROCEDURE HAS BEEN CHANGED TO ADDRESS RESPONSE TIME TESTING REQUIREMENTS.

[232] VOGTLE 1 DOCKET 50-424 LER 88-035 REV 01
UPDATE ON CONTROL ROOM ISOLATION OCCURS DURING SURVEILLANCE TESTING.
EVENT DATE: 111388 REPORT DATE: 103189 NSSS: WE TYPE: PWR

(NSIC 215776) ON 11/13/88, PLANT PERSONNEL WERE CONDUCTING TECHNICAL SPECIFICATION (T. S.) SURVEILLANCE TESTING PER PROCEDURE 14710-1, "REMOTE SHUTDOWN PANEL TRANSFER SWITCH AND CONTROL CIRCUIT 18 MONTH SURVEILLANCE TEST". WHILE RESETTING THE TRAIN A LOAD SEQUENCER, A MOMENTARY LOSS OF POWER TO RADIATION MONITOR 1RE-12116 RESULTED IN A CONTROL ROOM ISOLATION (CRI) ACTUATION AT 1230 CST. THE TRAIN B ESF CHILLER AND CONTROL ROOM HVAC FILTER FAN ACTUATED BUT TRAIN A ESF COMPONENTS WERE OUT OF SERVICE FOR THE TEST AND DID NOT ACTUATE. CONTROL ROOM OPERATORS VERIFIED THAT NO ABNORMAL RADIATION EXISTED AND RESET THE CRI SIGNAL AT 1435 CST. INVESTIGATION INDICATES THE CAUSE OF THE CRI WAS LOSS OF POWER TO RADIATION MONITOR 1RE-12116. THE LOSS OF POWER OCCURRED WHILE RESETTING THE SEQUENCER. RESETTING THE SEQUENCER CAUSED A MOMENTARY, LARGE CURRENT INRUSH. THE CURRENT INRUSH ACTIVATED A "ZIP" CIRCUIT IN AN INVERTER WHICH SHUT DOWN THE INVERTER AND INTERRUPTED POWER TO THE DISTRIBUTION PANEL THAT SUPPLIES THE SEQUENCER AND THE RADIATION MONITOR. CORRECTIVE ACTION INCLUDES RAISING THE SETPOINT WHICH ACTIVATES THE ZIP CIRCUIT DURING THE NEXT REFUELING OUTAGE AND PERFORMING SURVEILLANCE TESTING OF THE SEQUENCER WHICH WILL CONFIRM PROPER SYSTEM OPERATION.

[233] VOGTLE 1 DOCKET 50-424 LER 89-018
REACTOR TRIP FOLLOWING SPURIOUS CLOSURE OF MSIV DUE TO FUSE FAILURE.
EVENT DATE: 100289 REPORT DATE: 103089 NSSS: WE TYPE: PWR
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)
MERCOID CORP.

NAMCO CONTROLS
ROCKWELL-INTERNATIONAL
WESTINGHOUSE ELECTRIC CORP.

(NSIC 215754) ON 10-2-89, AT APPROXIMATELY 0136 CDT, THE NO. 1 STEAM GENERATOR (SG) TRAIN "A" MAIN STEAM ISOLATION VALVE (MSIV) FAILED CLOSED. AT 0137 CDT, AN AUTOMATIC REACTOR TRIP OCCURRED DUE TO SG NO. 1 REACHING ITS LOW-LOW WATER LEVEL SETPOINT. A TURBINE TRIP, MAIN FEEDWATER ISOLATION AND AUXILIARY FEEDWATER ACTUATION OCCURRED AS DESIGNED FOLLOWING THE TRIP. THE MSIV CLOSED DUE TO A BLOWN FUSE IN THE CONTROL LOGIC POWER SUPPLY. INVESTIGATIONS REVEALED THAT GROUNDING PROBLEMS EXISTED IN THE 125VDC CONTROL POWER DISTRIBUTION PANEL AND INDICATED THAT A GROUND COULD HAVE EXISTED IN A MSIV LIMIT SWITCH WHICH SHOWED SIGNS OF INTERNAL MOISTURE RELATED DETERIORATION AND ARCING. THE COMBINATION OF THE GROUNDS IN THE CONTROL POWER DISTRIBUTION PANEL AND THE SUSPECT GROUND IN THE MSIV LIMIT SWITCH LIKELY CAUSED THE FUSE TO BLOW. THE GROUNDING PROBLEMS WERE CORRECTED AND THE MSIV LIMIT SWITCH AND FUSE REPLACED. CORRECTIVE ACTION TO PREVENT RECURRENCE INCLUDES COMPLETING A PREVIOUSLY IDENTIFIED TASK OF SEALING THE MSIV LIMIT SWITCHES DURING THE NEXT REFUELING OUTAGE TO PREVENT WATER INTRUSION.

[234] WATERFORD 3 DOCKET 50-382 LER 89-018
FAILURE TO DECLARE MAIN STEAM SAFETY VALVE INDETERMINATE.
EVENT DATE: 092189 REPORT DATE: 102389 NSSS: CE TYPE: PWR

(NSIC 215706) ON SEPTEMBER 21, 1989, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% REACTOR POWER. AT 1241 HOURS, WHILE TESTING A MAIN STEAM SAFETY VALVE, THE LIFT PRESSURE WAS FOUND TO BE BELOW THE TECHNICAL SPECIFICATION (TS) ALLOWABLE VALUE. BECAUSE OF ABERRATIONS IN INDICATED MAIN STEAM PRESSURE WHEN REPOSITIONING A GAUGE THAT WAS IMPROPERLY INSTALLED, THE SURVEILLANCE WAS INVALIDATED. TEST EQUIPMENT (PRESSURE GAUGE AND CHART RECORDER) WERE CHECKED FOR PROPER CALIBRATION AND OPERATION BY 2000 HOURS. THE SURVEILLANCE TEST WAS PERFORMED AGAIN AT 2037 HOURS. THE VALVE WAS FOUND TO LIFT AT A HIGHER PRESSURE BUT STILL BELOW THE TS LIMIT. THE VALVE SETPOINT WAS OUT OF TOLERANCE LOW, IN EXCESS OF THE FOUR HOUR TS ACTION REQUIREMENT. THIS EVENT IS THEREFORE REPORTABLE AS A CONDITION PROHIBITED BY TS. ALTHOUGH EVENT HAS IDENTIFIED AREAS THAT REQUIRE IMPROVEMENTS, THERE WAS A GOOD FAITH EFFORT BY ALL INVOLVED PERSONNEL TO MEET THE TS REQUIREMENT BY COMPLETING THE TEST PROMPTLY WHILE INSURING ACCURATE TEST RESULTS. THE DESIGN BASIS FOR MAIN STEAM SAFETY VALVES (OVERPRESSURE PROTECTION) WAS MAINTAINED THROUGHOUT THIS EVENT. THEREFORE, THIS EVENT DID NOT RESULT IN AN INCREASED RISK TO THE HEALTH AND SAFETY OF THE PUBLIC OR PLANT PERSONNEL.

[235] WOLF CREEK 1 DOCKET 50-482 LER 89-018
FAILURE TO RECOGNIZE THE NEED TO TAKE ACTION TO PLACE DAMPERS IN SAFEGUARDS POSITION MAY HAVE RESULTED IN INOPERABILITY OF BOTH EMERGENCY EXHAUST SYSTEMS.
EVENT DATE: 082889 REPORT DATE: 092789 NSSS: WE TYPE: PWR

(NSIC 215596) ON 8/28/89, AT APPROXIMATELY 1026 CDT, IT WAS DISCOVERED THAT DAMPERS GLD036 AND GLD037, THE MOTOR OPERATED SUPPLY AIR AND ISOLATION DAMPERS TO THE NON-RADIOACTIVE TUNNEL IN THE AUX. BLDG. HEATING, VENTILATION, AND AIR CONDITIONING (HVAC) SYSTEM, WERE REMOVED FROM SERVICE IN THE OPEN POSITION. IT IS BELIEVED THAT THIS CONDITION MAY HAVE CAUSED BOTH EMERGENCY EXHAUST SYSTEMS TO BECOME INOPERABLE, A CONDITION PROHIBITED BY TECH SPECS, AND MAY HAVE PREVENTED THE SYSTEM FROM FULFILLING ITS SAFETY FUNCTION WHICH IS TO CONTROL THE RELEASE OF RADIOACTIVE MATERIAL. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO FAILURE TO RECOGNIZE THE NEED TO TAKE ACTION TO PLACE DAMPERS GLD036 AND GLD037 IN THEIR SAFEGUARDS POSITION PRIOR TO REMOVING THE DAMPERS' CONTROL POWER FUSES. TO PREVENT RECURRENCE, OPERATIONS MANAGEMENT HAS ISSUED A LETTER WHICH INCLUDES A LESSON PLAN ON TAGGING VENTILATION DAMPERS TO THE SHIFT SUPERVISORS AND OTHER

LICENSED PERSONNEL WHO MAY BE INVOLVED WITH TAGGING EVOLUTIONS. THE SHIFT SUPERVISORS ARE REQUIRED TO INSTRUCT ANY OF THEIR CREW MEMBERS WHO MAY BE INVOLVED IN CLEARANCE ORDERS ON THE SPECIFICS OF THE LESSONS PLAN BY OCTOBER 31, 1989.

[236] WOLF CREEK 1 DOCKET 50-482 LER 89-019
DISCOVERY OF CENTRIFUGAL CHARGING PUMP INOPERABILITY IN ONE TRAIN WHILE THE OTHER EQUIPMENT IN TRAIN WAS OUT OF SERVICE.
EVENT DATE: 091989 REPORT DATE: 101889 NSSS: WE TYPE: PWR
VENDOR: POTTER & BRUMFIELD

(NSIC 215659) ON 9/15/89, DURING PERFORMANCE OF A SURVEILLANCE TEST PROCEDURE, THE "A" TRAIN CENTRIFUGAL CHARGING PUMP (CCP) MINIMUM FLOW VALVE, BG HV-8110, DID NOT STROKE PROPERLY IN RESPONSE TO A SIMULATED FLOW SIGNAL. THE SURVEILLANCE TEST WAS SUSPENDED PENDING A REVIEW OF THE PROCEDURE METHODOLOGY AS THE PROCEDURE HAD RECENTLY BEEN REVISED. ON 9/19/89, AT 0420 CDT, A PREVENTATIVE MAINTENANCE OUTAGE BEGAN FOR SEVERAL "B" TRAIN COMPONENTS, INCLUDING THE EMERGENCY POWER SOURCE FOR THE "B" TRAIN CCP. AT APPROX. 1200 CDT ON 9/19, IT WAS DETERMINED THAT NO PROCEDURAL PROBLEM EXISTED, AND THAT AN EQUIPMENT PROBLEM ASSOCIATED WITH BG HV-8110 HAD LIKELY CAUSED THE SEPTEMBER 15 TEST DEFICIENCY. AT APPROX. 1755 CDT, IT WAS DETERMINED THAT WITH BG HV-8110 INOPERABLE, THE "B" TRAIN CCP COULD NOT BE CONSIDERED OPERABLE. BECAUSE THE "B" TRAIN CCP WAS ALSO INOPERABLE, THE UNIT ENTERED TECH SPEC 3.0.3. AT 1940 CDT, THE "A" TRAIN CCP WAS RESTORED TO SERVICE AFTER REPLACEMENT OF A FAULTY SLAVE RELAY, AND TECH SPEC 3.0.3 WAS EXITED AT THAT TIME. THIS EVENT OCCURRED AS A RESULT OF REACHING AN INCORRECT CONCLUSION WHEN INITIALLY EVALUATING THE TEST DEFICIENCY. THE STATION'S PHILOSOPHY OF A CONSERVATIVE APPROACH TO PROBLEM SOLVING WILL BE RE-EMPHASIZED TO LICENSED PERSONNEL. THE RESPONSIBILITIES OF TEST PERFORMERS HAS BEEN REITERATED TO INSTRUMENTATION AND CONTROL PERSONNEL.

[237] WPPSS 2 DOCKET 50-397 LER 89-030
HIGH PRESSURE CORE SPRAY SYSTEM INOPERABLE CAUSED BY SUPPRESSION POOL PUMP SUCTION VALVE FAILURE DUE TO MOTOR OPERATOR MANUFACTURING ERROR.
EVENT DATE: 072889 REPORT DATE: 082589 NSSS: GE TYPE: BWR
VENDOR: LIMITORQUE CORP.

(NSIC 215145) ON 7/28/89, IN ACCORDANCE WITH THE SUPPLY SYSTEM RESPONSE TO NOTICE OF VIOLATION "C" OF INSPECTION REPORT 89-13, THE REPORTABILITY DETERMINATION OF A EVENT WHICH OCCURRED ON 2/10/89, WAS CHANGED FROM NONREPORTABLE TO REPORTABLE PER THE REQUIREMENTS OF 10 CFR 50.73. ON 2/10/89 AT 051 HRS THE HIGH PRESSURE CORE SPRAY SUCTION VALVE FROM THE SUPPRESSION POOL HPCS-V-15 FAILED TO FULLY OPEN DURING PERFORMANCE OF A HPCS SURVEILLANCE PROCEDURE BECAUSE OF THE FAILURE OF THE ASSOCIATED LIMITORQUE MOTOR OPERATOR. AT 0510 HRS ON 2/10/89 THE TS ACTION STATEMENT WERE ENTERED FOR EMERGENCY CORE COOLING AND PRIMARY CONTAINMENT ISOLATION. THE HPCS-V-15 VALVE WAS MANUALLY CLOSED AT 0900 HRS ON 2/10/89. AN UNUSUAL EVENT (UE) WAS DECLARED BECAUSE CLOSURE OF THE VALVE COULD NOT BE VERIFIED TO SATISFY TECH SPEC ISOLATABLE PENETRATION REQUIREMENTS. THE CLOSURE WAS VERIFIED BY LEAK TESTING AND THE UE WAS TERMINATED A 1225 HOURS, 2/10/69. THE MOTOR OPERATOR WAS THEN REMOVED REBUILT REINSTALLED, TESTED FOR OPERABILITY, AND PLACED BACK INTO OPERATION ON 2/1-12/89. THE TECH SPEC ACTION STATEMENTS WERE EXITED AT 213 HRS ON FEBRUARY 12 WHEN THE MOTOR OPERATOR WAS DECLARED OPERABLE. THE ROOT CAUSE OF THE INOPERABLE OF THE HPCS-V-15 VALVE IS THAT THE MOTOR OPERATOR (HPCS-MO-15) WAS NOT MADE PER DESIGN BY THE MANUFACTURER.

[238] WPPSS 2 DOCKET 50-397 LER 89-036
INADEQUATE AVERAGE POWER RANGE MONITOR TECH SPEC SURVEILLANCE.
EVENT DATE: 090589 REPORT DATE: 100389 NSSS: GE TYPE: BWR

(NSIC 215574) AT 1215 HOURS, ON 9/5/89, THE PLANT OPERATING COMMITTEE (POC) REVIEWED AN INCONSISTENCY DOCUMENTED BY THE AVERAGE POWER RANGE MONITOR (APRM) PLANT SYSTEM ENGINEER BETWEEN THE WNP-2 PLANT TECH SPEC (TABLE 3.3.1-2, ITEM 2B) AND THE IMPLEMENTING PLANT SURVEILLANCE PROCEDURES (PPMS 7.4.3.1.3.5, .6, .7, AND .8). THE TECH SPEC REQUIRES THAT THE REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME FOR THE APRM FLOW BIASED SIMULATED THERMAL POWER UPSCALE FUNCTION BE CONFIRMED TO BE LESS THAN OR EQUAL TO 0.09 SECONDS NOT INCLUDING THE SIMULATED THERMAL POWER TIME CONSTANT OF 6 +/- 1 SECS. THE WNP-2 SURVEILLANCE PROCEDURES DID NOT PROVIDE FOR INDEPENDENT MEASUREMENT OF THESE TWO VALUES. THE PLANT SURVEILLANCE PROCEDURE REQUIRED THE MEASURED TIME RESPONSE TO BE LESS THAN OR EQUAL TO 7.09 SECS. THE POC CONCLUDED THE PLANT WAS NOT IN STRICT COMPLIANCE WITH THE TECH SPEC AND DIRECTED THAT THE REQUIRED TECH SPEC ACTION BE INITIATED TO PLACE THE PLANT IN AT LEAST STARTUP BY 1830 HOURS. AT 1307 HOURS, A WRITTEN REQUEST WAS MADE TO THE NRC FOR TEMPORARY RELIEF FROM THE TECH SPEC REQUIREMENTS. THIS RELIEF WAS GRANTED AT 1620 HOURS. FURTHER REVIEW OF THIS ISSUE RESULTED IN A FORMAL REQUEST FOR AN AMENDMENT TO THE TECH SPECS WHICH WAS RECEIVED ON 9/8/89. THE ROOT CAUSE OF THIS EVENT WAS LESS THAN ADEQUATE SURVEILLANCE PROCEDURES ON RESPONSE TIME TESTING OF THE APRM SYSTEM.

[239] WPPSS 2 DOCKET 50-397 LER 89-038
INADEQUATE PRIMARY CONTAINMENT INTEGRITY VERIFICATION.
EVENT DATE: 091389 REPORT DATE: 100689 NSSS: GE TYPE: BWR

(NSIC 215575) ON SEPTEMBER 13, 1989, A REPORTABILITY EVALUATION WAS APPROVED BY THE PLANT TECHNICAL MANAGER WHICH DIRECTED THAT AN EVENT WHICH BEGAN ON JANUARY 21, 1989, BE REPORTED PER 10CFR50.73. ON THE LATER DATE, PLANT EQUIPMENT OPERATORS DISCOVERED TWO SMALL 3/8 INCH VALVES WHICH SHOULD HAVE BEEN INCLUDED ON THE PRIMARY CONTAINMENT INTEGRITY VERIFICATION SURVEILLANCE. THE IMMEDIATE CORRECTIVE ACTION PLACED THESE VALVES ON THE SURVEILLANCE TO ALLOW VERIFICATION OF THEIR CLOSED CONDITION TO OCCUR ON A MONTHLY FREQUENCY. THE PLANT MANAGER ALSO DIRECTED THAT THE CONTAINMENT INTEGRITY PROCEDURE BE COMPARED WITH THE LOCAL LEAK RATE TESTING PROCEDURE TO IDENTIFY ANY OTHER MISSING VALVES. FOUR ADDITIONAL 1/2 INCH VALVES WERE DISCOVERED DURING THAT REVIEW. THE ROOT CAUSE OF THIS EVENT WAS LESS THAN ADEQUATE PROCEDURES THAT DID NOT IDENTIFY ALL THE CONTAINMENT ITEMS THAT REQUIRE VERIFICATION. FURTHER CORRECTIVE ACTION WILL INCLUDE A PHYSICAL WALK-DOWN OF ALL CONTAINMENT PENETRATIONS TO PROVIDE ASSURANCE THAT ALL ITEMS ARE NOW CONTAINED ON THE CHECKLIST. THIS EVENT POSED NO THREAT TO THE HEALTH AND SAFETY OF EITHER THE PUBLIC OR PLANT PERSONNEL.

[240] WPPSS 2 DOCKET 50-397 LER 89-039
INADEQUATE ELECTRICAL SEPARATION AND NON-FAILSAFE DESIGN OF REACTOR BUILDING EXHAUST AIR RADIATION MONITORING SYSTEM.
EVENT DATE: 091489 REPORT DATE: 101389 NSSS: GE TYPE: BWR

(NSIC 215653) ON SEPTEMBER 14, 1989 AN ELECTRICAL DESIGN ENGINEER IDENTIFIED THREE DISCREPANCIES WITH THE CURRENT CONFIGURATION OF THE REACTOR BUILDING EXHAUST AIR (REA) RADIATION MONITORING SYSTEM THAT DO NOT SATISFY THE DESIGN BASIS REQUIREMENTS. THEY CONSISTED OF INADEQUATE ELECTRICAL SEPARATION IN CONTROL ROOM CABINETS, ROUTING OF FAILSAFE CABLE IN NON- FAILSAFE RACEWAYS OUTSIDE OF THE POWER GENERATION CONTROL COMPLEX (PGCC), AND A NON-FAILSAFE DESIGN RESPONSE OF THE RADIATION MONITORS TO INOPERATIVE/DOWNSCALE CONDITIONS. THESE CONDITIONS WERE DISCOVERED BY THE ENGINEER DURING PREPARATION OF A POWER SUPPLY MODIFICATION TO THE REA RADIATION MONITORING SYSTEM. THE IMMEDIATE CORRECTIVE ACTIONS INCLUDE: 1) THE FAILSAFE CIRCUITS ROUTED IN NONFAILSAFE RACEWAYS WERE PLACED ON AN HOURLY FIRE TOUR TO MINIMIZE THE PROBABILITY OF A FIRE THAT COULD CAUSE A CIRCUIT FAULT, AND 2) THE REA RADIATION MONITOR DOWNSCALE ANNUNCIATOR RESPONSE PROCEDURE WAS REVISED TO REQUIRE OPERATOR ACTION TO PLACE THE AFFECTED TRIP MONITOR (REA-RIS-609A, -609B, -609C, AND/OR -609D) IN A TRIPPED CONDITION UPON RECEIPT OF A VALID DOWNSCALE CONDITION. THE ROOT CAUSES OF THE REA

RADIATION MONITORING SYSTEM BEING OUTSIDE OF THE PLANT DESIGN BASIS INCLUDE: 1) EQUIPMENT DESIGNATION WAS LESS THAN ADEQUATE BECAUSE OF INCORRECT ASSIGNMENT OF FAILSAFE CIRCUITS TO ELECTRICAL DIVISIONS BY THE NSSS.

[241] WPPSS 2 DOCKET 50-397 LER 89-040
STANDBY GAS TREATMENT SYSTEM CAPABILITY NOT WITHIN LICENSE BASIS CONSIDERATION FOR SECONDARY CONTAINMENT PERFORMANCE UNDER CERTAIN CONDITIONS DUE TO DESIGN.
EVENT DATE: 091989 REPORT DATE: 101889 NSSS: GE TYPE: BWR

(NSIC 215687) ON SEPTEMBER 19, 1989 IT WAS DETERMINED BY ENGINEERING ANALYSIS THAT UNDER CERTAIN METEOROLOGICAL CONDITIONS (MODERATE WIND AND LOW TEMPERATURE), COINCIDENT WITH A DBA LOCA AND ASSUMED FAILURE OF ONE TRAIN OF THE STANDBY GAS TREATMENT (SGT) SYSTEM, A SITUATION WOULD BE CREATED THAT IS NOT WITHIN THE LICENSING BASIS CONSIDERATION SECONDARY CONTAINMENT PERFORMANCE. THE ENGINEERING ANALYSIS WAS PERFORMED AS A FURTHER CORRECTIVE ACTION FOR LER 88-023. THE WNP-2 FSAR STATES THAT THE SECONDARY CONTAINMENT WILL BE MAINTAINED AT MINIMUM DIFFERENTIAL PRESSURE OF -0.25" W.G. FOLLOWING A POSTULATED LOCA, AND THAT THIS DIFFERENTIAL WILL BE ESTABLISHED WITHIN TWO MINUTES FOLLOWING THE ACCIDENT. RECENT ANALYSIS, BASED UPON STANDBY GAS TREATMENT, SECONDARY CONTAINMENT, STANDBY SERVICE WATER AND WEATHER MODELING, SHOWS THAT DURING POST-LOCA, OR ADVERSE WEATHER, DIFFERENTIAL PRESSURE OF THE SECONDARY CONTAINMENT MAY NOT ALWAYS MEET THE FSAR COMMITMENTS. CERTAIN COMBINATIONS OF POST-LOCA SINGLE ACTIVE FAILURES AND WINTER CONDITIONS ADVERSELY AFFECT SECONDARY CONTAINMENT AND, AS A RESULT, INCREASE SECONDARY CONTAINMENT LEAKAGE.

[242] WPPSS 4 DOCKET 50-513 LER 89-017
STEAM GENERATOR EDDY CURRENT TESTING RESULTS CLASSIFIED AS CATEGORY C-3.
EVENT DATE: 092589 REPORT DATE: 102389 NSSS: BW TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215718) SCHEDULED STEAM GENERATOR EDDY CURRENT TESTING (ECT) HAS BEEN CONDUCTED DURING THE CURRENT REFUELING OUTAGE IN ACCORDANCE WITH ASME SECTION XI AND TECHNICAL SPECIFICATION 4.10.1, "INSERVICE INSPECTION OF STEAM GENERATOR TUBES". ON SEPTEMBER 25, 1989, AT APPROXIMATELY 1330 HOURS, WITH THE PLANT IN MODE 6, IT WAS DETERMINED THAT RESOLVED ECT DATA PLACED STEAM GENERATOR #2 INTO THE C-3 CATEGORY. AS PLANNED, 100% OF STEAM GENERATOR #2 TUBES (3493 TUBES) WILL BE INSPECTED. SUBSEQUENTLY, ON OCTOBER 4, 1989 RESOLVED ECT RESULTS FOR NO. 4 STEAM GENERATOR PLACED IT IN CATEGORY C-3, THUS REQUIRING EXPANSION OF THE ECT PROGRAM TO 100 PERCENT OF THE TUBES IN ALL FOUR STEAM GENERATORS AS ORIGINALLY PLANNED. THE CAUSE OF THE TUBE DEGRADATION HAS NOT BEEN DETERMINED. ADDITIONAL INFORMATION ON ALL FOUR STEAM GENERATORS WILL BE PROVIDED IN A SUPPLEMENTAL REPORT AFTER ALL INSPECTIONS AND EVALUATIONS ARE COMPLETED. ALL TUBES WITH DEGRADATION GREATER THAN OR EQUAL TO THE TECHNICAL SPECIFICATION LIMITS WILL BE PLUGGED. THIS EVENT IS REPORTABLE PER 10CFR50.73(A)(2)(I)(B) SINCE IT INVOLVES A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[243] YANKEE ROWE DOCKET 50-029 LER 86-008 REV 01
UPDATE ON NO. 1 MAIN COOLANT PUMP SUCTION VALVE STEM FAILURE.
EVENT DATE: 061986 REPORT DATE: 092789 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELEC CORP.-NUCLEAR ENERGY SYS

(NSIC 215498) ON 6/19/86, WITH THE PLANT COOLING DOWN IN MODE 3 AT 0% POWER NO. 1 MAIN COOLANT PUMP FLOW INDICATION WAS ABNORMALLY LOW. THIS OCCURRED WHILE THE MAIN COOLANT PUMPS WERE BEING CYCLED TO MAINTAIN COOLANT TEMPERATURE EQUILIBRIUM IN PREPARATION TO ENTER MODE 4. THE PUMP WAS IMMEDIATELY SHUT DOWN. INVESTIGATION DETERMINED THE CAUSE OF THE PROBLEM TO BE FAILURE OF THE VALVE STEM ON NO. 1 LOOP HOT LEG ISOLATION VALVE (MC-MOV-325), ALLOWING THE VALVE GATE TO DROP INTO A PARTIALLY CLOSED POSITION. THE FAILURE OCCURRED WHEN A CRACK, WHICH

ORIGINATED AT A SECTION CHANGE IN THE VALVE STEM, PROPAGATED THROUGH THE STEM DIAMETER. THE PLANT CONTINUED THE COOLDOWN AND WAS PLACED IN MODE 5. THE VALVE STEM WAS REPLACED IN KIND AND, AFTER COMPLETION OF UNRELATED REPAIRS AND MODIFICATIONS (SEE LERS 86-06 AND 86-09, RESPECTIVELY), THE PLANT WAS RETURNED TO MODE 1. THE NRC WAS NOTIFIED OF THE EVENT VIA ENS AT 1650 HOURS ON 6/23/86. LABORATORY ANALYSIS OF THE FAILED VALVE STEM INDICATE THE MOST LIKELY CAUSE OF FAILURE WERE TENSILE STRESS ON THE STEM IN THE VALVE-OPEN POSITION AND IMPURITIES IN STAGNANT WATER TRAPPED IN THE ANNULUS AROUND THE VALVE STEM IN THE BACKSEAT POSITION.

[244] YANKEE ROWE DOCKET 50-029 LER 89-014
TECH SPEC VIOLATION CONCERNING ENTRY INTO HIGH RADIATION AREA.
EVENT DATE: 092589 REPORT DATE: 102589 NSSS: WE TYPE: PWR

(NSIC 215722) ON 9/25/89, AT 0900 HOURS, WHILE IN MODE 1 AT 100% POWER, A PLANT WORKER ENTERED A HIGH RADIATION AREA (HRA) IN A PIPE TRENCH UNDER THE PRIMARY AUXILIARY BUILDING CUBICLE CORRIDOR. THE INDIVIDUAL ENTERED THE PIPE TRENCH TO PERFORM A PRE-WORK INSPECTION FOR INSULATION REMOVAL AND WAS NOT WEARING AN ALARMING DOSIMETER. AN INVESTIGATION WAS CONDUCTED AND THE DETERMINATION WAS MADE THAT THE CONDITIONS OF ENTRY DID NOT COMPLY WITH TECH SPEC 6.12.B. THE ROOT CAUSE OF THIS EVENT WAS ATTRIBUTED TO PERSONNEL ERROR IN THAT APPROVED RADIATION PROTECTION (RP) PROCEDURES WERE NOT FOLLOWED CORRECTLY. IMMEDIATE CORRECTIVE ACTION INVOLVED ORDERING THE WORKER TO EXIT THE HRA. THE WORKER WAS SUSPENDED FOR 3 DAYS WITHOUT PAY FOR FAILING TO COMPLY WITH RP PROGRAM REQUIREMENTS. THIS INDIVIDUAL HAS BEEN RESTRICTED FROM THE PLANT RADIATION CONTROL AREA PENDING COMPLETION OF AN RP TRAINING COURSE. ADDITIONAL EVALUATIONS REVEAL NO PROGRAMMATIC DEFICIENCIES IN THE GENERAL EMPLOYEE TRAINING AND RP PROGRAMS. THIS EVENT IS CONSIDERED AN ISOLATED OCCURRENCE; TO CONFIRM THIS, A RADIATION WORK PERMIT COMPLIANCE OBSERVATION PROGRAM HAS BEEN TEMPORARILY ESTABLISHED. UNLESS THIS OBSERVATION PROGRAM IDENTIFIES AN ADVERSE TREND, NO FURTHER CORRECTIVE ACTION IS DEEMED NECESSARY AT THIS TIME. THERE WAS NO ADVERSE EFFECT TO THE PUBLIC HEALTH OR SAFETY.

[245] ZION 1 DOCKET 50-295 LER 89-001 REV 01
UPDATE ON OBN SERVICE WATER AREA VENT FAN AIRCRAFT CRASH DAMPER FOUND FAILED OPEN DUE TO FAULTY VALVE.
EVENT DATE: 011289 REPORT DATE: 101689 NSSS: WE TYPE: PWR
VENDOR: ASCO VALVES

(NSIC 215666) ON 12/19/88 AT 2100 HOURS WITH UNIT 1 AT 60% POWER, AN OPERATING DEPARTMENT B-MAN ON HIS SHIFTLY ROUNDS NOTICED THE OBN SERVICE WATER AREA VENT FAN (UA) AIRCRAFT CRASH DAMPER WAS OPEN WITH THE FAN OFF. A SHIFT FOREMAN WAS NOTIFIED AND, AFTER SEVERAL UNSUCCESSFUL ATTEMPTS TO CLOSE THE DAMPER, A WORK REQUEST (#Z-76749) WAS ISSUED TO THE MECHANICAL MAINTENANCE (MM) DEPARTMENT TO REPAIR AND CLOSE THE DAMPER. THE SHIFT CONTROL ROOM ENGINEER (SCRE) THEN STARTED AN HOURLY FIREWATCH FOR AN INOPERABLE FIRE BARRIER. THE TECH SPECS REQUIRE THE DAMPER TO BE OPERABLE OR TO BE IN ITS ACCIDENT POSITION. THE DAMPER WAS NOT CLOSED UNTIL TWO DAYS LATER. THERE WAS MINIMAL SAFETY SIGNIFICANCE DUE TO THIS INCIDENT DUE TO THE SHORT TIME THE DAMPER WAS OPEN. THE APPARENT CAUSE WAS A STICKING CONTROL AIR VALVE THAT DID NOT RECLOSE THE DAMPER FOLLOWING A PLANNED BRIEF BUS OUTAGE. THE VALVE WAS REPLACED AND THE DAMPER RETURNED TO NORMAL SERVICE.

[246] ZION 1 DOCKET 50-295 LER 89-004 REV 01
UPDATE ON FORCED SHUTDOWN DUE TO INOPERABLE CONTROL ROD POSITION INDICATION DUE TO REGULATOR FAILURE.
EVENT DATE: 030889 REPORT DATE: 110389 NSSS: WE TYPE: PWR
VENDOR: SOLA ELECTRIC COMPANY

(NSIC 215790) ON 3/8/89, ALL UNIT 1 ROD POSITION INDICATORS DROPPED TO A READING OF APPROXIMATELY 220 STEPS. IT WAS FOUND THAT A LINE VOLTAGE REGULATOR, IN SERIES WITH THE POWER SOURCE FOR THE ROD POSITION INDICATION SYSTEM, HAD FAILED. ALL INDICATOR CHANNELS WERE DECLARED INOPERABLE, AND OPERATORS PREPARED TO RAMP THE UNIT DOWN AS REQUIRED BY TECH SPECS. BECAUSE OF THE UNCERTAINTY OF ACTUAL ROD POSITION, AND THE CONCERN FOR MAINTAINING ADEQUATE SHUTDOWN MARGIN, THE UNIT OPERATOR WAS INSTRUCTED TO RAMP DOWN IN POWER USING ONLY CHEMICAL SHIM. AT ONE POINT IN THE SHUTDOWN, DUE TO THE DIFFICULTIES INHERENT IN USING ONLY BORON TO CONTROL REACTIVITY, REACTOR POWER DROPPED BELOW TURBINE POWER ENOUGH TO CAUSE COOLANT PRESSURE TO FALL BELOW THE DNB OPERATING LIMIT DEFINED IN PLANT TECH SPECS. THE FAILED REGULATOR WAS REPLACED WITH A NEW ONE; THE OLD REGULATOR WAS RETURNED TO THE MANUFACTURER AND ANALYZED FOR THE CAUSE OF THE FAILURE. THIS WAS DETERMINED TO BE A FAILED CAPACITOR AND ZENER DIODE.

[247] ZION 1 DOCKET 50-295 LER 89-014
FIRE DOORS TO CABLE SPREADING ROOMS FOUND OPEN WITHOUT A FIREWATCH PRESENT DUE TO PERSONNEL ERROR.
EVENT DATE: 091289 REPORT DATE: 101289 NSSS: WE TYPE: PWR

(NSIC 215667) ON 9/12/89 AT 1115 HOURS, AN OPERATOR FOUND THE UNIT 1 INNER CABLE SPREADING ROOM FIRE DOOR PROPPED OPEN AND THE U-1 OUTER CABLE SPREADING ROOM FIRE DOOR AJAR WITHOUT A FIRE WATCH PRESENT. THE DOORS WERE CLOSED AND LOCKED UPON DISCOVERY. IT WAS DETERMINED THAT CONTRACTORS WORKING IN THE AREA HAD OPENED THE DOORS TO FACILITATE INSTALLATION OF CABLES. THE UNIT WAS IN MODE 5, COLD SHUTDOWN, AT THE START OF A REFUELING OUTAGE. THE SAFETY SIGNIFICANCE WAS MINIMAL BECAUSE THE UNIT WAS IN COLD SHUTDOWN AND THE FIRE DETECTORS IN BOTH CABLE SPREADING ROOMS WERE OPERABLE. THE HALON AND CARDOX SYSTEMS WERE ALSO OPERABLE THROUGHOUT THE EVENT. CONTRACTOR PERSONNEL AT THE FOREMAN AND CRAFT LEVEL ARE INSTRUCTED ON THE REQUIREMENTS AND ACTIONS NECESSARY TO BLOCK OPEN A FIRE DOOR IN A NEW PROGRAM CALLED NON-STATION PERSONNEL ORIENTATION. ENGINEERING AND CONSTRUCTION DEPARTMENT PROVIDES THE TRAINING.

[248] ZION 1 DOCKET 50-295 LER 89-015
LOSS OF CONTAINMENT CLOSURE DURING REFUELING OPERATIONS DUE TO INADEQUATE PROCEDURES.
EVENT DATE: 092189 REPORT DATE: 102389 NSSS: WE TYPE: PWR

(NSIC 215726) THIS SPECIAL REPORT IS BEING SUBMITTED BECAUSE THIS EVENT VIOLATES AN INTERPRETATION OF TECH SPECS OUTLINED IN AN ON-SITE-REVIEW, DATED 2/3/78, WHICH COMMITTED TO MEETING THE INTENT OF STANDARDIZED TECH SPECS REGARDING CONTAINMENT INTEGRITY DURING CORE ALTERATIONS. UNIT 1 WAS IN MODE 6 FOR REFUELING AND THE CORE WAS BEING OFFLOADED. AUX. FEEDWATER CHECK VALVES WERE DISASSEMBLED FOR MAINTENANCE. AT APPROX. 2030 ON 9/20/89 STEAM GENERATOR DRAINING WAS COMMENCED BY OPENING THE BLOWDOWN AND ATMOSPHERIC RELIEF VALVES. AT APPROXIMATELY 1330 ON 9/21/89 IT WAS REALIZED THAT THE OPEN CHECK VALVES THROUGH THE ATMOSPHERICS CONSTITUTED A DIRECT CONTAINMENT VENT PATH TO ATMOSPHERE. THE SHIFT WAS CALLED AND INSTRUCTED TO SHUT THE ATMOSPHERICS. THE CAUSE OF THE EVENT WAS A COMBINATION OF PROCEDURE DEFICIENCY AND IMPROPER PLANNING THAT ALLOWED THE CONCURRENT ACTIVITIES. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL DUE TO THE FACT THAT NO RADIOACTIVE RELEASE OCCURRED DURING THE TIME THE VENT PATH EXISTED, AND NO DRIVING FORCE EXISTED TO CAUSE EXFILTRATION OF THE ACTIVITY HAD THERE BEEN A RELEASE.

[249] ZION 1 DOCKET 50-295 LER 89-017
CLOSURE OF CONTAINMENT ISOLATION VALVE 1FCV-PR24A DUE TO UNKNOWN CAUSE.
EVENT DATE: 100489 REPORT DATE: 110389 NSSS: WE TYPE: PWR

(NSIC 215788) UNIT 1 WAS DEFUELED. SAFEGUARDS WAS DEENERGIZED. AT APPROXIMATELY

1330 ON 10/4/89, NUMEROUS ALARMS ASSOCIATED WITH STEAM GENERATOR (S/G) LEVEL AND FLOW INDICATION WERE RECEIVED. SIMULTANEOUSLY, THE UNIT 1 PROCESS COMPUTER WENT DOWN, AND AT APPROXIMATELY THE SAME TIME THE FOLLOWING INSTRUMENTS WERE OBSERVED AS BEING 'FAILED ON-SCALE' (FAILED ON-SCALE MEANS THEY SHOULD HAVE INDICATED ZERO, BUT WERE INSTEAD READING AN INTERMEDIATE POSITION): 1LI-537, 1LI-529, 1LI-549, 1LI-519, 1LI-502B, 1LI-548, 1FI-540, 1FI-520. ABOUT THREE MINUTES LATER, IT WAS NOTED THAT 1FCV-PR24A HAD CLOSED. 1FCV-PR24A IS ONE OF TWO SERIES CONTAINMENT ISOLATION VALVES THAT ARE ALSO SUCTION VALVES FOR THE CONTAINMENT SYSTEM PARTICULATE IODINE NOBLE GAS (SPING) MONITOR. THE SPING CLOSES THE CONTAINMENT PURGE ISOLATION VALVES UPON DETECTING A HIGH RADIATION CONDITION. NO SPECIFIC CAUSE FOR THE CLOSURE OF 1FCV-PR24A COULD BE FOUND. THE SAFETY SIGNIFICANCE IS MINIMAL BECAUSE THE UNIT WAS DEFUELED, AND CONTAINMENT PURGE WAS NOT IN PROGRESS.

[250] ZION 2 DOCKET 50-304 LER 88-017 REV 01
UPDATE ON INOPERABLE REACTOR CAVITY VENT FAN 2A DUE TO A PROCEDURE DEFICIENCY.
EVENT DATE: 122788 REPORT DATE: 100289 NSSS: WE TYPE: PWR

(NSIC 215689) ON 12/27/88 AT 1400 HOURS UNIT 2 WAS IN HOT STAND-BY AT 1/2% POWER WITH TAVG ABOUT 550F, INSTRUMENT MAINTENANCE (IM) MECHANICS WERE ASSISTING TROUBLESHOOTING THE 2B REACTOR CAVITY VENT FAN (WHICH WAS SHUT OFF), AND DISCOVERED THAT THE 2A REACTOR CAVITY VENT FAN (VA) WAS RUNNING BACKWARDS. THE CONDITION OF THE 2A FAN RUNNING BACKWARDS RESULTED IN THE FAN BEING INOPERABLE, AND THUS FAILED TO MEET TECH SPEC LIMITING CONDITION FOR OPERATION 3.2.2.C.3.A WHICH REQUIRES ONE OF TWO REACTOR CAVITY VENT FANS TO BE OPERATING WHEN THE REACTOR COOLANT SYSTEM AVERAGE TEMPERATURE (TAVG) IS GREATER THAN 145F. THE CAUSE OF THE FAN RUNNING BACKWARDS IS SUSPECTED TO BE A WIRING ERROR. THE 2A REACTOR CAVITY FAN BREAKER HAD BEEN RECENTLY REBUILT AND REWIRED ON 12/12/88 WHICH COULD HAVE LED TO A REWIRING PROBLEM AND CAUSED REVERSE ROTATION OF THE MOTOR. UPON DISCOVERY, THE 2A FAN WAS TURNED OFF, AND THE 2B FAN WAS STARTED. AT 1700 HOURS OF 12/27/88, TWO OF THE THREE FIELD WIRES AT THE 2A FAN BREAKER WERE SWITCHED AND THE 2A FAN WAS RESTORED WITH PROPER AIR FLOW DIRECTION CONFIRMED. A PROCEDURE CHANGE WAS MADE REQUIRING POST-MAINTENANCE ROTATION VERIFICATION. THIS CHANGE AND THE INCIDENT ITSELF WERE DISCUSSED WITH ELECTRICAL MAINTENANCE PERSONNEL. THERE WERE NO ABNORMAL TEMPERATURE EXCURSIONS IN THE REACTOR CAVITY DURING THE TIME THAT THE 2A FAN WAS RUNNING BACKWARDS.

[251] ZION 2 DOCKET 50-304 LER 89-009
MISSED SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 090389 REPORT DATE: 100389 NSSS: WE TYPE: PWR

(NSIC 215554) AT 0630 HOURS ON 9/3/89 THE VENT STACK SPING FLOW MONITOR (2LP084) FAILED, RENDERING THE AUXILIARY BUILDING STACK RADIATION MONITOR INOPERABLE. THE SURVEILLANCE REQUIREMENTS (SHIFTLY SAMPLE AND BLOWER CHECK) WERE NOT PERFORMED FOR THE 1500-2300 SHIFT ON 9/3/89. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR. CORRECTIVE ACTION WILL BE DETERMINED FOLLOWING A HUMAN PERFORMANCE EVALUATION SYSTEM (HPES) INVESTIGATION. THE HEALTH AND SAFETY OF THE GENERAL PUBLIC WERE NOT COMPROMISED DUE TO THE FACT THAT PROCESS MONITORING DEVICES LOCATED UPSTREAM OF THE INOPERABLE MONITOR WERE OPERABLE THROUGHOUT THE EVENT.

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

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KEYWORD INDEX

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

KEYWORD INDEX

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VENDOR CODE INDEX

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202

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BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

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For month of November 1989

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11a. TYPE OF REPORT
Monthly Report

b. PERIOD COVERED (Inclusive dates)
November 1989

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
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