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Licensee Event Report (LER) Compilation

For month of December 1989

Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

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Abstract

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one month period identified on the cover of the document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting for revisions to those events occurring prior to 1984 are described in NRC Regulatory Guide 1.16 and NUREG-0161, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports. For those events occurring on and after January 1, 1984, LERs are being submitted in accordance with the revised rule contained in Title 10 Part 50.73 of the Code of Federal Regulations (10 CFR 50.73 - Licensee Event Report System) which was published in the Federal Register (Vol. 48, No. 144) on July 26, 1983. NUREG-1022, Licensee Event Report System - Description of Systems and Guidelines for Reporting, provides supporting guidance and information on the revised LER rule.

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

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[1] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 89-033
POWER RANGE NUCLEAR INSTRUMENTATION NOT CALIBRATED IN ACCORDANCE WITH FREQUENCY ESTABLISHED BY TECH SPECS DUE TO INADEQUATE PROCEDURAL GUIDANCE FOR SURVEILLANCE TEST PROGRAM IMPLEMENTATION.
EVENT DATE: 100989 REPORT DATE: 110889 NSSS: BW TYPE: PWR

(NSIC 215858) ON 10/9/89, IT WAS DETERMINED BY PLANT PERSONNEL THAT THE POWER RANGE NUCLEAR INSTRUMENTATION HAD NOT BEEN CALIBRATED TWICE DURING THE PREVIOUS WEEK AS REQUIRED BY THE TECH SPECS. DURING STEADY STATE POWER OPERATION, A HEAT BALANCE CALIBRATION OF THE POWER RANGE INSTRUMENTS IS PERFORMED EACH MONDAY AND FRIDAY IN ORDER TO MEET THE TECH SPEC SURVEILLANCE REQUIREMENT. HOWEVER, AFTER COMPLETION OF THE CALIBRATION ON MONDAY, 10/9, IT WAS DETERMINED THAT THE CALIBRATION WHICH HAD BEEN SCHEDULED FOR FRIDAY, 10/6, HAD NOT BEEN PERFORMED. IT WAS ALSO DETERMINED THAT, PRIOR TO 10/9, THE LAST HEAT BALANCE CALIBRATION HAD BEEN PERFORMED ON 10/2/89. THIS EVENT WAS NOT SAFETY SIGNIFICANT SINCE THE CALIBRATION PERFORMED ON 10/9 INDICATED THAT THE POWER RANGE INSTRUMENTS HAD REMAINED WITHIN THEIR CALIBRATION TOLERANCE AND NO ADJUSTMENTS WERE NECESSARY. THE CAUSE OF THIS EVENT WAS DETERMINED TO BE INADEQUATE PROCEDURAL GUIDANCE WITH RESPECT TO SPECIFIC RESPONSIBILITIES FOR IMPLEMENTATION OF THE SURVEILLANCE TEST PROGRAM. INTERIM CORRECTIVE ACTION CONSISTED OF ISSUANCE OF A MEMO SPECIFYING ADDITIONAL ACTIVITIES TO BE PERFORMED BY TEST COORDINATORS, SCHEDULERS, AND SUPERVISORS RESPONSIBLE FOR THE PERFORMANCE OF SURVEILLANCES.

[2] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 89-036
ELECTRICAL SHORT TO CHLORINE DETECTORS RESULTED IN AN UNEXPECTED ACTUATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM.
EVENT DATE: 101789 REPORT DATE: 111689 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 2 (PWR)

(NSIC 215860) AT 2228 HOURS ON 10/17/89, WHILE PERFORMING WIRING CHANGES ASSOCIATED WITH A MODIFICATION TO THE 'B' CHANNEL CORE PROTECTION CALCULATOR (CPC), A BREAKER (2RS2-17) IN THE 120 VAC VITAL ELECTRICAL DISTRIBUTION PANEL WHICH POWER 'B' CPC WAS CLOSED AND AN UNEXPECTED ACTUATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) OCCURRED. THE CREVS ACTUATED DUE TO AN INITIATION SIGNAL FROM TWO CHLORINE DETECTORS WHICH EXPERIENCED A LOW VOLTAGE CONDITION WHEN THE BREAKER WAS CLOSED. CONTROL ROOM OPERATIONS PERSONNEL RESET THE CHLORINE DETECTORS AND RESTORED CONTROL ROOM VENTILATION TO NORMAL. THE ROOT CAUSE OF THE POWER SUPPLY LOW VOLTAGE WAS AN ELECTRICAL SHORT IN 2RS2. THE CAUSE OF THE SHORT HAS BEEN REPAIRED. DURING THIS EVENT, THE CREVS ACTUATED AS DESIGNED ALTHOUGH NO ACTUAL HIGH CHLORINE LEVELS EXISTED. THEREFORE, THERE WAS NO ACTUAL SAFETY SIGNIFICANCE RELATED TO THIS EVENT. THE ACTUATION OF THE CREVS IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV) AND WAS REPORTED PER 10CFR50.72(B)(2)(II).

[3] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 89-017
PERSONNEL ERROR RESULTS IN LESS THAN THE REQUIRED TECH SPEC LOGARITHMIC POWER LEVEL NUCLEAR INSTRUMENTATION CHANNELS OPERABLE.
EVENT DATE: 100589 REPORT DATE: 110689 NSSS: CE TYPE: PWR

(NSIC 215900) ON 10/5/89, WHILE IN COLD SHUTDOWN CONTROL ROOM OPERATORS RECOGNIZED THAT THE NUMBER OF OPERABLE LOGARITHMIC (LOG) POWER LEVEL INSTRUMENTATION CHANNELS WAS LESS THAN REQUIRED BY TECH SPECS FOR THE EXISTING PLANT CONDITION. MODIFICATIONS WERE IN PROGRESS ON THE LOG POWER LEVEL CHANNELS TO REPLACE CABLE CONNECTORS BETWEEN INSTRUMENTS AND CONTROL ROOM CABINETS. THE SHIFT SUPERVISOR AUTHORIZED THE WORK TO BE PERFORMED ON THESE INSTRUMENTS, WITHOUT RECOGNIZING THE TECH SPEC REQUIREMENTS. AT THE TIME OF OCCURRENCE OF THIS EVENT, THE FUNCTION PERFORMED BY THE LOG POWER INSTRUMENTATION CHANNELS WAS TO PROVIDE A MEANS OF MONITORING REACTOR CORE REACTIVITY AND ANNUNCIATION OF ABNORMAL CONDITIONS. DURING THE TIME PERIOD LOG POWER WAS INOPERABLE OTHER CONTROL ROOM INSTRUMENTATION INCLUDING BORON DILUTION MONITORS AND STARTUP NUCLEAR INSTRUMENTATION CHANNELS WERE AVAILABLE FOR MONITORING CORE REACTIVITY CONDITIONS. THEREFORE, THERE WAS NO SAFETY SIGNIFICANCE AS A RESULT OF THIS EVENT. INITIAL TRAINING HAS BEEN PROVIDED TO OPERATIONS PERSONNEL AND ADDITIONAL TRAINING WILL BE PROVIDED ON THE SIMULATOR CONCERNING THIS EVENT. THIS EVENT IS

CONSIDERED OPERATIONS PROHIBITED BY TECH SPECS AND IS REPORTABLE PER 10CFR50.73(A)(2)(I)(B).

[4] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 89-018
PERSONNEL ERROR ALLOWING MAINTENANCE ACTIVITIES IN TWO PLANT PROTECTION SYSTEM CHANNELS SIMULTANEOUSLY RESULTED IN AN UNEXPECTED AUTOMATIC ACTUATION OF THE PLANT PROTECTION SYSTEM.
EVENT DATE: 101789 REPORT DATE: 111689 NSSS: CE TYPE: PWR

(NSIC 215901) ON 10/17/89, AN INADVERTENT PLANT PROTECTION SYSTEM (PPS) ACTUATION OCCURRED WHEN A 120 VAC VITAL ELECTRICAL DISTRIBUTION PANEL BREAKER WAS OPENED. SYSTEM HAD BEEN PARTIALLY DEENERGIZED PRIOR TO THIS OCCURRENCE DUE TO AN UNRELATED MAINTENANCE ACTIVITY ON A PRESSURIZER PRESSURE VARIABLE SETPOINT CARD. A REACTOR TRIP SIGNAL OCCURRED AND A MAIN STEAM ISOLATION SIGNAL (MSIS) WERE GENERATED AS A RESULT OF THE PPS ACTUATION. UPON RECLOSING OF THE BREAKER, SAFETY INJECTION, CONTAINMENT COOLING, CONTAINMENT ISOLATION AND CONTAINMENT SPRAY ACTUATION SIGNALS WERE GENERATED. SINCE PLANT WAS IN A REFUELING OUTAGE, WITH FUEL ASSEMBLIES STORED IN SPENT FUEL POOL, AT THE TIME OF OCCURRENCE OF THIS EVENT, A LIMITED AMOUNT OF ENGINEERED SAFETY FEATURES (ESF) WAS ALIGNED FOR AUTOMATIC ACTUATION. HOWEVER, THOSE COMPONENTS WHICH WERE INSERVICE ACTUATED AS DESIGNED. WHEN THE BREAKER WAS OPENED, STEAM GENERATOR PRESSURE INSTRUMENTATION FAILED LOW RESULTING IN MSIS AND WHEN BREAKER WAS RECLOSED, CONTAINMENT BUILDING PRESSURE INSTRUMENTATION SPIKED CAUSING OTHER ESF ACTUATIONS. PPS CHANNEL WHICH WAS DEENERGIZED WAS RETURNED TO SERVICE AND THE ESF ACTUATION SIGNALS WERE CLEARED. THE ROOT CAUSE WAS PERSONNEL ERROR, IN THAT MAINTENANCE ACTIVITIES WERE ALLOWED TO BE PERFORMED IN 2 PPS CHANNELS SIMULTANEOUSLY.

[5] ARNOLD DOCKET 50-331 LER 89-006 REV 01
UPDATE ON ISOLATION OF RCIC AND HPCI DUE TO A BROKEN THERMOCOUPLE LEAD AND POOR PERFORMANCE OF TEST SWITCHES IN STEAM LEAK DETECTION SYSTEM LOGIC.
EVENT DATE: 022489 REPORT DATE: 111089 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215934) ON 2/24/89, AN ISOLATION OF THE REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) OCCURRED DUE TO A SIGNAL FROM THE STEAM LEAK DETECTION SYSTEM (SLDS). TROUBLESHOOTING DETERMINED THE CAUSE TO BE A NEARLY BROKEN THERMOCOUPLE (TC) WIRE ON A TEMPERATURE DETECTION MODULE IN THE SLDS LOGIC. THE ROOT CAUSE WAS A COMBINATION OF THE NATURAL PROPERTIES OF THE SOLID CORE TC WIRE, 1970'S SURVEILLANCE TESTING METHODOLOGY, AND FINAL WIRE DAMAGE DURING RECENT REPLACEMENT OF THE MODULE WITH AN UPGRADED MODEL. THE WIRE WAS RE-TERMINATED. A TIME DELAY IN THE SLDS ISOLATION LOGIC WAS RAISED FROM ONE TO THREE SECONDS FOR THE RCIC AND HIGH PRESSURE COOLANT INJECTION (HPCI) SLDS LOGIC. RCIC WAS RETURNED TO SERVICE ON 3/2/89, AFTER SEVERAL DAYS OF MONITORING THE RCIC SLDS LOGIC FOR FURTHER PROBLEMS. ON 3/2/89, AN ISOLATION OF HPCI OCCURRED DUE TO A SIGNAL FROM THE SLDS LOGIC. TROUBLESHOOTING REVEALED AN INTERNAL PROBLEM WITHIN TEST SWITCHES BETWEEN THE TC'S AND SLDS TEMPERATURE DETECTION MODULES WAS RESULTING IN SIMULATED OPEN TC WIRE SIGNALS, WHICH WERE BEING DETECTED BY UPGRADED MODULES INSTALLED IN 1988. FOLLOWING REMOVAL OF THE TEST SWITCHES FROM ITS SLDS CIRCUITRY, HPCI WAS RETURNED TO SERVICE ON 3/3/89. THE TEST SWITCHES WERE REMOVED FROM THE RCIC AND REACTOR WATER CLEANUP SLDS LOGIC SHORTLY THEREAFTER.

[6] ARNOLD DOCKET 50-331 LER 89-013
MAIN STEAM ISOLATION VALVES HAVE UNACCEPTABLE LEAK TEST RESULTS DUE TO MISALIGNMENTS, EXCESSIVE CLEARANCES AND VALVE ORIENTATION.
EVENT DATE: 091589 REPORT DATE: 111089 NSSS: GE TYPE: BWR
VENDOR: ROCKWELL-INTERNATIONAL

(NSIC 215943) DURING A RECENT PLANT MAINTENANCE OUTAGE, LOCAL LEAK RATE TESTING WAS PERFORMED ON THREE MAIN STEAM ISOLATION VALVES (MSIVS) DUE TO PREVIOUS PERFORMANCE PROBLEMS, IN ACCORDANCE WITH DISCUSSIONS BETWEEN IOWA ELECTRIC AND THE NUCLEAR REGULATORY COMMISSION. AS ONE OF THE THREE VALVES TESTED FAILED TO MEET THE ESTABLISHED CRITERIA, THE REMAINING FIVE MSIVS WERE ALSO TESTED. A SECOND VALVE DID NOT MEET THE LEAKAGE CRITERIA, AND A THIRD, WHICH HAD PASSED THE

INITIAL TESTING, EXHIBITED EXCESSIVE LEAKAGE DURING A SECOND TEST SIX DAYS LATER. THE CAUSES WERE FOUND TO BE: A) INCORRECT SEAT CONTACT WITH LACK OF CONCENTRICITY DUE TO A SLIGHT MISALIGNMENT OF THE VALVE BORE WITH RESPECT TO THE VALVE SEAT, B) EXCESSIVE CLEARANCE BETWEEN THE VALVE BORE AND VALVE INTERNALS, AND C) THE VALVE'S 35 DEGREE CANTED ORIENTATION WHICH MAKES IT MORE SUSCEPTIBLE TO ALIGNMENT PROBLEMS. ONLY PARTIAL CONTACT WAS NOTED AT THE TOP OF THE SEAT UPON DISASSEMBLY FOR THIS VALVE. CORRECTIVE ACTIONS TAKEN INCLUDED RE-MACHINING OF VALVE DISCS AND SEATS AND INSTALLATION OF OVERSIZED INTERNALS. LONG TERM UPGRADES TO ALL MSIVS ARE PLANNED FOR THE 1990 REFUELING OUTAGE. THIS LER IS BEING SUBMITTED FOR INFORMATION ONLY.

[7] BEAVER VALLEY 1 DOCKET 50-334 LER 89-010
UNQUALIFIED FIRE DAMPERS.
EVENT DATE: 090789 REPORT DATE: 100989 NSSS: WE TYPE: PWR
VENDOR: SCHNEIDER INC.

(NSIC 215564) ON 9/7/89, UNDERWRITERS LABORATORIES (UL) NOTIFIED DUQUESNE LIGHT OF THE RESULTS OF THE FIRE TESTING FOR HORIZONTALLY MOUNTED PROTOTYPE FIRE DAMPERS APPLICABLE TO BEAVER VALLEY POWER STATION UNIT 1 CONDUCTED ON 9/6/89. THE TEST RESULTS INDICATED THAT THE DAMPERS FAILED TO COMPLY WITH THE REQUIREMENTS OF UL STANDARD 555-1968 FOR A NINETY (90) MINUTE FIRE ENDURANCE AND HOSE STREAM TEST. THE APPLICABLE FIRE DAMPERS WERE DECLARED INOPERABLE AND COMPENSATORY ACTIONS WERE TAKEN IN THE FORM OF FIRE WATCHES. AN ANALYSIS WAS CONDUCTED TO DETERMINE COMPLIANCE AS OUTLINED IN 10CFR50 APPENDIX R. IT WAS DETERMINED THAT REDUNDANT TRAINS OF EQUIPMENT REQUIRED FOR SAFE SHUTDOWN WERE POTENTIALLY AFFECTED, AND THEREFORE REPORTABLE IN ACCORDANCE WITH 10CFR50.73.A.2.II.A. FIRE WATCHES WILL BE MAINTAINED UNTIL THE DAMPERS ARE REPLACED OR ANALYSES PERFORMED TO SUPPORT THE PRESENT CONFIGURATIONS. THERE WERE NO MAJOR SAFETY IMPLICATIONS DUE TO THIS EVENT SINCE THE INSTALLED DUCTWORK AND FIRE DAMPERS, ALTHOUGH LACKING PROPER QUALIFICATIONS, WOULD AFFORD SOME PROTECTION TO PREVENT THE PROPAGATION OF FIRE. AUTOMATIC FIRE SUPPRESSION SYSTEMS AND EARLY WARNING SMOKE DETECTION SYSTEMS ARE INSTALLED ON AT LEAST ONE SIDE OF ALL AFFECTED BARRIERS TO MITIGATE THE EFFECTS OF A POSTULATED FIRE EVENT.

[8] BEAVER VALLEY 1 DOCKET 50-334 LER 89-012
INCORE INSTRUMENTATION THIMBLE TUBE WEAR.
EVENT DATE: 101089 REPORT DATE: 110889 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELEC CORP.-NUCLEAR ENERGY SYS

(NSIC 215877) ON 10/10/89, WHILE THE UNIT WAS IN ITS SEVENTH REFUELING OUTAGE, EDDY CURRENT TESTING IDENTIFIED NINE INCORE NUCLEAR INSTRUMENTATION GUIDE THIMBLE TUBES WITH DEGRADATION IN EXCESS OF SPECIFIED LIMITS. THE ANALYSIS ALSO PROJECTED DEGRADATION IN EXCESS OF 60% WALL THICKNESS FOR AN ADDITIONAL NINE TUBES BY THE END OF THE NEXT FUEL CYCLE. THIS DEGRADATION IS APPARENTLY DUE TO MECHANICAL WEAR OF THE THIMBLES AGAINST REACTOR VESSEL INTERNALS INDUCED BY THE COOLANT FLOW CHARACTERISTICS THROUGH THE VESSEL. FINITE ELEMENT ANALYSIS HAS DETERMINED THAT, BASED ON ASME CODE ALLOWABLE STRESS LIMITS, TUBE DEGRADATION OF UP TO 60% IS ACCEPTABLE. THE NINE TUBES WITH DEGRADATION BEYOND SPECIFIED LIMITS WERE REMOVED FROM SERVICE AND ISOLATED TO PREVENT LEAKAGE. THE NINE TUBES WITH PROJECTED DEGRADATIONS IN EXCESS OF 60% WERE REPOSITIONED TO PREVENT UNACCEPTABLE WALL THINNING. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. FAILURE OF A GUIDE THIMBLE TUBE IS BOUNDED BY BEAVER VALLEY UNIT 1 UFSAR ANALYSIS SECTION 14.3, "LOSS OF COOLANT ACCIDENT."

[9] BEAVER VALLEY 2 DOCKET 50-412 LER 88-002 REV 01
UPDATE ON REACTOR TRIP AND CONTROL ROOM EMERGENCY BOTTLED AIR PRESSURIZATION SYSTEM ACTUATION.
EVENT DATE: 012788 REPORT DATE: 103089 NSSS: WE TYPE: PWR
VENDOR: I-T-E CIRCUIT BREAKER

(NSIC 215807) DURING NORMAL PLANT TESTING THE 21C SERVICE WATER PUMP WAS SHUTDOWN. AT THIS TIME THE 21A SERVICE WATER PUMP DISCHARGE PRESSURE TRANSMITTER FAILED DOWNSCALE RESULTING IN THE AUTO START OF THE 21A EMERGENCY SERVICE WATER

PUMP. AT 0152 HOURS, THE 21A EMERGENCY SERVICE WATER PUMP WAS SHUTDOWN AND 62 MILLISECONDS LATER, THE 4KV BUS 2A SUPPLY BREAKER (ACB-42C) TRIPPED ON PHASE OVERCURRENT. THE LOSS OF THE 2A BUS CAUSED THE 21A REACTOR COOLANT PUMP TO TRIP ON UNDERVOLTAGE RESULTING IN A PLANT TRIP. THE AUTO-TRANSFER OF THE 2A BUS DID NOT OCCUR DUE TO THE OVERCURRENT TRIP. DE-ENERGIZING THE 2AE BUS CAUSED THE NO. 1 EMERGENCY DIESEL GENERATOR TO AUTOMATICALLY START AND ENERGIZE THE 4KV EMERGENCY BUS 2AE. DURING THIS TRANSIENT, THE RADIATION MONITORS FOR THE UNIT 2 CONTROL ROOM DE-ENERGIZED MOMENTARILY RESULTING IN ACTUATION OF THE CONTROL ROOM EMERGENCY BOTTLED AIR PRESSURIZATION SYSTEM (CREBAPS). THE SERVICE WATER PUMP DISCHARGE PRESSURE TRANSMITTER WAS CALIBRATED SATISFACTORILY AND IS SCHEDULED FOR REPLACEMENT. A MODIFICATION TO PREVENT A CREBAPS ACTUATION ON A LOSS OF POWER TO THE UNIT 2 CONTROL ROOM RADIATION MONITORS IS BEING EVALUATED. THE CAUSE OF THE ACB-42C TRIP IS UNDER INVESTIGATION. THE RESULTS OF THE INVESTIGATION IS PROVIDED IN THIS REPORT. THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC AS A RESULT OF THIS EVENT AS ALL PROTECTIVE SYSTEMS ACTUATION AS DESIGNED.

[10] BEAVER VALLEY 2 DOCKET 50-412 LER 89-014 REV 01
UPDATE ON LEAK COLLECTION VENTILATION FLOWPATH AUTOMATIC REALIGNMENT ACTUATION.
EVENT DATE: 050689 REPORT DATE: 103189 NSSS: WE TYPE: PWR
VENDOR: GOULD SWITCHGEAR DIVISION

(NSIC 215899) ON 5/6/89 AT 0210 HOURS, THE SUPPLY BREAKER TO THE J 480 VOLT ESSENTIAL BUS OPENED. THE ALTERNATE J BUS SUPPLY BREAKER DID NOT CLOSE. THIS DE-ENERGIZED THE J BUS AND ALL COMPONENTS POWERED OFF THE J BUS, INCLUDING NON-FILTERED VENTILATION EXHAUST RADIATION MONITOR (2RMR-RQ301). AS DESIGNED, THE MONITOR FAILED HIGH WHEN DE-ENERGIZED. THIS HIGH RADIATION SIGNAL INITIATED AN AUTOMATIC VENTILATION REALIGNMENT, DIVERTING THE VENTILATION FROM THE NON-FILTERED TO FILTERED FLOWPATH. OPERATORS FOUND NO APPARENT CAUSE FOR THE BREAKER TO TRIP. IT WAS DISCOVERED THAT THE RESET BUTTON ON THE SUPPLY BREAKER WAS IN THE "OUT" POSITION. FOLLOWING AN INVESTIGATION THE RESET BUTTON WAS RETURNED TO ITS NORMAL "IN" POSITION, AND THE SUPPLY BREAKER WAS CLOSED. THE J BUS INCLUDING RADIATION MONITOR (2RMR-RQ301) WAS RE-ENERGIZED AND THE VENTILATION SYSTEM WAS RETURNED TO ITS NORMAL FLOWPATH. AFTER REPLACING WITH A TESTED SPARE, MAINTENANCE SENT THE J BUS SUPPLY BREAKER TO THE VENDOR FOR ANALYSIS. THE BREAKER WAS FOUND TO HAVE A BROKEN CIRCUIT RUN IN THE POWER SHIELD WHICH CAUSED THE SPURIOUS TRIP. ALL RELAYS AND CIRCUITS ASSOCIATED WITH THE SUPPLY AND ALTERNATE BREAKERS WERE TESTED AND VERIFIED OPERABLE. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. THIS EVENT WAS BOUNDED BY BEAVER VALLEY UFSAR SECTION 6.5.3.2, "SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM".

[11] BRAIDWOOD 1 DOCKET 50-456 LER 89-013
INADEQUATE INCORPORATION OF ISOLATION REQUIREMENTS FOR THE STEAM GENERATOR BLOWDOWN SYSTEM DUE TO A PRESERVICE DESIGN DEFICIENCY.
EVENT DATE: 100589 REPORT DATE: 110389 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BRAIDWOOD 2 (PWR)

(NSIC 215923) ON 10/5/89 A DISCREPANCY WITH THE DESIGN OF THE STEAM GENERATOR BLOWDOWN (SD) SYSTEM WAS IDENTIFIED. THE SAFETY ANALYSIS REPORT (UFSAR) SPECIFIES SD ISOLATION ON THE INITIATION OF THE AUX. FEEDWATER (AF) SYSTEM. THIS MINIMIZES THE AF FLOW REQUIREMENTS. CURRENT DESIGN DOES NOT PROVIDE FOR AUTOMATIC ISOLATION ON ALL AF INITIATIONS. SD SYSTEM WAS ISOLATED. CHANGES WERE MADE TO EMERGENCY PROCEDURES REQUIRING SD ISOLATION ON AF INITIATION. AFTER THESE CHANGES SD OPERATION WAS PERMITTED. ENGINEERING EVALUATED THE AFFECTS ON THE ACCIDENT ANALYSIS OF THE UFSAR. IMPACTED SCENARIOS WERE LOSS OF NORMAL FEEDWATER (LONF), LOSS OF NON-EMERGENCY AC POWER (LOAC), AND FEEDWATER PIPE BREAK (FSPB). THIS WAS DUE TO THE LACK OF A SG LO LO WATER LEVEL SD ISOLATION ON AF INITIATION. CALCULATIONS WERE PERFORMED USING ACTUAL PLANT DATA AND TAKING CREDIT FOR MANUAL SD ISOLATION WITHIN 10 MINUTES FOR LONF AND LOAC EVENTS OR CREDIT FOR REACHING A PHASE A INITIATING SETPOINT WITHIN 6 MINUTES FOR FSPB EVENTS. IT WAS DETERMINED THAT THE UFSAR ACCEPTANCE CRITERIA WAS MET. BRAIDWOOD STATION WAS JUSTIFIED TO CONTINUE OPERATION PROVIDED THE BWEP'S WERE REVISED AND THE PERMANENT MODIFICATION IS COMPLETED ON A PRUDENT SCHEDULE. MODIFICATIONS ARE SCHEDULED. CAUSE WAS A PRESERVICE DESIGN DEFICIENCY FOR UNKNOWN REASONS. THE UFSAR WILL BE REVISED TO REFLECT THE CHANGES.

[12]   BRAIDWOOD 1   DOCKET 50-456   LER 89-012
CONTAINMENT VENTILATION ISOLATION AND FUEL HANDLING BUILDING FAN AUTO START ON MOMENTARY LOSS OF POWER TO AREA RADIATION MONITORS DUE TO PERSONNEL ERROR.
EVENT DATE: 101689   REPORT DATE: 111589   NSSS: WE   TYPE: PWR

(NSIC 215922) AT APPROXIMATELY 0522, ON 10/16/89 A NON-LICENSED OPERATOR (EA) WAS REMOVING A 120 VOLT AC CIRCUIT FROM SERVICE. THE SWITCH WAS LOCATED IN A DISTRIBUTION PANEL. WHILE ATTACHING A CARD TO SWITCH 6, THE EA INADVERTENTLY BUMPED SWITCH 4 TO THE OFF POSITION. SWITCH 4 WAS THE POWER SUPPLY FOR TWO AREA RADIATION MONITORS (ARM). ONE OF THE ARMS MONITORED THE FUEL HANDLING BUILDING (FHB), ONE MONITORED UNIT ONE CONTAINMENT. THE MOMENTARY LOSS OF POWER TO THE FHB ARM CAUSED A FHB CHARCOAL BOOSTER FAN TO AUTO START AND EXHAUST THROUGH THE CHARCOAL ADSORBERS. THE LOSS OF POWER TO THE CONTAINMENT ARM INITIATED A TRAIN A CONTAINMENT VENTILATION ISOLATION SIGNAL. THE TRAIN A ISOLATION VALVES CLOSED AND THE MINI FLOW PURGE SUPPLY AND EXHAUST FANS TRIPPED. THE EA IMMEDIATELY RETURNED SWITCH 4 TO THE ON POSITION. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. A CONTRIBUTING CAUSE TO THE EVENT WAS THE NARROW SWITCH SPACING OF THE DISTRIBUTION PANEL. THE FHB FAN WAS SECURED AND THE ISOLATION SIGNAL WAS RESET. THIS EVENT WILL BE COVERED IN AN TRAINING TAILGATE SESSION. DEVICES WHICH PROVIDE AN EFFECTIVE CARD ATTACHMENT PIECE WILL BE FABRICATED AND SUPPLIED TO OPERATING PERSONNEL. THERE WERE PREVIOUS EVENTS OF CONTAINMENT VENTILATION ISOLATION AND FHB BOOSTER FAN AUTO START DUE TO LOSS OF POWER TO ARMS.

[13]   BRAIDWOOD 2   DOCKET 50-457   LER 89-006
FAILURE TO IDENTIFY THAT CONTAINMENT SPRAY VALVE WAS REQUIRED TO BE LOCKED IN A THROTTLED POSITION DUE TO PROGRAMMATIC DEFICIENCY.
EVENT DATE: 031489   REPORT DATE: 111789   NSSS: WE   TYPE: PWR

(NSIC 215947) ON THE AFTERNOON SHIFT ON 3/13/89 A RO LICENSED OPERATOR (NSO) IDENTIFIED THE RETURN TO SERVICE POSITIONS FOR THE 2B CONTAINMENT SPRAY (CS) PUMP. ONE OF THE VALVES WAS THE 2CS021B, EDUCTOR 2B SPRAY ADDITIVE TANK SUCTION THROTTLE VALVE. THE NSO DETERMINED POSITIONS FROM THE PIPING AND INSTRUMENTATION DIAGRAMS (P&IDS). DIRECTLY BELOW THE VALVE SYMBOL APPEARED THE LETTERS "L.O." WHICH MEANT LOCKED OPEN. THERE WAS A NOTE ON THE P&ID SHEET THAT STATED THE VALVE SHOULD BE THROTTLED. BASED ON THE L.O. DESIGNATION THE LICENSED OPERATOR (EA) PLACED THE 2CS021B IN THE LOCKED OPEN POSITION. ON 10/18/89 AN EA DISCOVERED THAT THE 2CS021B WAS POSITIONED INCORRECTLY. THE VALVE WAS IMMEDIATELY REPOSITIONED. THE CAUSE OF THIS EVENT WAS THAT THE VALVE WAS NOT LABELED AS A THROTTLE VALVE. A CONTRIBUTING CAUSE WAS THAT THE OUT OF SERVICE PROGRAM DID NOT REQUIRE PLACING THE "AS FOUND" POSITIONS ON THE OUTAGE FORM. ALL "ACCESSIBLE" UNIT 2 LOCKED VALVES WERE POSITION VERIFIED. UNIT 1 LOCKED VALVES WILL BE VERIFIED PRIOR TO ITS RETURN TO OPERATION. LOCKED THROTTLED VALVES WILL BE PROVIDED WITH HIGH VISIBILITY LABELS. THE OUT OF SERVICE FORM HAS BEEN MODIFIED. THE OUT OF SERVICE PROGRAM WILL BE REVISED TO PROVIDE GUIDANCE CONERNING THROTTLE VALVES. THERE HAVE BEEN NO PREVIOUS OCCURRENCES.

[14]   BRAIDWOOD 2   DOCKET 50-457   LER 89-005
FAILURE TO CONSIDER ESF BUS OUTAGE EFFECTS ON OPPOSITE UNIT DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 100289   REPORT DATE: 110189   NSSS: WE   TYPE: PWR
OTHER UNITS INVOLVED: BRAIDWOOD 1 (PWR)

(NSIC 215924) AT 1638 ON 10/2/89 A UNIT 1 SAFETY RELATED BUS WAS REMOVED FROM SERVICE. THE UNIT 1 AND UNIT COMMON TECH SPEC ACTION STATEMENTS WERE ENTERED. THE REVIEW DID NOT CONSIDER THE IMPACT ON UNIT 2. THE BUS WAS PART OF THE AC POWER SOURCES REQUIREMENT FOR UNIT 2. THE ACTION STATEMENT PROVIDED FOR THE BUS TO BE INOPERABLE FOR 72 HOURS. THIS PROVISION SPECIFIED THAT OPERABILITY OF THE REMAINING SOURCES BE VERIFIED BY PERFORMING SURVEILLANCE REQUIREMENTS AT AN INCREASED FREQUENCY. PERFORMANCE OF A SPECIFICATION WHICH CONSISTS OF BREAKER ALIGNMENT AND BUS VOLTAGE VERIFICATION WAS REQUIRED WITHIN 1 HOUR AND EVERY 8 HOURS THEREAFTER. AT 2053 A SUPERVISOR DISCOVERED THAT THE ACTION STATEMENT HAD NOT BEEN ENTERED FOR UNIT 2 WHEN THE BUS WAS REMOVED FROM SERVICE. AT 2110 THE REQUIREMENTS OF THE ACTION STATEMENT WERE COMPLETED WITH ACCEPTABLE RESULTS. THIS EXCEEDED THE "WITHIN ONE HOUR" FREQUENCY BY 3 HOURS AND 32 MINUTES. THE

CAUSE OF THIS EVENT WAS A PROCEDURAL DEFICIENCY. THE OUT-OF-SERVICE PROCEDURE WAS STRUCTURED IN A UNIT SPECIFIC FORMAT. THIS CREATED A MIND SET THAT DID NOT KEY THE REVIEWER TO CONSIDER THE IMPACT ON THE OPPOSITE UNIT. THE SPECIFICATION WAS IMMEDIATELY PERFORMED. THE OUT-OF-SERVICE PROCEDURE WILL BE REVISED TO INCLUDE A CHECK OFF BLOCK FOR THE OPPOSITE UNIT. THERE HAVE BEEN NO PREVIOUS OCCURRENCES.

[15] BROWNS FERRY 1 DOCKET 50-259 LER 89-025
DESIGN ERRORS IN 250V DC ELECTRICAL SYSTEM RESULTS IN UNANALYZED CONDITION.
EVENT DATE: 062786 REPORT DATE: 110289 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 215827) THREE CONDITIONS WERE DISCOVERED ON JUNE 27, 1986 THAT COULD PLACE THE PLANT IN AN UNANALYZED CONDITION. THE FIRST CONDITION INVOLVES A SINGLE FAILURE OF ANY OF THE THREE MAIN PLANT 250V BATTERIES, IN CONJUNCTION WITH A LOSS OF OFFSITE POWER AND A RECIRCULATION LINE BREAK, WHICH COULD PREVENT ADEQUATE CORE COOLING BECAUSE OF A POTENTIAL TO OVERLOAD A DIESEL GENERATOR (DG). THE SECOND CONDITION INVOLVES THE LOSS OF MAIN PLANT 250V BATTERY NUMBER 1, WHICH COULD PREVENT LOAD SHEDDING FOR THE 480V SHUTDOWN BOARDS. THIS CONDITION, COINCIDENT WITH A UNIT 1 ACCIDENT SIGNAL, COULD POTENTIALLY OVERLOAD DGS A, B, C AND D. THE THIRD CONDITION INVOLVES AN INABILITY TO AUTOMATICALLY CLOSE THE INBOARD AND OUTBOARD RESIDUAL HEAT REMOVAL (RHR) PRIMARY CONTAINMENT ISOLATION VALVES (PCIV) ON UNITS 1, 2 AND 3 BECAUSE OF A LOSS OF MAIN PLANT 250V BATTERY 1. THE CAUSES OF THESE CONDITIONS ARE DESIGN ERRORS THAT WERE IMPLEMENTED BEFORE THE UNITS WERE LICENSED. CONDITION 1 WAS RESOLVED BY CHANGING THE CONTROL POWER FOR THE 480V SHUTDOWN BOARD BREAKER LOGIC FROM THE PLANT 250V BATTERIES TO BATTERIES ASSOCIATED WITH THE 4KV SHUTDOWN BOARDS. CONDITION 2 WILL BE RESOLVED BEFORE RESTART OF UNIT 1. FOR CONDITION 3, MOTIVE POWER FOR THE OUTBOARD UNIT 2 RHR PCIV WAS CHANGED TO ENSURE ITS AUTOMATIC CLOSURE WITH THE LOSS OF THE 250V BATTERY. THE MOTIVE POWER FOR THE RHR PCIVS ON UNITS 1 AND 3 WILL BE CHANGED BEFORE RESTART.

[16] BROWNS FERRY 1 DOCKET 50-259 LER 89-013
SEISMIC REANALYSIS OF THE REACTOR PRESSURE VESSEL.
EVENT DATE: 060189 REPORT DATE: 111689 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 215944) SEISMIC REANALYSIS OF THE BROWNS FERRY (BFN) REACTOR PRESSURE VESSEL (RPV) CALCULATED THAT THE CONTROL ROD DRIVE (CRD) HOUSINGS WILL EXCEED THE STRESS LIMITS SPECIFIED IN THE BFN FINAL SAFETY ANALYSIS REPORT (FSAR) DURING NORMAL POWER OPERATION COMBINED WITH A SEISMIC EVENT. THE REANALYSIS WAS PERFORMED USING AN UPDATED RPV MODEL AND FSAR SPECIFIED DAMPING FACTORS. THE ORIGINAL BFN RPV ANALYSIS WAS PERFORMED IN THE LATE 1970'S BY GENERAL ELECTRIC (GE). THE ORIGINAL CALCULATIONS HAVE NOT BEEN LOCATED BUT THE ANALYSIS RESULTS WERE DOCUMENTED IN THE BFN FSAR SHOWING THAT THE EXISTING CRD HOUSING ARRANGEMENT WAS IN COMPLIANCE WITH THE ORIGINAL FSAR LICENSING BASIS. ALTHOUGH THE STRESS LIMITS ARE NOW SHOWN TO BE EXCEEDED BY THE NEW ANALYSIS THERE ARE NO INDICATIONS THAT THE CRD HOUSINGS WOULD HAVE FAILED DURING A SEISMIC EVENT. THEREFORE, TVA AND GE CONCUR THAT THIS CONDITION DOES NOT CONSTITUTE A SIGNIFICANT SAFETY HAZARD. TVA HAS REFINED THE SEISMIC MODEL OF THE RPV AND WILL UPGRADE THE ACTUAL PLANT CONFIGURATION BY ADDING LATERAL RESTRAINTS TO THE CRD HOUSINGS. UNIT 2 WAS IN COLD SHUTDOWN WITH IRRADIATED FUEL IN THE VESSEL AND UNITS 1 AND 3 WERE DEFUELED WHEN THIS CONDITION WAS DISCOVERED. ALL THREE UNITS WERE AFFECTED BY THIS CONDITION.

[17] BROWNS FERRY 1 DOCKET 50-259 LER 89-027
DEENERGIZATION OF REACTOR PROTECTION SYSTEM BUS BY MOTOR GENERATOR CIRCUIT PROTECTOR OPERATIONS CAUSED BY INADEQUATE DESIGN OF PROTECTOR SETPOINTS.
EVENT DATE: 101689 REPORT DATE: 111589 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 215828) ON 10/16/89 AT 1658, UNIT 1 REACTOR PROTECTION SYSTEM (RPS) BUS 1B WAS DEENERGIZED RESULTING IN THE COMPLETION OF THE ACTUATION LOGIC FOR SEVERAL PLANT ENGINEERED SAFETY FEATURES (ESFS). ESFS AFFECTED BY THIS EVENT INCLUDE STANDBY GAS TREATMENT SYSTEM AND PRIMARY CONTAINMENT ISOLATIONS. INVESTIGATION REVEALED THAT RPS MOTOR GENERATOR (MG) SET CIRCUIT PROTECTORS HAD TRIPPED BECAUSE OF SHORT DURATION OUTPUT VOLTAGE FLUCTUATIONS. CAUSE OF THE VOLTAGE FLUCTUATIONS EXPERIENCED DURING THIS EVENT HAS NOT BEEN DETERMINED. HOWEVER, ROOT CAUSE OF THE RPS BUS DEENERGIZATION HAS BEEN PREVIOUSLY IDENTIFIED IN LER 259/89024. ROOT CAUSE OF THESE EVENTS IS AN INADEQUATE DESIGN OF THE SETPOINTS FOR THE CIRCUIT PROTECTORS FOR RPS MG SETS. THIS RESULTS IN THE LACK OF AN ADEQUATE OPERATING MARGIN FOR THE RPS MG SET VOLTAGE REGULATION. FOLLOWING THIS EVENT, THE VOLTAGE REGULATOR FOR RPS MG SET 1B WAS REPLACED TO ENSURE THAT IT WAS NOT CAUSING THE VOLTAGE FLUCTUATIONS. AS RESULT OF LER 259/89024, APPROPRIATE CORRECTIVE ACTIONS WERE IN PROGRESS AT THE TIME OF THIS EVENT. AN EVALUATION OF CIRCUIT PROTECTION SETPOINTS IS BEING PERFORMED TO DETERMINE APPROPRIATE CHANGES. ADDITIONALLY, DESIGN CHANGES HAVE BEEN INITIATED TO REPLACE THE VOLTAGE ADJUSTMENT POTENTIOMETERS FOR ALL RPS MG SETS AND TO ADD A TIME DELAY TO THE ACTUATION CIRCUITRY.

[18] BRUNSWICK 1 DOCKET 50-325 LER 89-016 REV 01
UPDATE ON FAILURE TO PERFORM TECH SPEC REQUIRED STROKE TIME TESTING ON PCIS VALVES.
EVENT DATE: 061489 REPORT DATE: 110189 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)

(NSIC 215868) ON JUNE 14, 1989, IT WAS DETERMINED THAT SOME OF THE VALVES LISTED IN TABLE 1 OF THIS LER WERE NOT BEING TESTED IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS; SPECIFICALLY, STROKE TIME TESTING. EER 89-0195 WAS WRITTEN TO DETERMINE THE ADEQUACY OF CURRENT TESTING AND CORRECTIVE ACTIONS NECESSARY FOR ANY DEFICIENCIES AS A RESULT OF THIS LER, IT WAS DETERMINED THAT ALTHOUGH THE BASIS EXISTS TO DELETE SOME VALVES FROM THE TECHNICAL SPECIFICATIONS, THE VALVES WILL BE STROKE TIME TESTED IN ACCORDANCE WITH THE TECHNICAL SPECIFICATIONS. IN ADDITION, IT WAS NOTED THAT THE 1-E11-F049 HAD NOT ALWAYS MET STROKE TIME REQUIREMENTS. THE ROOT CAUSE FOR THIS EVENT WAS THE FAILURE TO ESTABLISH PROPER PROCEDURAL CONTROLS FOR THE TECHNICAL SPCCIFICATION REQUIRED TESTING FOR THESE VALVES. CORRECTIVE ACTIONS INCLUDE ENSURING THE F049 MEETS STROKE TIME REQUIREMENTS AND REVISING PROCEDURES TO IMPLEMENT APPROPRIATE STROKE TIME TESTING FOR THE AFFECTED VALVES. THE SAFETY SIGNIFICANCE OF THIS EVENT IS CONSIDERED MINIMAL.

[19] BRUNSWICK 1 DOCKET 50-325 LER 89-017 REV 01
UPDATE ON SPURIOUS ISOLATION OF HIGH PRESSURE COOLANT INJECTION CHANNEL A CAUSED BY SUSPECTED FAILURE OF ROSEMOUNT 510 DU TRIP UNIT.
EVENT DATE: 062489 REPORT DATE: 110689 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 215869) ON JUNE 24, 1989, AT 2105 HOURS, A HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM, STEAM LINE BREAK DELTA PRESSURE HIGH SIGNAL, A CHANNEL, ISOLATION OCCURRED ON UNIT 1. AT THE TIME OF THE EVENT THE UNIT WAS IN COLD SHUTDOWN. NO OTHER ACTIVITIES WERE IN PROGRESS, AND NO PERSONNEL WERE FOUND IN THE AREA OF THE INSTRUMENTATION WHICH INITIATES THE REFERENCED SIGNAL. IT IS BELIEVED THE ISOLATION WAS CAUSED BY A SPURIOUS TRIP OF REDUNDANT DIVISION 1 ECCS TRIP UNIT 1-E41-PDTM-N004-1. THE REFERENCED TRIP UNIT IS A ROSEMOUNT 510 DU, AND IT HAS BEEN REPLACED. A SIMILAR EVENT WAS REPORTED IN LER 2-89-008. A ROOT CAUSE INVESTIGATION OF THIS EVENT HAS BEEN INCONCLUSIVE. LIMERICK IS ALSO INVESTIGATING SIMILAR PROBLEMS WITH ROSEMOUNT 510 DU TRIP UNITS AND CONTACTS HAVE BEEN ESTABLISHED WITH BOTH LIMERICK AND ROSEMOUNT. FINAL CORRECTIVE ACTIONS WILL BE BASED UPON CONTINUING EFFORTS BY THE VENDOR AND THE INDUSTRY. THIS EVENT WOULD NOT HAVE BEEN MORE SEVERE UNDER REASONABLE AND CREDIBLE ALTERNATIVE CONDITIONS AS THE FAILURE MODE CAUSED THE SYSTEMS TO ACTUATE AS DESIGNED.

[20]   BRUNSWICK 1   DOCKET 50-325   LER 89-019 REV 01
UPDATE ON FAILURE OF THE SERVICE WATER SYSTEM TO MEET DESIGN REQUIREMENTS.
EVENT DATE: 091489   REPORT DATE: 111489   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215870) 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. SUBSEQUENT EVALUATIONS HAVE DETERMINED THIS ITEM TO BE REPORTABLE PER 10CFR21. THIS EVENT IS CONSIDERED TO HAVE POSSIBLY HAD A MAJOR SAFETY IMPACT.

[21]   BRUNSWICK 1   DOCKET 50-325   LER 89-020
LOSS OF CONTROL POWER AND OPERABILITY OF THE 1-E41-F006 VALVE WHILE REMOVING THE INDICATOR LIGHT BULB FROM THE MCC BREAKER COMPARTMENT.
EVENT DATE: 101189   REPORT DATE: 110789   NSSS: GE   TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215871) AT 1035 ON 10/11/89, WITH UNIT 1 AT 100% POWER, THE HPCI F006 INJECTION VALVE MOTOR CONTROL CENTER INDICATING LIGHT SOCKET SHORTED AND BLEW THE CONTROL POWER FUSE FOR THE F006 VALVE AS THE BULB WAS BEING REMOVED. THIS RESULTED IN A LOSS OF CONTROL POWER TO THE F006 VALVE, WHICH WOULD HAVE PREVENTED THE VALVE FROM AUTOMATICALLY OPENING IF THE HPCI SYSTEM WAS NEEDED. THE EVENT WAS CAUSED BY MOVEMENT OF THE NEGATIVE SOCKET COIL INTO CONTACT WITH THE POSITIVE SOCKET TAB DURING BULB REMOVAL. THE COIL MOVEMENT DURING REMOVAL IS ATTRIBUTED TO INWARD FORCE APPLIED TO THE BULB DURING REMOVAL. WHEN THE COIL CAME INTO CONTACT WITH THE TAB, THE RESULTING SHORT CAUSED THE LOSS OF CONTROL POWER TO THE F006 VALVE. CORRECTIVE ACTIONS INCLUDE REPLACEMENT OF THE LAMP, SOCKET AND FUSE AND A SAMPLING OF VARIOUS PLANT MCCS FOR SIMILAR COIL DISTORTION. AS A RESULT OF THIS INSPECTION, ADDITIONAL CORRECTIVE ACTIONS WERE IDENTIFIED, INCLUDING A REVIEW OF THE EVENT WITH APPROPRIATE OPERATING PERSONNEL, EVALUATION OF THE FEASIBILITY OF REPLACING THE CURRENT SOCKET WITH A DIFFERENT TYPE SOCKET, AND DEVELOPMENT OF AN INSPECTION PLAN FOR SAFETY RELATED MCCS AND TRANSFORMER SOCKETS IN THE SWITCHYARD AREA THIS EVENT WOULD NOT BE CONSIDERED MORE SIGNIFICANT UNDER REASONABLE AND CREDIBLE ALTERNATIVE CONDITIONS.

[22]   BRUNSWICK 2   DOCKET 50-324   LER 89-017
UNPLANNED ACTUATION OF REACTOR LOW LEVEL INSTRUMENTATION DUE TO SUSPECTED PERTURBATION OF REACTOR INSTRUMENTATION LINES WHILE RETURNING C32-PT-N008 TO SERVICE FOLLOWING CALIBRATION.
EVENT DATE: 101089   REPORT DATE: 110689   NSSS: GE   TYPE: BWR

(NSIC 215867) AT 1525 HOURS ON 10/10/89 INADVERTENT UNIT 2 LOW WATER LEVEL (LL) NO. 2 AND 3 SIGNALS OCCURRED CAUSING A GROUP 1 ISOLATION, CORE SPRAY (CS) INITIATION SIGNAL, AND AUTO-STARTING OF THE UNITS 1 AND 2 COMMON EMERGENCY DIESEL GENERATORS (DGS). THE UNIT 2 1989-1990 REFUEL/MAINTENANCE OUTAGE WAS ONGOING WITH THE REACTOR (RX) DEFUELED. THE CS, RESIDUAL HEAT REMOVAL (RHR), RX WATER CLEANUP (RWCU), STANDBY GAS TREATMENT, AND THE RX BUILDING VENTILATION SYSTEMS WERE UNDER EQUIPMENT CLEARANCE. RETURN TO SERVICE OF RX FEEDWATER/RX PRESSURE TRANSMITTER (PT) C32-PT-N008 WAS ONGOING. THE CONTROL OPERATOR BECAME AWARE OF THIS EVENT THROUGH CONTROL ROOM INDICATION/ALARM ANNUNCIATION. AN RHR/LOW PRESSURE COOLANT INJECTION (LPCI) INITIATION SIGNAL AND A GROUP 3 OF THE RWCU SYSTEM WERE NOT CONFIRMED. THE ISOLATION/INITIATION SIGNALS WERE RESET AND BY 1818 HOURS THE DGS WERE RETURNED TO STANDBY. THIS EVENT HAD MINIMAL SAFETY

SIGNIFICANCE. THE LL SIGNALS ARE ATTRIBUTED TO PERTURBATION OF THE INSTRUMENT SENSING LEG COMMON TO N008 AND RX LL INSTRUMENTATION WHEN N008 WAS RETURNED TO SERVICE, PER OPIC-PT001. AN INSTRUCTION/PRECAUTIONARY STATEMENT WAS NOT PROVIDED WITH THE PROCEDURE TO ENSURE THE TEST PRESSURE ON N008 WAS EQUAL TO THE SYSTEM PRESSURE CONDITIONS PRIOR TO RETURN TO SERVICE.

[23] BYRON 1 DOCKET 50-454 LER 89-008 REV 01
UPDATE ON AUXILIARY FEEDWATER SUCTION PRESSURE SWITCHES FOUND OUT OF CALIBRATION DUE TO FAILURE TO CONSIDER HEAD CORRECTION.
EVENT DATE: 083089 REPORT DATE: 110189 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 2 (PWR)

(NSIC 215920) THERE WERE TWO RELATED EVENTS INVOLVING THE FUNCTION AND DESIGN OF THE AUX. FEEDWATER (AF) PUMP SUCTION PRESSURE SWITCHES (AFSPSS). FIRST EVENT: DURING AN INSTITUTE OF NUCLEAR POWER OPERATIONS EVALUATION OF THE AUX. FEEDWATER SYSTEM IT WAS NOTED THAT THE ONE AF SUCTION PRESSURE TRANSMITTER (AFSPT) CALIBRATION WAS NOT HEAD CORRECTED. A NUCLEAR WORK REQUEST WAS WRITTEN 2/9/89 TO MAKE THE HEAD CORRECTION. LATER IT WAS DETERMINED THAT THE REMAINING THREE TRANSMITTERS MAY ALSO NEED HEAD CORRECTION. PER A LETTER FROM SARGENT AND LUNDY, RECEIVED ON 8/30, IT WAS DETERMINED THAT THE SWITCH SETPOINTS, CORRESPONDING TO 3 OF THE 4 AFSPTS, HAD NOT BEEN WITHIN TECH SPEC ALLOWABLE VALUES. ROOT CAUSE OF THE FIRST EVENT WAS THAT DESIGN DOCUMENTS AND PROCEDURES ADDRESSING INSTRUMENT INSTALLATION AND CALIBRATION DID NOT CLEARLY INDICATE THAT THE AFSPTS REQUIRED HEAD CORRECTION. SECOND EVENT: WHILE INVESTIGATING THE IMPACT OF THE FIRST EVENT ON SAFETY, THE BASES OF THE ORIGINAL SETPOINTS WERE QUESTIONED. THE SPECIFIC QUESTION WAS WHETHER THE AF PUMPS WOULD RUNDOWN THE CONDENSATE STORAGE TANK (CST) SUPPLY AND INDUCE AIR THROUGH THE SUCTION LINE PRIOR TO SWITCHOVER TO THE SAFETY RELATED SOURCE OR PUMP TRIP. THE ROOT CAUSE OF THE SECOND EVENT WAS THAT THE INSTRUMENT SETPOINT CALIBRATIONS DID NOT CONSIDER THE CUMULATIVE EFFECT OF PUMP SUCTION PRESSURE DROP WITH INCREASED INSTRUMENT UNCERTAINTY.

[24] BYRON 1 DOCKET 50-454 LER 89-009
INADEQUATE INCORPORATION OF STEAM GENERATOR BLOWDOWN ISOLATION REQUIREMENTS AS ASSUMED IN CERTAIN ACCIDENT ANALYSIS CAUSED BY A PRESERVICE DESIGN IMPLEMENTATION DEFICIENCY.
EVENT DATE: 100589 REPORT DATE: 110389 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 2 (PWR)

(NSIC 215921) DURING A REVIEW OF EROSION/CORROSION CONCERNS FOR THE STEAM GENERATOR BLOWDOWN (SD) LINES, CONFLICTING INFORMATION REGARDING SIGNALS THAT INITIATE AUTOMATIC ISOLATION OF THE SD LINES WAS DISCOVERED BY BYRON SITE PERSONNEL. THE SD ISOLATION WAS CONFIRMED TO OCCUR ON A CONTAINMENT PHASE A SIGNAL (GENERATED ON RECEIPT OF ANY SAFETY INJECTION SIGNAL (SIS)) AND AN AUX. BUILDING HIGH TEMPERATURE SIGNAL (HIGH ENERGY LINE BREAK CONSIDERATION). HOWEVER, THE UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) INDICATED THAT AUTOMATIC SD ISOLATION WOULD ALSO OCCUR ON OTHER SIGNALS. DISCUSSIONS WITH WESTINGHOUSE INDICATED THAT SD ISOLATION ON A SIS AND 2 OF 4 STEAM GENERATOR (SG) LEVEL LOW-LOW SIGNAL WERE ASSUMED IN CERTAIN ACCIDENT ANALYSES. THE EFFECT OF NOT ISOLATING SD WHEN REQUIRED IS TO REDUCE THE AUXILIARY FEEDWATER (AF) FLOW PROVIDED TO THE SG FOR COOLING. AS INTERIM CORRECTIVE ACTION SD WAS ISOLATED AT 12:00 ON 10/05/89 UNTIL TEMPORARY PROCEDURE CHANGES WERE IMPLEMENTED THAT REQUIRED MANUAL ISOLATION OF BLOWDOWN WHENEVER A REACTOR TRIP OCCURS TO REPLICATE THE UFSAR ACCIDENT ANALYSIS. CAUSE OF EVENT WAS A PRESERVICE DESIGN IMPLEMENTATION DEFICIENCY. THE SD SYSTEM FUNCTIONAL REQUIREMENTS IDENTIFIED SEVERAL SIGNALS TO PROVIDE SD ISOLATION, HOWEVER ISOLATION ON SG LEVEL LOW-LOW WAS NOT PHYSICALLY INCORPORATED FOR UNKNOWN REASONS.

[25] CALVERT CLIFFS 2 DOCKET 50-318 LER 89-002 REV 01
UPDATE ON INOPERABLE FIRE BARRIER PENETRATION CAUSED BY LACK OF ADEQUATE ADMINISTRATIVE WORK CONTROLS RESULTS IN CONDITION PROHIBITED BY TECH SPECS.
EVENT DATE: 022889 REPORT DATE: 111089 NSSS: CE TYPE: PWR

(NSIC 215861) ON FEBRUARY 28, 1989 AT APPROXIMATELY 0900, IT WAS DISCOVERED THAT

CALVERT CLIFFS UNIT 2 WAS IN AN ACTION STATEMENT ASSOCIATED WITH TECHNICAL SPECIFICATION (TS) 3.7.12 "PENETRATION FIRE BARRIERS". IT WAS LATER DETERMINED THAT A FIRE BARRIER PENETRATION WAS INOPERABLE AND THE ACTION STATEMENT REQUIREMENTS WERE NOT BEING SATISFIED AND THAT THE UNIT WAS BEING OPERATED IN A CONDITION PROHIBITED BY THE PLANT'S TS. AT THE TIME OF DISCOVERY OF THE INCIDENT THE PLANT WAS OPERATING AT 100% RATED THERMAL POWER. THE CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE INADEQUATE ADMINISTRATIVE CONTROLS TO ASSURE THAT INTERNAL CONDUIT FIRE SEALS ARE MAINTAINED INTACT DURING MAINTENANCE ACTIVITIES WHICH AFFECT OR COULD HAVE THE POTENTIAL TO AFFECT A FIRE SEAL. NO ADVERSE SAFETY CONSEQUENCES RESULTED FROM THIS EVENT. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO RETURN THE PENETRATION TO AN OPERABLE STATUS BY 0920 THAT SAME DAY. THIS PLACED THE UNIT IN COMPLIANCE WITH TS. TO PREVENT RECURRENCE A MEMO HAS BEEN ISSUED TO ALL SITE SUPERVISORS DISCUSSING INTERNAL CONDUIT SEALS AND REFERENCING ASSOCIATED SEALING REQUIREMENTS AS WELL AS APPLICABLE PROCEDURES TO FOLLOW TO ASSURE THAT COMPLIANCE IS MAINTAINED. RECENTLY PERFORMED FIRE BARRIER SURVEILLANCE TESTS HAVE CONFIRMED THAT THIS DOES NOT APPEAR TO BE A WIDESPREAD OR RECURRING PROBLEM.

[26]   CATAWBA 2   DOCKET 50-414   LER 89-019
THREE AUXILIARY FEEDWATER PUMPS INOPERABLE DUE TO DEFECTIVE PROCEDURE AND INAPPROPRIATE ACTIONS.
EVENT DATE: 091289   REPORT DATE: 101189   NSSS: WE   TYPE: PWR

(NSIC 215582) ON SEPTEMBER 12, 1989, AT 1335 HOURS, A CONTROL ROOM OPERATOR (CRO) TURNED THE CLOSE-AUTO-OPEN SWITCHES FOR 2CA-15A (CA PUMP 2A SUCT FROM RN ISOL VALVE) AND 2CA-18B (CA PUMP 2B SUCT FROM RN ISOL VALVE) FROM THE AUTO POSITION TO THE CLOSE POSITION. THIS ACTION DISABLED THE AUTOMATIC OPENING CAPABILITY OF 2CA-15A AND 2CA-18B, RENDERING AUXILIARY FEEDWATER (CA) PUMPS 2A AND 2B INOPERABLE. THE CA TURBINE DRIVEN PUMP WAS ALREADY INOPERABLE, DUE TO MODIFICATION WORK IN PROGRESS. THE AUDIBLE ALARM FOR THE 1.47 BYPASS PANEL WAS NOT HEARD WHEN THE SWITCHES WERE TURNED. THE CRO SUBSEQUENTLY NOTICED THAT THE LIGHTS FOR "CA TRAIN A BYPASSED" AND "CA TRAIN B BYPASSED" ON THE 1.47 BYPASS PANEL WERE LIT. AT 1410 HOURS, AFTER REVIEWING THE APPROPRIATE LOGIC DIAGRAMS, THE CRO RETURNED THE SWITCHES TO THE AUTO POSITION. PT/2/A/4200/59, RN TO CA SUCTION PIPING FLUSH HAS A STEP TO "ENSURE THE FOLLOWING VALVES ARE CLOSED", AND LISTS 2CA-15A AND 2LA-18B. THIS INCIDENT IS ATTRIBUTABLE TO A DEFECTIVE PROCEDURE, DUE TO THE INTENT OF THE STEP BEING UNCLEAR, AND TO INAPPROPRIATE ACTIONS, FOR PLACING BOTH 2CA-15A AND 2CA-18B IN THE CLOSE POSITION, AND FOR TURNING DOWN THE VOLUME LEVEL ON THE 1.47 BYPASS PANEL AUDIBLE ALARM. PT/2/A/4200/59 HAS BEEN REVISED, AND AN OPERATOR UPDATE HAS BEEN ISSUED. UNIT 2 WAS IN MODE 1, POWER OPERATION, AT THE TIME OF THIS INCIDENT.

[27]   CRYSTAL RIVER 3   DOCKET 50-302   LER 89-035
INADEQUATE DESIGN CONTROLS LEAD TO SAFETY RELATED COMPONENTS BEING SUBJECTED TO VOLTAGES HIGHER THAN MAXIMUM RATED VOLTAGE.
EVENT DATE: 090689   REPORT DATE: 111089   NSSS: BW   TYPE: PWR

(NSIC 215855) ON THE BASIS OF A RECOMMENDATION FROM THE B&W OWNERS GROUP "SAFETY AND PERFORMANCE IMPROVEMENT PROGRAM", AN INVESTIGATION OF DC POWERED COMPONENTS WAS CONDUCTED. A DISCREPANCY WAS FOUND IN THE COMPONENTS' RATED VOLTAGES AND THE ACTUAL VOLTAGES SEEN BY THE COMPONENTS. ON 9/6/89 THIS WAS DETERMINED TO BE AN OPERATION OUTSIDE THE DESIGN BASIS FOR CRYSTAL RIVER UNIT 3. AT THE TIME THIS DETERMINATION WAS MADE, THE PLANT WAS IN MODE 5, COLD SHUTDOWN, FOR MAINTENANCE NOT RELATED TO THE DC POWER SYSTEM. EACH COMPONENT WAS EXAMINED TO DETERMINE IF IT WAS OPERABLE. A TOTAL OF 270 COMPONENTS WERE FOUND TO BE AFFECTED, AND 32 WERE DETERMINED TO BE INOPERABLE. ALL INOPERABLE COMPONENTS WERE REPLACED PRIOR TO THE UNIT START-UP ON 10/13/89. ALL OTHER AFFECTED DC COMPONENTS ARE BEING EVALUATED FOR FUTURE REPLACEMENT. THIS EVENT WAS CAUSED BY INADEQUATE CONTROL OF THE ENGINEERING DESIGN PROCESS BY FLORIDA POWER. FUTURE DESIGN PROCESSES WILL UTILIZE THE "DESIGN BASIS DOCUMENT" TO CORRECTLY SPECIFY THE REQUIRED VOLTAGE RANGE. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)II)(B).

[28] CRYSTAL RIVER 3 DOCKET 50-302 LER 89-033
INCORRECT DESIGN ASSUMPTION CAUSES INADEQUATE SETPOINT FOR SECOND LEVEL UNDERVOLTAGE RELAY SYSTEM AND LEADS TO OPERATION OUTSIDE THE PLANT DESIGN BASIS.
EVENT DATE: 090889 REPORT DATE: 111089 NSSS: BW TYPE: PWR

(NSIC 215854) CRYSTAL RIVER UNIT 3 WAS IN MODE 5 (COLD SHUTDOWN) REPLACING A FAULTED ENGINEERED SAFEGUARDS (ES) TRANSFORMER. ON 9/8/89, DURING THE DESIGN REVIEW OF THE REPLACEMENT 480 VOLT TRANSFORMER, DESIGN ENGINEERS DISCOVERED THAT THE SECOND LEVEL UNDERVOLTAGE RELAY (SLUR) SYSTEM SETPOINT FOR THE ENGINEERED SAFEGUARDS BUSES WAS NOT CONSERVATIVE. THE SETPOINT HAD BEEN CALCULATED WITHOUT CONSIDERING THE VOLTAGE DROP BETWEEN THE ENGINEERED SAFEGUARDS MOTOR CONTROL CENTERS AND THEIR ASSOCIATED LOADS. THE SETPOINT CALCULATED TO CORRECT THIS PROBLEM WAS INITALLY SET TOO HIGH. AN ES ACTUATION WHILE SUPPLYING NORMAL PLANT LOADS WOULD HAVE CAUSED A SPURIOUS SEPARATION FROM THE OFFSITE POWER SUPPLY. THESE ERRORS WERE CAUSED BY PERSONNEL ERROR. THE SLUR SETPOINT WAS RECALCULATED CONSIDERING THE VOLTAGE DROPS AND THE SPURIOUS SEPARATION ISSUE AND CHANGED ON 10/9/89. ADDITIONALLY, FLORIDA POWER WILL BE REVALIDATING THE MAJOR SAFETY RELATED ES BUS CALCULATIONS.

[29] CRYSTAL RIVER 3 DOCKET 50-302 LER 89-036
ERRORS IN THE SAFETY LISTING CAUSED PROCUREMENT AND INSTALLATION OF UNQUALIFIED EQUIPMENT RESULTING IN OPERATION OUTSIDE THE DESIGN BASIS.
EVENT DATE: 101889 REPORT DATE: 111789 NSSS: BW TYPE: PWR

(NSIC 215856) ON 10/18/89 AT 1810 CRYSTAL RIVER UNIT THREE WAS IN OPERATIONAL MODE 1 (POWER OPERATION). AT THAT TIME IT WAS DETERMINED THE PLANT WAS OPERATING OUTSIDE ITS DESIGN BASIS, BECAUSE BOTH OF THE BORATED WATER STORAGE TANK LEVEL TRANSMITTERS WHICH PROVIDE SIGNALS TO THE CONTROL ROOM WERE NOT SEISMICALLY QUALIFIED. UNQUALIFIED TRANSMITTERS HAD BEEN PURCHASED AND INSTALLED BECAUSE THE PLANT SAFETY LISTING, WHICH WAS USED TO DETERMINE THE PURCHASING SPECIFICATIONS, INCORRECTLY IDENTIFIED THE SAFETY FUNCTION OF THE TRANSMITTERS AS PRESSURE RETENTION ONLY. THE TRANSMITTERS HAVE BEEN REPLACED WITH QUALIFIED TRANSMITTERS. THE SAFETY LISTING IS BEING REVIEWED TO ENSURE THAT SIMILAR PROBLEMS DO NOT EXIST.

[30] DAVIS-BESSE 1 DOCKET 50-346 LER 89-015
REACTOR COOLANT SYSTEM FLOW TRANSMITTER ERRONEOUSLY DECLARED OPERABLE.
EVENT DATE: 092489 REPORT DATE: 112789 NSSS: BW TYPE: PWR

(NSIC 215945) ON 10/26/89, WHILE REVIEWING QUESTIONS RAISED ABOUT ACTIVITIES ASSOCIATED WITH THE CALIBRATION OF FTRC01A2, IT WAS CONCLUDED THAT THE TRANSMITTER WAS OUT OF TOLERANCE WHEN RPS CHANNEL 2 WAS RETURNED TO OPERABLE STATUS ON 9/24/89. RPS CHANNEL 2 WAS INOPERABLE FOR ABOUT 38 HOURS WITHOUT SATISFYING THE ACTION STATEMENT OF TECH SPEC 3.3.1.1. THE MOST PROBABLE CAUSE OF THE CALIBRATION PROBLEM WAS A LEAK-BY IN THE INSTRUMENT MANIFOLD THAT WAS INCORRECTLY SEEN AS A ZERO SHIFT. CONSEQUENTLY, WHEN THE TRANSMITTER WAS VALVED BACK INTO THE RCS FLOW PROCESS AND THE EQUALIZING VALVE TIGHTENED, THE TRANSMITTER OUTPUT WAS ERRONEOUSLY HIGH. ON 10/3/89, THE TRANSMITTER WAS RECALIBRATED SUCCESSFULLY. AN ENGINEERING REVIEW OF THE PROCESS READINGS WAS CONDUCTED AND THE CHANNEL RESTORED TO OPERABLE. INDIVIDUALS INVOLVED HAVE REVIEWED THE INCIDENT AND RESULTS FOR LESSONS LEARNED.

[31] DAVIS-BESSE 1 DOCKET 50-346 LER 89-014
INOPERABLE FIRE DETECTION ZONE 237 AND 323 DUE TO INSTALLATION OF PLASTIC SHEETING.
EVENT DATE: 092589 REPORT DATE: 111089 NSSS: BW TYPE: PWR

(NSIC 215875) ON 10/11/89, AT 1545 HOURS, A MODIFICATION SUPERVISOR REPORTED TO THE SHIFT SUPERVISOR THAT PLASTIC SHEETING WAS BLOCKING OFF SOME FIRE DETECTORS IN ROOM 237 (AUX. FEEDWATER PUMP 1-1 ROOM) AND ROOM 323 (B HIGH VOLTAGE SWITCHGEAR ROOM). FURTHER INVESTIGATION DETERMINED THAT THIS PLASTIC SHEETING HAD BEEN INSTALLED SINCE 9/25/89, TO PROTECT SURROUNDING EQUIPMENT DURING THE REMOVAL OF OLD AND INSTALLATION OF NEW STRUCTURAL STEEL FIRE PROOFING MATERIAL.

HOWEVER, THE PLASTIC SHEETING HAD IN EFFECT MADE THE DETECTORS INOPERABLE BY SEPARATING THEM FROM THE REST OF THE ROOM ATMOSPHERE. THE PLASTIC SHEETING WAS SUBSEQUENTLY REMOVED BY 1615 HOURS. SOME OF THESE ROOM FIRE BARRIERS HAD PREVIOUSLY BEEN DECLARED INOPERABLE, AND WITH NO DETECTION ON THE OPPOSITE SIDE OF SOME OF THESE BARRIERS, A CONTINUOUS FIRE WATCH WAS REQUIRED. SINCE THESE ROOMS WERE ONLY BEING CHECKED BY AN HOURLY FIRE WATCH PATROL, THE APPROPRIATE ACTION STATEMENT WAS NOT BEING MET. THE CAUSE WAS PERSONNEL OVERSIGHT. THE SUPERVISOR OF THE INSULATING CONTRACTOR PERSONNEL WHO INSTALLED THE PLASTIC SHEETING FAILED TO RECOGNIZE THAT HE WAS SEALING OFF THE DETECTORS WHICH RENDERED THEM INOPERABLE. THE MODIFICATION COORDINATORS AND SUPERVISORS WERE COUNSELED ON THIS INCIDENT. THE NEED TO BE SENSITIVE TO AND QUESTION THE EFFECTS OF THEIR ONGOING WORK ON PLANT EQUIPMENT AND SYSTEMS WAS STRESSED.

[32] DIABLO CANYON 1 DOCKET 50-275 LER 89-009
SAFETY INJECTION AND REACTOR TRIP FROM STEAM LINE DIFFERENTIAL PRESSURE SPURIOUS SIGNALS.
EVENT DATE: 100689 REPORT DATE: 110689 NSSS: WE TYPE: PWR
VENDOR: AGASTAT RELAY CO.
COPES-VULCAN, INC.
ITT-BARTON

(NSIC 215841) ON OCTOBER 6, 1989, AT 1302 PDT, WITH UNIT 1 IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER, AN AUTOMATIC SAFETY INJECTION/REACTOR TRIP WAS ACTUATED FROM A STEAM LINE DIFFERENTIAL PRESSURE SIGNAL. AT 1321 PDT, A 4-HOUR NON-EMERGENCY REPORT WAS MADE IN ACCORDANCE WITH 10 CFR 50.72(B)(2)(II). DURING THE REMOVAL FROM SERVICE PROCESS FOR CALIBRATION OF A PRESSURE TRANSMITTER FOR AN ATMOSPHERIC STEAM DUMP VALVE (ADV) CONTROL, PRESSURE OSCILLATIONS WERE CREATED IN THE COMMON SENSING LINE WITH A PROTECTION SET STEAM GENERATOR PRESSURE TRANSMITTER. THIS CAUSED REPEATED ACTUATIONS OF A STEAM LINE DIFFERENTIAL PRESSURE BISTABLE. THE STEAM LINE DIFFERENTIAL PRESSURE BISTABLE ACTUATION, COMBINED WITH A PREVIOUSLY TRIPPED STEAM PRESSURE BISTABLE, SATISFIED THE 2/3 COINCIDENCE LOGIC TO GENERATE A STEAM LINE DIFFERENTIAL PRESSURE SI SIGNAL. THE I&C REMOVAL FROM SERVICE/CALIBRATION PROCEDURE WAS INADEQUATE IN THAT IT DID NOT VERIFY THE CONDITION OF OTHER PROTECTION AND SAFEGUARDS CHANNEL BISTABLES TO ENSURE THAT COINCIDENT LOGIC COULD NOT BE SATISFIED. THE I&C PROCEDURES WILL BE REVISED TO ENSURE THAT WORK ON EQUIPMENT THAT SHARES A COMMON PROCESS TAP IS PERFORMED ONLY FOLLOWING ADEQUATE SITUATION EVALUATION AND AUTHORIZATION.

[33] DIABLO CANYON 1 DOCKET 50-275 LER 89-008
CONTROL ROOM VENTILATION SYSTEM SHIFT TO PRESSURIZATION MODE DURING TRANSFER OF SHIELDED CASK DUE TO INADEQUATE PROCEDURAL GUIDANCE.
EVENT DATE: 101289 REPORT DATE: 111089 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: DIABLO CANYON 2 (PWR)

(NSIC 215840) ON OCTOBER 12, 1989 AT 2140 PDT, WITH UNIT 1 IN MODE 6 AND UNIT 2 IN MODE 1 (POWER OPERATION) THE CONTROL ROOM VENTILATION SYSTEM AUTOMATICALLY SHIFTED FROM ITS NORMAL VENTILATION MODE TO THE PRESSURIZATION MODE, AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION. THIS ESF WAS DUE TO A HIGH RADIATION SIGNAL FROM UNIT 1 RADIATION DETECTORS RE-25 AND 26. THE HIGH RADIATION SIGNAL WAS GENERATED WHEN THE DETECTORS WERE EXPOSED TO RADIATION DURING THE TRANSFER OF RADIOACTIVE MATERIAL FROM THE REACTOR REFUELING CAVITY TO A SHIELDED CASK OUTSIDE THE CONTAINMENT EQUIPMENT HATCH. IN ACCORDANCE WITH 10CFR 50.72, A FOUR HOUR, NON-EMERGENCY REPORT WAS MADE ON OCTOBER 13, 1989 AT 0027 PDT. RADIATION PROTECTION MANAGEMENT WILL ISSUE MEMORANDA THAT ADVISES SUPERVISORS TO CONSIDER THE IMPACT OF THE MOVEMENT OF RADIOACTIVE MATERIAL ON INSTALLED PLANT RADIATION MONITORS. IN ADDITION, THE RADIATION PROCEDURE RCP D-200, "WRITING RADIATION WORK PERMITS," WILL BE REVISED TO INCLUDE SPECIFIC INSTRUCTIONS TO AVOID ESF ACTUATIONS THAT OCCUR DURING MOVEMENT OF RADIOACTIVE MATERIALS.

[34] DRESDEN 2 DOCKET 50-237 LER 89-028
CONTAINMENT COOLING SERVICE WATER PUMP SUCTION BAY WATER LEVEL REDUCTION DUE TO INTAKE STRUCTURE FLOW BLOCKAGE.
EVENT DATE: 090589 REPORT DATE: 110389 NSSS: GE TYPE: BWR

OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 215814) AT 0910 HOURS ON 9/5/89, WITH UNIT 2 AT 98.6% POWER AND UNIT 3 AT 88.1% POWER WHILE ATTEMPTING TO RUN THE 2A CONTAINMENT COOLING SERVICE WATER (CCSW) PUMP IN PREPARATION FOR DRESDEN OPERATING SURVEILLANCE (DOS) 1500-6, LOW PRESSURE COOLANT INJECTION (LPCI) SYSTEM PUMP OPERABILITY TEST, CCSW FLOW OSCILLATIONS WERE OBSERVED. IMMEDIATE INVESTIGATION FOUND THAT A LARGE AMOUNT OF DEBRIS HAD ACCUMULATED IN THE CRIBHOUSE INTAKE CANAL, RESULTING IN LOWERING OF THE CCSW SUCTION BAY WATER LEVEL. THE UNIT 2 AND UNIT 3 CCSW PUMPS WERE THEN DECLARED INOPERABLE AS A CONSERVATIVE MEASURE. THE 2A CIRCULATING WATER PUMP WAS SECURED TO REDUCE THE WATER HEAD DIFFERENTIAL ACROSS THE INTAKE STRUCTURE BAR RACKS, AND INTENSIVE EFFORTS WERE INITIATED TO REMOVE THE DEBRIS, WHICH HAD RESULTED FROM HEAVY RAINS AND FLOODING IN THE KANKAKEE RIVER BASIN AREA. THE UNIT 2 CCSW PUMPS WERE FUNCTIONALLY TESTED OPERABLE AT 1325 HOURS, AND THE UNIT 3 CCSW PUMPS WERE TESTED OPERABLE AT 1800 HOURS. SAFETY SIGNIFICANCE WAS MINIMAL DUE TO IMMEDIATE ACTIONS TO ENSURE ADEQUATE CCSW SUCTION BAY WATER LEVEL. THIS WAS THE FIRST OCCURRENCE OF INTAKE CANAL DEBRIS RESULTING IN SIGNIFICANT REDUCTION OF CCSW PUMP SUCTION BAY LEVEL.

[35] DRESDEN 2 DOCKET 50-237 LER 89-027
POSTULATED LPCI SWING BUS LOSS RESULTING FROM DIESEL GENERATOR VOLTAGE REGULATOR FAILURE DUE TO DESIGN DEFICIENCY.
EVENT DATE: 101289 REPORT DATE: 110989 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 215817) ON 10/12/89 WITH UNIT 2 AND UNIT 3 IN RUN MODE NEAR FULL RATED CORE THERMAL POWER, IT WAS DETERMINED THAT A POSTULATED FAILURE OF THE DIESEL GENERATOR (DG) VOLTAGE REGULATORS DURING A DESIGN BASIS LOSS OF COOLANT ACCIDENT (LOCA), CONCURRENT WITH A LOSS OF OFFSITE POWER (LOOP), COULD RENDER LOW PRESSURE COOLANT INJECTION (LPCI) SYSTEM MOTOR CONTROL CENTERS (MCCS) 29-7/28-7 (UNIT 2) OR 39-7/38-7 (UNIT 3) INCAPABLE OF PERFORMING THEIR INTENDED FUNCTION. IT IS POSTULATED THAT DURING THE LOCA/LOOP, OUTPUT VOLTAGE REGULATOR FAILURE OCCURS FOLLOWING THE START AND INITIAL LOADING OF UNIT DG. POSTULATED REGULATOR FAILURE IS SUCH THAT VOLTAGE IS MAINTAINED BELOW THAT REQUIRED TO CARRY LPCI MCC LOADS, BUT GREATER THAN THE VOLTAGE REQUIRED TO INITIATE THE AUTOMATIC LOW VOLTAGE MCC TRANSFER LOGIC. THIS THEN RESULTS IN THE LOSS OF LPCI SUBSYSTEM AND THE DIVISION II CORE SPRAY SYSTEM. ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO AN ORIGINAL CONSTRUCTION DESIGN DEFICIENCY. SAFETY SIGNIFICANCE OF THIS POSTULATED FAILURE IS MINIMAL BECAUSE THIS SEQUENCE OF EVENTS IS UNIQUE AND EXTREMELY REMOTE. CORRECTIVE ACTION INCLUDED DEVELOPMENT OF AN OPERATOR RESPONSE PROCEDURE, OPERATOR TRAINING, AND INITIATION OF DESIGN REVIEWS. THIS CONCERN HAD NOT BEEN PREVIOUSLY IDENTIFIED AT THE DRESDEN FACILITY. THIS EVENT IS BEING REPORTED AS A VOLUNTARY LER DUE TO ITS GENERIC APPLICABILITY.

[36] FARLEY 1 DOCKET 50-348 LER 89-005
CONTAINMENT INTEGRITY BREACH CAUSED BY PERSONNEL ERROR.
EVENT DATE: 093089 REPORT DATE: 110189 NSSS: WE TYPE: PWR

(NSIC 215876) ON 09/30/89 AND AGAIN ON 10/01/89, WITH THE UNIT SHUTDOWN FOR A REFUELING OUTAGE, CORE ALTERATIONS WERE PERFORMED WITHOUT HAVING ESTABLISHED CONTAINMENT REFUELING INTEGRITY AS REQUIRED BY TECH SPEC 3.9.4. CORE ALTERATIONS WERE PERFORMED AT 2200 ON 09/30/89 WHEN THE REACTOR VESSEL HEAD WAS LIFTED AND FROM 1245 TO 1948 ON 10/01/89 WHEN THE CONTROL ROD DRIVE MECHANISMS WERE UNLATCHED. DURING THESE TIMES THERE WAS A PATHWAY FROM CONTAINMENT TO THE OUTSIDE ATMOSPHERE THROUGH AT LEAST 1 STEAM GENERATOR (S/G) SECONDARY SIDE MANWAY AND S/G ATMOSPHERIC RELIEF VALVE. THE ACTION STATEMENT ASSOCIATED WITH TECH SPEC 3.9.4 REQUIRES THE IMMEDIATE SUSPENSION OF CORE ALTERATIONS IN CONTAINMENT WHENEVER CONTAINMENT REFUELING INTEGRITY IS NOT MAINTAINED. CORE ALTERATIONS CONTINUED SINCE IT WAS NOT REALIZED THAT CONTAINMENT REFUELING INTEGRITY WAS NOT IN EFFECT. THIS EVENT WAS CAUSED BY COGNITIVE PERSONNEL ERROR IN THAT THE SHIFT FOREMAN AND SHIFT SUPERVISOR WHO PERFORMED THE CONTAINMENT INTEGRITY PROCEDURE FAILED TO PROPERLY VERIFY THE POSITION OF THE ATMOSPHERIC RELIEF VALVES BY USING LOCAL INDICATION AS REQUIRED BY FNP-0-AP-16 (CONDUCT OF OPERATIONS - OPERATIONS GROUP). THE SHIFT FOREMAN AND SHIFT SUPERVISOR INVOLVED IN THIS EVENT HAVE BEEN

OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 215814) AT 0910 HOURS ON 9/5/89, WITH UNIT 2 AT 98.6% POWER AND UNIT 3 AT 88.1% POWER WHILE ATTEMPTING TO RUN THE 2A CONTAINMENT COOLING SERVICE WATER (CCSW) PUMP IN PREPARATION FOR DRESDEN OPERATING SURVEILLANCE (DOS) 1500-6, LOW PRESSURE COOLANT INJECTION (LPCI) SYSTEM PUMP OPERABILITY TEST, CCSW FLOW OSCILLATIONS WERE OBSERVED. IMMEDIATE INVESTIGATION FOUND THAT A LARGE AMOUNT OF DEBRIS HAD ACCUMULATED IN THE CRIBHOUSE INTAKE CANAL, RESULTING IN LOWERING OF THE CCSW SUCTION BAY WATER LEVEL. THE UNIT 2 AND UNIT 3 CCSW PUMPS WERE THEN DECLARED INOPERABLE AS A CONSERVATIVE MEASURE. THE 2A CIRCULATING WATER PUMP WAS SECURED TO REDUCE THE WATER HEAD DIFFERENTIAL ACROSS THE INTAKE STRUCTURE BAR RACKS, AND INTENSIVE EFFORTS WERE INITIATED TO REMOVE THE DEBRIS, WHICH HAD RESULTED FROM HEAVY RAINS AND FLOODING IN THE KANKAKEE RIVER BASIN AREA. THE UNIT 2 CCSW PUMPS WERE FUNCTIONALLY TESTED OPERABLE AT 1325 HOURS, AND THE UNIT 3 CCSW PUMPS WERE TESTED OPERABLE AT 1800 HOURS. SAFETY SIGNIFICANCE WAS MINIMAL DUE TO IMMEDIATE ACTIONS TO ENSURE ADEQUATE CCSW SUCTION BAY WATER LEVEL. THIS WAS THE FIRST OCCURRENCE OF INTAKE CANAL DEBRIS RESULTING IN SIGNIFICANT REDUCTION OF CCSW PUMP SUCTION BAY LEVEL.

[35] DRESDEN 2 DOCKET 50-237 LER 89-027
POSTULATED LPCI SWING BUS LOSS RESULTING FROM DIESEL GENERATOR VOLTAGE REGULATOR FAILURE DUE TO DESIGN DEFICIENCY.
EVENT DATE: 101289 REPORT DATE: 110989 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 215817) ON 10/12/89 WITH UNIT 2 AND UNIT 3 IN RUN MODE NEAR FULL RATED CORE THERMAL POWER, IT WAS DETERMINED THAT A POSTULATED FAILURE OF THE DIESEL GENERATOR (DG) VOLTAGE REGULATORS DURING A DESIGN BASIS LOSS OF COOLANT ACCIDENT (LOCA), CONCURRENT WITH A LOSS OF OFFSITE POWER (LOOP), COULD RENDER LOW PRESSURE COOLANT INJECTION (LPCI) SYSTEM MOTOR CONTROL CENTERS (MCCS) 29-7/28-7 (UNIT 2) OR 39-7/38-7 (UNIT 3) INCAPABLE OF PERFORMING THEIR INTENDED FUNCTION. IT IS POSTULATED THAT DURING THE LOCA/LOOP, OUTPUT VOLTAGE REGULATOR FAILURE OCCURS FOLLOWING THE START AND INITIAL LOADING OF UNIT DG. POSTULATED REGULATOR FAILURE IS SUCH THAT VOLTAGE IS MAINTAINED BELOW THAT REQUIRED TO CARRY LPCI MCC LOADS, BUT GREATER THAN THE VOLTAGE REQUIRED TO INITIATE THE AUTOMATIC LOW VOLTAGE MCC TRANSFER LOGIC. THIS THEN RESULTS IN THE LOSS OF LPCI SUBSYSTEM AND THE DIVISION II CORE SPRAY SYSTEM. ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO AN ORIGINAL CONSTRUCTION DESIGN DEFICIENCY. SAFETY SIGNIFICANCE OF THIS POSTULATED FAILURE IS MINIMAL BECAUSE THIS SEQUENCE OF EVENTS IS UNIQUE AND EXTREMELY REMOTE. CORRECTIVE ACTION INCLUDED DEVELOPMENT OF AN OPERATOR RESPONSE PROCEDURE, OPERATOR TRAINING, AND INITIATION OF DESIGN REVIEWS. THIS CONCERN HAD NOT BEEN PREVIOUSLY IDENTIFIED AT THE DRESDEN FACILITY. THIS EVENT IS BEING REPORTED AS A VOLUNTARY LER DUE TO ITS GENERIC APPLICABILITY.

[36] FARLEY 1 DOCKET 50-348 LER 89-005
CONTAINMENT INTEGRITY BREACH CAUSED BY PERSONNEL ERROR.
EVENT DATE: 093089 REPORT DATE: 110189 NSSS: WE TYPE: PWR

(NSIC 215876) ON 09/30/89 AND AGAIN ON 10/01/89, WITH THE UNIT SHUTDOWN FOR A REFUELING OUTAGE, CORE ALTERATIONS WERE PERFORMED WITHOUT HAVING ESTABLISHED CONTAINMENT REFUELING INTEGRITY AS REQUIRED BY TECH SPEC 3.9.4. CORE ALTERATIONS WERE PERFORMED AT 2200 ON 09/30/89 WHEN THE REACTOR VESSEL HEAD WAS LIFTED AND FROM 1245 TO 1948 ON 10/01/89 WHEN THE CONTROL ROD DRIVE MECHANISMS WERE UNLATCHED. DURING THESE TIMES THERE WAS A PATHWAY FROM CONTAINMENT TO THE OUTSIDE ATMOSPHERE THROUGH AT LEAST 1 STEAM GENERATOR (S/G) SECONDARY SIDE MANWAY AND S/G ATMOSPHERIC RELIEF VALVE. THE ACTION STATEMENT ASSOCIATED WITH TECH SPEC 3.9.4 REQUIRES THE IMMEDIATE SUSPENSION OF CORE ALTERATIONS IN CONTAINMENT WHENEVER CONTAINMENT REFUELING INTEGRITY IS NOT MAINTAINED. CORE ALTERATIONS CONTINUED SINCE IT WAS NOT REALIZED THAT CONTAINMENT REFUELING INTEGRITY WAS NOT IN EFFECT. THIS EVENT WAS CAUSED BY COGNITIVE PERSONNEL ERROR IN THAT THE SHIFT FOREMAN AND SHIFT SUPERVISOR WHO PERFORMED THE CONTAINMENT INTEGRITY PROCEDURE FAILED TO PROPERLY VERIFY THE POSITION OF THE ATMOSPHERIC RELIEF VALVES BY USING LOCAL INDICATION AS REQUIRED BY FNP-0-AP-16 (CONDUCT OF OPERATIONS - OPERATIONS GROUP). THE SHIFT FOREMAN AND SHIFT SUPERVISOR INVOLVED IN THIS EVENT HAVE BEEN

COUNSELED. THE CONTAINMENT INTEGRITY PROCEDURE HAS BEEN STRENGTHENED TO ENSURE CONTAINMENT INTEGRITY IS PROPERLY ESTABLISHED.

[37] FARLEY 2 DOCKET 50-364 LER 89-012
REACTOR TRIP CAUSED BY DESIGN ERROR IN THE DIGITAL ELECTRO-HYDRAULIC CONTROL SYSTEM.
EVENT DATE: 101889 REPORT DATE: 111489 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: FARLEY 1 (PWR)

(NSIC 215888) AT 1407 ON 10-18-89, WITH THE UNIT OPERATING AT 100% POWER, THE REACTOR TRIPPED DUE TO LO-LO STEAM GENERATOR LEVEL. THE LO-LO STEAM GENERATOR LEVEL OCCURRED WHEN THE TURBINE GENERATOR GOVERNOR VALVES CLOSED. THIS CLOSURE RESULTED FROM A DIGITAL ELECTRO-HYDRAULIC CONTROL (DEHC) SYSTEM DESIGN FEATURE ENHANCEMENT WHICH CAUSES THE GOVERNOR VALVE POSITION LIMITER TO LOWER TO ZERO AS A BACKUP TO CLOSING THE GOVERNOR VALVES WHEN BOTH OVERSPEED PROTECTION CHANNEL (OPC) POWER SUPPLIES FAIL. THIS EVENT WAS CAUSED BY A DESIGN ERROR IN THE DEHC SYSTEM. THIS ERROR ALLOWED A MOMENTARY DATA LOSS TO BE CREATED WHEN THE OPERATOR REQUESTED THE "RE-ENABLE HIGHWAY" FUNCTION AT THE DEHC OPERATORS/ALARMS CONSOLE. THIS RESULTED IN THE SAME SYSTEM CONTROL ACTION THAT IS TAKEN WHEN BOTH OPC POWER SUPPLIES FAIL. THE VALVE POSITION LIMITER FEATURE WHICH LOWERS TO ZERO AS A BACKUP TO CLOSING THE GOVERNOR VALVES WHEN THE OPC POWER SUPPLIES FAIL HAS BEEN REMOVED FROM THE UNIT 1 AND UNIT 2 DEHC SYSTEMS. WESTINGHOUSE HAS BEEN DIRECTED TO DETERMINE IF THERE ARE OTHER DEHC FEATURES AVAILABLE TO THE OPERATOR WHICH MAY CAUSE A TURBINE TRIP OR TURBINE VALVE CLOSURE.

[38] FARLEY 2 DOCKET 50-364 LER 89-013
PERSONNEL ERROR CAUSES REACTOR TRIP ON LO-LO STEAM GENERATOR LEVEL.
EVENT DATE: 101989 REPORT DATE: 111489 NSSS: WE TYPE: PWR

(NSIC 215889) AT 1823 ON 10-19-89, WITH THE UNIT OPERATING AT APPROXIMATELY TWO PERCENT POWER, THE REACTOR TRIPPED DUE TO LO-LO LEVEL IN THE 2C STEAM GENERATOR. THE UNIT OPERATOR IMPROPERLY TRANSFERRED FROM AUXILIARY FEEDWATER (AFW) TO MAIN FEEDWATER (MFW) DURING A PLANT STARTUP. THIS EVENT WAS CAUSED BY PERSONNEL ERROR IN THAT THE UNIT OPERATOR INCORRECTLY TRANSFERRED STEAM GENERATOR LEVEL CONTROL FROM AFW TO MFW. A CONTRIBUTING CAUSE WAS THAT THE PERSONNEL INVOLVED FAILED TO PROPERLY RESTORE AIR TO THE 2C MFW BYPASS VALVE. THE PERSONNEL INVOLVED HAVE BEEN COUNSELED CONCERNING IMPROPER FEEDWATER TRANSFER AND RESTORATION OF AIR TO FEEDWATER VALVES. A TRAINING CHANGE NOTICE DESCRIBING THIS EVENT WILL BE ISSUED TO ALL LICENSED PERSONNEL.

[39] FERMI 2 DOCKET 50-341 LER 87-045 REV 01
UPDATE ON LOW PRESSURE COOLANT INJECTION SWING BUS DESIGN FLAW IDENTIFIED BY PERSONNEL ERROR.
EVENT DATE: 090887 REPORT DATE: 110689 NSSS: GE TYPE: BWR
VENDOR: PORTEC INCORPORATED

(NSIC 215819) ON 9/8/87, AN OPERATOR, REMOVED A FUSE WHICH DEENERGIZED THE DC CONTROL POWER TO THE BUS 72C WHILE ATTEMPTING TO DEENERGIZE A COMPONENT FOR MAINTENANCE ACTIVITIES. THIS BUS IS THE NORMAL FEED TO THE LOW PRESSURE COOLANT INJECTION (LPCI) SWING BUS. THE LOSS OF DC CONTROL POWER RESULTED IN THE LOSS OF THE POWER SUPPLY TO THE SWING BUS AND CONSEQUENTLY TO THE LPCI LOOP SELECTION VALVES. SUBSEQUENTLY, THE DESIGN WAS REVIEWED BY NUCLEAR ENGINEERING AND IT WAS DETERMINED THAT THE DC CONTROL CIRCUITRY FOR MCC 72CF EQUIPMENT WAS INADEQUATE. THE LPCI OPERATION COULD BE PREVENTED BY EITHER OF TWO INDEPENDENT FAILURES EXTERNAL TO THE SWING BUS. THE DC CONTROL CIRCUITRY FOR THE MCC 72CF HAS BEEN REDESIGNED TO MEET THE PLANTS DESIGN BASIS. IN 10/89 A NEW SCENARIO WAS POSTULATED WHICH INVOLVED DEGRADATION OF THE VOLTAGE ON THE 480 V BUS SUPPLYING MCC 72CF SUCH THAT THE SWING BUS LOADS MIGHT NOT OPERATE PROPERLY AND THE BUS WOULD NOT TRANSFER TO ITS ALTERNATE SUPPLY.

[40] FERMI 2 DOCKET 50-341 LER 89-026
CONTROL CENTER HEATING VENTILATING AND AIR CONDITIONING SHIFTS TO RECIRCULATION MODE BECAUSE OF LOSS OF CONTROL POWER.
EVENT DATE: 101089 REPORT DATE: 110989 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215847) WHILE DETERMINING THE CAUSE OF AN INOPERABLE POSITION INDICATING LAMP, A MOMENTARY SHORT CIRCUIT CAUSED BY A FAILED CAMP FILAMENT OPENED THE CIRCUIT'S PROTECTIVE RELAY. THIS RESULTED IN A LOSS OF POSITION INDICATION FOR DIVISION II OF CONTROL CENTER HEATING VENTILATING AND AIR CONDITIONING SYSTEM (CCHVAC). PER DESIGN CHVAC SHIFTED TO RECIRCULATION MODE OF OPERATION. THE CAUSE OF THIS EVENT WAS THE FAILURE OF THE LAMP FILAMENT WHEN IT WAS BEING TIGHTENED. THIS DREW AN EXCESSIVE CURRENT WHICH OPENED THE FUSE. THE FUSE AND THE BURNED OUT LAMP WERE REPLACED AND THE SYSTEM RETURNED TO NORMAL MODE OF OPERATION. A MODIFICATION TO THE LOGIC WILL BE EVALUATED TO DETERMINE THE FEASIBILITY OF ELIMINATING THE INTERDEPENDENCE BETWEEN THE LOGIC AND THE POSITION INDICATION POWER SUPPLY IN CCHVAC.

[41] FERMI 2 DOCKET 50-341 LER 89-027
ENGINEERED SAFETY FEATURE ACTUATIONS OCCURRED DURING THE MEGGERING OF REACTOR PROTECTION CIRCUITS.
EVENT DATE: 101589 REPORT DATE: 111489 NSSS: GE TYPE: BWR

(NSIC 215849) AT 1430 HOURS, ON 10/15/89, AN ELECTRICIAN ATTEMPTED TO LOOSEN A WIRE IN A REACTOR PROTECTION SYSTEM PANEL (RPS) H11-P623. THE REACTOR BUILDING/HEATING VENTILATION AND AIR CONDITIONING (HVAC), CONTROL CENTER HVAC, AND DRYWELL FLOOR AND EQUIPMENT DRAIN SUMPS ISOLATED. THE STANDBY GAS TREATMENT SYSTEM, AND NON-INTERRUPTABLE CONTROL AIR COMPRESSORS STARTED IN RESPONSE TO THE LOOSENED WIRE. THE WIRE WAS IMMEDIATELY RE-TIGHTENED. WORK WAS STOPPED AND AN INVESTIGATION INTO THE RPS ACTUATION WAS INITIATED. ALL OF THE EQUIPMENT WAS RETURNED TO NORMAL OPERATION AT 1459 HOURS. THE WORK PACKAGE DID NOT CONTAIN ADEQUATE INFORMATION TO PERFORM THE JOB. THE SCHEMATIC PRINT CONTAINED IN THE WORK PACKAGE, DID NOT SHOW THAT THE WIRE LIFTED WAS USED AS A NEUTRAL SOURCE FOR THE ACTUATION CIRCUITS FOR THE SYSTEMS NOTED ABOVE. HOWEVER, THE SCHEMATIC DID HAVE THE APPROPRIATE PRINT REFERENCED ON IT. A CRITIQUE OF THIS EVENT WILL BE INCLUDED IN THE OPERATIONS, MAINTENANCE AND INSTRUMENT & CONTROLS REQUIRED READING PROGRAMS. THE LESSONS LEARNED WILL BE INCORPORATED INTO THE INITIAL TRAINING PROGRAMS FOR ELECTRICIANS AND INSTRUMENTATION AND CONTROL TECHNICIANS. THE EXPECTED COMPLETION DATE OF THESE CORRECTIVE ACTIONS IS 11/30/89.

[42] FERMI 2 DOCKET 50-341 LER 89-028
SAFETY RELIEF VALVES FAIL THEIR SET PRESSURE TOLERANCE TEST.
EVENT DATE: 102089 REPORT DATE: 112089 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 215850) THE MAIN STEAM SYSTEM IS EQUIPPED WITH FIFTEEN SAFETY RELIEF VALVES (SRVS). TECHNICAL SPECIFICATIONS REQUIRE THAT HALF OF THE SRVS BE PROVEN OPERABLE AT LEAST ONCE EVERY EIGHTEEN MONTHS BY PERFORMING A SET PRESSURE TEST. FIFTEEN SRVS WERE REMOVED AND SENT TO WYLE LABORATORIES TO MEET THE SURVEILLANCE REQUIREMENT. WYLE LABORATORIES HAS VERBALLY NOTIFIED DETROIT EDISON THAT 8 OF THE SRVS FAILED THEIR SET PRESSURE TEST. THE CAUSE OF THIS EVENT IS CURRENTLY UNDER REVIEW BY DETROIT EDISON (DECO) AND GENERICALLY BY THE BOILING WATER REACTOR OWNERS GROUP (BWROG) SRV SET POINT DRIFT COMMITTEE. ALL VALVES REMOVED FROM THE PLANT FOR TESTING ARE BEING REFURBISHED, CLEANED, RETESTED AND RECERTIFIED TO BE WITHIN ACCEPTED TOLERANCES PRIOR TO RETURN TO FERMI 2 FROM WYLE LABORATORIES.

[43] FITZPATRICK DOCKET 50-333 LER 89-007 REV 01
UPDATE ON SURVEILLANCE OF FIRE BARRIER PENETRATIONS MISSED DUE TO MISINTERPRETATION OF BARRIERS REQUIRING SURVEILLANCE.
EVENT DATE: 050989 REPORT DATE: 111389 NSSS: GE TYPE: BWR

(NSIC 215872) ON 5/9/89 DURING NORMAL OPERATION AT 100% RATED POWER, IT WAS

DISCOVERED THAT SURVEILLANCE OF SOME FIRE BARRIER PENETRATIONS REQUIRED BY TECHNICAL SPECIFICATION 4.12.F WAS NOT DONE WITHIN THE ALLOWED TIME FOR A NUMBER OF FIRE BARRIERS. THE MISSED SURVEILLANCE WAS COMPLETED FOR THE IDENTIFIED BARRIERS AND DEFICIENCIES HAVE BEEN CORRECTED. THE EVENT WAS CAUSED BY CHANGES TO PROCEDURES BASED ON A REFERENCE MANUAL WHICH DID NOT INCLUDE ALL REQUIRED FIRE BARRIERS. CORRECTIVE ACTIONS INCLUDE: 1) ESTABLISHING FIRE WATCHES AND COMPLETION OF REQUIRED SURVEILLANCE, 2) CORRECTION OF DEFICIENCIES, 3) CORRECTION OF SURVEILLANCE PROCEDURES, AND 4) REVISION OF THE REFERENCE MANUAL. LER-86-001, 86-002, 87-017, 87-022, AND 88-006 ARE SIMILAR MISSED SURVEILLANCE EVENTS.

[44] FITZPATRICK DOCKET 50-333 LER 89-015 REV 01
UPDATE ON 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: 110789 NSSS: GE TYPE: BWR
VENDOR: HAMMEL DAHL

(NSIC 215873) 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 OF IRON OXIDE SLUDGE IN THE VALVE OPERATING INTERNAL 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.

[45] FITZPATRICK DOCKET 50-333 LER 89-018
AIR IN PRESSURE SENSOR TRANSMITTER GENERATES FALSE HIGH STEAM FLOW SIGNAL ISOLATING HIGH PRESSURE COOLANT INJECTION SYSTEM DURING A SURVEILLANCE TEST.
EVENT DATE: 100889 REPORT DATE: 110689 NSSS: GE TYPE: BWR

(NSIC 215874) A ROUTINE SURVEILLANCE TEST OF THE OPERABILITY OF THE HIGH PRESSURE COOLANT INJECTION (HPCI) (BJ) SYSTEM WAS IN PROGRESS ON 10/8/89 AT 14% POWER DURING START-UP FOLLOWING A PLANNED THREE-WEEK MAINTENANCE OUTAGE. AT 10:26 A.M., A HPCI HIGH STEAM FLOW SIGNAL CLOSED THE HPCI ISOLATION VALVES FOR THE OUTBOARD STEAM SUPPLY AND STEAM LINE WARMING. OPERATORS VERIFIED THE ABSENCE OF STEAM LEAKAGE. THE SURVEILLANCE TESTS REQUIRED TO BE PERFORMED WHEN HPCI IS INOPERABLE WERE INITIATED. INSPECTION OF THE DIFFERENTIAL PRESSURE TRANSMITTER, WHICH PROVIDES THE HIGH STEAM FLOW SIGNAL, FOUND THE CALIBRATION WAS ACCURATE. DURING RECALIBRATION A SMALL QUANTITY OF AIR WAS OBSERVED TO VENT FROM THE PRESSURE INSTRUMENT SENSING LINES. IT IS BELIEVED THAT PRESENCE OF NON-CONDENSIBLE, BUT COMPRESSIBLE AIR IN THE SENSING LINES, COMBINED WITH THE FAST START TRANSIENT, RESULTED IN OSCILLATIONS AND THE FALSE HIGH STEAM FLOW SIGNAL. A FIRM HYPOTHESIS FOR THE SOURCE OF ENTRY OF NON-CONDENSIBLES INTO THE SYSTEM WAS NOT ESTABLISHED. UPON SATISFACTORY PERFORMANCE OF THE SURVEILLANCE TEST, HPCI WAS RETURNED TO SERVICE AT 6:30 P.M. THE PRESSURE TRANSMITTER WILL AGAIN BE CALIBRATED AND VENTED DURING THE NEXT SCHEDULED OUTAGE. LER-86-015 DESCRIBES A RCIC ISOLATION DUE TO FALSE HIGH STEAM FLOW DUE TO AIR IN THE TRANSMITTER.

[46] FT. ST. VRAIN DOCKET 50-267 LER 85-003 REV 01
UPDATE ON BATTERY CELL FAILURE IDENTIFIED DURING LOSS OF OUTSIDE POWER TEST.
EVENT DATE: 030485 REPORT DATE: 051289 NSSS: GA TYPE: HTGR

(NSIC 215936) CAUSE - UNKNOWN. ON 12/18/84, AT 1200 HOURS, WITH THE REACTOR SHUTDOWN AND THE PCRV DEPRESSURIZED, THE LOSS OF OUTSIDE POWER AND TURBINE TRIP SEMI-ANNUAL SURVEILLANCE, WAS INITIATED BY STATION PERSONNEL. DURING THIS TEST, BOTH DIESEL GENERATOR TIE BREAKERS FAILED TO CLOSE. THIS EVENT WAS PREVIOUSLY REPORTED TO NRC LER 84-014 ON 1/17/85. INVESTIGATIONS INTO THE CAUSE OF THAT EVENT IDENTIFIED A FAULTY CELL ON STATION BATTERY 1A AS BEING PARTIALLY RESPONSIBLE FOR THE TEST FAILURE. THE BATTERY WAS IDENTIFIED AS A COMPONENT FAILURE IN THAT REPORT. SUBSEQUENT DISCUSSIONS WITH NRC PERSONNEL RESULTED IN REPORTING THE STATION BATTERY 1A FAILURE IN THIS LER, SEPARATE FROM LER 84-014. THE STATION BATTERIES HAVE BEEN REPLACED WITH LEAD-ANTIMONY BATTERIES DURING THE 1988 HELIUM CIRCULATOR REPAIR OUTAGE. THE BATTERY LOAD PROFILES HAVE BEEN REVIEWED AND REVISED. EACH OF THE NEW BATTERIES HAS PASSED A SERVICE DISCHARGE TEST USING THE NEW LOAD PROFILES. TECH SPEC AMENDMENT 864 WAS APPROVED TO ACCOMODATE THE CHARGING REQUIREMENTS OF THE NEW BATTERIES. PUBLIC SERVICE COMPANY OF COLORADO AND THE EXIDE CORPORATION HAVE NOT BEEN ABLE TO IDENTIFY A DEFINITIVE ROOT CAUSE(S) OF THE PREMATURE DEGRADATION OF THE LEAD-CALCIUM BATTERIES.

[47]   FT. ST. VRAIN   DOCKET 50-267   LER 88-009 REV 01
UPDATE ON REACTOR SCRAM ON HIGH HOT REHEAT TEMPERATURE FOLLOWING HELIUM CIRCULATOR TRIP.
EVENT DATE: 050688   REPORT DATE: 080489   NSSS: GA   TYPE: HTGR

(NSIC 215937) CAUSE - INSTRUMENT FAILURE. AT 1234 HOURS ON 5/6/88, WITH THE PLANT OPERATING AT 79% POWER, A REACTOR SCRAM ON HIGH HOT REHEAT STEAM TEMPERATURE OCCURRED FOLLOWING A TRIP OF "B" HELIUM CIRCULATOR. THE CAUSE OF THIS EVENT WAS A MALFUNCTION OF AN ANALOG AVERAGING INSTRUMENT IN THE OVERALL PLANT CONTROL SYSTEM WHICH RESULTED IN AN ANALOG AVERAGE HOT REHEAT STEAM TEMPERATURE INDICATION 35F BELOW THE ACTUAL TEMPERATURE, THEREBY PREVENTING THE COLD REHEAT ATTEMPERATION BOOST CIRCUIT FROM OPERATING PROPERLY TO PREVENT THE SCRAM. A DISCREPANCY BETWEEN HOT REHEAT STEAM TEMPERATURES INDICATED BY THE PLANT PROTECTIVE SYSTEM (PPS) AND ACTUAL HOT REHEAT STEAM TEMPERATURES ALSO CONTRIBUTED TO THE CAUSE. THE HOT REHEAT STEAM TEMPERATURE CIRCUIT OF THE OVERALL PLANT CONTROL SYSTEM WAS FUNCTIONALLY TESTED AND PERFORMED AS DESIGNED. ALTHOUGH THE MALFUNCTION OF THE ANALOG AVERAGING INSTRUMENT IN THE OVERALL PLANT CONTROL SYSTEM APPEARS TO BE INTERMITTENT, IT IS READILY DETECTABLE BASED ON OPERATOR OBSERVATIONS. OPERATORS HAVE BEEN MADE AWARE OF THE INDICATIONS THAT WILL EXIST, AND THE ACTIONS TO BE TAKEN, IF THE PROBLEM REAPPEARS. THE CAUSE OF THE DISCREPANCY BETWEEN PPS INDICATED AND ACTUAL HOT REHEAT STEAM TEMPERATURES HAS BEEN ATTRIBUTED TO THE CALIBRATION METHODOLOGY, AND SEVERAL IMPROVEMENTS HAVE BEEN MADE.

[48]   FT. ST. VRAIN   DOCKET 50-267   LER 88-012 REV 02
UPDATE ON ERROR DISCOVERED IN COMPUTER CODE USED TO DEVELOP E.Q. TEMPERATURE PROFILES.
EVENT DATE: 081688   REPORT DATE: 051089   NSSS: GA   TYPE: HTGR

(NSIC 215938) CAUSE - ERROR IN PROGRAM. IN A LETTER FROM GENERAL ATOMICS (GA) TO PUBLIC SERVICE CO. OF COLORADO (PSC) DATED 5/10/88, GA INFORMED PSC THAT AN ERROR HAD BEEN DISCOVERED IN THE OPEN BUILDING CONTEMPT-G PROGRAM USED IN DEVELOPING THE FSV ENVIRONMENTAL QUALIFICATION TEMPERATURE PROFILES FOR EQUIPMENT QUALIFICATION. THE ERROR CAUSED THE CONTEMPT-G PROGRAM TO CALCULATE THE LOWER PEAK TEMPERATURES FOR CERTAIN LINE BREAK SCENARIOS. GA CONTENDED THAT WITH THE ERROR CORRECTED, CERTAIN PREVIOUSLY ANALYZED AND ACCEPTABLE SCENARIOS NOW HAD PEAK TEMPERATURES OUTSIDE OF THE NRC APPROVED COMPOSITE PROFILES. PSC CONTRACTED NUMERICAL APPLICATIONS INC. (NAI) TO ANALYZE THE SCENARIOS USED BY GA TO DEVELOP FSV BUILDING TEMPERATURE PROFILES. NAI ANALYSIS RESULTS CONFIRMED GA'S TEMPERATURE PROFILES OBTAINED FROM THE CONTEMPT-G PROGRAM WERE NON-CONSERVATIVE. UPON RECEIPT OF NAI'S ANALYSIS RESULTS, PSC NOTIFIED THE NRC ON 8/16/88. NEW TEMPERATURE PROFILES HAVE BEEN DEVELOPED BY NAI. PSC HAS VERIFIED THAT ALL E.Q. EQUIPMENT QUALIFIED TO THE ORIGNAL GA TEMPERATURE PROFILE IS ALSO QUALIFIED TO THE NEW NAI PROFILE. ALSO, PSC HAS MODIFIED THE STEAM LINE RUPTURE DETECTION ISOLATION SYSTEM (SLRDIS) TO ACTUATE AT A FIXED SETPOINT OF LESS THAN OR EQUAL TO

171F. THIS SUPPLEMENTAL REPORT PROVIDES FINAL STATUS OF THE E.Q. BINDER REVIEW, TEMPERATURE PROFILE DEVELOPMENT, AND SLRDIS MODIFICATIONS.

[49] FT. ST. VRAIN DOCKET 50-267 LER 88-015 REV 02
UPDATE ON LOSS OF POWER TO 480V ESSENTIAL BUSES DURING LOSS OF OUTSIDE ELECTRIC POWER TEST.
EVENT DATE: 101188 REPORT DATE: 082589 NSSS: GA TYPE: HTGR

(NSIC 216018) CAUSE - DIRTY/STICKING CONTACTS. ON 10/11/88, WITH THE REACTOR SHUTDOWN FOR HELIUM CIRCULATOR MAINTENANCE AND THE PCRV PRESSURIZED TO 85 PSIA, THE LOSS OF OUTSIDE ELECTRIC POWER AND TURBINE TRIP SURVEILLANCE TEST WAS INITIATED. THE DIESEL GENERATOR ENGINES OF BOTH DIESEL GENERATOR SETS STARTED AUTOMATICALLY, AS DESIGNED, AND AUTOMATIC LOAD SHEDDING WAS COMPLETED FROM THE THREE ESSENTIAL 480V BUSES. HOWEVER, BOTH DIESEL GENERATOR OUTPUT BREAKERS FAILED TO CLOSE AND POWER WAS LOST TO THE 480V ESSENTIAL BUSES. OUTSIDE POWER WAS MANUALLY RESTORED TO THE 4160V BUSES AND THE NON-ESSENTIAL 480V BUSES. AFTER ABOUT 39 MINUTES, POWER WAS RESTORED TO THE 480V ESSENTIAL BUSES. THE FAILURE OF THE DIESEL GENERATOR OUTPUT BREAKERS TO CLOSE WAS MOST LIKELY DUE TO A CONTACT PROBLEM THAT CORRECTED ITSELF DURING THE TEST. PART OF THE LOGIC HAD BEEN DEFEATED DURING THE TEST AND NO SINGLE FAILURE WAS FOUND THAT ALONE COULD HAVE PREVENTED THE DIESEL GENERATORS FROM SUPPLYING ESSENTIAL LOADS IN AN ACTUAL LOSS OF OUTSIDE POWER EVENT. ALSO, POWER WAS RESTORED TO THE ESSENTIAL BUSES WITHIN FSAR TIME REQUIREMENTS, AND THIS EVENT WAS DETERMINED NOT REPORTABLE BUT IS BEING REPORTED AS A VOLUNTARY REPORT. THIS SUPPLEMENT REVISES THE CORRECTIVE ACTION.

[50] FT. ST. VRAIN DOCKET 50-267 LER 89-007
LOOP I SHUTDOWN AND TURBINE TRIP CAUSED BY LOSS OF INSTRUMENT BUS.
EVENT DATE: 042189 REPORT DATE: 052289 NSSS: GA TYPE: HTGR

(NSIC 215939) CAUSE - OPERATOR ERROR/EQUIPMENT MALFUNCTION. AT 0628 HOURS ON 4/21/89 WITH THE REACTOR OPERATING AT ABOUT 50% POWER, THE MAIN TURBINE GENERATOR TRIPPED IMMEDIATELY FOLLOWED BY A TRIP OF A SINGLE REACTOR COOLANT LOOP (LOOP I). THESE UNEXPECTED SYSTEM TRIPS WERE CAUSED BY A TEMPORARY LOSS OF ELECTRICAL POWER TO THE 'C' ESSENTIAL INSTRUMENT BUS. THIS ELECTRICAL POWER LOSS OCCURRED WHILE AN EQUIPMENT OPERATOR WAS TRANSFERRING THE 'C' ESSENTIAL INSTRUMENT BUS TO ITS ALTERNATE POWER SUPPLY AS PART OF AN EQUIPMENT CLEARANCE ON THE 'C' BATTERY CHARGER. WHILE MANUALLY TRANSFERRING THE 'C' ESSENTIAL BUS TO ITS ALTERNATE POWER SUPPLY, THE BUS BECAME DE-ENERGIZED FOR APPROXIMATELY 20 SECONDS. THIS TEMPORARY LOSS OF POWER TO 'C' ESSENTIAL BUS RESULTED IN A TRIP OF THE MAIN TURBINE GENERATOR AND LOOP I. REACTOR POWER WAS REDUCED AND STABILIZED AT 23%. CORE COOLING WAS MAINTAINED BY LOOP II THROUGHOUT THIS EVENT. SINCE THIS EVENT RESULTED IN AN UNEXPECTED ACTUATION OF THE PLANT PROTECTIVE SYSTEM LOOP I SHUTDOWN FUNCTION, IT IS BEING REPORTED IN ACCORDANCE WITH THE REQUIREMENTS OF 10 CFR 50.73(A)(2)(IV). ROOT CAUSE INVESTIGATION OF THIS EVENT HAS IDENTIFIED TWO PROBABLE CAUSES, OPERATOR ERROR OR EQUIPMENT MALFUNCTION. HOWEVER, NEITHER OF THESE COULD BE POSITIVELY IDENTIFIED AS THE CAUSE OF THIS EVENT.

[51] FT. ST. VRAIN DOCKET 50-267 LER 89-008
STACK MONITOR RT-4801 ALARM SETPOINT FOUND OUT OF CALIBRATION.
EVENT DATE: 042789 REPORT DATE: 052689 NSSS: GA TYPE: HTGR

(NSIC 215940) CAUSE - CALCULATION ERROR. ON 4/27/89 AT 2010 HOURS WITH THE REACTOR AT ABOUT 28% POWER, IT WAS CONFIRMED THAT BACKUP RADIATION MONITOR RT-4801 ALARM SETPOINT WAS OUT OF CALIBRATION SUCH THAT IT WOULD NOT HAVE INITIATED AN ALARM IN ACCORDANCE WITH THE TECH SPEC SETPOINT REQUIREMENTS. FAILURE TO MAINTAIN CALIBRATION IN ACCORDANCE WITH THE REQUIREMENTS ESTABLISHED IN THE FSV TECH SPECS CONSTITUTES A CONDITION PROHIBITED BY THE TECH SPECS AND IS BEING REPORTED HEREIN PER 10 CFR 50.73(A)(2)(I)(B). THIS OUT OF CALIBRATION CONDITION WAS DUE TO A CALCULATION ERROR IN "SETPOINT CHANGE REPORT" 86-213 DATED 10/7/86. PSC RECORDS HAVE IDENTIFIED 3 INTERVALS WHEN THIS BACKUP MONITOR WAS RELIED UPON TO FULFILL TECH SPEC REQUIREMENTS WHEN THE PRIMARY RADIATION MONITOR WAS OUT OF SERVICE. THE RADIATION MONITOR RT-4801 ALARM SETPOINT WAS RECALIBRATED AND RETURNED TO OPERABLE STATUS ON 4/26/89. DURING THIS PERIOD,

THERE WERE NO PLANT RELEASES THAT EXCEEDED 10 CFR 20 LIMITS AND THE OVERALL PERFORMANCE AND CONTROL OF THE RADIATION MONITORING SYSTEM HAS NOT BEEN DEGRADED.

[52] FT. ST. VRAIN DOCKET 50-267 LER 89-009
FAILURE OF REGION 3 CONTROL ROD TO SCRAM.
EVENT DATE: 042789 REPORT DATE: 053089 NSSS: GA TYPE: HTGR

(NSIC 215941) CAUSE - EXCESSIVE LOADING OF BEARING. ON 4/27/89, WITH THE REACTOR OPERATING AT APPROXIMATELY 26.5% POWER, THE CONTROL ROD PAIR IN REGION 3 FAILED TO SCRAM DURING PERFORMANCE OF A WEEKLY PARTIAL SCRAM SURVEILLANCE TEST. ALTHOUGH THE CONTROL ROD PAIR WAS CAPABLE OF BEING INSERTED WITH ITS MOTOR DRIVE, THIS CONDITION IS NOT SPECIFICALLY ALLOWED BY THE TECH SPECS AND THE PLANT WAS SHUT DOWN. THIS CONDITION IS REPORTABLE PER THE REQUIREMENTS OF 10 CFR 50.73(A)(2)(I)(A). THIS EVENT WAS CAUSED BY EXCESSIVE SHIMS IN THE CONTROL ROD DRIVEN GEAR TRAIN WHICH RESULTED IN EXCESSIVE LOADING OF THE FIRST STAGE BEARING. THIS EXCESSIVE LOADING PRODUCED PREMATURE BEARING WEAR AND PREVENTED FREE GEAR TRAIN MOVEMENT. THE EXCESSIVE SHIMS WERE CAUSED BY A MEASUREMENT ERROR DURING REFURBISHMENT. THIS EVENT IS AN ISOLATED OCCURRENCE, WITH NO GENERIC IMPLICATIONS. THE CONTROL ROD DRIVE ASSEMBLY WAS REPLACED ON 5/1/89 AND PLANT OPERATIONS WERE RESUMED.

[53] FT. ST. VRAIN DOCKET 50-267 LER 89-011
OPERATION IN EXCESS OF THE 82% POWER LIMIT.
EVENT DATE: 062289 REPORT DATE: 072189 NSSS: GA TYPE: HTGR

(NSIC 215942) CAUSE - INADEQUATE PROCEDURE. AT 0217 HOURS, ON 6/23/89, WITH THE REACTOR OPERATING AT POWER, IT WAS DISCOVERED THAT REHEAT STEAM ATTEMPERATION FLOW HAD NOT BEEN ACCOUNTED FOR IN THE SECONDARY HEAT BALANCE CALCULATION OF REACTOR POWER. THE SHIFT SUPERVISOR TOOK IMMEDIATE ACTION AND UPDATED THE SECONDARY HEAT BALANCE CALCULATION TO ACCOUNT FOR THIS ATTEMPERATION FLOW. AT 0218 HOURS REACTOR POWER (AS CALCULATED BY THE UPDATED SECONDARY HEAT BALANCE) INDICATED 83.4%, WHICH EXCEEDED FORT ST. VRAIN'S MAXIMUM AUTHORIZED OPERATING LIMIT OF 82%. AT 0219 HOURS REACTOR OPERATORS BEGAN REDUCING REACTOR POWER. AT 0228 REACTOR POWER WAS BELOW 82%. AN INVESTIGATION OF THIS EVENT DETERMINED THAT REACTOR POWER WAS IN EXCESS OF 82% FOR APPROXIMATELY FOUR HOURS. AFTER CONFIRMING THAT REACTOR POWER DID ACTUALLY EXCEED 82%, THE NRC OPERATIONS CENTER WAS NOTIFIED AT 1215 HOURS ON 6/23/89 IN ACCORDANCE WITH THE REQUIREMENTS OF 10 CFR 50.72(B)(1)II)(B). THE ROOT CAUSE FOR THIS EVENT WAS IDENTIFIED TO BE INADEQUATE PROCEDURES. APPROPRIATE PROCEDURE CHANGES HAVE BEEN IMPLEMENTED TO ADDRESS THESE INADEQUACIES.

[54] FT. ST. VRAIN DOCKET 50-267 LER 89-012
UNANALYZED GAS WASTE RELEASE FROM GAS WASTE SURGE TANK.
EVENT DATE: 062889 REPORT DATE: 072489 NSSS: GA TYPE: HTGR

(NSIC 216016) CAUSE - PERSONNEL ERROR. ON 6/28/89, AT 0155 HOURS WITH THE REACTOR OPERATING AT 79% POWER, IT WAS DETERMINED THAT AN UNANALYZED GAS WASTE RELEASE HAD OCCURRED FROM THE 1A GAS WASTE SURGE TANK. THIS RELEASE STARTED AT 0135 WHEN PLANT PERSONNEL INTENDED TO INITIATE A RELEASE FROM THE 1B GAS WASTE SURGE TANK. IT WAS TERMINATED AT 0155 WHEN THE 1A GAS WASTE SURGE TANK PRESSURE WAS OBSERVED TO BE DECREASING. LATER SAMPLING OF THE 1A SURGE TANK DETERMINED RELEASE ACTIVITY TO BE WITHIN ALLOWABLE LIMTS. SINCE FORT ST. VRAIN TECH SPEC ELCO 8.1.1D REQUIRES THE CONTENTS OF THE GAS WASTE SURGE TANKS TO BE SAMPLED AND ANALYZED FOR GASEOUS RADIOACTIVITY PRIOR TO RELEASE, THIS EVENT CONSTITUTES OPERATION IN VIOLATION OF THE TECH SPECS AND IS BEING REPORTED PER 10 CFR 50.73(A)(2)(I)(B). PERSONNEL ERROR CAUSED THIS EVENT. NO PROCEDURAL DEFICIENCIES WERE IDENTIFIED. EQUIPMENT OPERATORS FAILED TO PERFORM THE INITIAL VALVE LINEUP IN ACCORDANCE WITH THE RELEASE PROCEDURE, NPAP-19, AND ALSO FAILED TO PERFORM A THOROUGH INDEPENDENT VERIFICATION OF THE VALVE LINEUP AS REQUIRED.

[55] FT. ST. VRAIN DOCKET 50-267 LER 89-013
12 CABLE TRAYS IN THE AUXILIARY ELECTRIC ROOM FILLED ABOVE DESIGN LIMIT.
EVENT DATE: 081189 REPORT DATE: 090889 NSSS: GA TYPE: HTGR

(NSIC 216017) CAUSE - COMPUTER PROGRAM ERROR. FOLLOWING A RECENT AUDIT, PSC IDENTIFIED 12 CABLE TRAYS IN THE AUX. ELECTRIC ROOM THAT ARE FILLED ABOVE THEIR DESIGN LIMIT. AN ERROR WAS DISCOVERED IN THE CABLE TRAY COMPUTER PROGRAM USED TO CALCULATE CABLE TRAY FILL. THIS ERROR CAUSED THE CABLE TRAY COMPUTER PROGRAM TO CALCULATE LOWER THAN ACTUAL TRAY FILLS FOR THESE 12 CABLE TRAYS. ON 8/11/89 AT 1150 HOURS, PSC NOTIFIED THE NRC IN ACCORDANCE WITH THE REQUIREMENTS OF 10CFR50.72(B)(1)(II)(B). PSC HAS CORRECTED THE PROGRAM ERROR AND IS REASSESSING THE CABLE TRAY FILL OF THESE 12 CABLE TRAYS AGAINST THE DESIGN BASIS FOR FSV. THIS ASSESSMENT WILL BE COMPLETED BY 10/25/89. DUE TO THE LONG PERIOD OF TIME THAT THE PROGRAM ERROR HAD EXISTED IN THE "CABLE TRAY LOADING COMPUTER PROGRAM", IT IS DIFFICULT TO SPECIFICALLY IDENTIFY ALL THE VARIOUS FACTORS THAT CONTRIBUTED TO THIS ISSUE. THE "ROOT CAUSE" FOR THIS EVENT WAS THAT A COMPARATIVE FEATURE IN THE CABLE TRAY LOADING PROGRAM INADVERTENTLY OMITTED CERTAIN CABLES FROM THE TRAY FILL CALCULATIONS.

[56] FT. ST. VRAIN DOCKET 50-267 LER 89-014
DEGRADED FIRE BARRIER PENETRATION SEALS.
EVENT DATE: 081889 REPORT DATE: 091589 NSSS: GA TYPE: HTGR

(NSIC 216019) CAUSE - SEALS DEFECTIVE. ON 8/18/89, WITH THE REACTOR SHUTDOWN, FOUR FIRE BARRIER PENETRATION SEALS (BT11, BT21, BT27, AND BT125) WERE DETERMINED TO BE DEFICIENT AS FOUND DURING ROUTINE MAINTENANCE ACTIVITIES. ON 8/31/89, TWO ADDITIONAL FIRE BARRIER PENETRATION SEALS (BT126 AND BT127) WERE DETERMINED TO BE DEFICIENT AS FOUND DURING ROUTINE MAINTENANCE ACTIVITIES. THE SEALS ARE LOCATED IN BUILDING 10 AND WERE FOUND NOT TO MEET THE FOAM "CELL STRUCTURE" CRITERIA OF THE SEAL SPECIFICATION. THESE IMPAIRMENTS WERE THE RESULT OF IMPROPER CURING DURING INSTALLATION DUE TO AN INADEQUATE PROCEDURE. COMPENSATORY MEASURES IN EACH CASE INCLUDED IMMEDIATELY ESTABLISHING A FIREWATCH AND INITIATING REPAIRS ON EACH OF THE IMPAIRED FIRE BARRIER PENETRATION SEALS. THIS EVENT CONSTITUTES OPERATION IN VIOLATION OF TECH SPEC LCO 4.10.4, FIRE BARRIER PENETRATION SEALS, AND IS BEING REPORTED PER 10 CFR 50.73(A)(2)(I)(B).

[57] FT. ST. VRAIN DOCKET 50-267 LER 89-015
REGION 19 CONTROL ROD DRIVE FAILED TO FULLY INSERT ON SCRAM SIGNAL.
EVENT DATE: 081889 REPORT DATE: 091789 NSSS: GA TYPE: HTGR

(NSIC 216020) CAUSE - BOLT LODGED IN GUIDE TUBE. AT 1820 HOURS ON 8/17/89, WITH THE PLANT OPERATING AT ABOUT 79% POWER AND 256 MWE, THE REGION 19 CONTROL ROD PAIR FAILED TO PROPERLY INSERT DURING THE WEEKLY PARTIAL SCRAM TEST (SR 4.1.1.B.1/2-W). THIS INSERTION FAILURE WAS IMMEDIATELY INDICATED BY ACTUATION OF A "SLACK CABLE" ALARM FOR REGION 19. THE SLACK CABLE ALARM WAS CLEARED AND SCRAM TESTING WAS COMPLETED ON THE REMAINING ROD PAIRS. AT 1900 HOURS THE REGION 19 ROD PAIR WAS RETESTED BUT AGAIN FAILED TO INSERT. FOLLOWING EXTENSIVE TROUBLESHOOTING EFFORTS AND ANALYSIS OF REGION 19 BACK-EMF DATA, THE REGION 19 CONTROL ROD PAIR WAS DECLARED IMMOVABLE AND AN ORDERLY PLANT SHUTDOWN WAS INITIATED IN ACCORDANCE WITH THE REQUIREMENTS OF INTERIM TECH SPEC LCO 3.1.1. THE NRC OPERATIONS CENTER WAS NOTIFIED AT 0451 HOURS ON 8/18/89. PLANT SHUTDOWN WAS COMPLETE AT 1555 HOURS ON 8/18. THROUGH VISUAL INSPECTION OF THE REGION 19 CRD, IT HAS BEEN DETERMINED THAT A FAILED CLEVIS BOLT HEAD BECAME LODGED BETWEEN ONE OF THE CONTROL ROD ABSORBER STRINGS AND ITS GUIDE TUBE, THEREBY PREVENTING ROD INSERTION. GA INTERNATIONAL SERVICES CORP. IS CURRENTLY PERFORMING A FAILURE ANALYSIS ON THE FAILED BOLT HEAD. PSC HAS DECIDED NOT TO RESTART THE PLANT.

[58] FT. ST. VRAIN DOCKET 50-267 LER 89-016
LOOP II SHUTDOWN INITIATED BY PPS DUE TO LOW BEARING WATER SURGE TANK LEVEL.
EVENT DATE: 081889 REPORT DATE: 091589 NSSS: GA TYPE: HTGR

(NSIC 216021) CAUSE - EQUIPMENT MALFUNCTION AND OPERATOR ERROR. ON 8/18/89, WHILE PERFORMING AN ORDERLY PLANT SHUTDOWN FROM 80% REACTOR POWER TO REPAIR A

FAULTY CONTROL ROD DRIVE (SEE LER 89-015), AN AUTOMATIC PPS LOOP II SHUTDOWN OCCURRED. THIS LOOP II SHUTDOWN WAS INITIATED AT 0719 HOURS WITH THE REACTOR OPERATING AT ABOUT 65% POWER. FORCED CORE COOLING WAS MAINTAINED THROUGHOUT THIS EVENT BY OPERATION OF THE LOOP I COOLANT SYSTEM. THIS LOOP SHUTDOWN WAS NOT PART OF A PREPLANNED SEQUENCE. LOW WATER LEVEL IN THE LOOP II BEARING WATER SURGE TANK CAUSED BOTH LOOP II BEARING WATER PUMPS TO TRIP AND SUBSEQUENTLY A TRIP OF BOTH LOOP II HELIUM CIRCULATORS. A TRIP OF 2 CIRCULATORS IN A LOOP INITIATES AN AUTOMATIC LOOP SHUTDOWN. THIS LOW LEVEL CONDITION RESULTED FROM A COMBINATION OF EQUIPMENT MALFUNCTION AND OPERATOR ERROR. THE LOOP II BEARING WATER SURGE TANK LEVEL CONTROLLER, LC-2136, FAILED TO PROPERLY RESPOND TO THE DECREASING TANK LEVEL. AT 0700 HRS, CONTROL ROOM OPERATORS WERE ALERTED TO THE LOW LOOP II SURGE TANK LEVEL CONDITION BY ACTUATION OF ALARM I-02B WINDOW 4-5. NO ACTION WAS TAKEN TO INVESTIGATE OR CORRECT THIS LOW LEVEL CONDITION. SEVERAL MINUTES AFTER RECEIVING THE INITIAL ALARM, THE REACTOR OPERATOR CHECKED THE WRONG ALARM (I.E., LOOP I SURGE TANK LEVEL), WHICH WAS CLEAR, AND INCORRECTLY ASSUMED THE LOW LEVEL CONDITION HAD CORRECTED ITSELF. THE LOOP II SURGE TANK LOW LEVEL ALARM WAS STILL ACTUATED AND TANK LEVEL CONTINUED TO DROP.

[59] FT. ST. VRAIN DOCKET 50-267 LER 89-017
REACTOR SCRAM SIGNAL GENERATED BY WIDE RANGE NUCLEAR CHANNELS III AND V WHILE REACTOR SHUTDOWN.
EVENT DATE: 081889 REPORT DATE: 091589 NSSS: GA TYPE: HTGR

(NSIC 216022) CAUSE - INDUCED ELECTRICAL NOISE. ON 8/18/89, AFTER COMPLETING A CONTROLLED PLANT SHUTDOWN FROM 80% REACTOR POWER (SEE LER 89-015), WIDE RANGE NUCLEAR CHANNELS (WRC) III AND V UPSCALED AND TRIPPED IN RESPONSE TO INDUCED ELECTRICAL NOISE AT 1638 HOURS. THIS TRIP OF WRCS III AND V COMPLETED THE 2 OF 3 MINIMUM ACTUATION LOGIC AND GENERATED A REACTOR SCRAM SIGNAL. SINCE A MANUAL REACTOR SCRAM WAS ALREADY ACTUATED, NO CONTROL ROD MOVEMENT RESULTED FROM THIS SPURIOUS WRC SCRAM SIGNAL. THIS EVENT WAS NOT PART OF A PREPLANNED SEQUENCE OR EVOLUTION AND THEREFORE IS BEING REPORTED HEREIN IN ACCORDANCE WITH THE REQUIREMENTS OF 10 CFR 50.73(A)(2)(IV). ON 8/29/89, PUBLIC SERVICE COMPANY ANNOUNCED THAT FORT ST. VRAIN WILL NOT BE RESTARTED, THUS MARKING THE END OF THE PLANT'S OPERATIONAL PHASE. THE NUCLEAR STARTUP CHANNELS WILL BE UTILIZED DURING DEFUELING OPERATIONS FOR AUTOMATIC PROTECTION AGAINST REACTIVITY TRANSIENTS. PSC HAS DETERMINED THAT MODIFICATIONS TO PROTECT THE WIDE RANGE CHANNELS AGAINST ELECTRONIC NOISE ARE NOT JUSTIFIED, AND NO FURTHER CORRECTIVE ACTION IS PLANNED.

[60] FT. ST. VRAIN DOCKET 50-267 LER 89-018
CRACKING IN INCONEL MAIN STEAM RINGHEADERS.
EVENT DATE: 082789 REPORT DATE: 092689 NSSS: GA TYPE: HTGR

(NSIC 216023) CAUSE - UNDETERMINED. ON 8/25/89, PSC DISCOVERED A LEAK IN THE MAIN STEAM RINGHEADER OF ONE OF THE LOOP I STEAM GENERATOR MODULES. UPON FIRTHER INVESTIGATION, ON 8/27/89, 37 CRACK INDICATIONS WERE FOUND IN THE INCONEL ALLOY 800 RINGHEADERS IN 8 OF THE 12 MODULES, AND THE CONDITION WAS DETERMINED TO HAVE GENERIC IMPLICATIONS. PSC'S INVESTIGATION INTO THE CAUSE OF THE CRACK INDICATIONS IS IN PROGRESS. THIS PRELIMINARY REPORT IS PROVIDED TO DESCRIBE THE CURRENT STATUS OF THIS INVESTIGATION. THE ROOT CAUSE OF THE CRACKING IN THE INCONEL MAIN STEAM RINGHEADERS IS CURRENTLY UNDETERMINED. BASED ON A PRELIMINARY EVALUATION, IT APPEARS LIKELY THAT THE CRACKING MECHANISM WAS DUE TO CREEP OR LOSS OF DUCTILITY. A METALLOGRAPHIC EXAMINATION IS IN PROGRESS TO CONFIRM THIS PRELIMINARY CONCLUSION. BASED ON THE EXTENT OF THE RINGHEADER CRACKING AND OTHER OPERATIONAL CONSIDERATIONS, ON 8/29/89, PSC DECIDED TO SHUT DOWN FORT ST. VRAIN PERMANENTLY. WELD REPAIR ACTIVITIES HAVE BEEN PERFORMED TO THE EXTENT NECESSARY TO ENSURE REDUNDANT DECAY HEAT REMOVAL CAPABILITY DURING SHUTDOWN AND DEFUELING CONDITIONS.

[61] FT. ST. VRAIN DOCKET 50-267 LER 89-019
MOISTURE INGRESS TO INSTRUMENT AIR SYSTEM.
EVENT DATE: 090689 REPORT DATE: 100689 NSSS: GA TYPE: HTGR

(NSIC 216024) CAUSE - FAILED CHECK VALVE AND OPERATOR ERROR. AT 0200 HOURS ON

9/6/89, WITH THE PLANT SHUT DOWN, OPERATIONS PERSONNEL DISCOVERED THAT STEAM HAD ENTERED THE 'A' INSTRUMENT AIR HEADER DURING ACTIVITIES ASSOCIATED WITH STARTUP OF THE OUTSIDE AUXILIARY BOILER. BOTH SERVICE AIR AND STEAM LINES WERE CONNECTED TO THE BOILER FUEL OIL ATOMIZATION LINE, PER NORMAL STARTUP PROCEDURES. THE FAILURE OF A CHECK VALVE ALLOWED STEAM TO FLOW INTO THE SERVICE AIR LINE. FURTHER FAILURE OF A CONTROL/ISOLATION VALVE ALLOWED STEAM TO THEN ENTER 'A' INSTRUMENT AIR HEADER. ALL INSTRUMENTS THAT WERE INITIALLY AFFECTED WERE BLOWN DOWN AND RETURNED TO SERVICE. A COMPREHENSIVE PROGRAM OF CHECKING FOR MOISTURE IN CRITICAL INSTRUMENTATION AND VALVES HAS BEEN COMPLETED. REPAIRS HAVE ALSO BEEN INITIATED FOR VALVE DEFICIENCIES IDENTIFIED. THE ROOT CAUSE OF THIS EVENT WAS INSUFFICIENT ATTENTION TO PROCEDURAL DETAILS BY OPERATIONS PERSONNEL. BOTH THE ATOMIZING STEAM VALVE (V-84885) AND ATOMIZING AIR VALVE (V-84881) FOR THE OAB WERE INADVERTENTLY LEFT OPEN. THIS IS CONTRARY TO INSTRUCTIONS SPECIFIED IN SOP 84-02 FOR SHUTTING DOWN THE OAB, ALTHOUGH IT IS ACKNOWLEDGED THAT V-84881 WAS LEAKING BY AND WOULD NOT HAVE PROVIDED THE REQUIRED ISOLATION EVEN IF IT HAD BEEN SHUT. A MAJOR CONTRIBUTING CAUSE WAS THE FAILURE OF THE ATOMIZING AIR LINE CHECK VALVE (V-84882). AN INSPECTION OF THE CHECK VALVE REVEALED THE SEAT DISC HAD BROKEN OFF.

[62]   HATCH 1   DOCKET 50-321   LER 89-012
LEAKING RELIEF VALVE LIFTS CAUSING RWCU ISOLATION.
EVENT DATE: 100389   REPORT DATE: 103189   NSSS: GE   TYPE: BWR
VENDOR: CONSOLIDATED VALVE CORP.

(NSIC 215863) ON 10/03/89, AT APPROXIMATELY 2315 CDT, UNIT 1 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2436 MWT (APPROXIMATELY 100% OF RATED THERMAL POWER). AT THAT TIME PLANT OPERATORS RECEIVED INDICATION THAT THE REACTOR WATER CLEANUP (RWCU) SYSTEM WAS EXPERIENCING HIGH DIFFERENTIAL FLOW. THIS RESULTED IN THE ISOLATION OF THE GROUP 5 PRIMARY CONTAINMENT ISOLATION SYSTEM VALVES. PLANT EQUIPMENT OPERATORS INSPECTED THE RWCU SYSTEM AND DETERMINED THAT NO LEAKAGE HAD OCCURRED OUTSIDE THE SYSTEM. WHEN NO SYSTEM PROBLEMS COULD BE IDENTIFIED WHICH WOULD HAVE CAUSED THE ISOLATION, OPERATORS CONCLUDED THAT SYSTEM OPERATION COULD BE SAFELY RESUMED, AND THE SYSTEM WAS RETURNED TO SERVICE BY APPROXIMATELY 0400 CDT ON 10/04/89. A SUBSEQUENT AND MORE DETAILED INVESTIGATION REVEALED ELEVATED TEMPERATURE ON THE TAILPIPE OF RELIEF VALVE 1G31-F3058. THIS VALVE WAS APPARENTLY LEAKING TO THE RADWASTE SYSTEM AND THUS NO WATER OR STEAM HAD BEEN OBSERVED DURING THE INITIAL INSPECTION. THE ROOT CAUSE OF THIS EVENT WAS CONCLUDED TO BE COMPONENT FAILURE. ELEVATED TAILPIPE TEMPERATURE SHOWED RELIEF VALVE 1G31-F3058 WAS LEAKING PAST ITS VALVE SEAT. THE LEAKAGE EXPOSED AN INCREASED PROPORTION OF VALVE SEAT SURFACE TO HIGH PRESSURE WATER AND FLASHING STEAM, THUS DECREASING THE LIFT PRESSURE SETPOINT, LEADING TO A SPURIOUS BLOWDOWN OF THE RWCU SYSTEM, FOLLOWED BY THE ISOLATION CAUSED BY THE LEAK DETECTION SYSTEM.

[63]   HATCH 1   DOCKET 50-321   LER 89-013
PERSONNEL ERROR RESULTS IN MISSED TECH SPECS SURVEILLANCE.
EVENT DATE: 100389   REPORT DATE: 103189   NSSS: GE   TYPE: BWR

(NSIC 215864) ON 10/03/89, AT APPROXIMATELY 0305 CDT, UNIT ONE WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2433 CMWT (APPROXIMATELY 100 PERCENT OF RATED THERMAL POWER). AT THAT TIME, REACTOR WATER CLEANUP (RWCU, EIIS CODE CE) SYSTEM VALVE 1G31-F020 WAS CLOSED RENDERING THE INSERVICE REACTOR COOLANT CONTINUOUS IN-LINE CONDUCTIVITY MONITOR (EIIS CODE KN) INOPERABLE. HOWEVER, COMPENSATORY ACTIONS WERE NOT TAKEN AS REQUIRED BY UNIT 1 TECHNICAL SPECIFICATIONS SECTION 4.6.F.2 TO ENSURE THAT REACTOR COOLANT CONDUCTIVITY WAS MONITORED ON A CONTINUOUS OR PERIODIC BASIS. THE CONDITION WAS IDENTIFIED AT APPROXIMATELY 2315 CDT, ON 10/03/89, AT WHICH TIME LIMITING CONDITION FOR OPERATION (LCO) 1-89-468 WAS INITIATED TO ENSURE THAT PERIODIC ANALYSIS OF THE REACTOR COOLANT FOR CONDUCTIVITY WAS PERFORMED AS REQUIRED BY THE TECHNICAL SPECIFICATIONS. THE ROOT CAUSE OF THE EVENT WAS COGNITIVE PERSONNEL ERROR. LICENSED PERSONNEL FAILED TO RECOGNIZE THAT CLOSING OF THE VALVE RENDERED THE CONDUCTIVITY MONITOR INOPERABLE AND THAT COMPENSATORY ACTIONS WERE REQUIRED. CORRECTIVE ACTIONS FOR THE EVENT INCLUDED THE INITIATION OF AN IN-LINE CONDUCTIVITY SURVEILLANCE AS REQUIRED BY THE TECHNICAL SPECIFICATIONS AND WILL INCLUDE THE COUNSELING OF INVOLVED PERSONNEL.

[64] HATCH 1 DOCKET 50-321 LER 89-015
D/G 1B INOPERABLE DUE TO INSTALLATION OF WRONG PART.
EVENT DATE: 100989 REPORT DATE: 110689 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 2 (BWR)

(NSIC 215366) ON 10/9/89 AT APPROXIMATELY 0900 CDT, UNIT 1 WAS IN THE RUN MODE AT APPROXIMATELY 2436 CMWT (APPROXIMATELY 100% OF RATED THERMAL POWER) AND UNIT 2 WAS IN THE REFUEL MODE WITH ALL FUEL REMOVED FROM THE CORE. AT THAT TIME, PROCEDURE 34SV-R43-002-2S, "DIESEL GENERATOR 1B MONTHLY TEST," WAS BEING PERFORMED. WHILE ATTEMPTING A LOCAL, MANUAL START OF DIESEL GENERATOR (D/G) 1R43-S001B PER PROCEDURE, THE D/G FAILED TO START. SUBSEQUENT INVESTIGATION REVEALED THAT APPROXIMATELY TWO GALLONS OF LUBRICATING OIL HAD ACCUMULATED BETWEEN THE PISTONS OF THE NUMBER 2 CYLINDER, HYDRAULICALLY LOCKING THE PISTONS. THE OIL ACCUMULATED AS A RESULT OF AN INCORRECT TYPE OF PUMP THAT HAD BEEN INSTALLED IN THE D/G'S STANDBY CIRCULATING LUBRICATING OIL SYSTEM ON 9/5/89. MONTHLY OPERABILITY TESTING OF D/G 1R43-S001B WAS COMPLETED SATISFACTORILY ON 9/15/89, TEN DAYS AFTER THE INCORRECT PUMP WAS INSTALLED AND PLACED IN SERVICE. THEREFORE, IT IS CONCLUDED D/G 1R43-S001B BECAME INOPERABLE AT SOME INDETERMINATE POINT BETWEEN 9/25/89 AND 10/9/89. THE OTHER FOUR D/GS WERE NOT AFFECTED BY THIS EVENT. THE ROOT CAUSE OF THIS EVENT IS PERSONNEL ERROR. THE INCORRECT MODEL NUMBER HAD BEEN ASSIGNED TO THE WAREHOUSE STOCK NUMBER FOR THE CIRCULATING LUBRICATING OIL PUMP MPL NUMBER. CONSEQUENTLY, THE INCORRECT TYPE OF PUMP WAS ISSUED AND INSTALLED. A CORRECTIVE ACTION FOR THIS EVENT INCLUDED REPLACING THE PUMP.

[65] HATCH 1 DOCKET 50-321 LER 89-014
PERSONNEL ERROR LEADS TO GROUP 5 ISOLATION OF PRIMARY CONTAINMENT ISOLATION SYSTEM.
EVENT DATE: 101089 REPORT DATE: 110889 NSSS: GE TYPE: BWR

(NSIC 215865) ON 10/10/89, AT APPROXIMATELY 1020 CDT, UNIT 1 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2436 MWT (APPROXIMATELY 100% OF RATED THERMAL POWER). AT THAT TIME, PLANT EQUIPMENT OPERATORS (PEOS), IN ACCORDANCE WITH PROCEDURE, OPENED VALVE 1G31-F052A TO PLACE INTO SERVICE THE REACTOR WATER CLEANUP (RWCU) SYSTEM FILTER/DEMINERALIZER (F/D) 1G31-D002A. OPENING THE VALVE CAUSED A PRESSURE TRANSIENT WHICH ACTUATED THE RWCU SYSTEM HIGH DIFFERENTIAL FLOW ALARM. A PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) GROUP 5 ISOLATION THEN OCCURRED PER DESIGN. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. MISCOMMUNICATION RESULTED IN THE INABILITY TO QUICKLY RECOGNIZE AND MITIGATE THE DIFFERENTIAL FLOW CONDITION, IN THAT THE PEO WHO OPENED THE VALVE MISTAKENLY BELIEVED COMMUNICATION HAD BEEN ESTABLISHED WITH THE MAIN CONTROL ROOM PRIOR TO THE ATTEMPT TO PLACE THE F/D INTO SERVICE. A CONTRIBUTING FACTOR IS THE NON-FAULT TOLERANT DESIGN OF THE RWCU SYSTEM F/DS, IN THAT OPENING THE F/D ISOLATION VALVE CAN RESULT IN A PRESSURE/FLOW TRANSIENT. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDE COUNSELLING THE INVOLVED PERSONNEL, ISSUING A MEMORANDUM EMPHASIZING THE NEED TO USE THE VERBAL REPEAT BACK TECHNIQUE FOR CONFIRMING COMMANDS BEFORE EXECUTING ACTIONS, AND SCHEDULING A DESIGN CHANGE TO THE RWCU SYSTEM.

[66] HOPE CREEK 1 DOCKET 50-354 LER 89-018
TECH SPEC REQUIRED READINGS NOT TAKEN DUE TO MISCOMMUNICATION OF REQUIREMENTS DUE TO PERSONNEL ERROR.
EVENT DATE: 092989 REPORT DATE: 110289 NSSS: GE TYPE: BWR

(NSIC 215883) ON OCTOBER 3, 1989, IT WAS DETERMINED THAT TWO TECHNICAL SPECIFICATION REQUIRED READINGS WERE NOT TAKEN AS REQUIRED WHEN THE NORTH PLANT VENT (OFFGAS) MONITORING SYSTEM WAS INOPERABLE ON 9/29/89. THIS CONDITION WAS DISCOVERED DURING A SHIFT TURNOVER MEETING. IMMEDIATELY UPON DISCOVERY THE REQUIRED READINGS WERE TAKEN, AND STEPS IMPLEMENTED TO ENSURE PROPER RECORDING OF THE READINGS UNTIL THE MONITORING SYSTEM WAS RETURNED TO SERVICE. THE ROOT CAUSE OF THIS EVENT WAS A LACK OF ADEQUATE COMMUNICATION IN THAT THE EQUIPMENT OPERATOR RESPONSIBLE FOR TAKING THE READINGS WAS NOT INFORMED TO BEGIN TAKING THE READINGS WHEN THE MONITORING SYSTEM WAS DECLARED INOPERABLE. CORRECTIVE ACTIONS CONSIST OF REVIEWING THIS EVENT WITH ALL SHIFT PERSONNEL, AND COUNSELLING FOR THE

INDIVIDUALS WHO WERE RESPONSIBLE FOR ENSURING THAT THE REQUIRED READINGS WERE TAKEN.

[67] HOPE CREEK 1 DOCKET 50-354 LER 89-019
TECH SPEC VIOLATION DUE TO USE OF INADEQUATE INSTRUMENTATION ON CORE SPRAY PUMPS FOR ASME SECTION XI TESTING DUE TO INADEQUATE DESIGN CHANGE PACKAGE.
EVENT DATE: 100489 REPORT DATE: 110389 NSSS: GE TYPE: BWR

(NSIC 215884) ASME CODE IWP-4120 STATES, "THE FULL SCALE RANGE OF EACH INSTRUMENT (USED FOR PROCESS MEASUREMENT DURING THE TEST) SHALL BE THREE TIMES THE REFERENCE VALUE OR LESS." THE CORE SPRAY SYSTEM FLOW INDICATORS UTILIZED FOR PUMP PERFORMANCE MONITORING IN THESE TESTS HAVE A SCALE OF 0 TO 10,000 GPM. THE BASELINE RATED FLOW OF EACH CORE SPRAY PUMP IS 3200 GPM, HENCE, THE MAXIMUM RANGE ALLOWED ON ASSOCIATED FLOW INSTRUMENTATION (FOR TEST PURPOSES) IS 9600 GPM. BECAUSE TECH SPEC 4.0.5 MANDATES ADHERENCE TO ASME BOILER AND PRESSURE VESSEL CODE SECTION XI, THIS CONDITION CONSTITUTES A TECH SPEC VIOLATION. ALL CORE SPRAY PUMPS WERE DECLARED INOPERABLE ON 10/4/89, AND THE REQUIRED QUARTERLY INSERVICE TESTS WERE COMPLETED WITH PORTABLE TEST EQUIPMENT MEETING THE SECION XI REQUIREMENTS. THE PRIMARY CAUSE OF THIS OCCURRENCE WAS AN INADEQUATE DESIGN CHANGE PACKAGE (DCP) WHICH WAS UTILIZED TO DETERMINE BASELINE DATA FOR ASME SECTION XI REQUIRED SINGLE PUMP TESTING OF CORE SPRAY PUMPS. THE IMPLICATIONS OF ASME SECTION XI INSTRUMENTATION REQUIREMENTS WERE NOT RECOGNIZED DURING THE PREPARATION AND REVIEW OF THIS DCP. THE PRIMARY CORRECTIVE ACTION FOR THIS EVENT CENTERS ON THE CREATION OF AN IST INSTRUMENT VERIFICATION PROJECT. THE SCOPE OF THIS PROJECT INCLUDES VERIFICATION THAT ALL IST REQUIRED INSTRUMENTATION MEETS THE REQUIREMENTS.

[68] HOPE CREEK 1 DOCKET 50-354 LER 89-020
ENGINEERED SAFETY FEATURES ACTUATION DURING PERFORMANCE OF SENSOR CALIBRATION DUE TO PERSONNEL ERROR.
EVENT DATE: 101189 REPORT DATE: 111389 NSSS: GE TYPE: BWR

(NSIC 215886) ON OCTOBER 11, 1989 DURING PERFORMANCE OF AN I&C SENSOR CALIBRATIN PROCEDURE, A PRESSURE SPIKE IN THE ASSOCIATED REACTOR VESSEL LEVEL REFERENCE LEG RESULTED IN AN ENGINEERED SAFETY FEATURES (ESF) ACTUATION OF SEVERAL PLANT SYSTEMS/COMPONENTS. THE PRESSURE SPIKE WAS INDUCED IN THE REFERENCE LEG DURING THE PROCESS OF FILLING AND VENTING A RECENTLY REPLACED PRESSURE TRANSMITTER. THE ROOT CAUSE OF THIS INCIDENT WAS A PERSONNEL ERROR ON THE PART OF THE TECHNICIANS PERFORMING THE SENSOR CALIBRATION IN NOT PROPERLY FOLLOWING THE ASSOCIATED PROCEDURE FOR THE FILLING AND VENTING EVOLUTION. CORRECTIVE ACTIONS CONSIST OF COUNSELLING FOR THE TECHNICIANS INVOLVED IN THE INCIDENT.

[69] HOPE CREEK 1 DOCKET 50-354 LER 89-021
DEVIATION FROM ELECTRICAL SEPARATION CRITERIA BETWEEN TRANSIENT MONITORING CIRCUITRY AND REACTOR PROTECTION SYSTEM PANEL CIRCUITRY.
EVENT DATE: 101389 REPORT DATE: 111389 NSSS: GE TYPE: BWR

(NSIC 215885) ON 10/13/89, THE SENIOR NUCLEAR SHIFT SUPERVISOR (SNSS, SRO LICENSED) WAS INFORMED BY I&C SYSTEMS ENGINEERING THAT AN ENGINEERING REVIEW OF TWO DESIGN CHANGES AFFECTING THE GENERAL ELECTRIC TRANSIENT ANALYSIS RECORDING SYSTEM (GETARS) CONCLUDED THAT CLASS 1E ELECTRICAL SEPARATION CRITERIA HAD NOT BEEN MET IN TWO REACTOR PROTECTION SYSTEM (RPS) PANELS. POWER FROM AN EXTERNAL CLASS 1E ENGINEERED SAFETY FEATURES (ESF) UNINTERRUPTABLE POWER SUPPLY (UPS) WAS CONNECTED TO A GETARS MULTIPLEXER, AND THAT SOME MONITORED RPS CONTACTS DOWNSTREAM OF THE MULTIPLEXER WERE ALSO POWERED FROM THIS SAME SOURCE. THIS CONFIGURATION IS IN VIOLATION OF SEPARATION CRITERIA AS ESTABLISHED BY REG GUIDE 1.75. ACTIONS WERE IMMEDIATELY TAKEN TO RECTIFY THE ELECTRICAL SEPARATION DEVIATIONS - PRIMARILY, REMOVING THE POWER SUPPLIES WHICH DID NOT CONFORM TO SEPARATION CRITERIA, AND RE-POWERING AFFECTED GETARS COMPONENTS FROM INTERNAL RPS PANEL POWER SUPPLIES. INVESTIGATION INTO THE CIRCUMSTANCES BEHIND THIS EVENT IS NOT YET COMPLETE, PENDING RESOLUTION OF THE ENGINEERING ANALYSIS OF THIS EVENT. THE ENGINEER WHO PERFORMED THE ANALYSIS IS CURRENTLY UNAVAILABLE, DUE TO AN AUTOMOBILE ACCIDENT, TO RESOLVE ISSUES RELATED TO THIS INCIDENT. A SUPPLEMENT TO

THIS REPORT WILL BE SUBMITTED NO LATER THAN 12/1/89 DETAILING THE CIRCUMSTANCES BEHIND THIS OCCURRENCE.

[70] HOPE CREEK 1 DOCKET 50-354 LER 89-022
LOSS OF SHUTDOWN COOLING DUE TO REACTOR PROTECTION SYSTEM ACTUATION DUE TO USE OF DEFICIENT PROCEDURE.
EVENT DATE: 101689 REPORT DATE: 111589 NSSS: GE TYPE: BWR

(NSIC 215887) ON 10/16/89 AT 1503 HOURS, THE "B" CHANNEL RPS ELECTRIC PROTECTION ASSEMBLIES (EPA'S) OPENED, RESULTING IN A "B" CHANNEL HALF SCRAM, A "B" AND "D" NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) ISOLATION AND A LOSS OF "B" RHR WHICH HAD BEEN OPERATING IN SHUTDOWN COOLING MODE. FOLLOWING THIS EVENT, TROUBLESHOOTING WAS INITIATED WHICH INCLUDED MONITORING THE "B" EPA WITH EXTERNAL TEST EQUIPMENT. DURING THIS TROUBLESHOOTING INTERVAL, NINE (9) OTHER IDENTICAL OR VERY SIMILAR EVENTS OCCURRED. THE TROUBLESHOOTING IDENTIFIED THE CAUSE OF THE EPA TRIPS TO BE AN UNDERVOLTAGE SIGNAL FROM THE MOTOR-GENERATOR (MG) SET. AFTER THE MG OUTPUT VOLTAGE WAS ADJUSTED USING A DIGITAL VOLT METER, NO FURTHER NSSSS ISOLATIONS HAVE OCCURRED. THESE EVENTS WERE OF MINIMAL SAFETY SIGNIFICANCE. THE ROOT CAUSE WAS DETERMINED TO BE RESETTING THE MG SETS VOLTAGE OUTPUT USING THE PANEL METER. A CONTRIBUTING CAUSE WAS DEFICIENT PROCEDURES. CORRECTIVE ACTIONS INCLUDE PROCEDURE CHANGES AND TRENDING OF MG SETS OUTPUT VOLTAGE.

[71] LA SALLE 1 DOCKET 50-373 LER 89-012 REV 01
UPDATE ON REACTOR CORE ISOLATION COOLING HI STEAM FLOW ISOLATION SWITCH FAILED DIAPHRAGM.
EVENT DATE: 030989 REPORT DATE: 110889 NSSS: GE TYPE: BWR
VENDOR: STATIC-O-RING

(NSIC 215907) ON 3/9/89, AT 2045 HOURS, DURING PERFORMANCE OF LASALLE INSTRUMENT SURVEILLANCE LIS-RI-101, "UNIT 1 STEAM LINE HIGH FLOW REACTOR CORE ISOLATION COOLING (RCIC) ISOLATION CALIBRATIONAL TEST," PRESSURE DIFFERENTIAL SWITCH 1E31-N013BB WAS FOUND TO HAVE A DIAPHRAGM LEAK. UNIT 1 WAS IN OPERATIONAL CONDITION 1 (RUN) AT 100% POWER LEVEL. THE SETPOINT FOR THIS SWITCH WAS FOUND WITHIN THE ACTION LIMIT AND THE LIMITING CONDITION FOR OPERATION (LCO). THIS SWITCH FUNCTIONS WITH SIMILAR SWITCH 1E31-N013BA, TO PROVIDE DIVISION II (INBOARD) ISOLATION OF THE RCIC/RESIDUAL HEAT REMOVAL STEAM LINE AND TO INITIATE A RCIC TURBINE TRIP. 1E31-N013BB IS CONNECTED IN REVERSE PARALLEL TO 1E31-N013BA. THE DESIGN FUNCTION OF 1E31-N013BB IS TO INITIATE AN INBOARD CONTAINMENT ISOLATION IN THE EVENT OF AN INSTRUMENT LINE BREAK. PARTICULARLY A BREAK IN THE INSTRUMENT LINE COMING FROM THE HIGH PRESSURE SIDE OF THE STEAM FLOW ELBOW LEADING TO THE HIGH PRESSURE SIDE OF 1E31-N013BA AND TO THE LOW PRESSURE SIDE OF 1E31-N013BB. RCIC SYSTEM HAD BEEN DECLARED INOPERABLE ON 3/9/89 AT 1300 HOURS IN ORDER TO PERFORM THE REQUIRED SURVEILLANCE. THE HIGH PRESSURE CORE SPRAY SYSTEM REMAINED OPERABLE THROUGHOUT THE DURATION OF THIS EVENT. A REPLACEMENT SWITCH WAS INSTALLED, CALIBRATED AND FUNCTIONALLY TESTED SATISFACTORILY. THE CAUSE OF FAILURE WAS DUE TO A TEAR FOUND IN THE DIAPHRAGM.

[72] LA SALLE 1 DOCKET 50-373 LER 89-024
UNSEALED OPENINGS IN CONTROL ROOM DUE TO ORIGINAL CONSTRUCTION.
EVENT DATE: 101389 REPORT DATE: 111389 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 2 (BWR)

(NSIC 215908) ON 10/13/89 WITH UNIT 1 DEFUELED AND UNIT 2 IN COLD SHUTDOWN AN UNSEALED OPENING IN THE MAIN CONTROL ROOM FLOOR WAS DISCOVERED. SUBSEQUENTLY ON 10/30/89 WITH UNIT 1 DEFUELED AND UNIT 2 IN OPERATIONAL CONDITION 1 (RUN) ANOTHER UNSEALED OPENING IN THE MAIN CONTROL ROOM WEST WALL WAS DISCOVERED. BASED ON A VISUAL INSPECTION OF THE OPENINGS AND THE FACT THAT THEY WERE NOT INTENDED PENETRATIONS, IT WAS DETERMINED THAT THE OPENINGS HAD EXISTED SINCE THE BARRIERS WERE CONSTRUCTED. WORK TO SEAL THESE OPENINGS WAS QUICKLY PERFORMED. IN ADDITION, SMOKE TESTS AND VISUAL INSPECTIONS OF THE ENTIRE CONTROL ROOM FLOOR, WALL, AND CEILINGS WERE PERFORMED TO VERIFY THAT NO OTHER OPENINGS WHICH COULD COMPROMISE THE FIRE RATING OF THE BARRIERS EXISTED. DURING THIS INSPECTION, IT WAS OBSERVED THAT A CONTROL ROOM PENETRATION HAD AN EXCESSIVE GAP BETWEEN A

VENTILATION DUCT PASSING THROUGH IT AND THE WALL. THE REASON FOR THIS WAS THAT THE ANGLE IRON DESIGNED TO SEAL THIS GAP HAD MOVED AWAY FROM THE WALL. A WORK REQUEST TO REPAIR THE PENETRATION WAS IMMEDIATELY WRITTEN. NO OTHER DEFICIENCIES IN THE BARRIER WAS OBSERVED. THIS EVENT IS REPORTABLE PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(I) DUE TO A DEVIATION FROM THE PLANT TECH SPECS.

[73] LA SALLE 2 DOCKET 50-374 LER 89-014
PRIMARY CONTAINMENT ISOLATION SYSTEM GROUP 4 ISOLATION DURING GROUND ISOLATION.
EVENT DATE: 102089 REPORT DATE: 111789 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 1 (BWR)

(NSIC 215909) ON 10/20/89 AT 0123 HOURS WITH UNIT 2 IN OPERATIONAL CONDITION 1 (RUN MODE) A PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS, PC) (JM) GROUP 4 ISOLATION OCCURRED. THIS RESULTED IN THE CLOSURE OF THE UNIT 2 REACTOR BUILDING VENTILATION (VR) (VA) ISOLATION DAMPERS, TRIP OF THE UNIT 2 RUNNING VR SUPPLY AND EXHAUST FANS AND AUTO START OF THE UNIT 2 STANDBY GAS TREATMENT TRAIN (SBGT, VG) (BH). THE REACTOR BUILDING ISOLATION DAMPERS SERVE TO ISOLATE THE SECONDARY CONTAINMENT (CS)(NG). THE PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) GROUP 4 ISOLATION OCCURRED DURING THE PERFORMANCE OF A DC GROUND ISOLATION. THE GROUND ISOLATION WAS PERFORMED WITHOUT USING EXISTING PROCEDURES BECAUSE THE EXISTING PROCEDURES WERE NOT WRITTEN FOR THE CROSSTIED CONDITION THAT THE DIVISION 2 DC BUSES WERE CURRENTLY IN. THIS RESULTED IN DE-ENERGIZING THE DIVISION 2 PCIS LOGIC DC POWER SUPPLY. AFTER FURTHER INVESTIGATION THE GROUND WAS DETERMINED TO BE THE DC POWER SUPPLY TO THE VISUAL ANNUNCIATOR LOGIC PANEL 1PA08J. THE POWER SUPPLY WAS REPLACED. THIS EVENT IS REPORTABLE TO THE NRC PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV) DUE TO THE ACTUATION OF AN ENGINEERED SAFETY FEATURE SYSTEM.

[74] LIMERICK 1 DOCKET 50-352 LER 89-010 REV 01
UPDATE ON REFUEL FLOOR SECONDARY CONTAINMENT ISOLATION DUE TO LOW NEGATIVE DIFFERENTIAL PRESSURE AS A RESULT OF AN APPARENT DESIGN DEFICIENCY.
EVENT DATE: 020489 REPORT DATE: 111589 NSSS: GE TYPE: BWR

(NSIC 215878) ON FEBRUARY 4, 1989, AT 1818 HOURS, ISOLATION OF THE REFUEL FLOOR (RF) SECONDARY CONTAINMENT AND INITIATION OF THE STANDBY GAS TREATMENT SYSTEM (SGTS), ENGINEERED SAFETY FEATURES (ESF), OCCURRED ON LOW NEGATIVE DIFFERENTIAL PRESSURE BETWEEN THE RF SECONDARY CONTAINMENT AND THE OUTSIDE ENVIRONMENT. A REACTOR ENCLOSURE (RE) SUPPLY DUCT HIGH TEMPERATURE CONDITION SIGNAL CAUSED ISOLATION OF STEAM TO THE RF VENTILATION SUPPLY FAN HEATING COILS. THE RF VENTILATION SUPPLY FANS TRIPPED ON LOW TEMPERATURE AND THE RF VENTILATION EXHAUST FANS CONTINUED TO OPERATE UNTIL THE RF DIFFERENTIAL PRESSURE REACHED THE EXHAUST FAN TRIP SETPOINT. WITH THE NORMAL SUPPLY AND EXHAUST FANS OFF, THE RF DIFFERENTIAL PRESSURE DECAYED TO GREATER THAN NEGATIVE 0.1 INCHES H2O. THE RF SECONDARY CONTAINMENT AND SGTS THEN OPERATED AS DESIGNED TO RESTORE ACCEPTABLE NEGATIVE DIFFERENTIAL PRESSURE BETWEEN THE RF SECONDARY CONTAINMENT AND OUTSIDE ENVIRONMENT. SGTS WAS SECURED AND RF VENTILATION PLACED IN RESET TO ALLOW TROUBLESHOOTING. THE STEAM SUPPLY TO THE RF AND RE VENTILATION SUPPLY FAN ROOM ISOLATION VALVE WAS BYPASSED AND A TEMPORARY CIRCUIT ALTERATION (TCA) WAS INSTALLED TO BYPASS THE LOW TEMPERATURE TRIP OF THE RF VENTILATION SUPPLY FANS. THE RF SECONDARY CONTAINMENT ISOLATION WAS RESET AND THE NORMAL RF VENTILATION SYSTEM RETURNED TO SERVICE.

[75] LIMERICK 1 DOCKET 50-352 LER 89-054 REV 01
UPDATE ON TECH SPEC VIOLATION DUE TO MISSED SURVEILLANCE OF CHANNEL 'D' REACTOR HIGH LEVEL TRIP (FEEDWATER/MAIN TURBINE) BECAUSE OF A PROCEDURAL DEFICIENCY.
EVENT DATE: 102589 REPORT DATE: 111589 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 215880) ON OCTOBER 25, 1989, PLANT STAFF ENGINEERS DISCOVERED THAT THE DAILY CHANNEL CHECK SURVEILLANCE REQUIREMENTS FOR THE 'D' CHANNEL REACTOR HIGH LEVEL TRIP (FEEDWATER/MAIN TURBINE) WAS NOT BEING MET AND THE ASSOCIATED TECHNICAL SPECIFICATIONS (TS) ACTIONS WERE NOT INITIATED. THIS RESULTED IN A CONDITION PROHIBITED BY TS. THIS CONDITION WAS CAUSED BY A PROCEDURAL DEFICIENCY

AND HAS EXISTED SINCE INITIAL PLANT OPERATION OF BOTH UNIT 1 AND UNIT 2. THE ORIGINAL PLANT DESIGN DID NOT INCLUDE A LEVEL INDICATOR FOR FEEDWATER/MAIN TURBINE TRIP SYSTEM CHANNEL 'D'. THE CONSEQUENCES OF THIS EVENT WERE MINIMAL AS THE REMAINING CHANNELS WERE SUFFICIENT TO PROVIDE THE TRIP FUNCTION. THE DEFICIENT DAILY SURVEILLANCE LOG PROCEDURES WERE REVISED AND THE REQUIRED CHANNEL CHECK FOR TRIP SYSTEM CHANNEL 'D' WAS SUCCESSFULLY COMPLETED AT APPROXIMATELY 2345 ON OCTOBER 25, 1989 FOR UNIT 1 AND UNIT 2.

[76] LIMERICK 2 DOCKET 50-353 LER 89-004
INOPERABILITY OF REACTOR ENCLOSURE COOLING WATER RADIATION MONITOR IN EXCESS OF TS ALLOWED TIME DUE TO A PERSONNEL ERROR.
EVENT DATE: 072589 REPORT DATE: 082889 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215138) ON JULY 25, 1989, THE UNIT 2 REACTOR ENCLOSURE COOLING WATER (RECW) RADIATION MONITOR WAS DETERMINED BY PLANT STAFF TO HAVE BEEN INOPERABLE FOR A TIME IN EXCESS OF THAT ALLOWED BY TECHNICAL SPECIFICATIONS (TS) WITHOUT THE ASSOCIATED ACTION BEING TAKEN RESULTING IN A CONDITION PROHIBITED BY TS. ON JUNE 26, 1989, THE MONITOR WAS DECLARED INOPERABLE DUE TO AN ERRATIC INDICATOR, AND CHEMISTRY BEGAN THE TS ACTION OF SAMPLING. ON JULY 19, 1989, AN INVALID RECW RADIATION BACKGROUND MEASUREMENT WAS OBTAINED BY UNIT 2 INSTRUMENTATION AND CONTROL (I&C) PERSONNEL AND THEN INAPPROPRIATELY UTILIZED IN PERFORMANCE OF THE FUNCTIONAL ST. ON JULY 21, 1989, THE SYSTEM WAS DECLARED OPERABLE BY SHIFT SUPERVISION AFTER REVIEW TO VERIFY NO OUTSTANDING STS OR MAINTENANCE REQUEST FORMS. ON JULY 25, 1989, PLANT STAFF DISCOVERED THAT. PERFORMANCE OF THE FUNCTIONAL ST RESULTED IN NON-CONSERVATIVE ALARM SETPOINTS FOR THE MONITOR. SHIFT SUPERVISION WAS NOTIFIED, THE SYSTEM WAS DECLARED INOPERABLE, AND CHEMISTRY PERFORMED REQUIRED SAMPLING. THE CAUSE OF THIS EVENT WAS A PERSONNEL ERROR COMPOUNDED BY A PROCEDURAL DEFICIENCY AND THE EXISTANCE OF A UNIT 2 STARTUP I&C GROUP WHICH WAS SEPARATE FROM THE UNIT 1 I&C GROUP. THE I&C PERSONNEL INVOLVED WERE COUNSELED, CHEMISTRY STS SIMILAR TO AND INCLUDING THE ONE ASSOCIATED WITH THE EVENT WILL BE REVISED, AND THE TWO SEPAR

[77] LIMERICK 2 DOCKET 50-353 LER 89-009
EIGHT REACTOR CORE ISOLATION COOLING SYSTEM ACTUATIONS.
EVENT DATE: 082089 REPORT DATE: 110789 NSSS: GE TYPE: BWR

(NSIC 215881) BETWEEN AUGUST 20, 1989 AND SEPTEMBER 25, 1989, EIGHT REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM ACTUATION CYCLES OCCURRED UNDER THE PURVIEW OF THE UNIT 2 START UP TEST PROGRAM. THESE RCIC SYSTEM TESTS WERE COMPLETED SATISFACTORILY DURING UNIT 2 TEST CONDITIONS 1 AND 2. THIS SPECIAL REPORT IS BEING SUBMITTED PURSUANT TO TECHNICAL SPECIFICATION (TS) REPORTING REQUIREMENT 6.9.2, AS REQUIRED BY TS ACTION 3.7.3.B.

[78] MAINE YANKEE DOCKET 50-309 LER 89-004
PLANT SHUTDOWN DUE TO CONTAINMENT PURGE VALVE LEAKAGE IN EXCESS OF TECH SPEC LIMITS.
EVENT DATE: 101089 REPORT DATE: 110989 NSSS: CE TYPE: PWR
VENDOR: ALLIS CHALMERS

(NSIC 215857) AT 1230 ON 10/10/89, MAINE YANKEE INITIATED A PLANT SHUTDOWN AS REQUIRED BY PLANT TECH SPECS. A TYPE C CONTAINMENT LEAKAGE RATE TEST ON THE CONTAINMENT VENTILATION AND PURGE INLET PENETRATION DEMONSTRATED A LEAKAGE RATE THAT RESULTED IN A COMBINED CONTAINMENT LEAKAGE RATE IN EXCESS OF THE TECH SPEC LIMIT. ADDITIONAL TESTING SHOWED EXCESSIVE LEAKAGE PAST THE SEAT OF THE OUTBOARD PURGE ISOLATION VALVE. THE SHUTDOWN CONTINUED DUE TO THE INABILITY TO QUANTIFY THE LEAKAGE PAST THE INBOARD ISOLATION. THE PLANT WAS PLACED IN HOT SHUTDOWN AT 1740. UPON THE REPAIR OF THE OUTBOARD ISOLATION VALVE SEAT, THE INBOARD ISOLATION VALVE LEAKAGE RATE WAS DETERMINED TO BE LESS THAN THE TECH SPEC CONTAINMENT LEAKAGE RATE LIMIT. NO SPECIFIC FAILURE MECHANISM WAS IDENTIFIED. THE SEATS ON THE INBOARD AND OUTBOARD ISOLATION VALVES WERE ADJUSTED TO OBTAIN A SATISFACTORY LEAKAGE RATE.

[79] MAINE YANKEE DOCKET 50-309 LER 89-005
EMERGENCY CORE COOLING SYSTEM (ECCS) VALVE NOT FULLY CLOSED.
EVENT DATE: 101589 REPORT DATE: 111489 NSSS: CE TYPE: PWR

(NSIC 215859) DURING PLANT HEATUP ON 10/15/89, WHILE SHUTTING DOWN THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM, OPERATORS IDENTIFIED LEAKAGE PAST THE B TRAIN RHR PUMP SUCTION VALVE, RH-7. DURING NORMAL OPERATION, THE RHR PUMP SERVES AS THE B TRAIN LOW PRESSURE SAFETY INJECTION (LPSI) PUMP. TECH SPECS REQUIRE RH-7 TO BE LOCKED SHUT FOR EMERGENCY CORE COOLING SYSTEM (ECCS) OPERATION. ALSO, RH-7 IS ONE OF THE LOCKED SHUT CONTAINMENT ISOLATION BARRIERS FOR THE RHR PENETRATION. INVESTIGATION DETERMINED THAT THE POSITION INDICATION SLOT WAS TOO SHORT ON THE RH-7 HANDWHEEL PEDESTAL. AS A RESULT, THE POSITION INDICATION PIN WAS RESTRAINED BY THE BOTTOM OF THE SLOT, PREVENTING FULL VALVE CLOSURE. THE POSITION INDICATION PIN HAS BEEN REMOVED. TECH SPEC REQUIREMENTS FOR CONTAINMENT INTEGRITY WERE MET BECAUSE THE RHR CONTAINMENT PENETRATION WAS ISOLATED IN ACCORDANCE WITH THE PLANT TECH SPECS FOR CONTAINMENT INTEGRITY. THE ECCS FUNCTION OF THE VALVE WAS MET BY THE CONTAINMENT ISOLATION VALVES AND THE A TRAIN RHR PUMP SUCTION VALVE, RH-6. ECCS VALVES WITH SIMILAR LOCAL POSITION INDICATION ARRANGEMENTS WERE CHECKED AND NONE WERE FOUND WITH VALVE POSITION ADVERSELY IMPACTED. THE POSITION INDICATION SLOTS FOR 3 OTHER VALVES HAVE BEEN ELONGATED, TO PREVENT IMPACTING BY THE POSITION INDICATION PINS. POST-MAINTENANCE FUNCTIONAL TESTING REQUIREMENTS WILL BE REVISED BY THE END OF THE 1990 REFUELING OUTAGE.

[80] MCGUIRE 1 DOCKET 50-369 LER 88-019 REV 02
UPDATE ON FLOW RATES LISTED IN FSAR COULD NOT BE ACHIEVED FOR THE UNIT 2 HYDROGEN SKIMMER SYSTEM DUE TO DEFECTIVE PROCEDURE AND EQUIPMENT CONFIGURATION.
EVENT DATE: 071988 REPORT DATE: 103189 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 215806) ON 11/08/87, DURING A UNIT 1 CONTAINMENT CLOSEOUT INSPECTION, NRC NOTICED SOME HYDROGEN SKIMMER (VX) SYSTEM DAMPERS APPEARED TO BE CLOSED. NRC QUESTIONED OPERATIONS ABOUT THE POSITIONS OF THE DAMPERS AND OPERATIONS INSPECTED AND VERIFIED THAT EACH UNIT 1 VX SYSTEM DAMPERS WAS IN ITS PRE-OPERATIONAL POSITION. BECAUSE OF NRC CONCERNS, OPERATIONS AGREED TO PERFORM A FLOW BALANCE TEST ON THE UNIT 2 VX SYSTEM DURING THE 1988 REFUELING OUTAGE. ON 07/19/88, PERFORMANCE TOOK "AS FOUND" FLOW MEASUREMENTS ON THE UNIT 2 VX SYSTEM AND FOUND THAT SOME DAMPER COMPARTMENT FLOWS DID NOT MEET FLOW REQUIREMENTS. DESIGN WAS CONSULTED TO EVALUATE THE TEST RESULTS, AND REQUESTED FLOW MEASUREMENTS BE TAKEN AGAIN USING A MORE ACCURATE MEASURING DEVICE. THE REQUIRED FLOW DISTRIBUTIONS FOR INDIVIDUAL COMPARTMENTS WERE REANALYZED. THE RESULT OF THE RE-ANALYSIS WAS A SIGNIFICANT LOWERING OF THE FLOW RATES REQUIRED TO LIMIT POTENTIAL LOCAL HYDROGEN CONCENTRATION TO LESS THAN FOUR PERCENT BY VOLUME. DESIGN CONCLUDED THAT THE VX SYSTEM WITH ITS PRESENT FLOW BALANCE CONDITION IS SUFFICIENT AND THE VX SYSTEM IS CONSIDERED OPERABLE. THIS EVENT IS ASSIGNED A CAUSE OF DEFECTIVE PROCEDURE BECAUSE THE PRE-OPERATIONAL FLOW BALANCE TEST PROCEDURE DEMONSTRATED THE VX SYSTEM WOULD MEET THE TECH SPEC REQUIREMENTS; THEREFORE, DESIGN WAIVED THE FINAL TEST.

[81] MCGUIRE 1 DOCKET 50-369 LER 89-024
MAIN STEAM ISOLATION VALVES WERE TESTED IN A NON-CONSERVATIVE MANNER BECAUSE OF AN INSTALLATION DEFICIENCY.
EVENT DATE: 090889 REPORT DATE: 111089 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)
VENDOR: ATWOOD & MORRILL CO., INC.

(NSIC 215903) ON MAY 16, 1989, GENERAL OFFICE (GO) PERFORMANCE PERSONNEL WROTE PROBLEM INVESTIGATION REPORT (PIR) 0-M89-0122 AS A RESULT OF A REVIEW OF INFORMATION NOTICE (IEN) 88-51. IEN 88-51 DESCRIBED A FAILURE OF MAIN STEAM ISOLATION VALVES (MSIVS) TO CLOSE WITHOUT AIR ASSISTANCE. IT WAS DISCOVERED THAT MCGUIRE TIMED THE MSIVS TO CLOSE WITH AIR ASSISTANCE. SAFETY ANALYSIS CALCULATIONS TAKE NO CREDIT FOR AIR ASSISTANCE TO CLOSE. ON MAY 26, 1989, AN OPERABILITY EVALUATION WAS ISSUED STATING THAT THE MSIVS WOULD CLOSE AS REQUIRED WITHOUT AIR ASSISTANCE AND THE MSIVS SHOULD BE STROKE TIMED WITHOUT AIR

ASSISTANCE TO CLOSE DURING THE NEXT REFUELING OUTAGES. ON SEPTEMBER 8, 1989, THE MSIV STROKE TIMING PERIODIC TEST WAS PERFORMED WITHOUT THE AIR ASSISTANCE AND 3 MSIVS FAILED TO CLOSE WITHIN 5 SECONDS AS REQUIRED. PIR 0-M89-0239 WAS WRITTEN ON THE SAME DAY. AS A RESULT OF THIS PIR, AN OPERABILITY EVALUATION WAS ISSUED ON SEPTEMBER 11, 1989, STATING THAT THE MSIVS WERE OPERABLE WITH THE AIR ASSIST TO CLOSE FEATURE. THE MSIVS WERE TESTED WITH AIR ASSIST TO CLOSE AND THEY ALL SUCCESSFULLY PASSED THE MSIV VALVE STROKE TIMING TEST. THIS EVENT IS ASSIGNED A CAUSE OF INSTALLATION DEFICIENCY BECAUSE BRASS GUIDE SCREWS IN THE ACTUATORS WERE TIGHTENED EXCESSIVELY CAUSING THE MSIV ACTUATORS TO BIND WHEN CLOSING.

[82] MCGUIRE 1 DOCKET 50-369 LER 89-025
TURBINE DRIVEN AUXILIARY FEEDWATER PUMP AUTOMATICALLY STARTED BECAUSE OF AN INAPPROPRIATE ACTION.
EVENT DATE: 091889 REPORT DATE: 101889 NSSS: WE TYPE: PWR

(NSIC 215651) ON SEPTEMBER 18, 1989, AT APPROXIMATELY 1130, PERFORMANCE (PRF) PERSONNEL WERE PERFORMING A VALVE STROKE TIMING TEST FOR VALVE 1CA-27A, MOTOR DRIVEN AUXILIARY FEEDWATER (CA) PUMP 1A RECIRCULATION. IN THE PROCESS OF THE TEST, PRF TECHNICIAN A WAS PLACING A JUMPER ACROSS SLIDING LINK B-15 IN THE CA PUMP 1A AUXILIARY PANEL. THE JUMPER CAME LOOSE AND INADVERTENTLY MADE CONTACT WITH SLIDING LINK B-14 DIRECTLY ABOVE IT. THE TURBINE DRIVEN (T/D) CA PUMP AUTOMATICALLY STARTED AS A RESULT OF THIS CONTACT. OPERATIONS (OPS) PERSONNEL ATTEMPTED TO RESET THE T/D CA PUMP WITHOUT SUCCESS. AT 1140, OPS PERSONNEL INITIATED AN EMERGENCY WORK REQUEST TO HAVE INSTRUMENTATION AND ELECTRICAL (IAE) PERSONNEL TROUBLESHOOT THE ELECTRICAL CIRCUITRY OF THE T/D CA PUMP. THE PROBLEM WAS REPAIRED AND OPS PERSONNEL SECURED THE T/D CA PUMP AT 1246. THIS EVENT IS ASSIGNED A CAUSE OF INAPPROPRIATE ACTION BECAUSE PROPER EXECUTION OF THE TEST STEP FAILED DURING INSTALLATION OF THE JUMPER BECAUSE THE JUMPER TYPE CHOSEN FAILED TO HOLD. UNIT 1 WAS IN MODE 1, POWER OPERATION, AT 100 PERCENT POWER DURING THE EVENT. THIS EVENT HAS BEEN REVIEWED WITH APPROPRIATE PRF PERSONNEL AND THEY HAVE BEEN INSTRUCTED TO USE A DIFFERENT TYPE JUMPER IN FUTURE TESTS WHEN POSSIBLE. APPROPRIATE PROCEDURAL CHANGES WILL BE MADE TO PRF TEST PROCEDURES TO PREVENT RECURRENCE OF SIMILAR EVENTS.

[83] MCGUIRE 1 DOCKET 50-369 LER 89-031
TECH SPECS 3.0.3 WAS ENTERED ON UNIT 1 AND UNIT 2 DUE TO AN INAPPROPRIATE ACTION DURING MAINTENANCE ON THE CONTROL AREA VENTILATION SYSTEM.
EVENT DATE: 101289 REPORT DATE: 111389 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 215932) ON OCTOBER 12, 1989, INSTRUMENTATION AND ELECTRICAL (IAE) PERSONNEL WERE PERFORMING MAINTENANCE ON THE CHLORINE DETECTORS FOR THE CONTROL AREA VENTILATION (VC) SYSTEM. AT 1435, IAE PERSONNEL WERE RETURNING THE CHLORINE DETECTORS TO SERVICE BY RECONNECTING POWER LEADS. ONE OF THE POWER LEADS WAS ACCIDENTALLY DROPPED AND TOUCHED THE GROUNDING SCREW. THIS CAUSED THE AC POWER SUPPLY FUSE TO BLOW, RESULTING IN THE AUTOMATIC ISOLATION OF THE FOUR OUTSIDE AIR INTAKES ON THE VC SYSTEM. THIS RESULTED IN UNITS 1 AND 2 ENTERING TECHNICAL SPECIFICATION (TS) 3.0.3. OPERATIONS PERSONNEL REMOVED POWER FROM THE OUTSIDE AIR INTAKE VALVES AND THEN MANUALLY OPENED THE VALVES, THEREBY EXITING TS 3.0.3. HOWEVER, UNITS 1 AND 2 THEN ENTERED TS 3.3.3.1 WHICH REQUIRES THAT TWO CHANNELS OF THE VC AIR INTAKE RADIATION MONITORS BE OPERABLE. THE RADIATION MONITORS COULD STILL DETECT RADIATION; HOWEVER, THEY COULD NOT CLOSE THE VC SYSTEM OUTSIDE AIR INTAKE VALVES BECAUSE THE POWER HAD BEEN REMOVED. IAE PERSONNEL REPLACED THE FUSE. THE INTAKE VALVES WERE RETURNED TO SERVICE, AND TS 3.3.3.1 WAS EXITED. UNIT 1 AND UNIT 2 WERE IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER AT THE TIME OF THIS EVENT. THIS EVENT IS ASSIGNED A CAUSE OF INAPPROPRIATE ACTION BECAUSE THE ACTION TAKEN WAS ACCIDENTAL.

[84] MCGUIRE 2 DOCKET 50-370 LER 89-010
A LEAK OCCURRED ON CONTAINMENT SPRAY HEAT EXCHANGER 2A AFTER VALVE STROKE TIMING BECAUSE OF A DEFICIENT PROCEDURE AND AN INAPPROPRIATE ACTION.
EVENT DATE: 090589 REPORT DATE: 102389 NSSS: WE TYPE: PWR
VENDOR: DELTA SOUTHERN CO.

GARLOCK, INC.
WESCHLER ELECTRIC CORP.

(NSIC 215904) ON 9/5/89, AT APPROX. 1609, CONTROL ROOM (CR) PERSONNEL OPENED VALVE 2NS-18A, CONTAINMENT SPRAY (NS) PUMP 2A SUCTION FROM CONTAINMENT SUMP BLOCK, FOR STROKE TIME MEASUREMENT. CR PERSONNEL THEN NOTICED PRESSURIZER LEVEL DECREASING AND PRESSURIZER RELIEF TANK LEVEL INCREASING AND CLOSED VALVE 2NS-18A AT APPROX. 1655, A LEAK WAS REPORTED IN THE AUX. BLDG 716' ELEVATION PIPE CHASE. ABOUT A MINUTE LATER LEVEL BEGAN DECREASING IN THE REFUELING WATER STORAGE TANK. AFTER SEVERAL VALVE MANIPULATIONS, CR PERSONNEL DETERMINED THE PROBLEM WAS WITH NS HEAT EXCHANGER (HX) 2A. A VISUAL INSPECTION AT 1900 REVEALED A LEAK ON THE HX AT THE LOWER HEAD GASKET. CR PERSONNEL CLOSED VALVE 2NS-20A, NS PUMP 2A SUCTION FROM REFUELING WATER STORAGE TANK BLOCK, WHICH ISOLATED THE LEAK. LATER ESTIMATES CONCLUDED THAT APPROXIMATELY 10,000 GALLONS OF WATER LEAKED THROUGH THE GASKET. THIS EVENT IS ASSIGNED A CAUSE OF PROCEDURE DEFICIENCY BECAUSE OF AMBIGUOUS AND INCOMPLETE INFORMATION CONCERNING STROKING OF VALVE 2NS-18A. THIS EVENT IS ALSO ASSIGNED A CAUSE OF INAPPROPRIATE ACTION BECAUSE OF LACK OF ATTENTION TO DETAIL. THE GASKET WAS REPLACED AND THE AFFECTED NS SYSTEM PIPING WAS ANALYZED AND RETURNED TO SERVICE. UNIT 2 WAS IN MODE 5, COLD SHUTDOWN, WITH THE REACTOR COOLANT SYSTEM AT 305 PSI AND 139.5 DEGREES-F, AND RESIDUAL HEAT REMOVAL PUMP 2A IN OPERATION.

[85] MILLSTONE 1 DOCKET 50-245 LER 89-020
CONTAMINATED MATERIAL OUTSIDE THE RADIOLOGICAL CONTROL AREA.
EVENT DATE: 090689 REPORT DATE: 110689 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: MILLSTONE 2 (PWR)
SEABROOK 1 (PWR)

(NSIC 215815) THIS LICENSEE EVENT REPORT (LER) IS SUBMITTED FOR INFORMATION ONLY. ON SEPTEMBER 6, 1989, IN RESPONSE TO A PREVIOUS EVENT INVOLVING THE RELEASE OF A RADIOACTIVELY CONTAMINATED HYDROLASER MACHINE FROM THE SITE, A RADIOLOGICAL SURVEY WAS INITIATED WHICH IDENTIFIED CONTAMINATED TOOLS AND EQUIPMENT OUTSIDE OF THE RADIOLOGICAL CONTROL AREA (RCA) AND OUTSIDE OF THE MILLSTONE SITE PROTECTED AREA. THIS LER DISCUSSES SHORT AND LONG TERM CORRECTIVE ACTIONS AND FINDINGS TO DATE.

[86] MILLSTONE 1 DOCKET 50-245 LER 89-019
FAILURE TO COMPLETE SURVEILLANCE IN REQUIRED TIME.
EVENT DATE: 100389 REPORT DATE: 110289 NSSS: GE TYPE: BWR

(NSIC 215824) ON 10/3/89 AT 1835 HOURS WITH THE PLANT AT 100% POWER IT WAS DETERMINED THAT THE SURVEILLANCE TEST FOR THE ISOLATION CONDENSER ISOLATION INSTRUMENT FUNCTIONAL AND CALIBRATION TEST WAS PERFORMED BEYOND THE TIME INTERVAL REQUIRED BY THE PLANTS TECH SPECS. THIS SURVEILLANCE TEST WAS REQUIRED TO BE PERFORMED PRIOR TO 9/25/89. THE ACTUAL TEST PERFORMANCE DATE HOWEVER, WAS 10/3/89. THE RESULTS OF THIS TEST MET THE PRESCRIBED ACCEPTANCE CRITERIA.

[87] MILLSTONE 1 DOCKET 50-245 LER 89-021
REACTOR SCRAM ON TURBINE STOP VALVE CLOSURE.
EVENT DATE: 101989 REPORT DATE: 111789 NSSS: GE TYPE: BWR
VENDOR: COPES-VULCAN, INC.
CRANE COMPANY

(NSIC 215816) ON 10/19/89 AT 1515 HOURS WITH THE PLANT AT 70% POWER (530F AND 1000 PSIG) A FULL REACTOR SCRAM OCCURRED AS A RESULT OF A MAIN TURBINE TRIP (TURBINE STOP VALVES GREATER THAN 10% CLOSURE). THE MAIN TURBINE TRIP WAS THE RESULT OF A HIGH REACTOR WATER LEVEL TURBINE TRIP SIGNAL (+48"). THE HIGH REACTOR WATER CONDITION WAS THE RESULT OF "A" FEEDWATER REGULATING VALVE BECOMING STUCK IN THE OPEN POSITION WHICH OCCURRED WHILE RETURNING "B" FEEDWATER REGULATING VALVE TO SERVICE. DURING THE LEVEL DECREASE WHICH FOLLOWED THE SCRAM, STANDBY GAS TREATMENT SYSTEM INITIATED AS EXPECTED. ALL SYSTEMS FUNCTIONED AS EXPECTED. NO SAFETY CONSEQUENCES RESULTED FROM THIS EVENT.

[88] MILLSTONE 2 DOCKET 50-336 LER 89-008
INCOMPLETE SURVEILLANCE REQUIREMENTS FOR TESTING OF ANNUNCIATOR CIRCUITS.
EVENT DATE: 100289 REPORT DATE: 110189 NSSS: CE TYPE: PWR

(NSIC 215848) ON 10/2/89 AT 1300 HOURS WHILE OPERATING AT 92% NORMAL POWER OPERATION, A MANAGEMENT REVIEW OF A REPORTABLE OCCURRENCE SUBMITTED BY ANOTHER LICENSEE DETERMINED THAT MILLSTONE 2 WAS NOT PERFORMING A FUNCTIONAL CHECK OF CONTROL BOARD ANNUNCIATORS DURING TECH SPEC "CHANNEL FUNCTIONAL TESTS". THIS WAS CONSIDERED A CONDITION PROHIBITED BY THE PLANT'S TECH SPECS. THE FAILURE TO PERFORM A FUNCTIONAL TEST OF THE ANNUNCIATORS HAS NO SAFETY CONSEQUENCES SINCE THE TRIP AND OPERATIONAL FUNCTIONS OF THE EQUIPMENT WERE BEING VERIFIED AS REQUIRED. ADDITIONALLY, THE ANNUNCIATOR PORTION OF THE CIRCUIT PERFORMS NO SAFETY FUNCTION. ACTION WAS TAKEN TO COMMENCE A REVIEW OF ALL APPLICABLE INSTRUMENTATION AND CONTROL PROCEDURES.

[89] MILLSTONE 3 DOCKET 50-423 LER 89-023
INOPERABLE HYDRANT HOSE HOUSES DUE TO INSUFFICIENT FIRE HOSE INVENTORY.
EVENT DATE: 100689 REPORT DATE: 110689 NSSS: WE TYPE: PWR

(NSIC 215916) ON 10/6/89, AT APPROXIMATELY 1400 HOURS, IN MODE 1 AT 100% POWER, OPERATIONS DEPARTMENT (OD) SHIFT PERSONNEL DISCOVERED THAT ALL UNIT 3 HYDRANT HOSE HOUSES (HHHS) DID NOT HAVE THE REQUIRED FIRE HOSE LENGTHS. SHIFT PERSONNEL HAD REMOVED FIRE HOSES FROM 3 OF 11 HHHS ON 10/2/89 TO SUPPORT A NON-FIRE RELATED EMERGENCY. DURING PERFORMANCE OF A HOSE HOUSE FLUSH SURVEILLANCE ON 10/6, THE DEFICIENCY WAS INITIALLY IDENTIFIED FOR THE 3 HHHS. FURTHER INVESTIGATIONS DETERMINED THAT ALL HHHS DID NOT HAVE THE REQUIRED HOSE LENGTHS. ROOT CAUSES OF THIS EVENT WERE PERSONNEL ERROR AND PROCEDURAL DEFICIENCY. OD PERSONNEL REMOVED FIRE HOSES ON 10/2, WITHOUT TAKING COMPENSATORY ACTION. THE BUILDING SERVICES DEPARTMENT "HHH" SURVEILLANCE PERFORMED ON 10/5, IDENTIFIED THE DEFICIENCY IN THE 3 HHHS AND INCORRECTLY EVALUATED THE INVENTORY IN THE REMAINING 8 HHHS. THE PROCEDURE DID NOT REQUIRE IMMEDIATE NOTIFICATION OF THE SHIFT SUPERVISOR (SS) UPON DISCREPANCY IDENTIFICATION. IMMEDIATE CORRECTIVE ACTION WAS TO RESTORE ALL HHHS TO OPERABLE STATUS. A MEMO WAS ISSUED TO ALL OD SHIFT PERSONNEL PRESCRIBING THE USAGE REQUIREMENTS FOR HHH EQUIPMENT. CAUTION SIGNS WILL BE POSTED ON ALL HHHS BY 10/30/89. THE APPLICABLE SURVEILLANCE PROCEDURE HAS BEEN REVISED TO REQUIRE A SENIOR LICENSED OPERATOR REVIEW OF PROCEDURE REVISIONS AND IMMEDIATE NOTIFICATION OF THE SS UPON DISCREPANCY IDENTIFICATION.

[90] MILLSTONE 3 DOCKET 50-423 LER 89-024
FAILURE TO ENTER AN MSIV ACTION STATEMENT DUE TO PROCEDURAL DEFICIENCIES.
EVENT DATE: 101089 REPORT DATE: 110989 NSSS: WE TYPE: PWR
VENDOR: SULZER BROTHERS INC.

(NSIC 215918) ON 10/10/89, AT APPROXIMATELY 1200, AT 100% POWER IN MODE 1, INCONCLUSIVE TEST RESULTS FOR THE "C" MSIV WERE OBTAINED, WHILE PERFORMING THE QUARTERLY MAIN STEAM ISOLATION VALVE (MSIV) PARTIAL (SOFT) STROKE SURVEILLANCE. THE TECH SPEC FOR MSIV OPERABILITY WAS NOT ENTERED WHEN THE STROKE TIME OF THE 2A SOLENOID EXCEEDED THE 5 SECOND ACCEPTANCE CRITERIA, DUE TO CONCERN OVER THE VALIDITY OF THE TEST DATA. PLANT PERSONNEL DETERMINED THAT ADDITIONAL DATA WAS NEEDED TO DETERMINE MSIV OPERABILITY. UPON RECEIPT OF THIS ADDITIONAL DATA, THE "C" MSIV WAS DEEMED OPERABLE, AT 0830 ON 10/11/89. LATER INVESTIGATIONS CONCLUDED THAT THE MSIV TECH SPECS SHOULD HAVE BEEN ENTERED WHEN THE INITIAL TEST DATA WAS OBTAINED. ROOT CAUSE OF THE EVENT WAS INADEQUATE PROCEDURAL GUIDANCE FOR IDENTIFYING CORRECT ACCEPTANCE CRITERIA ON MSIV SOLENOID OPERATION. THE MOST PROBABLE CAUSE OF THE SLOW 2A SOLENOID STROKE WAS DETERMINED TO BE PARTICULATE DEPOSITION. IMMEDIATE CORRECTIVE ACTION WAS TO PERFORM THE PARTIAL STROKE TEST ON THE "C" MSIV. THE "C" MSIV WAS TESTED 4 SUBSEQUENT TIMES. ALL MSIVS WILL BE TESTED MONTHLY FOR THREE TESTS TO ENSURE OPERABILITY. INCREASED PROCEDURAL GUIDANCE ON DESIRED MSIV SOFT STROKE SURVEILLANCE ACCEPTANCE CRITERIAL WILL BE IMPLEMENTED BY 11/30/89.

[91] MILLSTONE 3 DOCKET 50-423 LER 89-025
LIFTING OF PRESSURIZER SAFETIES ABOVE THE ALLOWED TOLERANCE.
EVENT DATE: 101289 REPORT DATE: 111389 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE

(NSIC 215917) ON 10/12/89, WYLE LABORATORIES NOTIFIED NORTHEAST UTILITIES OF AN ANOMALY IN THE TEST RESULTS ON TWO PREVIOUSLY INSTALLED PRESSURIZER SAFETY VALVES. THE SUBJECT VALVES ARE REQUIRED BY TECH SPECS TO BE OPERABLE WITH A SETTING OF 2500 PSIA +/- 1% (2475-2525 PSIA) DURING MODES 1, 2, AND 3. BENCH TESTING AT WYLE LABORATORY REVEALED ONE VALVE (S/N N56964-07-0102) INITIALLY LIFTED AT 2534 PSIA AND THE OTHER VALVE (S/N N56964-07-0059) INITIALLY LIFTED AT 2553 PSIA. AS THE VALVES WERE NOT INSTALLED IN THE PLANT AT THE TIME, NO OPERATOR ACTION WAS REQUIRED IN RESPONSE TO THIS EVENT. THE ROOT CAUSE FOR PRESSURIZER SAFETY S/N 102 LIFTING HIGHER THAN THE 1% TOLERANCE IS UNKNOWN. THE ROOT CAUSE FOR PRESSURE SAFETY S/N 59 LIFTING HIGHER THAN TOLERANCE IS BELIEVED TO BE AN IMPROPERLY MACHINED DISC INSERT. AN INSPECTION OF THE S/N 59 REVEALED THAT A SURFACE ON THE DISC INSERT WHICH MATES TO THE DISC HOLDER WAS NOT FLAT. THIS ALLOWED THE DISC TO ROCK AND POSSIBLY RESEAT IN A DIFFERENT LOCATION AFTER EACH LIFT. THE DIFFERENT LOCATIONS CAN AFFECT SPRING TENSION AND IN TURN THE LIFT PRESSURE. THE DISC INSERT FOR S/N 59 WAS REPLACED WITH ONE SPECIFICALLY INSPECTED TO ASSURE THERE WAS A FLAT, POLISHED MATING SURFACE FOR THE DISC HOLDER. ALSO, PARTS CONSIDERED CRITICAL TO THE PROPER OPERATION OF SAFETY S/N 59 WERE THOROUGHLY INSPECTED FOR PROPER FIT AND FUNCTION.

[92] MONTICELLO DOCKET 50-263 LER 89-024
PROCEDURAL INADEQUACY RESULTS IN REACTOR PROTECTION SYSTEM ACTUATION.
EVENT DATE: 100289 REPORT DATE: 103189 NSSS: GE TYPE: BWR

(NSIC 215830) WITH THE PLANT SHUTDOWN AND FUEL LOADING IN PROGRESS, A REACTOR PROTECTION SYSTEM (RPS) TRIP OCCURRED DUE TO A SPURIOUS INCREASE IN THE SIGNAL FROM INTERMEDIATE RANGE MONITOR (IRM) 18. THE IRMS WERE IN THE NON-COINCIDENT TRIP MODE DUE TO FUEL LOADING BEING IN PROGRESS. NO CONTROL ROD MOVEMENT RESULTED BECAUSE ALL OPERABLE CONTROL RODS WERE FULLY INSERTED AT THE TIME OF THE TRIP. THE TRIP WAS RESET. A SHORT TIME LATER, A SECOND RPS TRIP OCCURRED DUE TO ANOTHER SPURIOUS INCREASE IN THE IRM 18 SIGNAL. THIS TRIP WAS ALSO RESET. AS BEFORE, NO CONTROL ROD MOVEMENT RESULTED FROM THE TRIP. IRM 18 WAS THEN PLACED IN BYPASS, REMOVING IT FROM THE RPS LOGIC AS ALLOWED BY TECHNICAL SPECIFICATIONS. AFTER RETURNING THE IRMS TO COINCIDENT TRIP LOGIC, THE BYPASS WAS REMOVED FROM IRM 18. THE PROCEDURE USED TO CONTROL FUEL MOVEMENT WITHIN THE REACTOR IS BEING MODIFIED TO INCORPORATE NON-COINCIDENT SOURCE RANGE MONITOR (SRM) TRIP LOGIC, BUT RETAIN COINCIDENT IRM TRIP LOGIC. TRAINING WILL BE PROVIDED ON THE NEED TO IDENTIFY AND EVALUATE MEASURES WHICH CAN BE ESTABLISHED TO PREVENT RECOGNIZED POTENTIAL EVENTS.

[93] MONTICELLO DOCKET 50-263 LER 89-026
INCOMPLETE UNDERSTANDING OF NON-COINCIDENT LOGIC CONFIGURATION RESULTS IN REACTOR PROTECTION SYSTEM ACTUATION DURING BREAKER MAINTENANCE.
EVENT DATE: 100489 REPORT DATE: 110389 NSSS: GE TYPE: BWR

(NSIC 215831) WITH THE PLANT SHUT DOWN FOR REFUELING AND FUEL LOADING TEMPORARILY SUSPENDED, A REACTOR PROTECTION SYSTEM TRIP OCCURRED DUE TO THE DE-ENERGIZATION OF REACTOR PROTECTION SYSTEM BUS A. REACTOR PROTECTION SYSTEM BUS A WAS DE-ENERGIZED BY THE REMOVAL OF THE SOURCE BREAKER FOR MOTOR CONTROL CENTER 111, WHICH SUPPLIED POWER TO REACTOR PROTECTION SYSTEM MOTOR-GENERATOR SET A. THE DE-ENERGIZATION OF REACTOR PROTECTION SYSTEM BUS A RESULTED IN A FULL REACTOR PROTECTION SYSTEM TRIP BECAUSE THE REACTOR PROTECTION SYSTEM SHORTING LINKS HAD BEEN REMOVED, ENABLING THE SOURCE RANGE MONITORS TO INITIATE A NON-COINCIDENT REACTOR PROTECTION SYSTEM TRIP. NO CONTROL ROD MOVEMENT RESULTED BECAUSE ALL OPERABLE CONTROL RODS WERE FULLY INSERTED AT THE TIME OF THE TRIP. REACTOR PROTECTION SYSTEM BUS A WAS RE-ENERGIZED, AND THE TRIP WAS RESET MOTOR CONTROL CENTER 111 SOURCE BREAKER REMOVAL HAD BEEN ALLOWED DUE TO A LACK OF UNDERSTANDING OF THE FULL IMPLICATIONS OF REACTOR PROTECTION SYSTEM SHORTING LINK REMOVAL. AN OPERATIONS MEMO WAS ISSUED TO INFORM OPERATIONS PERSONNEL OF THE IMPLICATIONS OF THE REMOVAL OF THE REACTOR PROTECTION SYSTEM SHORTING LINKS. PLANT PROCEDURES

ARE BEING REVISED TO ENSURE THAT ALL OPERATIONS PERSONNEL UNDERSTAND THE IMPLICATIONS OF SHORTING LINK REMOVAL DURING FUTURE FUEL HANDLING OPERATIONS.

[94] MONTICELLO DOCKET 50-263 LER 89-027
INADEQUATE REVIEW OF CIRCUIT ISOLATION RESULTS IN REACTOR WATER CLEANUP ISOLATION VALVE CLOSURE.
EVENT DATE: 101089 REPORT DATE: 110989 NSSS: GE TYPE: BWR

(NSIC 215832) DURING PERFORMANCE OF A MODIFICATION PROCEDURE THE REACTOR WATER CLEANUP HI FILTER-DEMINERALIZER INLET TEMPERATURE CIRCUIT WAS DE-ENERGIZED DUE TO MOMENTARILY LIFTING A WIRE LEAD FROM A TERMINAL BLOCK TO ALLOW TERMINATION OF A SECOND WIRE LEAD. THIS RESULTED IN CLOSURE OF THE REACTOR WATER CLEANUP PRIMARY CONTAINMENT ISOLATION VALVES. THE WIRE LEADS WERE RETERMINATED, THE ISOLATION WAS RESET, AND REACTOR WATER CLEANUP RETURNED TO SERVICE. THE PLANT WAS SHUTDOWN FOR REFUELING AT THE TIME OF THE EVENT. THE ROOT CAUSE OF THIS EVENT IS A COGNITIVE PERSONNEL ERROR. INADEQUATE ATTENTION WAS GIVEN TO OTHER CIRCUITS THAT COULD BE AFFECTED BY THE MODIFICATION ISOLATION. THE NEED TO REVIEW REDUNDANT DRAWINGS AND ADEQUATELY VERIFY AS-BUILT CONFIGURATION FOR CIRCUIT ISOLATIONS INVOLVING LIFTED LEADS OR JUMPERS WILL BE EMPHASIZED DURING TRAINING.

[95] MONTICELLO DOCKET 50-263 LER 89-028
MODIFICATION REVIEW IDENTIFIED POTENTIAL FOR DEGRADED ECCS CAPABILITY.
EVENT DATE: 101389 REPORT DATE: 111089 NSSS: GE TYPE: BWR

(NSIC 215833) A CONCERN WAS IDENTIFIED BY PLANT ENGINEERING STAFF DURING THE REVIEW OF A MODIFICATION TO CORRECT A DEFICIENCY IN THE POWER SUPPLY TO THE LOW PRESSURE COOLANT INJECTION VALVES. THE CONCERN IS A POTENTIAL DEGRADED CONDITION OF THE ELECTRICAL POWER (I.E., OVERVOLTAGE, UNDERVOLTAGE OR ABNORMAL FREQUENCY) SUPPLIED BY THE EMERGENCY DIESEL GENERATOR DURING A LOSS-OF-COOLANT ACCIDENT WITH LOSS OF OFF-SITE POWER. THIS COULD MAKE THE SELECTED LPCI INJECTION VALVES AND ONE DIVISION OF CORE SPRAY INOPERABLE. THE PLANT DESIGN BASIS INCLUDED ONLY COMPLETE LOSS OF VOLTAGE EVENTS. DEGRADED VOLTAGE OR FREQUENCY FOR ON-SITE POWER SOURCES WERE NOT POSTULATED FAILURE MODES AND WERE NOT PART OF THE DESIGN BASIS. ADDITIONAL RELAYS HAVE BEEN ADDED TO MONITOR THE LPCI SWING BUS FOR ABNORMAL POWER CONDITIONS. UNDER DEGRADED POWER CONDITIONS, THESE ADDITIONAL RELAYS WILL INITIATE A SWING BUS TRANSFER FROM DIVISION I (NORMAL SUPPLY) TO DIVISION II (ALTERNATE SUPPLY) FOR THE BUS SUPPLYING THE LPCI INJECTION VALVES. THIS LER IS BEING SUBMITTED DUE TO ITS POTENTIAL GENERIC INTEREST.

[96] MONTICELLO DOCKET 50-263 LER 89-029
DEGRADATION OF SECONDARY CONTAINMENT DUE TO DESIGN DEFICIENCIES AND INADEQUATE TESTING PROCEDURES.
EVENT DATE: 101489 REPORT DATE: 111389 NSSS: GE TYPE: BWR

(NSIC 215834) ON OCTOBER 14, 1989, WHILE THE PLANT WAS SHUTDOWN FOR REFUELING, SECONDARY CONTAINMENT FAILED THE TECHNICAL SPECIFICATION 4.7.C.1.(A) CAPABILITY TEST. SECONDARY CONTAINMENT WAS DECLARED INOPERABLE. ALL WORK REQUIRING SECONDARY CONTAINMENT WAS IMMEDIATELY STOPPED. INSPECTIONS OF THE SECONDARY CONTAINMENT REVEALED SEVERAL DEGRADED CONDITIONS, INCLUDING BYPASS FLOW FROM THE REACTOR BUILDING TO THE REACTOR BUILDING PLENUM. SYSTEM DESIGN DEFICIENCIES, INADEQUATE TESTING AND INADEQUATE MAINTENANCE PROCEDURES CAUSED THIS EVENT. THE DEGRADED CONDITIONS WERE REPAIRED. OPERATING AND TESTING PROCEDURES WERE CHANGED TO ELIMINATE THE STANDBY GAS TREATMENT SYSTEM BYPASS FLOW. MAINTENANCE PROCEDURES WILL BE WRITTEN TO PRECLUDE SECONDARY CONTAINMENT DEGRADATION. A DESIGN CHANGE IS BEING CONSIDERED TO MINIMIZE BYPASS FLOW.

[97] NINE MILE POINT 1 DOCKET 50-220 LER 89-013
REACTOR SCRAM DUE TO VOLTAGE SURGE ON REACTOR PROTECTION SYSTEM BUS 11.
EVENT DATE: 092989 REPORT DATE: 102989 NSSS: GE TYPE: BWR

(NSIC 215812) ON 9/29/89, NINE MILE POINT UNIT 1 (NMP1) WAS IN A REFUELING OUTAGE WITH THE MODE SWITCH IN "SHUTDOWN", AND THE CORE OFFLOADED. AT 1315 HOURS NMP1

EXPERIENCED A FULL REACTOR SCRAM DUE TO A MOMENTARY DROP IN VOLTAGE ON REACTOR PROTECTION SYSTEM (RPS) BUS 11. THE VOLTAGE DROP IN RPS BUS 11 WAS DUE TO A LEAD WIRE BEING SHORTED TO GROUND DURING REPLACEMENT OF A POWER SUPPLY ON THE ROD POSITION INDICATION SYSTEM (RPIS). IMMEDIATE CORRECTIVE ACTION WAS TAKEN BY OPERATIONS PERSONNEL BY RESETTING THE SCRAM AT 1317 HOURS. THE ROOT CAUSE FOR THIS EVENT WAS A FAILURE TO PROPERLY PLAN AND EVALUATE THE WORK PRIOR TO STARTING PHYSICAL WORK ACTIVITIES IN THE FIELD. 10CFR50.72(B)(2)(II) NOTIFICATION WAS MADE AT 1355 HOURS.

[98]   NINE MILE POINT 1   DOCKET 50-220   LER 89-014
REDUNDANT SAFETY SYSTEMS INOPERABLE DUE TO THE LACK OF A COMPLETE PROGRAM TO CALIBRATE NON-TECH SPEC INSTRUMENTATION.
EVENT DATE: 100489   REPORT DATE: 110389   NSSS: GE   TYPE: BWR

(NSIC 215818) ON AUGUST 30, 1989, WITH THE REACTOR SHUTDOWN AND THE CORE OFFLOADED, IT WAS DISCOVERED THAT THE HEATING ELEMENT TEMPERATURE CONTROLLERS FOR THE NINE MILE POINT UNIT 1 (NMP1) REACTOR BUILDING EMERGENCY VENTILATION SYSTEM CHARCOAL FILTER DUCT HEATERS WERE NOT IN A SCHEDULED CALIBRATION PROGRAM. ON OCTOBER 4, 1989, AFTER TESTING, IT WAS DETERMINED THAT THE CONTROLLERS WERE SET BELOW DESIGN CRITERIA WHICH MAY HAVE RESULTED IN THE REDUNDANT COMPONENTS BECOMING INOPERABLE. IMMEDIATE CORRECTIVE ACTION INCLUDED CALIBRATION OF THE DEVICES AND ESTABLISHMENT OF A CALIBRATION FREQUENCY FOR THESE PARTICULAR COMPONENTS. LONG TERM CORRECTIVE ACTION WILL INCLUDE THE COMPLETION OF A PREVIOUSLY DEVELOPED PROGRAM FOR THE CALIBRATION AND PERIODIC TESTING OF NON-TECHNICAL SPECIFICATION BALANCE OF PLANT INSTRUMENTATION WHICH MAY AFFECT SYSTEM AND COMPONENT OPERABILITY. ADDITIONALLY, A LESSONS LEARNED TRANSMITTAL WILL BE DEVELOPED TO COMMUNICATE THE CONSEQUENCES OF SUCH A CONDITION.

[99]   NINE MILE POINT 2   DOCKET 50-410   LER 89-022
ENGINEERED SAFETY FEATURE INITIATION DUE TO ELECTRICAL FAULT.
EVENT DATE: 100789   REPORT DATE: 110689   NSSS: GE   TYPE: BWR
VENDOR: ASEA ELECTRIC, INC.

(NSIC 215891) ON 10/7/89, AT 2317 HOURS WITH THE REACTOR AT APPROXIMATELY 100% POWER AND THE MODE SWITCH IN "RUN" (OPERATIONAL CONDITION 1), NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION. SPECIFICALLY, BOTH TRAINS OF THE STANDBY GAS TREATMENT SYSTEM WERE INITIATED ON A LOSS OF VOLTAGE TO THE REACTOR BUILDING VENTILATION ABOVE/BELOW REFUEL FLOOR LOW FLOW RELAYS. THE IMMEDIATE CORRECTIVE ACTION WAS THAT OPERATIONS INITIATED AN INVESTIGATION. THE IMMEDIATE CAUSE WAS IDENTIFIED AS A FAULT IN A CABLE FEEDING A UNIT SUBSTATION WHICH CAUSED THE POWER SUPPLY BREAKER TO TRIP. THE INTERMEDIATE CAUSE WAS CABLE INSULATION DETERIORATION DUE TO CORONA EFFECTS ON THE HIGH SIDE (13800 VOLT) INPUT CABLES TO A 600 VOLT UNIT SUBSTATION. THE ROOT CAUSE WAS DETERMINED TO BE INCOMPLETE DOCUMENTATION FOR INSTALLATION. A WORK REQUEST WAS ISSUED TO REPLACE THE SHORTED CABLE AND ENSURE ADEQUATE CABLE CLEARANCE.

[100]   NINE MILE POINT 2   DOCKET 50-410   LER 89-032
VIOLATION OF TECH SPEC DUE TO PERSONNEL ERROR.
EVENT DATE: 101189   REPORT DATE: 111389   NSSS: GE   TYPE: BWR

(NSIC 215893) ON OCTOBER 12, 1989, AT 1100 HOURS IT WAS DETERMINED THAT ON OCTOBER 11, 1989, A LIMITING CONDITION FOR OPERATION (LCO) AS DEFINED BY TECHNICAL SPECIFICATION (TS) 3.3.2 HAD BEEN EXCEEDED BY 58 MINUTES AT NINE MILE POINT UNIT 2 (NMP2). AT THE TIME THE LCO WAS EXCEEDED NMP2 WAS OPERATING AT 99% POWER WITH THE MODE SWITCH IN "RUN". THE LCO WAS EXCEEDED WHEN THE DIVISION I REACTOR WATER CLEANUP DIFFERENTIAL FLOW TRANSMITTER (2WCS*FT69X) WAS ALLOWED TO BE INOPERABLE FOR 3 HOURS 58 MINUTES WITHOUT PERFORMING THE PROPER TS ACTION ITEM AT TIME 3 HOURS. THE CAUSE OF THIS CONDITION WAS COGNITIVE PERSONNEL ERROR DUE TO IMPROPER APPLICATION OF TECHNICAL SPECIFICATION 3.3.2. THE INITIAL CORRECTIVE ACTION WAS PERFORMED ON OCTOBER 11, 1989, AT 2319 BY PLACING 2WCS*FT69X BACK IN SERVICE. ADDITIONAL CORRECTIVE ACTION INCLUDED GENERATING A TRAINING

MODIFICATION RECOMMENDATION DEALING WITH TS ACTION STATEMENTS AND THEIR IMPACT ON EACH OTHER.

[101] NINE MILE POINT 2 DOCKET 50-410 LER 89-034
TECH SPEC VIOLATION DUE TO AN INOPERABLE RADIATION MONITOR CAUSED BY PERSONNEL ERROR.
EVENT DATE: 101289 REPORT DATE: 111389 NSSS: GE TYPE: BWR

(NSIC 215894) WHILE PERFORMING ROUTINE CHECKS OF THE DIGITAL RADIATION MONITORING SYSTEM (DRMS) ON OCTOBER 12, 1989, AT 1600 HOURS, NIAGARA MOHAWK RADIATION PROTECTION (RP) AND OPERATIONS (OPS) PERSONNEL DISCOVERED THAT THE "B" BELOW REFUEL FLOOR RADIATION MONITOR (2HVR*CAB32B) WAS NOT IN OPERATION. PRIOR TO THIS EVENT, THE UNIT WAS IN MODE 1, OPERATING AT 100% POWER. THE ROOT CAUSE OF THE EVENT WAS COGNITIVE PERSONNEL ERROR DUE TO A RADIATION PROTECTION (RP) TECHNICIAN SHUTTING OFF THE WRONG MONITOR AFTER RESIDUAL HEAT REMOVAL (RHR) "B" LOOP WAS SECURED FROM SUPPRESSION POOL SPRAY MODE. A CONTRIBUTING CAUSE WAS INATTENTION TO DETAIL, SPECIFICALLY, OPERATIONS PERSONNEL DID NOT RECOGNIZE THE SECURED RADIATION MONITOR. IMMEDIATE CORRECTIVE ACTION WAS TO RESTART THE MONITOR SAMPLE PUMP AND RESTORE IT TO SERVICE (OCTOBER 12, 1989, AT 1638). LATER INVESTIGATION REVEALED THAT THE SAMPLE FLOW PUMP HAD NOT BEEN IN SERVICE SINCE 0431 HOURS ON OCTOBER 12, 1989, A PERIOD OF APPROXIMATELY 12 HOURS. THIS RESULTED IN A VIOLATION OF TECHNICAL SPECIFICATION (TS) 3.3.2-1.3.B.

[102] NINE MILE POINT 2 DOCKET 50-410 LER 89-031
REACTOR WATER CLEANUP SYSTEM ISOLATION DUE TO EQUIPMENT MALFUNCTION.
EVENT DATE: 101389 REPORT DATE: 111389 NSSS: GE TYPE: BWR

(NSIC 215892) ON OCTOBER 13, 1989, AT 1746 HOURS, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED THE ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF), SPECIFICALLY, ISOLATION OF THE REACTOR WATER CLEANUP (WCS) SYSTEM ON AN ERRONEOUS HIGH DIFFERENTIAL FLOW SIGNAL. AT THE TIME OF THE EVENT, THE PLANT WAS IN "HOT SHUTDOWN" (OPERATIONAL CONDITION 3) WITH REACTOR TEMPERATURE AT APPROXIMATELY 418 DEGREES FAHRENHEIT AND REACTOR PRESSURE AT APPROXIMATELY 307 POUNDS PER SQUARE INCH GAUGE. THE CAUSE OF THIS EVENT WAS DETERMINED TO BE AN EQUIPMENT MALFUNCTION. A CONTRIBUTING CAUSE TO THIS EVENT WAS A PROCEDURE DEFICIENCY. CORRECTIVE ACTIONS INCLUDED: ISSUING WORK REQUESTS (WRS) TO TROUBLESHOOT THE SOURCE OF THE HIGH DELTA FLOW ISOLATION SIGNAL; REPLACING AN OUT OF CALIBRATION FLOW TRANSMITTER; REPAIRING SEVERAL LEAKING INSTRUMENT SENSING LINE DRAIN VALVES; AND REVISING AN OPERATING PROCEDURE TO INCLUDE STEPS WHICH WILL HELP PRECLUDE RECURRENCE OF THIS EVENT.

[103] NINE MILE POINT 2 DOCKET 50-410 LER 89-035
ENGINEERED SAFETY FEATURE INITIATION DUE TO PERSONNEL ERROR.
EVENT DATE: 101389 REPORT DATE: 111389 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215895) ON OCTOBER 13, 1989, AT 0949 HOURS NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED AN ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF), SPECIFICALLY AN AUTOMATIC REACTOR SCRAM AS A RESULT OF A MAIN TURBINE TRIP. AT THE TIME OF THE EVENT, THE REACTOR WAS OPERATING AT 54% RATED POWER WITH THE MODE SWITCH IN "RUN" (OPERATIONAL CONDITION 1). THE ROOT CAUSE WAS DETERMINED TO BE PERSONNEL ERROR IN THAT AN ADEQUATE EVALUATION OF PLANT IMPACT WAS NOT PERFORMED. A CONTRIBUTING CAUSE WAS FAILURE OF ADMINISTRATIVE PROCEDURES TO ADDRESS A COMPREHENSIVE PLANT IMPACT REVIEW. CONTROL ROOM OPERATORS CARRIED OUT IMMEDIATE CORRECTIVE ACTIONS. OTHER CORRECTIVE ACTIONS INCLUDE REVISION OF PLANT IMPACT EVALUATION FORM AND REPAIR OF MOTOR GENERATOR LOGIC CIRCUIT. THE RESPONSIBLE CHIEF SHIFT OPERATOR (CSO) WAS DIRECTED TO WRITE A SPECIAL REPORT CONCERNING THIS EVENT. MEETINGS WERE HELD TO STRESS THE IMPORTANCE OF PLANT IMPACT EVALUATIONS. ADMINISTRATIVE PROCEDURES WILL BE REVISED. A LESSONS LEARNED TRANSMITTAL WILL BE ISSUED.

[104] NINE MILE POINT 2 DOCKET 50-410 LER 89-036
REACTOR SCRAM ON HIGH NEUTRON FLUX DUE TO PERSONNEL ERROR/PROCEDURE DEFICIENCY.
EVENT DATE: 101889 REPORT DATE: 111789 NSSS: GE TYPE: BWR

(NSIC 215896) ON OCTOBER 18, 1989, AT 06:15:41 HOURS, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED AN UPSCALE TRIP OF THE INTERMEDIATE RANGE MONITORS (IRMS) RESULTING IN AN AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM (RPS) AND A FULL REACTOR SCRAM. AT THE TIME OF THE EVENT, THE REACTOR MODE SWITCH WAS IN THE START/HOT STANDBY POSITION WITH REACTOR POWER IN THE INTERMEDIATE RANGE (1%). THE ROOT CAUSE FOR THIS EVENT WAS DETERMINED TO BE INADEQUATE CONTROL OF ACTIVITIES ASSOCIATED WITH THE PLANT SHUTDOWN ON THE PART OF CONTROL ROOM PERSONNEL. A CONTRIBUTING FACTOR WAS A PROCEDURAL DEFICIENCY. CORRECTIVE ACTIONS INCLUDE (1) INSTRUCTING OPERATING SHIFT PERSONNEL ON THIS EVENT, (2) REVISING THE OPERATIONS TRAINING PROGRAM, AND (3) REVISING PLANT SHUTDOWN PROCEDURE N2-OP-101C.

[105] NINE MILE POINT 2 DOCKET 50-410 LER 89-037
VIOLATION OF TECH SPEC DUE TO INADEQUATE CONTROLS ON APPENDIX "R" VALVE.
EVENT DATE: 101889 REPORT DATE: 111789 NSSS: GE TYPE: BWR

(NSIC 215897) ON OCTOBER 18, 1989, AT 2300 HOUR DURING AN END OF DAY REVIEW OF THE DAILY CHECKS PROCEDURE (N2-OSP-LOG-D001), IT WAS DISCOVERED THAT NINE MILE POINT UNIT 2 (NMP2) HAD NOT BEEN IN COMPLIANCE WITH PLANT TECHNICAL SPECIFICATIONS (T.S.). SPECIFICALLY, WHILE NMP2 WAS IN "HOT SHUTDOWN" (OPERATIONAL CONDITION 3) THE ASSOCIATED AC CIRCUIT FOR DRYWELL DRAIN (DER) SYSTEM MOTOR OPERATED VALVE 2DER*MOV128 HAD NOT BEEN DE-ENERGIZED WITHIN THE REQUIRED TIME FRAME. T.S. REQUIRES THIS CIRCUIT BREAKER TO BE TRIPPED (OPENED) WITHIN ONE HOUR IF PLANT IS IN OPERATING CONDITIONS 1, 2, OR 3. THE CAUSE OF THIS EVENT WAS HUMAN PERFORMANCE PROBLEMS AS A RESULT OF INADEQUATE ADMINISTRATIVE CONTROLS ON AN MOV WHICH HAD BOTH 10CFR50 APPENDIX "R" AND TECHNICAL SPECIFICATION REQUIREMENTS REGARDING ITS POWER SUPPLY. INITIAL CORRECTIVE ACTIONS WERE TO IMMEDIATELY DE-ENERGIZE THE CIRCUIT AND PLACE NEW HOLD-OUT TAGS ON THE EQUIPMENT. ADDITIONAL CORRECTIVE ACTIONS INCLUDE REVIEWING OTHER RE-ENERGIZED CIRCUITS TO VERIFY TS REQUIREMENTS HAD NOT BEEN VIOLATED, AND REVISING OPERATING PROCEDURES TO PRECLUDE REOCCURENCE OF THIS TYPE OF EVENT.

[106] NINE MILE POINT 2 DOCKET 50-410 LER 89-038
ACTUATION OF DIVISION II EMERGENCY DIESEL GENERATOR DUE TO HUMAN PERFORMANCE PROBLEM.
EVENT DATE: 102089 REPORT DATE: 111789 NSSS: GE TYPE: BWR

(NSIC 215898) ON OCTOBER 20, 1989, AT 1202 HOURS, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED AN UNSCHEDULED MANUAL INITIATION OF THE DIVISION (DIV.) II EMERGENCY DIESEL GENERATOR (DG). AT THE TIME OF THE EVENT, THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) WITH REACTOR TEMPERATURE AND PRESSURE AT APPROXIMATELY 134 DEGREES FAHRENHEIT AND 3.3 POUNDS PER SQUARE INCH GAUGE, RESPECTIVELY. ALSO AT THE TIME OF THE EVENT, THE DIV. III EMERGENCY DG'S MONTHLY SURVEILLANCE TEST WAS IN PROGRESS. THE ROOT CAUSE OF THIS EVENT WAS HUMAN PERFORMANCE PROBLEMS. THE LICENSED OPERATOR, WHILE ATTEMPTING TO START THE DIV. III DG, ERRONEOUSLY STARTED THE DIV. II DG. CORRECTIVE ACTIONS INCLUDED: (1) DISPATCHED AN OPERATOR TO DIV. II DG TO VERIFY PROPER OPERATION IN ACCORDANCE WITH ITS OPERATING PROCEDURE; (2) SECURED AND PLACED THE DIV. II DG IN STANDBY IN ACCORDANCE WITH ITS OPERATING PROCEDURE; (3) TEMPORARILY RELIEVED THE RESPONSIBLE OPERATOR FROM HIS LICENSED DUTIES; (4) COMPLETED THE IN PROGRESS SURVEILLANCE TEST UTILIZING A DIFFERENT LICENSED OPERATOR; AND (5) REVIEWED EVENT WITH SHIFT OPERATING CREWS.

[107] OCONEE 1 DOCKET 50-269 LER 89-016
DESIGN OVERSIGHT RESULTS IN POTENTIAL FOR UNANALYZED BREACH OF CONTAINMENT ISOLATION DURING A SIMULTANEOUS LOCA/SEISMIC EVENT.
EVENT DATE: 101789 REPORT DATE: 111689 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 215836) ON OCTOBER 17, 1989 DESIGN ENGINEERING (DE) NOTIFIED THE STATION THAT DURING A SIMULTANEOUS LOCA/SEISMIC EVENT AN UNANALYZED BREECH OF CONTAINMENT COULD RESULT IN THE DISCHARGE OF CONTAINMENT ATMOSPHERE TO THE ENVIRONMENT. THE PROBLEM WAS IDENTIFIED AS A RESULT OF DISCUSSIONS BETWEEN STATION AND DE PERSONNEL. THE PROBLEM HAS EXISTED SINCE NUCLEAR STATION MODIFICATION (NSM) 1261 INSTALLED NON-SEISMIC REACTOR BUILDING AUXILIARY COOLERS (RBACS) IN ALL THREE CONTAINMENT BUILDINGS, BETWEEN 1980 AND 1982. THE DESIGN OF NSM 1261 WAS BASED ON LPSW DESIGN PRESSURE VERSUS ACTUAL SYSTEM OPERATING PRESSURE. PORTIONS OF THE NSM WERE CONSEQUENTLY NOT DESIGNED TO SEISMIC REQUIREMENTS. THESE PORTIONS ARE LOCATED INSIDE CONTAINMENT AND IF THEY WERE TO FAIL WHILE REACTOR BUILDING (RB) PRESSURE WAS AT THE MAXIMUM DESIGN PRESSURE (59.0 PSIG) DURING AN ACCIDENT, RB PRESSURE COULD FORCE CONTAINMENT ATMOSPHERE INTO THE RUPTURED LPSW PIPING, OUT TO THE ENVIRONMENT. IMMEDIATE CORRECTIVE ACTION WAS TO ISOLATE THE "B" REACTOR BUILDING COOLING UNIT (RBCU) WHICH PLACED EACH UNIT IN A 7 DAY LIMITING CONDITION OF OPERATION (LCO). ROOT CAUSE IS DESIGN DEFICIENCY, DESIGN OVERSIGHT DUE TO FAILURE TO RECOGNIZE THE POTENTIAL INTERACTION BETWEEN HIGH RB PRESSURE DURING A LOCA/SEISMIC EVENT AND THE RELATIVELY LOWER LPSW PRESSURE.

[108] OYSTER CREEK DOCKET 50-219 LER 89-022
TECH SPEC VIOLATION DUE TO ABSENCE OF AN SRO IN THE CONTROL ROOM CAUSED BY PERSONNEL ERROR.
EVENT DATE: 100589 REPORT DATE: 110689 NSSS: GE TYPE: BWR

(NSIC 215813) ON OCTOBER 5, 1989 AT 2153 HOURS, THE SRO LICENSED GROUP SHIFT SUPERVISOR (GSS) LEFT THE GSS OFFICE WITHOUT INFORMING THE SRO LICENSED GROUP OPERATING SUPERVISOR (GOS). AT 2154 HOURS THE GOS LEFT THE CONTROL ROOM WITHOUT NOTICING THE GSS WAS NOT IN THE CONTROL ROOM. AS A RESULT, THERE WAS NO SRO IN THE CONTROL ROOM FOR A SIX MINUTE PERIOD, UNTIL APPROXIMATELY 2200 HOURS. THIS IS A VIOLATION OF TECHNICAL SPECIFICATION 6.2.2.C THAT REQUIRES A SRO BE PRESENT IN THE CONTROL ROOM AT ALL TIMES WHEN THERE IS FUEL IN THE VESSEL EXCEPT WHEN THE REACTOR IS IN COLD SHUTDOWN OR REFUEL MODES. THIS EVENT IS REPORTABLE BASED ON 10CRF50.73(A)(2)(I)(B). CORRECTIVE ACTIONS INCLUDED THE IMMEDIATE RETURN OF THE GOS TO THE CONTROL ROOM WHEN IT WAS DISCOVERED THE GOS WAS NOT IN THE CONTROL ROOM AND PLANT OPERATIONS MANAGEMENT DISCUSSING THE OCCURRENCE WITH THE INDIVIDUALS INVOLVED.

[109] PALISADES DOCKET 50-255 LER 89-022
RADIOLOGICAL CONSEQUENCES OF A SAFETY INJECTION REFUELING WATER TANK RUPTURE.
EVENT DATE: 072389 REPORT DATE: 101989 NSSS: CE TYPE: PWR

(NSIC 215634) ON JULY 23, 1989 IT WAS DETERMINED THAT NO LICENSING BASIS EXISTED WHICH DEFINED THE RADIOLOGICAL CONSEQUENCES OF A SAFETY INJECTION REFUELING WATER (SIRW) TANK (BP;TK1) RUPTURE. CONSEQUENTLY, NO RADIONUCLIDE CONCENTRATION LIMITS HAVE EVER BEEN ESTABLISHED. THE POTENTIAL FOR HAVING AN UNIDENTIFIED LICENSING BASIS WAS IDENTIFIED BY CORPORATE HEALTH PHYSICS AND PLANT LICENSING PERSONNEL WHILE REVIEWING DOCUMENTATION ASSOCIATED WITH RADIOACTIVE RELEASES FROM THE SIRW TANK. THE REACTOR WAS CRITICAL WITH THE PLANT AT 80 PERCENT OF RATED POWER WHEN IT WAS DETERMINED THAT NO LICENSING BASIS EXISTED. THE SIRW TANK IS LOCATED ABOVE THE MAIN CONTROL ROOM AT APPROXIMATELY THE 643 FOOT ELEVATION AND IS NOT PHYSICALLY ENCLOSED BY ANY OTHER PLANT STRUCTURE. WHILE IT WAS RECOGNIZED DURING BOTH ORIGINAL PLANT LICENSING AND THE SYSTEMATIC EVALUATION PROGRAM THAT THE TANK WAS NOT DESIGNED TO WITHSTAND TORNADO WIND FORCES OR RESULTANT MISSILES, THE ONLY CONSEQUENCE DETERMINATION PERFORMED INVOLVED POTENTIAL EFFECTS ON OTHER EQUIPMENT DUE TO DELUGE AND NON-AVAILABILITY OF A BORON INJECTION SOURCE. A LICENSING ANALYSIS PERFORMED HAS DETERMINED THE CRITERION OF 10CFR100.11 TO BE APPLICABLE. PRELIMINARY DOSE CONSEQUENCE ANALYSES FOR TANK RUPTURE, BASED ON PAST TANK CONCENTRATIONS GOING BACK TO 1975 HAVE RESULTED IN DOSES ONLY A FRACTION OF 10CFR100.11.

[110] PALO VERDE 1 DOCKET 50-528 LER 89-007 REV 01
UPDATE ON PRESSURIZER SAFETY RELIEF VALVE SETPOINTS OUT OF TOLERANCE.
EVENT DATE: 041289 REPORT DATE: 100689 NSSS: CE TYPE: PWR
VENDOR: DRESSER INDUSTRIES, INC.

(NSIC 215660) ON 6/12/89 BETWEEN 1435 MST AND 1630 MST, PALO VERDE UNIT 1 WAS IN MODE 4 (HOT SHUTDOWN) WITH A STEAM BUBBLE BEING MAINTAINED IN THE PRESSURIZER TO ALLOW ASME SECTION XI TESTING OF THE PRESSURIZER RELIEF VALVES WHEN TWO OF THE FOUR PRESSURIZER CODE SAFETY VALVES WERE DISCOVERED OUT OF THE TECH SPEC TOLERANCE OF 2500 POUNDS PER SQUARE INCH-ABSOLUTE (PSIA) PLUS OR MINUS 1% (25 PSI). TECH SPEC 3.4.2.1 REQUIRES THAT A MINIMUM OF ONE PRESSURIZER CODE SAFETY VALVE BE OPERABLE IN MODE 4. THE CAUSE OF THE EVENT IS UNDER EVALUATION AND WILL BE REPORTED IN A SUPPLEMENT TO THIS REPORT. AS IMMEDIATE CORRECTIVE ACTION, THE TWO VALVES WERE ADJUSTED AND RETESTED SATISFACTORILY. THE ACTION TO PREVENT RECURRENCE IS UNDER EVALUATION AND WILL BE DESCRIBED IN A SUPPLEMENT TO THIS REPORT. ADDITIONALLY, A CHANGE TO THE TECH SPEC REQUIREMENT FOR THE SETPOINT TOLERANCE IS BEING PURSUED.

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

(NSIC 215787) ON 7/31/89, PALO VERDE UNIT 1 WAS IN A REFUELING OUTAGE WITH THE CORE OFF-LOADED TO THE SPENT FUEL POOL. ON THE NIGHT SHIFT OF 7/31/89, 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 7/31/89, 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 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 7/31/89, 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.

[112] PALO VERDE 1 DOCKET 50-528 LER 89-015
SWITCHYARD FIRE CAUSED BY SHUNT REACTOR FAILURE DUE TO A LEAD FAILURE.
EVENT DATE: 090189 REPORT DATE: 110289 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 2 (PWR)
PALO VERDE 3 (PWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215930) AT APPROXIMATELY 1618 MST ON 9/1/89, PALO VERDE UNIT 1 WAS IN A REFUELING OUTAGE WITH THE CORE OFF-LOADED, PALO VERDE UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT 100% POWER AND PALO VERDE UNIT 3 WAS IN MODE 5 (COLD SHUTDOWN) WHEN THE "B" PHASE SHUNT REACTOR ON THE DEVERS TRANSMISSION LINE AT THE PVNGS SWITCHYARD FAILED CATASTROPHICALLY. THE FAILURE IMPOSED A GROUND FAULT ON THE DEVERS 500 KV LINE. BREAKERS ISOLATING THE DEVERS LINE FROM THE PVNGS SWITCHYARD AND DEVERS SUBSTATION TRIPPED AS DESIGNED TO ISOLATE THE FAULT AND DEENERGIZE THE DEVERS LINE. THE FAILURE ALSO RESULTED IN THE RELEASE OF A LARGE VOLUME OF OIL AND FIRE. ALL OTHER PORTIONS OF THE SWITCHYARD REMAINED IN SERVICE DURING AND FOLLOWING THE EVENT. THE CAUSE OF THE SHUNT REACTOR FAILURE WAS A LEAD FAILURE. TO PREVENT RECURRENCE, SALT RIVER PROJECT (OPERATING MANAGER FOR THE SWITCHYARD) WILL APPROVE AND REVIEW THE RESULTS OF SWITCHYARD COMPONENT TESTING PRIOR TO ACCEPTANCE OF EQUIPMENT FOR INSTALLATION IN THE SWITCHYARD. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED PURSUANT TO 10CFR50.73.

[113] PALO VERDE 2 DOCKET 50-529 LER 89-007
MAIN STEAM SAFETY VALVE SETPOINTS DISCOVERED OUT-OF-TOLERANCE.
EVENT DATE: 100389 REPORT DATE: 110189 NSSS: CE TYPE: PWR
VENDOR: DRESSER INDUSTRIES, INC.

(NSIC 215913) ON OCTOBER 3, 1989 WHILE UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER, THE EVALUATION OF AUGMENTED ASME SURVEILLANCE TESTING WAS CONDUCTED WHICH DETERMINED THAT PRIOR TO ADJUSTMENT THIRTEEN (13) OF THE TWENTY (20) MAIN STEAM SAFETY VALVES (MSSV)(SB)(RV) RELIEF SETTINGS HAD BEEN OUT OF THE TOLERANCE LIMITS SPECIFIED IN TECHNICAL SPECIFICATION (TS) 3.7.1.1 AND THE TESTING REQUIREMENTS ESTABLISHED BY APS. THIS TESTING AND ADJUSTMENT WAS PERFORMED DURING THE PERIOD OF SEPTEMBER 19 THROUGH 22, 1989 WHILE UNIT 2 WAS IN MODE 3 (HOT STANDBY), TO VERIFY THE RELIEF SETTINGS OF THE MAIN STEAM SAFETY VALVES (MSSV)(SB)(RV). THIS TESTING AND ADJUSTMENT WAS BEING CONDUCTED AS A RESULT OF THE RELIEF SETTINGS THAT WERE OUT OF TOLERANCE IN UNITS 1 AND 2 AS REPORTED IN LICENSEE EVENT REPORTS (LER) 528/88-014-01, 528/89-010-00, AND 529/89-002-00. THE VARIANCES IDENTIFIED IN THE AS-FOUND DATA OF THE SETPOINTS HAVE BEEN DISCUSSED WITH THE VALVE VENDOR AND ARE CONSIDERED TO BE WITHIN THE DESIGN TOLERANCE OF THE VALVES. AS CORRECTIVE ACTION THE VALVES HAVE BEEN RESET AND APPROPRIATE TESTING CONDUCTED. AS CORRECTIVE ACTION TO PREVENT RECURRENCE, APS IS PURSUING AN AMENDMENT TO THE TS TO INCREASE THE TOLERANCE ON THE MSSV SETPOINT. PREVIOUS SIMILAR EVENTS WERE REPORTED IN LER'S 528/88-014-01, 528/89-010-00, AND 529/89-002-00.

[114] PEACH BOTTOM 2 DOCKET 50-277 LER 89-022
HIGH PRESSURE COOLANT INJECTION TRIP SOLENOID RENDERED INOPERABLE DUE TO THE POWER LEAD BECOMING CAUGHT IN THE PANEL DOOR LATCH AND DISCONNECTED FROM ITS TERMINAL STRIP.
EVENT DATE: 100389 REPORT DATE: 110289 NSSS: GE TYPE: BWR

(NSIC 215842) ON OCTOBER 3, 1989, AT 1420 HOURS THE UNIT 2 REMOTE HIGH PRESSURE COOLANT INJECTION (HPCI) STOP VALVE TRIP FUNCTIONS WERE RENDERED INOPERABLE. TECHNICIANS PERFORMING MAINTENANCE IN A HPCI ELECTRICAL PANEL DISCOVERED AN UNTERMINATED LEAD IN THE CIRCUIT TO THE HPCI TRIP SOLENOID. THE LEAD HAD BECOME ENTANGLED WITH THE PANEL DOOR INNER LATCHING MECHANISM, SUCH THAT WHEN THE TECHNICIANS OPENED THE PANEL DOOR THE LEAD PULLED OUT OF ITS LUG ON THE TERMINAL STRIP. THE ROOT CAUSE OF THE EVENT WAS LEADS LOOSELY HANGING INSIDE THE PANEL. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. THE LEAD WAS REPAIRED AND THE HPCI TRIP FUNCTIONS TESTED SATISFACTORILY. THE LEADS IN THE UNIT 2 AND UNIT 3 HPCI PANELS WERE INSPECTED AND SECURED AS NECESSARY TO PREVENT RECURRENCE. THERE WAS ONE PREVIOUS SIMILAR LER.

[115] PEACH BOTTOM 2 DOCKET 50-277 LER 89-023
MECHANICAL BINDING OF MSIV "DC" SOLENOID PILOT VALVE CAUSES REACTOR SCRAM DURING TESTING.
EVENT DATE: 100589 REPORT DATE: 110689 NSSS: GE TYPE: BWR
VENDOR: AUTOMATIC VALVE COMPANY

(NSIC 215843) AT 1806 HOURS ON 10/5/89, WITH UNIT 2 AT 99.5% THERMAL POWER, ST 1.3A-2 "PCIS GROUP I LOGIC SYSTEM FUNCTIONAL TEST" WAS IN PROGRESS. AS PART OF THE TEST, THE OUTBOARD MAIN STEAM ISOLATION VALVE (MSIV) AC SOLENOID PILOT VALVES WERE DE-ENERGIZED, WHICH RESULTED IN THE UNEXPECTED CLOSURE OF THE "D" OUTBOARD MSIV. SUBSEQUENTLY, AN AUTOMATIC REACTOR SCRAM OCCURRED AT 1806 HOURS DUE TO AN APRM HIGH FLUX SIGNAL UPON THE RESULTANT RAPID INCREASE IN REACTOR PRESSURE. A REACTOR WATER CLEAN UP ISOLATION OCCURRED SHORTLY AFTER THE SCRAM. OTHER SAFETY SYSTEMS OPERATED AS DESIGNED. THE CAUSE OF THE "D" MSIV CLOSURE WAS THE BINDING OF ITS DC SOLENOID PILOT VALVE PLUNGER IN THE VENTED POSITION THUS ALLOWING THE MSIV TO CLOSE WHEN THE AC SOLENOID VALVE WAS DE-ENERGIZED. THE ROOT CAUSE OF THE FAILURE WAS INCOMPLETE TECHNICAL GUIDANCE SUPPLIED BY THE SOLENOID MANUFACTURER FOR THE INSTALLATION OF THE PLUNGER SPRING IN THE SOLENOID VALVE. THE CORRECT TECHNICAL INFORMATION HAS BEEN RECEIVED FROM THE MANUFACTURER AND THE MAINTENANCE PROCEDURE USED TO INSTALL THESE SOLENOID VALVES ON THE MSIV'S WILL BE REVISED TO REFLECT THIS INFORMATION. THE "D" DC SOLENOID VALVE WAS REPLACED. ST 1.3A-2 WAS COMPLETED SATISFACTORILY PRIOR TO RETURNING UNIT 2 TO SERVICE. THE UNIT 3 SOLENOID VALVES WILL BE INSPECTED PRIOR TO UNIT 3 RESTART.

[116] PEACH BOTTOM 2 DOCKET 50-277 LER 89-024
LOCAL POWER RANGE MONITOR (LPRM) SPIKE CAUSES REACTOR SCRAM SIGNAL WHILE IN HOT SHUTDOWN.
EVENT DATE: 100689 REPORT DATE: 110689 NSSS: GE TYPE: BWR

(NSIC 215844) AT 1307 HOURS ON 10/6/89 WITH UNIT 2 IN HOT SHUTDOWN, THE REACTOR PROTECTION SYSTEM (RPS) INITIATED A FULL REACTOR SCRAM SIGNAL. THE FULL REACTOR SCRAM WAS A RESULT OF A CHANNEL "B" RPS SCRAM SIGNAL BEING RECEIVED IN CONJUNCTION WITH A CHANNEL "A" RPS SCRAM SIGNAL ALREADY INSERTED. THE "B" RPS SCRAM SIGNAL WAS THE RESULT OF A "D" AVERAGE POWER RANGE MONITOR (APRM) HI-HI FLUX SIGNAL CAUSED BY LOCAL POWER RANGE MONITOR (LPRM) 40-33A SPIKING UPSCALE. THE "A" CHANNEL HALF SCRAM WAS INSERTED AS A RESULT OF THE "E" INTERMEDIATE RANGE MONITOR (IRM) BEING INOPERABLE AND BLOCKED IN STANDBY. IMMEDIATE CORRECTIVE ACTION WAS TO BYPASS LPRM 40-33A. THE "A" HALF SCRAM SIGNAL WAS RESET WHEN THE PREVIOUSLY INOPERABLE "E" IRM WAS REPAIRED AND RETURNED TO SERVICE. THE PROXIMATE CAUSE OF THIS EVENT WAS THE OUTPUT SIGNAL FOR THE LPRM 40-33A SPIKING HIGH. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. IT IS PLANNED TO PERFORM AN INVESTIGATION OF THE CAUSE OF THE SPIKING OUTPUT OF LPRM 4033A. THE RESULTS OF THIS INVESTIGATION AND THE RESULTING CORRECTIVE ACTIONS WILL BE REPORTED IN A SUPPLEMENT TO THIS LER. THERE WAS ONE PREVIOUS SIMILAR EVENT.

[117] PEACH BOTTOM 2 DOCKET 50-277 LER 89-025
INOPERABLE AUTOMATIC DEPRESSURIZATION SYSTEM DUE TO CIRCUITRY CONTAINING NON-SAFETY RELATED COMPONENT CAUSED BY INADEQUATE ORIGINAL DESIGN.
EVENT DATE: 100789 REPORT DATE: 110289 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 215845) ON 10/7/89, PECO DETERMINED THAT NON-SAFETY RELATED BELLOWS LEAK DETECTING PRESSURE SWITCHES (BLDPS) INSTALLED ON MAIN STEAM RELIEF VALVES (MSRVS) COULD PREVENT THE MANUAL AND AUTOMATIC OPENING OF MULTIPLE MSRVS DURING DESIGN BASIS CONDITIONS. THIS WOULD POTENTIALLY RENDER BOTH THE AUTOMATIC AND MANUAL FUNCTION OF THE AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) INOPERABLE. UNDER CERTAIN SEISMIC OR ENVIRONMENTAL CONDITIONS, THE POSITIVE SIDE OF THE MSRV LOGIC CIRCUITS COULD BECOME GROUNDED BECAUSE OF THE NON-SAFETY RELATED BLDPS AND IF A SINGLE FAILURE IN THE NEGATIVE SIDE OF THE CIRCUITRY OCCURS, MULTIPLE ADS (MSRV) VALVES COULD BE RENDERED INOPERABLE. THERE WERE NO ACTUAL SAFETY CONSEQUENCES THAT OCCURRED AS A RESULT OF THIS CONDITION. THE ROOT CAUSE OF THIS EVENT WAS LESS THAN ADEQUATE ORIGINAL DESIGN OF THE MSRV CIRCUITRY. A TEMPORARY PLANT ALTERATION (TPA) WAS INSTALLED ON UNIT 2, THEREBY RETURNING ADS TO AN OPERABLE STATUS ON 10/9/89. A SIMILAR TPA OR MODIFICATION WILL BE INSTALLED ON UNIT 3 PRIOR TO UNIT 3 RESTART. AN APPROPRIATE MODIFICATION WILL BE PERFORMED TO PERMANENTLY FIX THIS CONDITION. THERE HAVE BEEN NO PREVIOUS SIMILAR LERS.

[118] PEACH BOTTOM 2 DOCKET 50-277 LER 89-026
CONTROL ROOM EMERGENCY VENTILATION SYSTEM ACTUATION DUE TO RADIATION MONITOR SETPOINT ANOMALY DURING SURVEILLANCE TESTING.
EVENT DATE: 101289 REPORT DATE: 110889 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 215846) ON OCTOBER 12, 1989, AT 0900, A CONTROL ROOM EMERGENCY VENTILATION ACTUATION OCCURRED. THE ACTUATION OCCURRED DUE TO A MOMENTARY FALSE HIGH RADIATION SIGNAL FROM THE CONTROL ROOM VENTILATION "B" RADIATION MONITOR. THE ACTUATION OCCURRED WHILE A TECHNICIAN WAS RESTORING THE MONITOR HIGH RADIATION SETPOINT TO NORMAL RANGE, FOLLOWING PERFORMANCE OF A SURVEILLANCE TEST, USING THE THUMBWHEEL SETPOINT SWITCH. THE THUMBWHEEL APPARENTLY GENERATED A SPURIOUS MISREPRESENTATIVE SETPOINT VALUE BELOW NORMAL BACKGROUND RADIATION LEVEL. THE CAUSE OF THIS ANOMALY IS ATTRIBUTED TO SENSITIVITY OF THE SWITCH TO INTERRUPTION OF CONTACT OF THE SWITCH CONTACTS WHEN CHANGING SWITCH POSITION. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. ATTEMPTS TO RECREATE THE EVENT DID NOT REPRODUCE THE ACTUATION. THE SWITCH CONTACTS WERE CLEANED AS A PRECAUTION. THE APPLICABLE SURVEILLANCE TESTS WILL BE REVISED TO INCLUDE A NOTE OF CAUTION REGARDING THUMBWHEEL MOVEMENT. THERE WERE NO PREVIOUS SIMILAR LERS.

[119] 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 9/19 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.

[120] PILGRIM 1 DOCKET 50-293 LER 88-002 REV 01
UPDATE ON FULL SCRAM TRIP SIGNAL DURING SURVEILLANCE AND RESULTING INCOMPLETE ACTUATIONS.
EVENT DATE: 011788 REPORT DATE: 110989 NSSS: GE TYPE: BWR
VENDOR: BORG-WARNER CORP.
GENERAL ELECTRIC CO.

(NSIC 215804) ON JANUARY 17, 1988 AT 0113 HOURS, A FULL REACTOR PROTECTION SYSTEM (RPS) SCRAM TRIP SIGNAL OCCURRED DURING A SURVEILLANCE TEST. THE AUTOMATIC ACTUATIONS RESULTING FROM THE TRIP SIGNAL WERE INCOMPLETE IN THAT PORTIONS OF THE "B" TRAINS OF THE PRIMARY CONTAINMENT SYSTEM (PCS) AND SECONDARY CONTAINMENT SYSTEM (SCS) DID NOT ACTUATE AS DESIGNED. A CONTROL ROD DRIVE (CRD) SYSTEM VENT VALVE DID NOT CLOSE AUTOMATICALLY. FOLLOWING IMMEDIATE INVESTIGATION, THE RPS TRIP SIGNAL AND ISOLATIONS WERE RESET AT 0135 HOURS. THE CAUSE FOR THE TRIP SIGNAL WAS INADEQUACY OF THE PROCEDURE BEING USED FOR THE SURVEILLANCE. THE CAUSE FOR THE INCOMPLETE AUTOMATIC ACTUATIONS WAS HIGH CONTACT RESISTANCE OF TWO CONTACTS OF A LOGIC RELAY. THE CAUSE FOR THE CRD SYSTEM VENT VALVE NOT CLOSING AUTOMATICALLY WAS MECHANICAL BINDING OF THE VALVE DUE TO EXCESSIVE CORROSION BUILDUP BETWEEN THE VALVE BONNET AND VALVE STEM. CORRECTIVE ACTIONS CONSISTED OF REVISING THE SURVEILLANCE PROCEDURE AND REPLACING THE LOGIC RELAY. CORRECTIVE ACTIONS FOR THE CRD SYSTEM VENT VALVE INCLUDED REMOVAL OF THE EXCESSIVE CORROSION BUILDUP BETWEEN THE VALVE BONNET AND VALVE STEM THAT CAUSED THE VALVE TO BIND. THIS EVENT OCCURRED DURING AN EXTENDED OUTAGE WITH NEGLIGIBLE CORE DECAY HEAT AND WITH THE MODE SELECTOR SWITCH IN THE SHUTDOWN POSITION. THERE WERE NO CONTROL RODS IN THE WITHDRAWN POSITION AT THE TIME OF THE TRIP SIGNAL.

[121] POINT BEACH 2 DOCKET 50-301 LER 89-005
INTERMEDIATE RANGE HIGH FLUX TRIP SIGNAL DUE TO A LOOSE CONNECTION FOR THE INPUT/OUTPUT CABLE.
EVENT DATE: 100689 REPORT DATE: 110689 NSSS: WE TYPE: PWR
VENDOR: POWER DESIGNS INC.

(NSIC 215852) ON OCTOBER 6, 1989, AT 10:02 A.M., WHILE IN THE REFUELING SHUTDOWN WITH THE REACTOR TRIP BREAKERS OPEN, AN INTERMEDIATE RANGE HIGH LEVEL TRIP SIGNAL WAS UNEXPECTEDLY GENERATED DURING THE COURSE OF A ROUTINE SOURCE RANGE CHANNEL CALIBRATION. SUBSEQUENT INVESTIGATION REVEALED A CONNECTION FOR THE INPUT/OUTPUT CABLE TO INTERMEDIATE RANGE CHANNEL N35 POWER SUPPLY NQ203 WAS APPROXIMATELY 3/4 TURN LOOSE. POWER SUPPLY NQ203 FEEDS THE INTERMEDIATE RANGE BISTABLE. THE CONNECTION WAS TIGHTENED AND THE CONNECTIONS FOR THE REMAINING INTERMEDIATE RANGE AND SOURCE RANGE DETECTORS WERE CHECKED. THE OTHER CONNECTIONS FOR THIS UNIT

CHECKED "TIGHT." A MAINTENANCE WORK REQUEST WAS ISSUED TO COMPLETE THE INVESTIGATION AND CHECK THE TIGHTNESS OF INTERMEDIATE AND SOURCE RANGE CONNECTIONS ON UNIT 1 AFTER THE REFUELING SHUTDOWN IN SPRING 1990.

[122] POINT BEACH 2 DOCKET 50-301 LER 89-006
DEGRADATION OF STEAM GENERATOR TUBES.
EVENT DATE: 101589 REPORT DATE: 111489 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215853) UNIT 2 WAS SHUT DOWN FOR REFUELING 15 ON SEPTEMBER 22, 1989. EDDY CURRENT EXAMINATION OF THE STEAM GENERATOR TUBES WAS CONDUCTED FROM OCTOBER 4 TO OCTOBER 15, 1989, USING A DIGITAL MULTI-FREQUENCY EDDY CURRENT SYSTEM. IN THE A STEAM GENERATOR HOT LEG, INSPECTION RESULTS INDICATED 4 TUBES DEGRADED EQUAL TO OR GREATER THAN 40% OF THE WALL THICKNESS, ONE TUBE WITH AN UNDEFINED SIGNAL, 20 TUBES WITH AXIAL INDICATIONS IN THE TUBESHEET AREA, AND 2 RESTRICTED TUBES. ALL 27 OF THESE TUBES WERE PLUGGED. IN THE B STEAM GENERATOR A TOTAL OF 7 TUBES WERE DEGRADED EQUAL TO OR GREATER THAN 40% OF THE TUBE WALL THICKNESS, AND 2 TUBES HAD AXIAL INDICATIONS IN THE TUBESHEET AREA. SIX OF THESE TUBES WERE MECHANICALLY PLUGGED AND THE REMAINING THREE WERE INCLUDED IN THE SLEEVING PROGRAM. A TOTAL OF 298 TUBES WERE PREVENTIVELY SLEEVED IN THE COLD LEG TO ADDRESS WASTAGE CONCERNS. THE 800 PSID LEAK TEST REVEALED 4 MECHANICALLY PLUGGED SLEEVED TUBES AND TWO SLEEVES LEAKING SLIGHTLY IN THE B HOT LEG WHILE 3 OTHER TUBES SHOWED DAMPNESS. ONE EXPLOSIVELY PLUGGED TUBE AND THREE SLEEVED TUBES WERE LEAKING SLIGHTLY IN THE A HOT LEG. THE LEAKING MECHANICAL PLUGS WERE REPAIRED BY THE PLUG-IN-PLUG REPAIR AND THE OTHERS WERE LEFT AS FOUND BECAUSE OF THE MINIMAL LEAKRATE (<1 DROP/2 MIN) OR THE LACK OF ANY INDICATIONS DURING EDDY CURRENT TESTING. THE LEAKING SLEEVES WERE LEFT AS FOUND AS THE SLEEVES ARE LEAK LIMITING BY DESIGN.

[123] QUAD CITIES 1 DOCKET 50-254 LER 87-017 REV 01
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 10FR50.73(A)(2)(IV) AND (A)(2 (V).

[124] QUAD CITIES 1 DOCKET 50-254 LER 89-016
NEW FUEL ASSEMBLY DROPPED IN FUEL POOL WHEN REFUEL BRIDGE FUEL GRAPPLE RELEASED DUE TO PERSONNEL ERROR AND LACK OF PROCEDURAL GUIDANCE.
EVENT DATE: 092189 REPORT DATE: 101889 NSSS: GE TYPE: BWR

(NSIC 215823) ON SEPTEMBER 21, 1989. UNIT ONE WAS IN THE SHUTDOWN MODE WITH ALL FUEL REMOVED FROM THE REACTOR VESSEL. AT 1410 HOURS, DURING THE TRANSFER OF NEW FUEL FROM THE NEW FUEL STORAGE VAULT TO THE FUEL POOL, FUEL ASSEMBLY LYT191 WAS RELEASED FROM THE REFUELING GRAPPLE AND FELL UPON SPENT FUEL RACKS. THE GRAPPLE CONTROL SWITCH WAS INADVERTENTLY LEFT IN THE "RELEASE" POSITION AFTER ATTEMPTING TO UNLATCH. THE UNLATCHING FAILURE WAS DUE TO THE ADJACENT ASSEMBLY NOT BEING FULLY SEATED. THE CAUSE OF FUEL ASSEMBLY DROP WAS A COMBINATION OF PERSONNEL ERROR AND PROCEDURAL DEFICIENCY. CORRECTIVE ACTION INCLUDED A REFUEL BRIDGE HOIST CIRCUITRY MODIFICATION TO PREVENT RAISING A FUEL ASSEMBLY WITH THE GRAPPLE CONTROL SWITCH IN "RELEASE." THE FUEL HANDLING PROCEDURES WERE REVISED TO ASSURE PROPER FUEL ASSEMBLY SEATING AND PROPER POSITIONING OF THE GRAPPLE CONTROL

SWITCH. THIS REPORT IS SUBMITTED TO COMPLY WITH THE REQUIREMENTS OF 10CFR20 405(A)(1)(IV).

[125] QUAD CITIES 1 DOCKET 50-254 LER 89-018
POTENTIAL SINGLE FAILURE OF THE DIESEL GENERATOR VOLTAGE REGULATOR WHICH COULD RESULT IN A LOSS OF ALL BUT ONE EMERGENCY CORE COOLING SYSTEM LOOP.
EVENT DATE: 101289 REPORT DATE: 110989 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 215825) ON OCTOBER 12, 1989, AN EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE NOTIFICATION WAS MADE TO THE NUCLEAR REGULATORY COMMISSION (NRC), NOTIFYING THEM OF A SCENARIO INVOLVING THE PLANT POSSIBLY BEING OUTSIDE THE DESIGN BASIS. IT WAS DETERMINED THAT DURING A LOSS OF OFFSITE POWER (LOOP), IN CONJUNCTION WITH A LOSS OF COOLANT ACCIDENT (LOCA), IT IS POSSIBLE THAT THE UNIT DIESEL GENERATOR (DG)(EK) VOLTAGE REGULATOR (RG) COULD FAIL IN SUCH A MANNER THAT THE ONLY EMERGENCY CORE COOLING SYSTEM (ECCS) AVAILABLE WOULD BE ONE LOOP OF THE CORE SPRAY (CS) SYSTEM (BM). THE SCENARIO IS OF MINIMAL SAFETY SIGNIFICANCE BECAUSE OF THE LOW PROBABILITY OF SUCH AN OCCURRENCE, CALCULATED TO BE ON THE ORDER OF 1E-8/YR. REGULAR SURVEILLANCES AND MONITORING OF DG PARAMETERS DURING OPERATION, AND A NEW PROCEDURE MITIGATE THE AFFECTS OF SUCH AN OCCURRENCE. OPTIONS TO ELIMINATE THIS DESIGN DEFICIENCY ARE UNDER INVESTIGATION. THIS REPORT IS BEING SUBMITTED AS A VOLUNTARY REPORT.

[126] RIVERBEND 1 DOCKET 50-458 LER 89-032
RCIC ISOLATION DURING PERFORMANCE OF A SURVEILLANCE TEST PROCEDURE DUE TO PERSONNEL ERROR.
EVENT DATE: 090789 REPORT DATE: 100489 NSSS: GE TYPE: BWR

(NSIC 215595) AT 1412 ON 9/7/89 WITH THE UNIT IN OPERATIONAL CONDITION 1 AT 100 PERCENT POWER, AN UNPLANNED ENGINEERED SAFETY FEATURE (ESF) ACTUATION OCCURRED DURING THE PERFORMANCE OF SURVEILLANCE TEST PROCEDURE (STP-207-4536) WHEN AN INSTRUMENT AND CONTROL (I&C) TECHNICIAN FAILED TO PROPERLY LIFT A LEAD. THE STEP REQUIRES THE LEAD TO BE LIFTED TO PREVENT AN ISOLATION FROM OCCURRING DURING THE TEST. HOWEVER, DUE TO HUMAN ERROR, THE REACTOR CORE ISOLATION COOLING (RCIC) OUTBOARD CONTAINMENT ISOLATION VALVE CLOSED WHEN THE TRIP SIGNAL WAS INITIATED. IMMEDIATELY UPON DISCOVERY OF THE ERROR, THE TRIP SIGNAL WAS RESET, THE SYSTEM WAS RESTORED 78 MINUTES LATER AND THE STP WAS SUCCESSFULLY COMPLETED. PERSONNEL IN THE I&C DEPARTMENT WILL RECEIVE A BRIEFING ON THE EVENT AND WILL BE CAUTIONED THAT THE UTMOST ATTENTION TO DETAIL IS REQUIRED WHILE PERFORMING STPS. TECHNICIANS INVOLVED WERE COUNSELED ON THEIR ERROR. IN ADDITION, ADM-0015, "STATION SURVEILLANCE TEST PROGRAM", WILL BE REVISED TO REQUIRE THAT THE READER VERIFY THAT AN ACTION IS COMPLETED PRIOR TO SIGNING OFF A STEP. ALL SYSTEMS PERFORMED PER THEIR DESIGN IN RESPONSE TO THE TRIP SIGNAL. NO ACTUAL CONDITION EXISTED FOR WHICH ISOLATION OF THE RCIC SYSTEM IS REQUIRED. THEREFORE, THERE WAS NO ADVERSE IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT.

[127] ROBINSON 2 DOCKET 50-261 LER 89-011
AUXILIARY FEEDWATER SYSTEM FLOWRATE COULD EXCEED LIMITS OF ACCIDENT ANALYSIS.
EVENT DATE: 100689 REPORT DATE: 110689 NSSS: WE TYPE: PWR

(NSIC 215829) WITH THE PLANT IN COLD SHUTDOWN, CALCULATIONS WERE PERFORMED IN SUPPORT OF A PLANT MODIFICATION TO INCREASE THE DIAMETER OF THE AUXILIARY FEEDWATER (AFW) PUMP SUCTION PIPING. THESE CALCULATIONS SHOWED THAT A MAIN STEAMLINE BREAK CONCURRENT WITH A LOSS OF NON-SAFETY GRADE POWER TO THE STEAM DRIVEN (SD) AFW PUMP DISCHARGE FLOW CONTROL VALVE COULD RESULT IN EXCEEDING THE LIMITS OF THE MAIN STEAMLINE BREAK ACCIDENT ANALYSIS. THE RESULTS OF THESE CALCULATIONS WERE CONVEYED TO SITE PERSONNEL AT 1610 HOURS ON OCTOBER 6, 1989. AT 1654 HOURS, PURSUANT TO THE REQUIREMENTS OF 10CF50.72(B)(2)(I), A FOUR HOUR NON-EMERGENCY EVENT NOTIFICATION WAS MADE TO THE NRC VIA THE EMERGENCY NOTIFICATION SYSTEM (ENS). TO CORRECT THIS SITUATION, A PLANT MODIFICATION WAS DEVELOPED AND IMPLEMENTED TO DECREASE THE FLOWRATE SETPOINT FOR THE DISCHARGE FLOW CONTROL VALVE AND INSTALL A MECHANICAL DEVICE TO LIMIT VALVE TRAVEL IN THE

OPEN DIRECTION. THIS MODIFICATION WILL ENSURE THAT THE VALVE OPERATING RANGE WILL COMPLY WITH THE LIMITS PROVIDED IN THE ACCIDENT ANALYSES. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(II)(B).

[128] SALEM 1 DOCKET 50-272 LER 89-018 REV 01
UPDATE ON SSPS CABINET CONNECTIONS UNSATISFACTORY DUE TO INADEQUATE INITIAL FABRICATION.
EVENT DATE: 050289 REPORT DATE: 110989 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215839) ON 5/2/89, DURING TESTING FOLLOWING MODIFICATIONS TO THE SOLID STATE PROTECTION SYSTEM (SSPS), SEVERAL WIRES WERE DISCOVERED TO NOT BE PROPERLY CONNECTED TO THE CIRCUIT BOARD CONNECTORS. SSPS CABINETS 100% VISUAL INSPECTION AND 100% PULL-TESTING WAS PERFORMED. RESULTS SHOWED THAT 119 OF 2644 CLIPS WERE UNSATISFACTORY FOR TRAIN A AND 103 OF 2644 CLIPS WERE UNSATISFACTORY FOR TRAIN B. DISCUSSION WITH THE TECHNICIAN INDICATES THE WIRES IN THE SSPS CABINET WERE IN CONTACT WITH THE CIRCUIT BOARD, AND IT WAS ONLY THROUGH PHYSICAL MOVEMENT THAT THEY HAD BECOME DISCONNECTED. REVIEW OF MODIFICATION WORK PERFORMED ON SSPS SHOWED THAT MANY OF THE AREAS IDENTIFIED AS HAVING UNSATISFACTORY CONNECTIONS HAVE NOT BEEN MODIFIED SINCE THEY WERE ORIGINALLY DELIVERED BY WESTINGHOUSE. THE ROOT CAUSE OF THIS OCCURRENCE HAS BEEN ATTRIBUTED TO INADEQUATE CONNECTION OF THE TERMI-POINT CLIPS DURING INITIAL SSPS FABRICATION PERFORMED BY WESTINGHOUSE. WESTINGHOUSE HAS STATED THAT THEY WERE UNAWARE OF ANY SIMILAR OCCURRENCES INVOLVING SSPS, BUT INTENDED TO ISSUE A TECHNICAL BULLETIN ON THE SUBJECT. FULL VISUAL INSPECTION AND A COMPLETE PULL-TEST OF EACH TRAIN OF THE SSPS INPUT, LOGIC AND OUTPUT CABINETS HAS BEEN COMPLETED. ALL CONNECTIONS THAT DID NOT PASS THE INSPECTION AND/OR TEST WERE REPLACED AND PULL-TESTED SATISFACTORILY.

[129] SALEM 1 DOCKET 50-272 LER 89-029
TECH SPEC 3.3.2.1 TABLE 3.3-8F ACTION 21 NON-COMPLIANCE DUE TO INADEQUATE PROCEDURES.
EVENT DATE: 090489 REPORT DATE: 102789 NSSS: WE TYPE: PWR

(NSIC 215837) ON 10/16/89 IT WAS DETERMINED THAT TECH SPEC 3.3.2.1 TABLE 3.3-3.8F ACTION 21 WAS NOT COMPLIED WITH HISTORICALLY. TABLE 3.3-3.8F ADDRESSES THE REQUIREMENT TO HAVE THE MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS (MDAFPS) START ON A LOSS OF MAIN FEEDWATER SIGNAL. THIS IS ONLY APPLICABLE IN MODE 1, "POWER OPERATION". THE LOSS OF MAIN FEEDWATER SIGNAL IS GENERATED UPON THE TRIPPING OF BOTH STEAM GENERATOR FEEDWATER PUMPS (SGFPS). THE DESIGN OF THE TRIP CIRCUITRY IS SUCH THAT IF THE 125VDC STEAM GENERATOR & TURBINE CONTROL BREAKER (1CDC2AX30) IS CLEARED AND TAGGED (C/T), THE CIRCUIT CONTINUITY TO THE 5X TRIP RELAY IS EFFECTIVELY BROKEN. SUBSEQUENTLY THE 5X RELAY CONTACT CAN NOT BE CLOSED TO ENABLE THE AUTOMATIC START OF THE MDAFPS EVEN IF A TRIP SIGNAL IS PUT INTO THE CIRCUIT. PART OF THE SGFP "STANDARD TAGOUT", AS IDENTIFIED IN THE TAGGING REQUEST INFORMATION SYSTEM (TRIS), IS TO C/T THE 1CDC2AX30 BREAKER WHICH WILL RENDER THE AUTOMATIC START FEATURE OF THE MDAFPS, ON LOSS OF MAIN FEEDWATER, INOPERABLE. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INADEQUATE PROCEDURES. THE DESIGN OF THE CIRCUITRY WAS NOT FULLY CONSIDERED IN DEFINING THE STANDARD TAGOUT OF A SGFP. THIS MAY HAVE OCCURRED DUE TO THE DIFFERENCE IN IDENTIFICATION OF THE BREAKER BETWEEN THE TRIS AND THE ASSOCIATED ENGINEERING PRINTS. THE ENGINEERING PRINTS IDENTIFY THE BREAKER AS 1CCDC30-CT.

[130] SALEM 1 DOCKET 50-272 LER 89-030
TECH SPEC ACTION STATEMENT 3.0.5 ENTRY DUE TO HUMAN FACTORS DESIGN CONCERNS.
EVENT DATE: 101189 REPORT DATE: 110189 NSSS: WE TYPE: PWR

(NSIC 215838) ON 10/11/89 AT 1403 HOURS, TECH SPEC ACTION STATEMENT 3.0.5 WAS ENTERED. NO. 11 CONTAINMENT FAN COIL UNIT (CFCU) WAS INOPERABLE (CLEARED AND TAGGED) DUE TO REQUIRED MAINTENANCE WHEN 1C DIESEL GENERATOR (D/G) (EK) WAS INADVERTENTLY TRIPPED. THE TECH SPEC 3.6.2.3 ACTION STATEMENTS DO NOT APPLY WHEN 2 GROUPS OF CFCUS AND ONE CONTAINMENT SPRAY HEADER ARE INOPERABLE. TECH SPEC ACTION STATEMENT 3.0.5 APPLIES SINCE ONE OF THE GROUPS OF CFCUS AND THE

CONTAINMENT SPRAY HEADER WERE DECLARED INOPERABLE SOLELY DUE TO INOPERABILITY OF THEIR EMERGENCY TRIP PUSH BUTTON WAS INADVERTENTLY PUSHED. A WORKER HAD GONE BETWEEN ERECTED SCAFFOLDING AND THE D/G CONTROL PANEL AND IN SO DOING BRUSHED UP AGAINST THE PUSH BUTTON, ACTUATING THE D/G TRIP. THE D/G EMERGENCY PUSH BUTTON EXTENDS APPROXIMATELY 3 INCHES OUT FROM THE CONTROL PANEL. SUBSEQUENT REVIEW OF THIS EVENT HAS IDENTIFIED THAT BETTER HUMAN FACTORS DESIGN CONSIDERATIONS COULD HAVE PREVENTED THIS EVENT. A FIFTEEN (15) MINUTE OPERABILITY RUN WAS SUCCESSFULLY COMPLETED FOR 1C D/G. SUBSEQUENTLY, ON 10/11/89 AT 1423 HOURS, TECH SPECS ACTION STATEMENT 3.0.5 WAS EXITED. THE WORKER INVOLVED IN THIS EVENT WAS COUNSELED ON THE NEED TO EXERCISE CAUTION WHEN WORKING NEAR PLANT EQUIPMENT. DUE TO THE RECURRING NATURE OF THIS EVENT, ENGINEERING WILL REVIEW POSSIBLE EQUIPMENT MODIFICATIONS TO PREVENT INADVERTENT ACTUATION OF THE D/G EMERGENCY TRIP PUSH BUTTON.

[131] SALEM 2 DOCKET 50-311 LER 89-016
TECH SPEC ACTION STATEMENT 3.0.3 ENTRY - 3 OF 4 MAIN STEAM ISOLATION VALVES INOPERABLE.
EVENT DATE: 101489 REPORT DATE: 111389 NSSS: WE TYPE: PWR
VENDOR: HOPKINSONS LIMITED

(NSIC 215862) ON 10/14/89, DURING A CONTROLLED SHUTDOWN (FOR MAINTENANCE), 3 OF THE 4 MAIN STEAM ISOLATION VALVES (MSIVS) FAILED TO CLOSE WITHIN 5 SECONDS, CONTRARY TO THE REQUIREMENTS OF TECHNICAL SPECIFICATION SURVEILLANCE 4.7.1.5. SINCE THE ACTION STATEMENTS FOR TECHNICAL SPECIFICATION 3.7.1.5 ONLY APPLY IF NO MORE THAN ONE (1) MSIV IS INOPERABLE AND THREE (3) OF THE MSIVS WERE DECLARED INOPERABLE, TECHNICAL SPECIFICATION ACTION STATEMENT 3.0.3 WAS ENTERED. THE ROOT CAUSE OF THIS EVENT HAS NOT BEEN DETERMINED TO DATE. IN THE PAST, THE HYDRAULIC BYPASS VALVES HAVE BEEN RESPONSIBLE FOR MSIV CLOSURE TIME CONCERNS; THEREFORE, THE HYDRAULIC BYPASS VALVES FOR EACH OF THE 3 MSIVS WHICH DID NOT CLOSE WITHIN 5 SECONDS, WERE DISASSEMBLED AND INSPECTED. NO EQUIPMENT PROBLEMS/CONCERN, WERE IDENTIFIED. INVESTIGATION FOR THE ROOT CAUSE OF THIS EVENT IS CONTINUING. A PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH THE INCREASED CLOSURE TIME HAS BEEN INITIATED. A SUPPLEMENTAL REPORT WILL BE ISSUED TO DOCUMENT THE COMPLETION OF INVESTIGATIONS. THE SUPPLEMENTAL REPORT WILL ADDRESS THE RESULTS OF THE PROBABILISTIC RISK ASSESSMENT, ANY ADDITIONAL CORRECTIVE ACTIONS REQUIRED AND THE ROOT CAUSE OF THIS EVENT. THE MSIVS WERE DECLARED OPERABLE UPON SUCCESSFULLY COMPLETING A RE-TEST OF THE TECHNICAL SPECIFICATION SURVEILLANCE AT 1715 HOURS ON NOVEMBER 4, 1989. THE UNIT WAS RETURNED TO MODE 1 (POWER OPERATION).

[132] SAN ONOFRE 1 DOCKET 50-206 LER 88-020 REV 01
UPDATE ON STEAM GENERATOR WIDE RANGE LEVEL INDICATION SYSTEM CONTRARY TO POST-TMI DESIGN REQUIREMENTS.
EVENT DATE: 120888 REPORT DATE: 112089 NSSS: WE TYPE: PWR

(NSIC 215931) ON 12/8/88, WITH UNIT 1 IN MODE 5, IT WAS DETERMINED THAT THE DESIGN REQUIREMENTS OF POST-TMI ACTION PLAN, ITEM II.E.1.2, PART 2, HAD NOT BEEN FULLY IMPLEMENTED IN THE DESIGN OF THE STEAM GENERATOR WIDE-RANGE LEVEL (SGWRL) INDICATION SYSTEM. THIS SYSTEM SERVES AS ONE OF TWO REDUNDANT MEANS OF PROVIDING AUX. FEEDWATER FLOW INDICATION DURING POST-ACCIDENT CONDITIONS. SCE COMMITTED TO UPGRADE THE SGWRL INSTRUMENTATION IN 1981-1982 TO MEET POST-TMI DESIGN REQUIREMENTS BY: 1) PROVIDING FOR ENVIRONMENTAL QUALIFICATION (EQ) OF APPROPRIATE COMPONENTS, AND 2) PROVIDING A POWER SUPPLY HAVING A BATTERY BACK-UP. DURING DESIGN CHANGE WORK BEING PERFORMED ON THE NUCLEAR INSTRUMENTATION SYSTEM DURING THE CYCLE 10 REFUELING OUTAGE, IT WAS DISCOVERED THAT THE THREE SGWRL INDICATORS (ONE PER STEAM GENERATOR), WERE NOT POWERED FROM A BATTERY-BACKED POWER SOURCE, NOR WERE THE ASSOCIATED LEVEL TRANSMITTERS, WHICH ARE INSTALLED IN THE CONTAINMENT BUILDING, ENVIRONMENTALLY QUALIFIED. THIS CONDITION IS ALSO CONTRARY TO TECH SPEC 3.5.6, WHICH REQUIRES THE SGWRL INDICATION SYSTEM TO BE OPERABLE FOR POST-ACCIDENT MONITORING. THE ROOT CAUSE OF THIS EVENT WAS WEAKNESSES IN THE COMMITMENT MANAGEMENT PROGRAM, AS IT WAS ADMINISTERED IN 1979-1982. ONE WEAKNESS INCLUDED NOT SPECIFICALLY IDENTIFYING EACH DISCRETE COMMITMENT FOUND WITHIN THE SOURCE DOCUMENT.

[133] SAN ONOFRE 1 DOCKET 50-206 LER 89-022 REV 01
UPDATE ON NON-CONSERVATIVE TECH SPEC FOR OVERPRESSURE MITIGATION SYSTEM.
EVENT DATE: 091489 REPORT DATE: 110189 NSSS: WE TYPE: PWR

(NSIC 215809) ON 9/14/89, AT 0845, WITH UNIT 1 AT 91% POWER, AN ENGINEERING REVIEW OF 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 < 40 PSIG, ARE NON-CONSERVATIVE. AT LOW RCS PRESSURES, THE LOW 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 THE TS 3.20 BASIS, PRIOR TO EXCEEDING 10CFR50, APP. G, RCS PRESSURE LIMITS AT A 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 THE PRESSURIZER LEVEL IS < 50%. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO PAST WEAKNESSES IN SCE'S ENGINEERING AND TECHNICAL SUPPORT TO SAN ONOFRE, WHICH IS DESCRIBED IN DETAIL IN OUR 10/3/88 SUBMITTAL TO THE NRC ADDRESSING THIS SUBJECT. THE CORRECTIVE ACTIONS IDENTIFIED IN THAT SUBMITTAL ARE ALSO APPLICABLE TO THE CAUSES OF THIS CONDITION.

[134] SAN ONOFRE 2 DOCKET 50-361 LER 88-036 REV 01
UPDATE ON SPENT FUEL HANDLING MACHINE OPERATION WITH POST-ACCIDENT CLEAN-UP UNITS INOPERABLE DUE TO INADEQUATE PROCEDURAL CONTROLS.
EVENT DATE: 121688 REPORT DATE: 110189 NSSS: CE TYPE: PWR

(NSIC 215805) ON 2/3/89, WHILE REVIEWING FUEL TRANSHIPMENT ACTIVITIES, IT WAS NOTED THAT THE SPENT FUEL HANDLING MACHINE (SFHM) MAY HAVE BEEN OPERATED OVER THE FUEL STORAGE POOL WHILE THE POST-ACCIDENT CLEAN-UP UNITS (PACU) WERE NOT OPERABLE. THIS IS CONTRARY TO TECH SPEC 3.9.12, ACTION STATEMENT DURING PERIODS OF NO FUEL MOVEMENT WITHIN THE FUEL HANDLING BUILDING (FHB) (SYSTEM ND), EACH OF THE SIX FHB SPENT FUEL HATCH (COMPONENT DR) PANELS WERE SECURED WITH ONLY 4 OF THE APPROX. 75 SCREWS THE DESIGN PROVIDES, THEREBY REDUCING THE SEISMIC CAPABILITY OF THE HATCH BELOW THAT FOR WHICH THE FHB AND PACUS ARE DESIGNED. IN THIS CONFIGURATION, THE PACUS ARE CONSIDERED INOPERABLE SINCE A DESIGN BASIS EARTHQUAKE COULD RESULT IN A LOSS OF FHB LEAKAGE INTEGRITY. SUBSEQUENT INVESTIGATION OF THE TRANSHIPMENT ACTIVITIES REVEALED THAT WITH THE PACUS INOPERABLE ON 12/16 AND 12/23/88. THE SFHM WAS OPERATED OVER THE SPENT FUEL STORAGE POOL WITH A 40 POUND LOAD BEAM FOR SURVEILLANCE TESTING. CLOSURE OF THE HATCH WAS COMMUNICATED TO OPERATING PERSONNEL WHO, BELIEVING THAT THE HATCH AND THUS THE PACUS WERE OPERABLE IN THE ABOVE INSTANCES, SUBSEQUENTLY AUTHORIZED PERFORMANCE OF THE SFHM SURVEILLANCE. THE ROOT CAUSE FOR THESE EVENTS WAS INADEQUATE PROCEDURE CONTROLS AND INCOMPLETE UNDERSTANDING OF REQUIREMENTS FOR CLOSURE OF FHB HATCH.

[135] SAN ONOFRE 2 DOCKET 50-361 LER 89-018
DELINQUENT TECH SPEC SURVEILLANCE ON 4.16 KV EMERGENCY BUS LOSS OF VOLTAGE SIGNAL.
EVENT DATE: 081589 REPORT DATE: 092589 NSSS: CE TYPE: PWR

(NSIC 215370) AT APPROX. 1300 ON 8/24/89, DURING A REVIEW OF TECH SPEC SURVEILLANCES, IT WAS DISCOVERED THAT THE UNIT 2 TRAIN "B" LOSS OF VOLTAGE SIGNAL (LOVS) CHANNEL CALIBRATION AND CHANNEL FUNCTIONAL TEST, REQUIRED BY TECH SPEC 4.3.2.1, WERE DELINQUENT AS OF 8/15/89. ACTION WAS INITIATED TO PERFORM THE SURVEILLANCES. PLANT MANAGEMENT, CONSIDERING THAT A SHUTDOWN BASED ON MISSED SURVEILLANCES WOULD BE AN OVERLY CONSERVATIVE ACTION WHEN THE EQUIPMENT COULD BE PROVEN OPERABLE UPON COMPLETION OF THE SURVEILLANCES, USED INFORMATION PROVIDED IN GENERIC LETTER 87-09 TO ALLOW 24 HOURS FOR PERFORMANCE OF THE SURVEILLANCES. AT 1056 ON 8/25/89, WITH UNIT 2 OPERATING AT 100% POWER, PERFORMANCE OF THE LOVS FUNCTIONAL TESTING REQUIRED THAT A PORTION OF THE LOVS PROTECTION BE DEFEATED, RESULTING IN A VOLUNTARY ENTRY INTO TECH SPEC 3.0.3. AT 1102, FOLLOWING REINSTATMENT OF THE LOVS PROTECTION, TECH SPEC 3.0.3 WAS EXITED. AT APPROXIMATELY 1115 ON 8/25/89, PLANT MANAGEMENT WAS INFORMED BY THE NRC REGION V

THAT GENERIC LETTER 87-09 WAS INAPPROPRIATELY APPLIED. THEREFORE, THE LOVS TRAIN "B" INSTRUMENTATION WAS DECLARED INOPERABLE AND TECH SPEC 3.0.3 WAS ENTERED. AT 1201, TECH SPEC 3.0.3 WAS EXITED FOLLOWING THE COMPLETION OF A SATISFACTORY TECH SPEC SURVEILLANCE. THE ROOT CAUSE OF THE DELINQUENT SURVEILLANCE IS PERSONNEL ERROR.

[136] SAN ONOFRE 3 DOCKET 50-362 LER 89-010
FUEL HANDLING ISOLATION SYSTEM TRAIN "A" ACTUATION DUE TO POWER SUPPLY FAILURE.
EVENT DATE: 091089 REPORT DATE: 101089 NSSS: CE TYPE: PWR
VENDOR: NUCLEAR MEASUREMENTS CORP.

(NSIC 215588) AT 0045 ON 9/10/89, FUEL HANDLING ISOLATION SYSTEM (FHIS) TRAIN "A" ACTUATED DUE TO LOSS OF POWER ON THE PARTICULATE/IODINE CHANNEL. AFTER DETERMINATION THAT FHIS ACTUATION WAS SPURIOUS, FHIS TRAIN "A" WAS RESET AT 0115. MONITOR WAS PLACED IN BYPASS AND THE FUEL HANDLING BUILDING (FHB) VENTILATION SYSTEM WAS RETURNED TO NORMAL. THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE RADIATION LEVELS REMAINED NORMAL AND ALL FHIS TRAIN "A" COMPONENTS FUNCTIONED AS DESIGNED. THE REDUNDANT FHIS TRAIN "B" REMAINED OPERABLE THROUGHOUT THE EVENT. THE LOSS OF POWER TO THE RADIATION MONITOR MODULE WAS CAUSED BY AN OVERCURRENT CONDITION DUE TO A SHORT IN THE MODULE'S POWER SUPPLY CIRCUIT WHICH RESULTED IN A BLOWN FUSE, INTERRUPTING CURRENT TO THE MODULE. THE ROOT CAUSE IS MANUFACTURING DEFICIENCIES WITH NUCLEAR MEASUREMENT CORPORATION CRM 74/75 INSTRUMENT MODULES. A NYLON SCREW, WHICH SECURES THE -15 VDC VOLTAGE REGULATOR TO A METAL PLATE HEAT SINK, WITH MICA INSULATION IN BETWEEN THE REGULATOR AND HEAT SINK, WAS FOUND BROKEN DUE TO THERMAL AGING. THIS PERMITTED THE REGULATOR TO SHIFT SLIGHTLY AND ALLOWED A BURR ON THE METAL HEAT SINK TO PENETRATE THE MICA INSULATION. THE BURR ESTABLISHED THE SHORT CIRCUIT FROM THE PLATE TO THE VOLTAGE REGULATOR. A PREVIOUS FHIS ACTUATION DUE TO A SIMILAR MODULE POWER SUPPLY FAILURE WAS REPORTED IN LER 88-011 (DOCKET 50-361).

[137] SEQUOYAH 1 DOCKET 50-327 LER 88-028 REV 01
UPDATE ON HIGH AIRBORNE ACTIVITY LEVEL IN THE AUXILIARY BUILDING RESULTED IN THE SUSPENSION OF FIRE WATCH PATROLS.
EVENT DATE: 072388 REPORT DATE: 091589 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 215227) ON 7/23/88, WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN), THE HOURLY FIRE WATCH PATROL THROUGH THE 734 FT ELEVATION OF THE AUX. BLDG. (REFUELING FLOOR) WAS SUSPENDED FROM 0700 EDT TO 1500 EDT BECAUSE OF AN UNEXPECTED INCREASE IN THE AIRBORNE RADIOACTIVITY IN THAT AREA. THE SUBJECT FIRE WATCH PATROL WAS REQUIRED BY ACTION STATEMENT (A) TO LIMITING CONDITION FOR OPERATION (LCO) 3.7.12 AS A COMPENSATORY MEASURE FOR 3 BREACHED FIRE BARRIERS. FIRE BARRIER BREACHING PERMITS WERE ISSUED AS PART OF A WORKPLAN TO REROUTE A CABLE TO PRECLUDE A POTENTIAL "APPENDIX R" CABLE INTERACTION AND ENSURE THE AVAILABILITY OF REACTOR COOLANT SYSTEM (RCS) LETDOWN FOLLOWING A POSTULATED FIRE. THE IMMEDIATE CAUSE OF THIS EVENT WAS THE HIGH LEVEL OF AIRBORNE RADIOACTIVITY IN THE AUX. BLDG. THIS CONDITION REPRESENTED AN OVERRIDING PERSONNEL SAFETY CONCERN AND ACCESS TO THE AREA WAS SUBSEQUENTLY RESTRICTED. THE ROOT CAUSE WAS VALVE LEAKS THAT ALLOWED NITROGEN TO LEAK INTO THE RCS DURING SAFETY INJECTION SYSTEM TESTING. THE INLEAKAGE CAUSED A RISE IN THE INDICATED RCS LEVEL AND PROMPTED OPERATORS TO OPEN THE REACTOR VESSEL HEAD VENTS. THE HEAD VENT LINE CONNECTS WITH THE PRESSURIZER SAFETY VALVE DISCHARGE LINE TO VENT TO THE PRESSURIZER RELIEF TANK. THERE WAS AN INCREASE IN THE ACTIVITY LEVEL OF THE CONTAINMENT ATMOSPHERE.

[138] SEQUOYAH 2 DOCKET 50-328 LER 89-013
INCORRECT SMOKE DETECTORS LOCATED IN ANNULUS FIRE ZONE 374 DUE TO PERSONNEL ERROR.
EVENT DATE: 091489 REPORT DATE: 101389 NSSS: WE TYPE: PWR
VENDOR: PYROTRONICS

(NSIC 215642) ON 9/14/89, WITH UNITS 1 AND 2 IN MODE 1 AT 100% POWER, 2,235 POUNDS PER SQUARE INCH GAUGE, 578F, IT WAS DISCOVERED THAT THE MINIMUM NUMBER OF OPERABLE PHOTOELECTRIC FIRE DETECTORS WAS NOT MAINTAINED FOR FIRE ZONE 374 IN THE ANNULUS AREA OF UNIT 2, AS REQUIRED BY TECH SPEC (TS) LIMITING CONDITION FOR

OPERATION (LCO) 3.3.3.8 AND SHOWN ON TABLE 3.3-11. DURING THE PERFORMANCE OF SURVEILLANCE INSTRUCTION (SI) 234.7, "TECHNICAL SPECIFICATION FIRE DETECTORS," THREE FIRE DETECTORS WERE IDENTIFIED AS IONIZATION-TYPE DETECTORS. THE REMAINING 19 DETECTORS WERE IDENTIFIED AS PHOTOELECTRIC-TYPE FIRE DETECTORS. TS TABLE 3.3-11 REQUIRES A MINIMUM NUMBER OF 20 PHOTOELECTRIC FIRE DETECTORS BE OPERABLE IN FIRE ZONE 374. WITH ONLY 19 PHOTOELECTRIC FIRE DETECTORS INSTALLED AND OPERABLE, SQN HAD OPERATED IN A CONDITION PROHIBITED BY TSS. THE THREE IONIZATION-TYPE DETECTORS WERE DECLARED INOPERABLE, AND LCO 3.3.3.8 WAS ENTERED AT 0223 EASTERN DAYLIGHT TIME (EDT) ON SEPTEMBER 15, 1989. AN HOURLY FIRE WATCH WAS ESTABLISHED, A CONDITION ADVERSE TO QUALITY REPORT WAS INITIATED, AND A WORK REQUEST TO REPLACE THE THREE INCORRECT INSTRUMENTS WAS WRITTEN AND IMPLEMENTED. LCO 3.3.3.8 WAS EXITED AT 2141 EDT ON 9/15/89, AFTER THE DETECTORS WERE REPLACED AND SI-234.7 WAS SUCCESSFULLY PERFORMED. THE ROOT CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE PERSONNEL ERROR.

[139] SHOREHAM DOCKET 50-322 LER 86-038 REV 01
UPDATE ON RBSVS "A" SIDE INITIATION DURING AN I&C SURVEILLANCE PROCEDURE WHEN A TECHNICIAN ACCIDENTLY BRUSHED A RELAY WITH A LIFTED LEAD.
EVENT DATE: 100486 REPORT DATE: 012987 NSSS: GE TYPE: BWR

(NSIC 202621) ON 10-4-86 AT 1044, A RBSVS "A" SIDE INITIATION OCCURRED DURING AN I&C SURVEILLANCE TEST WHEN AN I&C TECHNICIAN ACCIDENTLY BRUSHED A LIFTED LEAD AGAINST A RELAY. THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) AT THE TIME WITH THE MODE SWITCH IN SHUTDOWN AND ALL RODS INSERTED IN THE CORE. THE TECHNICIAN WAS PERFORMING A CHANNEL FUNCTIONAL TEST ON THE "A" REACTOR BUILDING DIFFERENTIAL PRESSURE TRANSMITTER, 1T46*PDT043A, IN ACCORDANCE WITH SP44.650.16 (REACTOR BUILDING DIFFERENTIAL PRESSURE - LOW CHANNEL FUNCTIONAL TEST). AS HE WAS LIFTING THE LEADS TO PREVENT AN INADVERTENT RBSVS AND CRAC INITIATION, THE TECHNICIAN ACCIDENTLY BRUSHED ONE OF THE LEADS AGAINST A RELAY IN THE RBSVS INITIATION LOGIC CIRCUITRY. UNAWARE OF WHETHER OR NOT THE RELAY HAD CHANGED STATE, HE STOPPED WORK AND NOTIFIED THE CONTROL ROOM WHO INFORMED HIM THAT CONDITIONS HAD NOT CHANGED. HE THEN PROCEEDED WITH THE TEST. AS HE RETERMINATED THE LIFTED LEADS, THE REACTOR BUILDING NORMAL VENTILATION SYSTEM (RBNVS) ISOLATED AND AN "A" SIDE RBSVS INITIATION OCCURRED. THE CONTROL ROOM INSTRUCTED THE TECHNICIAN TO RETURN THE LEADS TO THE AS FOUND POSITION, SECURED THE RBSVS, AND RETURNED THE RBNVS TO NORMAL AT 1052. TO PREVENT RECURRENCE, PROCEDURAL REVISIONS WILL BE IMPLEMENTED.

[140] SOUTH TEXAS 1 DOCKET 50-498 LER 89-020
UNPLANNED ENGINEERED SAFETY FEATURES ACTUATIONS DUE TO AN INVERTER FAILURE.
EVENT DATE: 101189 REPORT DATE: 111089 NSSS: WE TYPE: PWR
VENDOR: ELGAR, CORP.

(NSIC 215927) ON 10/11/89, UNIT 1 WAS IN MODE 3 PRIOR TO RESTART FROM A REFUELING OUTAGE. AT APPROXIMATELY 1533 HOURS, THE INVERTER WHICH FEEDS THE CHANNEL IV CLASS 1E VITAL AC DISTRIBUTION PANEL FAILED. THIS CAUSED ENGINEERED SAFETY FEATURES ACTUATIONS OF THE CONTROL ROOM, REACTOR CONTAINMENT BUIDLING AND FUEL HANDLING BUILDING HVAC SYSTEMS DUE TO LOSS OF POWER TO THEIR RESPECTIVE RADIATION MONITORS. THE CAUSE OF THIS EVENT WAS FAILURE OF A BRIDGE RECTIFIER CIRCUIT ON THE INVERTER DC TO DC CONVERTER BOARD. THE CIRCUIT APPEARED TO HAVE FAILED DUE TO EXCESSIVE OUTPUT VOLTAGE OVER AN EXTENDED PERIOD OF TIME WHICH OVERHEATED THE COMPONENTS. THE INVERTER HAS BEEN REPAIRED AND THE DC TO DC CONVERTER VOLTAGE ADJUSTED. PREVENTIVE MAINTENANCE PROCEDURES WILL BE REVISED TO REQUIRE PERIODIC DC TO DC CONVERTER BOARD OUTPUT VOLTAGE ADJUSTMENTS. INVERTERS WILL BE ADDED TO THE EXISTING PLANT THERMOGRAPHY PROGRAM WHICH WILL ASSIST IN IDENTIFYING EQUIPMENT THAT COULD BE APPROACHING A SIMILAR FAILURE.

[141] SOUTH TEXAS 2 DOCKET 50-499 LER 89-025
TECH SPEC VIOLATION DUE TO FAILURE TO PROPERLY CALIBRATE POWER RANGE NUCLEAR INSTRUMENTATION.
EVENT DATE: 100689 REPORT DATE: 111389 NSSS: WE TYPE: PWR

(NSIC 215929) ON 10/6/89, UNIT 2 WAS IN MODE 1 AT 100% POWER. AT APPROX. 1406

HOURS, DURING THE PERFORMANCE OF POWER RANGE NUCLEAR INSTRUMENTATION CALIBRATIONS, ONE CHANNEL WAS INCORRECTLY CALIBRATED. THIS CONDITION, WHICH RENDERED THE CHANNEL INOPERABLE, WAS NOT DISCOVERED UNTIL APPROXIMATELY 2340 HOURS. FAILURE TO PLACE THE INOPERABLE CHANNEL IN THE TRIPPED CONDITION WITHIN SIX HOURS OF ITS BECOMING INOPERABLE IS A VIOLATION OF TECH SPEC 3.3.1. THE CAUSE OF THIS EVENT WAS THE ASSIGNMENT OF A TECHNICIAN TO PERFORM THE CALIBRATION WHO WAS NOT ADEQUATELY TRAINED. IN ADDITION, NO MEANS EXISTED TO CONFIRM THAT THE CALIBRATION WAS CORRECT. A MAINTENANCE QUALIFICATION PROGRAM HAS BEEN DEVELOPED TO ENSURE THAT ONLY QUALIFIED INDIVIDUALS WILL BE ASSIGNED TO PERFORM SPECIFIC MAINTENANCE TASKS. A METHOD TO VERIFY THAT THE POWER RANGE NUCLEAR INSTRUMENTATION IS PROPERLY CALIBRATED WILL BE INCORPORATED IN THE APPROPRIATE PROCEDURES.

[142] SOUTH TEXAS 2 DOCKET 50-499 LER 89-026
REACTOR TRIP DUE TO A DROPPED CONTROL ROD.
EVENT DATE: 101389 REPORT DATE: 111389 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215928) ON OCTOBER 13, 1989, UNIT 2 WAS IN MODE 1 AT 100% POWER. AT 1745 HOURS, A REACTOR TRIP OCCURRED DUE TO THE DETECTION OF HIGH NEUTRON FLUX NEGATIVE RATE ON 2 OF 4 POWER RANGE NEUTRON MONITORING CHANNELS. THE PLANT WAS BROUGHT TO A STABLE CONDITION IN MODE 3 WITH NO UNEXPECTED POST-TRIP TRANSIENTS. THE CAUSE OF THE EVENT IS BELIEVED TO BE AN INTERMITTENT HIGH RESISTANCE CONNECTION ON A STATIONARY GRIPPER DIODE IN THE ROD CONTROL SYSTEM WHICH CAUSED ROD F-8 IN CONTROL BANK A TO DROP. THE DIODE HAS BEEN REPLACED. THE REMAINING STATIONARY GRIPPER DIODES ON BOTH UNITS WILL BE INSPECTED DURING THE NEXT SCHEDULED MAINTENANCE OUTAGE ON EACH UNIT.

[143] SUSQUEHANNA 1 DOCKET 50-387 LER 89-002 REV 01
UPDATE ON OPERATOR ERROR CAUSED FEEDWATER FLOW TRANSIENT AND REACTOR SCRAM.
EVENT DATE: 011289 REPORT DATE: 111389 NSSS: GE TYPE: BWR

(NSIC 215911) AT 0415 HOURS ON 1/12/89, WITH UNIT 1 OPERATING AT APPROXIMATELY 20% POWER, A REACTOR SCRAM OCCURRED DUE TO ACTUATION OF THE REACTOR PROTECTION SYSTEM (RPS). OPERATIONS WAS IN THE PROCESS OF TRANSFERRING FROM STARTUP LEVEL CONTROL TO AUTO FEEDWATER LEVEL CONTROL WHEN CONTROL OF LEVEL WAS LOST DUE TO A RAPID INCREASE IN FEEDWATER FLOW RATE. THE REACTOR LEVEL REACHED THE +54" LEVEL WHICH RESULTS IN A TRIP OF THE MAIN TURBINE. THE LARGE COLD WATER ADDITION CAUSED REACTOR POWER TO INCREASE PAST 24%, WHICH RESULTED IN THE RPS ACTUATION UPON TURBINE TRIP. THE REQUIRED PLANT EQUIPMENT RESPONSE DURING THE TRANSIENT WAS PER DESIGN. THE CAUSE OF THE EVENT WAS ATTRIBUTED TO COGNITIVE OPERATOR ERROR. A COOLDOWN OF 101F WAS EXPERIENCED OVER THE FIRST HOUR FOLLOWING THE SCRAM. THIS EXCEEDED THE TECH SPEC MAXIMUM COOLDOWN RATE OF 100F PER HOUR DURING A ONE HOUR PERIOD, BUT WAS NOT IMMEDIATELY IDENTIFIED. AS A RESULT, TECH SPEC ACTION REQUIREMENTS WERE NOT PROPERLY IMPLEMENTED. AN ENGINEERING ANALYSIS CONCLUDED THAT NO ADVERSE EFFECTS ON THE REACTOR COOLANT SYSTEM STRUCTURAL INTEGRITY OCCURRED AS A RESULT OF THE TEMPERATURE DEVIATION. TRAINING WAS CONDUCTED FOR ALL LICENSED OPERATORS PRIOR TO ASSUMING SHIFT DUTIES BEFORE THE NEXT STARTUP. ENHANCEMENTS FOR CLARITY PURPOSES WERE MADE TO THE OPERATING PROCEDURE.

[144] SUSQUEHANNA 1 DOCKET 50-387 LER 89-024
DIESEL GENERATOR 'C' CRANKCASE EXPLOSION DUE TO HEAVY SCORING OF THE NO. 5 RIGHT PISTON AND CYLINDER LINER.
EVENT DATE: 100789 REPORT DATE: 110389 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)

(NSIC 215910) AT 2303 HOURS ON 10/7/89, WITH UNIT 1 OPERATING AT 98% POWER AND UNIT 2 IN REFUELING, A CRANKCASE EXPLOSION OCCURRED ON THE 'C' EMERGENCY DIESEL GENERATOR (D/G) DURING ITS 18-MONTH 24 HOUR RUN SURVEILLANCE TEST. THE D/G WAS SHUT DOWN FROM THE CONTROL ROOM, DECLARED INOPERABLE AND AN LCO WAS ENTERED IN ACCORDANCE WITH TECH SPEC 3.8.1.1. THE 'E' D/G WAS SUBSTITUTED FOR THE 'C' D/G AND THE LCO WAS CLEARED ON 10/8/89. FAILURE OF A SINGLE D/G IS NOT REPORTABLE

PER 10CFR50.72. HOWEVER, BECAUSE THIS WAS THE SECOND D/G TO EXPERIENCE A CRANKCASE EXPLOSION IN THREE WEEKS (THE 'B' D/G EXPERIENCED A CRANKCASE EXPLOSION ON 9/16/89), IT WAS JUDGED PRUDENT TO NOTIFY THE NRC PER 10CFR50.72(B)(2)(III) IN THE EVENT THAT A COMMON MODE OF DIESEL FAILURE COULD EXIST. INVESTIGATIONS TO DATE AND TECHNICAL REVIEWS SHOW NO COMMON ROOT CAUSE BETWEEN THE TWO RECENT FAILURES OR WITH THREE PREVIOUS CRANKCASE EXPLOSIONS, NOR ANY REASON TO BELIEVE THAT THE D/G'S CANNOT PERFORM THEIR DESIGN SAFETY FUNCTIONS. THE 'C' D/G CRANKCASE EXPLOSION WAS ATTRIBUTED TO HEAVY SCORING OF THE NO. 5 RIGHT PISTON AND CYLINDER LINER, WHICH GENERATED THE HEAT NECESSARY FOR THE EXPLOSION. REPAIRS WERE MADE, POST MAINTENANCE AND SURVEILLANCE TESTING WAS SUCCESSFULLY COMPLETED AND THE 'C' D/G WAS DECLARED OPERABLE ON 10/23/89.

[145] SUSQUEHANNA 2 DOCKET 50-388 LER 89-005 REV 01
UPDATE ON INADVERTENT CROSS-TIE OF REACTOR BUILDING HVAC ZONES I AND III.
EVENT DATE: 052089 REPORT DATE: 111789 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 1 (BWR)

(NSIC 215935) ON 5/20/89, WITH UNIT 2 OPERATING AT 100% POWER AND UNIT 1 IN THE REFUELING CONDITION, IT WAS DISCOVERED THAT THE REACTOR BUILDING HEATING, VENTILATING AND AIR CONDITIONING (HVAC) SYSTEM ZONES I AND III HAD BEEN CROSS-TIED FROM 5/11/89 THROUGH 5/20/89. SEVERAL ITEMS CONTRIBUTED TO THIS INCIDENT: RAILROAD ACCESS BAY DESIGN WHICH NECESSITATES COMPLEX ADMINISTRATIVE CONTROLS; THE LACK OF DETAILS AND ACCURACY IN THE EQUIPMENT RELEASE FORM (ERF) PREPARED FOR THIS EVOLUTION; AND COGNITIVE PERSONNEL ERROR ON THE PART OF THE UNIT SUPERVISOR IN FAILING TO IDENTIFY THE ERF'S DEFICIENCIES AND THE IMPACT ON UNIT 2 SECONDARY CONTAINMENT INTEGRITY. THE OPERATIONS SECTION COMPLETED TRAINING FOR ALL LICENSED OPERATORS, EMPHASIZING THE IMPORTANCE OF THOROUGH ERF REVIEW. A STANDARDIZED ERF FOR ALL RAILROAD ACCESS BAY EVOLUTIONS INVOLVING THE REMOVAL OF WALLS/FLOOR PLUGS WAS DEVELOPED AND INCORPORATED INTO MAINTENANCE AND CONSTRUCTION PLANNING GUIDES TO ENHANCE THE WORK GROUP/OPERATIONS INTERFACE FOR THESE EVOLUTIONS. TRAINING FOR MAINTENANCE AND CONSTRUCTION PLANNERS WAS COMPLETED CONCERNING THESE ENHANCEMENTS. PP&L IS EVALUATING THE FEASIBILITY OF PLANT MODIFICATIONS FOR ALERTING OPERATORS IF THE SUBJECT DAMPERS ARE MISPOSITIONED.

[146] SUSQUEHANNA 2 DOCKET 50-388 LER 89-011
INSTRUMENT AIR LEAK DUE TO AN IMPROPERLY MADEUP FITTING RESULTS IN INITIATION OF A FULL RPS ACTUATION.
EVENT DATE: 100889 REPORT DATE: 110789 NSSS: GE TYPE: BWR

(NSIC 215912) AT 0741 ON 10/8/89, WITH THE UNIT IN A REFUELING OUTAGE AND ALL FUEL OFF LOADED, A FULL REACTOR PROTECTION SYSTEM (RPS) ACTUATION WAS AUTOMATICALLY INITIATED IN RESPONSE TO A HIGH WATER LEVEL IN THE SCRAM DISCHARGE VOLUME (SDV). NO CONTROL ROD MOTION OCCURRED SINCE ALL RODS WERE FULLY INSERTED AT THE TIME OF THE EVENT. A LOW PRESSURE CONDITION IN THE CONTROL ROD DRIVE (CRD) SYSTEM AIR SUPPLY HEADER, WHICH OCCURRED WHEN A SECTION OF AIR TUBING PULLED OUT OF ITS COMPRESSION FITTING, CAUSED SEVERAL CRD SYSTEM AIR OPERATED VALVES TO CYCLE, PER DESIGN, WHICH RESULTED IN A VALVE CONFIGURATION THAT CAUSED THE LEVEL IN THE SDV TO INCREASE. THE RPS ACTUATION OCCURRED WHEN THE WATER LEVEL IN THE SDV REACHED THE HIGH LEVEL TRIP SETPOINT. FURTHER INVESTIGATION DETERMINED THAT THE TUBING HAD NOT BEEN PROPERLY INSTALLED IN THE COMPRESSION FITTING. THE TUBING TO FITTING JOINT WAS PROPERLY REWORKED AND CHECKED FOR AIR LEAKAGE. ALL FITTINGS ASSOCIATED WITH A PARTICULAR PREVENTATIVE MAINTENANCE ACTIVITY (PM), PERFORMED IN JULY 1989 AND SUSPECTED TO HAVE BEEN A CONTRIBUTING FACTOR IN THIS EVENT, WERE ALSO CHECKED FOR AIR LEAKS. ALTHOUGH THIS EVENT WAS CONCLUDED TO BE AN ISOLATED CASE, AN ADDITIONAL STEP TO LEAK CHECK APPROPRIATE AIR FITTINGS IS BEING ADDED TO THE POST MAINTENANCE TESTING SECTION OF THE PROCEDURE WHICH GOVERNS THE PM ACTIVITY MENTIONED ABOVE.

[147] SUSQUEHANNA 2 DOCKET 50-388 LER 89-012
RPS ACTUATION RECEIVED DURING RESTORATION OF DIV. I +/- 24 VDC BATTERY SYSTEM.
EVENT DATE: 101589 REPORT DATE: 111389 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215915) ON 10/15/89 AT 0534 HOURS WITH UNIT 2 IN CONDITION 5, REFUELING, AN UNPLANNED ENGINEERED SAFETY FEATURE ACTUATION OCCURRED WHEN A FULL REACTOR PROTECTION SYSTEM (RPS) ACTUATION WAS RECEIVED DURING RESTORATION OF THE DIVISION I +/- 24 VDC BATTERY SYSTEM FOLLOWING SYSTEM MAINTENANCE. WHEN THE POSITIVE BATTERY CHARGER WAS PLACED IN THE EQUALIZE MODE THE OVERVOLTAGE RELAY TRIPPED, SENDING A TRIP SIGNAL TO THE BATTERY CHARGER'S OUTPUT BREAKER. THE OUTPUT BREAKER OPENED AND SINCE THE BATTERY WAS NOT CONNECTED, THE 24 VDC POSITIVE BUS DEENERGIZED. THIS IN TURN CAUSED A DIVISION I INTERMEDIATE RANGE MONITOR UPSCALE/INOP TRIP SIGNAL WHICH ACTUATED A NUCLEAR MONITORING SYSTEM TRIP OF THE RPS. WITH THE SHORTING LINKS REMOVED, THIS SINGLE CHANNEL TRIP RESULTS IN A FULL RPS ACTUATION. THE CAUSE OF THE EVENT IS ATTRIBUTED TO THE TYPE OF RELAYS USED IN THE +/- 24 VDC OVERVOLTAGE TRIP CIRCUIT. THE RELAYS USED ARE GE NGV RELAYS SET AT THE UPPER END OF THEIR RANGE. WITH THIS TYPE OF RELAY THERE IS AN OVERLAP BETWEEN THE RELAY OPERATING RANGE AND THE BATTERY CHARGER EQUALIZING VOLTAGE. THIS TYPE OF RELAY STILL PROVIDES EQUIPMENT OPERABILITY PROTECTION IN THE EVENT OF AN ACTUAL OVERVOLTAGE CONDITION, HOWEVER NUISANCE BATTERY CHARGER TRIPS, WHEN IN THE EQUALIZE MODE, ARE POSSIBLE.

[148] TROJAN DOCKET 50-344 LER 89-016 REV 01
UPDATE ON INADEQUATE MANAGEMENT OVERSITE, PROCEDURE DEFICIENCIES AND PERSONNEL ERRORS RESULT IN POWER OPERATION WITH AN INOPERABLE RECIRCULATION SUMP.
EVENT DATE: 071789 REPORT DATE: 102789 NSSS: WE TYPE: PWR

(NSIC 215778) ON JULY 17, 1989 THE PLANT WAS IN MODE 5 (COLD SHUTDOWN) WITH REACTOR COOLANT SYSTEM (RCS) CONDITIONS OF 360 PSIG AND 185 DEGREES F WHEN AN INSPECTION OF THE CONTAINMENT RECIRCULATION SUMP DETERMINED THAT THE WIRE MESH SCREEN ON THE TOP OF THE CONTAINMENT RECIRCULATION SUMP TRASH RACK WAS NOT INSTALLED. THIS INSPECTION WAS BEING CONDUCTED DUE TO SIGNIFICANT AMOUNTS OF DEBRIS DISCOVERED WITHIN THE SCREENED AREA OF THE CONTAINMENT RECIRCULATION SUMP. THE SCREEN IS PART OF THE DESIGN OF THE CONTAINMENT RECIRCULATION SUMP AND THEREFORE THE PLANT WAS OUTSIDE ITS DESIGN BASIS. THE REASON FOR THE MESH SCREEN NOT BEING INSTALLED WAS APPARENTLY A FAILURE TO COMPLETE ALL CONSTRUCTION ACTIVITIES. FAILURE TO DETECT THIS CONDITION EARLIER THAN 1989 IS ATTRIBUTED TO INADEQUATE INSPECTIONS OF THE CONTAINMENT RECIRCULATION SUMP, AS A RESULT OF INEFFECTIVE MANAGEMENT AND SUPERVISORY OVERSIGHT OF CONTAINMENT RECIRCULATION SUMP MAINTENANCE, SURVEILLANCE, ENGINEERING AND QUALITY ACTIVITIES . CORRECTIVE ACTIONS WERE TO INSTALL THE MISSING SCREEN, REPAIR DAMAGED PORTIONS OF THE EXISTING SCREEN, REMOVE THE FOREIGN MATERIALS FROM THE AREA, REVISE THE INSPECTION PROCEDURE, AND INSPECT CONTAINMENT USING THE REVISED PROCEDURE. A NUCLEAR DIVISION IMPROVEMENT PLAN HAS BEEN DEVELOPED TO IMPLEMENT ACTIONS.

[149] TROJAN DOCKET 50-344 LER 89-028
PERSONNEL ERROR IN PREPARING PROCEDURE RESULTS IN MISSED ROD POSITION SURVEILLANCE.
EVENT DATE: 080189 REPORT DATE: 111689 NSSS: WE TYPE: PWR

(NSIC 215946) DURING A REVIEW OF PERIODIC OPERATING TEST 24-1, "SHIFT OPERATING ROUTINES", DATA ON 10/17/89 IT WAS DISCOVERED THAT THE SURVEILLANCE REQUIRED BY TECH SPEC 4.1.3.2, "POSITION INDICATION CHANNELS SURVEILLANCE REQUIREMENTS", WAS NOT PERFORMED AT FOUR HOUR INTERVALS FOR THE SHUTDOWN BANKS' WHEN THE PLANT COMPUTER ROD POSITION DEVIATION MONITOR ALARM WAS INOPERABLE FROM 8/1 TO 8/4/89. SHUTDOWN BANKS' POSITIONS WERE RECORDED AT THE SPECIFIED 12 HOUR INTERVAL. THE ROD POSITION DEVIATION MONITOR ALARM WAS INOPERABLE AS THE RESULT OF NOT RESTORING A COMPUTER VARIABLE TO THE PROPER VALUE FOLLOWING A COMPUTER OUTAGE. THE FORM USED TO RECORD THE ROD POSITION DATA ONLY LISTED THE CONTROL BANKS DUE TO A PERSONNEL ERROR IN NOT PROVIDING SPACE FOR RECORDING THE SHUTDOWN BANKS POSITION. THE PROCEDURE WAS DEVIATED TO INCLUDE THE SHUTDOWN BANKS AND A TRAINING INFORMATION BULLETIN WAS ISSUED ON THE INCREASED SURVEILLANCE WHEN THE ROD DEVIATION ALARM IS NOT OPERABLE. A CHANGE WAS ALSO MADE TO THE PLANT COMPUTER SOFTWARE TO RESTORE THE COMPUTER VARIABLE TO THE CORRECT VALUE ON A COMPUTER RESTART. THE REQUIRED TECH SPEC 4.1.3.2 TWELVE HOUR SURVEILLANCE OF THE SHUTDOWN BANKS WAS PERFORMED SATISFACTORILY DURING THE AUGUST 1 TO AUGUST 4, 1989 TIME PERIOD. AS NO POSITION DEVIATIONS BEYOND THE ALLOWABLE TEN STEPS OCCURRED

DURING THIS TIME PERIOD THIS EVENT DID NOT HAVE ANY EFFECT ON PUBLIC HEALTH AND SAFETY.

[150] TURKEY POINT 3 DOCKET 50-250 LER 88-015 REV 01
UPDATE ON POST ACCIDENT HYDROGEN MONITOR SYSTEM DEFICIENCIES DUE TO PROCEDURE AND ADMINISTRATIVE CONTROL WEAKNESS.
EVENT DATE: 072988 REPORT DATE: 110989 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)
VENDOR: COMSIP DELPHI INC.

(NSIC 215802) ON JULY 29, 1988, TURKEY POINT UNITS 3 AND 4 WERE OPERATING AT 100% POWER WHEN A QUALITY ASSURANCE AUDIT IDENTIFIED THAT CONTRARY TO THE REQUIREMENTS OF TECHNICAL SPECIFICATION (TS) SURVEILLANCE REQUIREMENT 4.18, PLANT SURVEILLANCES DID NOT VERIFY THREE VALVES ON EACH UNIT TO BE IN THE REQUIRED POSITION EVEN THOUGH THEY ARE ACCESSIBLE. THESE VALVES ARE IN THE FLOW PATH BETWEEN THE HYDROGEN MONITORS AND THE CONTAINMENT BUILDING. ALSO IDENTIFIED WAS A LACK OF DIRECTION TO OPEN A VALVE IN EACH UNIT TO INITIATE B TRAIN OPERATION. THE OMISSION OF TWO OF THE THREE VALVES FROM THE FLOWPATH VERIFICATION WAS DUE TO PROCEDURAL WEAKNESS IN THE REVIEW OF PLANT CHANGES/MODIFICATIONS. THE THIRD VALVE IN EACH UNIT WAS NOT VERIFIED DUE TO INADEQUATE ADMINISTRATIVE CONTROLS. THESE VALVES WERE INSTALLED TO FACILITATE WORK ASSOCIATED WITH A PLANT MODIFICATION, HOWEVER WERE NOT ADEQUATELY CONTROLLED. AN INSTRUCTION WAS DEVELOPED TO PROVIDE A STRUCTURED AND CONSISTENT REVIEW PROCESS FOR PLANT MODIFICATIONS TO ASSURE AFFECTED PROCEDURES ARE IDENTIFIED. PROCEDURES WERE REVISED TO VERIFY VALVE POSITION FOR THE PAHM VALVES IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS. ADMINISTRATIVE CONTROLS WHICH LED TO NOT VERIFYING ONE VALVE IN EACH UNIT ARE NO LONGER IN EFFECT. THE TWO B TRAIN PAHM VALVES (ONE PER UNIT) HAVE BEEN REMOVED FROM THE PAHM SYSTEM.

[151] TURKEY POINT 3 DOCKET 50-250 LER 88-018 REV 01
UPDATE ON EMERGENCY CORE COOLING SYSTEM PUMP SUCTION ISOLATION DUE TO PROCEDURE INADEQUACY.
EVENT DATE: 082288 REPORT DATE: 111389 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)

(NSIC 215803) ON AUGUST 22, 1988, TURKEY POINT UNITS 3 AND 4 WERE OPERATING IN MODE 1 AT 100 PERCENT POWER WHEN IT WAS DETERMINED THAT DURING THE PERFORMANCE OF THE MONTHLY EMERGENCY CORE COOLING SYSTEM (ECCS) VALVE CYCLING SURVEILLANCE, THE REFUELING WATER STORAGE TANK (RWST) SUPPLY TO THE ECCS WAS ISOLATED FROM BOTH TRAINS OF THE ECCS INCLUDING RESIDUAL HEAT REMOVAL (RHR), CONTAINMENT SPRAY (CS) AND HIGH PRESSURE SAFETY INJECTION (HPSI). ALSO DURING THE SURVEILLANCE, BOTH TRAINS OF RHR WERE ISOLATED FROM THE RWST DURING CYCLING OF TWO OTHER VALVES. THE ROOT CAUSE WAS DETERMINED TO BE PROCEDURAL INADEQUACY IN THAT TRANSLATION OF THE STANDARD TECHNICAL SPECIFICATION REQUIREMENTS INTO 0-ADM-021, "TECHNICAL SPECIFICATION IMPLEMENTATION PROCEDURE," DID NOT ADEQUATELY REFLECT THE TURKEY POINT DESIGN. SOME WESTINGHOUSE PLANTS HAVE PARALLEL RWST VALVES UNLIKE THE TURKEY POINT UNITS WHICH HAVE VALVES IN SERIES. CORRECTIVE ACTIONS INCLUDED REVISING THE SURVEILLANCE PROCEDURE AND 0-ADM-021.

[152] TURKEY POINT 3 DOCKET 50-250 LER 89-013
BORIC ACID TRANSFER PUMP NOT DECLARED OUT OF SERVICE PRIOR TO REFILLING THE SEAL POT RESULTING IN NO FLOW PATH FROM A BORIC ACID TANK TO THE REACTOR COOLANT SYSTEM.
EVENT DATE: 091089 REPORT DATE: 100989 NSSS: WE TYPE: PWR
VENDOR: DURAMETALLIC CORP.
GOULDS PUMPS INC.

(NSIC 215531) ON SEPTEMBER 10, 1989, AT 1345, UNIT 3 ENTERED TECHNICAL SPECIFICATION 3.0.1 FOR 19 MINUTES WHEN THE 3A BORIC ACID TRANSFER PUMP (BATP) WAS DECLARED OUT OF SERVICE. AT THE TIME OF THE EVENT, THE 3A BATP WAS THE ONLY BATP ALIGNED TO TAKE SUCTION FROM THE A BORIC ACID TANK (BAT) AND DISCHARGE TO THE UNIT 3 CHARGING PUMPS. AT 1345 WHILE REFILLING THE 3A BATP SEAL POT, MAINTENANCE PERSONNEL ACCIDENTALLY DAMAGED THE NITROGEN PRESSURE INDICATOR.

SINCE NITROGEN PRESSURE ON THE SEAL POT COULD NOT BE MONITORED, THE 3A BATP WAS DECLARED OUT OF SERVICE. THIS RESULTED IN LOSS OF A FLOW PATH FROM THE A BAT TO THE UNIT 3 REACTOR COOLANT SYSTEM. THE 3B BATP WAS ALIGNED TO TAKE SUCTION FROM THE A BAT AND DISCHARGE TO THE UNIT 3 CHARGING PUMPS. UNIT 3 EXITED TECHNICAL SPECIFICATION 3.0.1 AT 1404 ON SEPTEMBER 10, 1989. FURTHER REVIEW REVEALED THAT THE BATPS ARE TECHNICALLY "INOPERABLE" WHEN THE NITROGEN PRESSURE INDICATORS ARE REMOVED TO REFILL THE SEAL POT. OPERATIONS PERSONNEL WERE NOT AWARE THAT THE BATPS ARE CONSIDERED INOPERABLE WHEN THE SEAL POT NITROGEN PRESSURE INDICATOR IS REMOVED; THEREFORE, THE BATPS WERE NOT DECLARED OUT OF SERVICE PRIOR TO REFILLING THE SEAL POT. AN ENTRY HAS BEEN MADE IN THE OPERATIONS NIGHT ORDER BOOK TO CONVEY THIS OPERABILITY CONCERN TO CONTROL ROOM OPERATORS.

[153] TURKEY POINT 3 DOCKET 50-250 LER 89-015
CHARGING PUMPS DECLARED INOPERABLE DUE TO INADEQUATE TROUBLESHOOTING TECHNIQUES USED TO DETERMINE THE CAUSE FOR REDUCED CHARGING FLOW TO THE REACTOR COOLANT SYSTEM.
EVENT DATE: 101989 REPORT DATE: 111689 NSSS: WE TYPE: PWR
VENDOR: UNION PUMP COMPANY

(NSIC 215821) ON 10/19/89, AT 1030, WITH UNIT 3 IN MODE 1 AT 45% POWER, ALL THREE CHARGING PUMPS WERE DECLARED OUT OF SERVICE (OOS). AN EVALUATION OF REDUCED CHARGING FLOW TO THE REACTOR COOLANT SYSTEM (RCS) REVEALED THAT NONE OF THE PUMPS SINGULARLY COULD MAINTAIN PROPER PRESSURIZER LEVEL WITH A 45 GPM LETDOWN ORIFICE IN SERVICE. ONLY TWO CHARGING PUMPS ARE REQUIRED TO BE OPERABLE FOR UNIT OPERATION. TECHNICAL SPECIFICATION (TS) 3.6.D.1 ALLOWS 1 OF 2 OPERABLE CHARGING PUMPS TO BE OOS FOR 24 HOURS. FAILURE TO MEET TS 3.6.D.1 LIMITING CONDITION FOR OPERATION (LCO) PLACED UNIT 3 IN TS 3.0.1 WHICH REQUIRES THE UNIT TO BE IN HOT STANDBY WITHIN 7 HOURS. INADEQUATE TROUBLESHOOTING TECHNIQUES USED TO INVESTIGATE THE REDUCED CHARGING PUMP FLOW CONDITION LED TO THE INCORRECT DETERMINATION THAT THE 3A AND 3C CHARGING PUMPS WERE INOPERABLE. THE 3B CHARGING PUMP INTERNAL DISCHARGE VALVES AND VALVE GUIDES WERE FOUND TO BE WORN AND THE CENTER SUCTION VALVE GUIDE WAS FOUND TO BE "BACKED OUT." THIS ALLOWED FLOW FROM THE 3A AND 3C CHARGING PUMPS TO BE RECIRCULATED THROUGH THE IDLE 3B CHARGING PUMP. THIS ALSO AFFECTED PERFORMANCE OF THE 3B CHARGING PUMP. UPON ISOLATION OF THE 3B CHARGING PUMP FOR VENTING, CHARGING FLOW INCREASED. THE 3A AND 3C CHARGING PUMPS WERE DECLARED OPERABLE.

[154] TURKEY POINT 4 DOCKET 50-251 LER 89-013
MISSED SURVEILLANCE ON INTAKE COOLING WATER ISOLATION VALVES TO THE TURBINE PLANT COOLING WATER HEAT EXCHANGERS DUE TO PERSONNEL ERROR.
EVENT DATE: 102089 REPORT DATE: 111489 NSSS: WE TYPE: PWR

(NSIC 215822) ON OCTOBER 20, 1989, WITH UNIT 4 IN MODE 3 (HOT STANDBY), FPL QUALITY ASSURANCE PERSONNEL DISCOVERED THAT QUARTERLY INSERVICE TESTING OF INTAKE COOLING WATER (ICW) ISOLATION VALVES POV-4-4882 AND POV-4-4883 TO THE TURBINE PLANT COOLING WATER (TPCW) HEAT EXCHANGERS HAD NOT BEEN PERFORMED. PLANT CHANGE/MODIFICATION (PC/M) 88-346, WHICH WAS IMPLEMENTED AND TURNED OVER TO OPERATIONS ON APRIL 30, 1989, IMPOSED QUARTERLY IN SERVICE TESTING REQUIREMENTS ON THESE VALVES. FAILURE TO PERFORM THE QUARTERLY INSERVICE TEST IS IN VIOLATION OF TECHNICAL SPECIFICATION 4.0.3 WHICH ADDRESSES SURVEILLANCE REQUIREMENTS FOR INSERVICE INSPECTION OF ASME CODE CLASS 1, 2, AND 3 COMPONENTS. THE CAUSE OF THE MISSED SURVEILLANCE WAS COGNITIVE ERRORS BY CONTRACT PERSONNEL. FAILURE TO IDENTIFY AND DOCUMENT THE PROCEDURES REQUIRING REVISION AFTER TURNOVER OF PC/M 88-346 AND FAILURE TO ELEVATE UNRESOLVED PROCEDURE REVISION COMMENTS TO A HIGHER LEVEL OF MANAGEMENT RESULTED IN THE MISSED SURVEILLANCE. THE VALVES WERE TESTED AND DETERMINED TO MEET THEIR INTENDED SAFETY FUNCTION. THE VALVES ARE BEING ADDED TO OPERATING PROCEDURE (OP) 0209.1, "VALVE EXERCISING PROCEDURE." THE INDIVIDUAL RESPONSIBLE FOR THIS CONDITION HAS BEEN COUNSELLED.

[155] VOGTLE 1 DOCKET 50-424 LER 88-047
ERROR IN PROCEDURE LEADS TO TECH SPEC 3.0.3 ENTRY.
EVENT DATE: 061488 REPORT DATE: 111089 NSSS: WE TYPE: PWR

(NSIC 215808) ON 6-14-88, BETWEEN APPROXIMATELY 1400 AND 1430 CDT, WHILE THE UNIT WAS AT 100% POWER, HANDSWITCHES FOR MANUAL ACTUATION OF CONTAINMENT ISOLATION - PHASE A AND CONTAINMENT VENTILATION ISOLATION WERE TESTED. EACH HANDSWITCH WAS TAKEN OUT OF SERVICE, TESTED, AND RETURNED TO SERVICE. ON 10-13-89, WHILE PREPARING TO PERFORM THE TEST, A SYSTEM ENGINEER IDENTIFIED AN ERROR IN THE PROCEDURE WHICH RESULTED IN SIMULTANEOUSLY DISABLING BOTH HANDSWITCHES. THIS CONDITION IS IN CONFLICT WITH THE REQUIREMENTS OF TECHNICAL SPECIFICATION (TS) TABLE 3.3-2 WHICH REQUIRES BOTH HANDSWITCHES TO BE OPERABLE IN MODES 1, 2, 3, AND 4 (AND AT SPECIFIC TIMES IN MODE 6). ALTHOUGH DURING THE PREVIOUS TEST, LCO ENTRIES HAD BEEN MADE FOR EACH HANDSWITCH BEING OUT OF SERVICE, IT WAS NOT RECOGNIZED THAT BOTH HANDSWITCHES WERE OUT OF SERVICE AT THE SAME TIME. THIS CONDITION RESULTS IN ENTRY INTO TS 3.0.3. THE PROCEDURE ERROR WHICH ALLOWED SIMULTANEOUS DISABLING OF THE HANDSWITCHES HAS BEEN CORRECTED.

[156] VOGTLE 2 DOCKET 50-425 LER 89-027
REACTOR TRIP ON HIGH FLUX RATE DUE TO ROD DROP.
EVENT DATE: 101189 REPORT DATE: 110389 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215919) ON 10/11/89, AT 2333 CDT, AN AUTOMATIC REACTOR TRIP OCCURRED WITH THE REACTOR IN STABLE OPERATION AT 58% OF RATED THERMAL POWER. ALL AUTOMATIC SAFETY FEATURES FUNCTIONED AS REQUIRED AND THE REACTOR WAS STABILIZED IN MODE 3 WITHOUT INCIDENT. NO ANNUNCIATOR OR OTHER WARNING OF A PROBLEM PRECEDED THE REACTOR TRIP. FOLLOWING A REVIEW OF COMPUTER PRINTOUTS OF DATA ASSOCIATED WITH THE TRIP, THE FIRST OUT ANNUNCIATOR WAS IDENTIFIED AS A HIGH FLUX RATE TRIP ANNUNCIATOR. OPERABILITY TESTING OF THE CONTROL RODS THEN INDICATED THAT A PROBLEM EXISTED WITH ROD K-2 IN CONTROL BANK B. INVESTIGATION OF THE CONTROL ROD CIRCUITRY IDENTIFIED A FAILED DIODE WHICH HAD APPARENTLY RESULTED IN A LOSS OF CURRENT TO THE STATIONARY GRIPPER COIL. THIS ALLOWED THE ROD TO DROP INTO THE CORE AND INITIATE A NEGATIVE FLUX RATE TRIP. CORRECTIVE ACTION INCLUDED REPLACING THE DIODE FOR ROD K-2.

[157] VOGTLE 2 DOCKET 50-425 LER 89-028
ARCING POWER CABLE LEADS TO CONTAINMENT VENTILATION ISOLATION.
EVENT DATE: 101689 REPORT DATE: 111489 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215925) ON 10/16/89, A TECHNICIAN WAS PREPARING TO REPLACE A FAULTY CIRCUIT BOARD IN A CONTAINMENT VENT EFFLUENT RADIATION MONITOR PANEL. WHILE PERFORMING THIS WORK, HE CONTACTED A POWER CABLE AND ARCING OCCURRED AT THE TERMINAL CONNECTION. THE ARCING RESULTED IN POWER FLUCTUATIONS AT THE INPUT/OUTPUT CIRCUIT BOARD WHICH SUBSEQUENTLY FAILED. THIS LED TO A CONTAINMENT VENTILATION ISOLATION (CVI) ACTUATION AT 0825 CDT. CONTROL ROOM OPERATORS VERIFIED THAT NO ABNORMAL RADIATION CONDITION EXISTED IN THE CONTAINMENT BUILDING ATMOSPHERE AND RESET THE APPROPRIATE VALVES AND DAMPERS AND THE CVI SIGNAL. THE CAUSE OF THIS EVENT WAS IN INADEQUATE DESIGN. THE SCREW ON THE RADIATION MONITOR TERMINAL BLOCK WAS TOO SHORT TO ADEQUATELY ENGAGE THE THREADED OPENING AND PROVIDE A TIGHT, PERMANENT CONNECTION WITH THE ATTACHED POWER CABLE. WHEN THE TECHNICIAN'S HAND CONTACTED THE CABLE, THE CONNECTION WAS LOOSENED AND ARCING OCCURRED. THIS SCREW AND A SIMILAR SCREW IN UNIT 1 HAVE BEEN REPLACED.

[158] WATERFORD 3 DOCKET 50-382 LER 89-002 REV 01
UPDATE ON INADEQUATE QUALIFICATION OF INSTRUMENT AIR TUBING DURING INITIAL CONSTRUCTION.
EVENT DATE: 121884 REPORT DATE: 111389 NSSS: CE TYPE: PWR

(NSIC 215905) AT 1954 HOURS ON JANUARY 25, 1989, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 82% POWER WHEN DESIGN ENGINEERING (DE) PERSONNEL DISCOVERED A DISCREPANCY WITH THE SAFETY CLASSIFICATION OF INSTRUMENT AIR (IA) TUBING THAT SUPPLIES THE ACTUATORS OF SAFETY INJECTION SUMP OUTLET ISOLATION VALVES SI 602A&B. TUBING SUPPORTS WERE INSTALLED SEISMICALLY QUALIFIED, BUT THE TUBING AS SHOWN IN DESIGN DOCUMENTS WAS INSTALLED AS NON-NUCLEAR SAFETY (NNS) DURING INITIAL CONSTRUCTION. BECAUSE SI 602A&B ARE REQUIRED TO OPEN AFTER A LOCA

COINCIDENT WITH A SAFE SHUTDOWN EARTHQUAKE (SSE), THIS EVENT IS REPORTABLE AS A CONDITION OUTSIDE THE PLANT'S DESIGN BASIS. THE ROOT CAUSE OF THIS EVENT WAS THE INSUFFICIENT DOCUMENTATION OF CODE INTERPRETATIONS DURING INITIAL CONSTRUCTION. AN INVESTIGATION REVEALED THAT SAFETY CLASS TUBING AND WELD FILLER MATERIAL WERE USED. THE TUBING WELDS PASSED PENETRANT TESTS, AND RECORDS WERE REVIEWED TO ENSURE THE WELDERS WERE QUALIFIED TO PERFORM SAFETY RELATED WELDING. THE TUBING IS CONSIDERED TO BE CLASSIFIED AS SAFETY CLASS 3 AND WOULD PERFORM ITS DESIGN FUNCTION. THUS, THIS EVENT DID NOT THREATEN THE HEALTH OR SAFETY OF THE PUBLIC OR PLANT PERSONNEL.

[159] WATERFORD 3 DOCKET 50-382 LER 89-019
LOSS OF VOLTAGE TO A SAFETY BUS DUE TO PERSONNEL ERROR.
EVENT DATE: 101289 REPORT DATE: 111389 NSSS: CE TYPE: PWR

(NSIC 215906) AT 1326 HOURS ON OCTOBER 12, 1989, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS IN MODE 6, REFUELING, WHEN AN ELECTRICIAN INADVERTENTLY OPENED THE 4.16 KV BUS 3A2 METERING POTENTIAL TRANSFORMER (PT) FUSE DRAWER INITIATING A LOSS OF VOLTAGE SIGNAL FOR THE 3A2 ELECTRICAL BUS. THE ABOVE INDICATED LOSS OF VOLTAGE CAUSED THE FEEDER BREAKER FROM BUS 3A2 AND SUPPLY BREAKER TO SAFETY BUS 3A3S TO OPEN AND DIESEL GENERATOR A LOAD SEQUENCER TO RESET. THE METERING PT FUSE DRAWER WAS CLOSED AND ALL BREAKERS RETURNED TO THE APPROPRIATE LINEUP BY 1335 HOURS. BECAUSE OF THE INADVERTENT ACTUATION OF AN ENGINEERED SAFETY FEATURE (LOSS OF VOLTAGE CIRCUIT), THIS EVENT IS REPORTABLE PER 10 CFR 50.73(A)(2)(IV). THE ROOT CAUSE OF THIS EVENT IS PERSONNEL ERROR. CONTRIBUTING TO THIS EVENT WERE UNNECESSARY WORK INSTRUCTIONS AND INADEQUATE LABELING AND CONTROL OF THE METERING PT FUSE DRAWER. BECAUSE THERE WAS NO LOSS OF POWER TO THE OPERATING SHUTDOWN COOLING PUMP OR OTHER REQUIRED SAFETY RELATED EQUIPMENT, THIS EVENT DID NOT THREATEN THE HEALTH AND SAFETY OF THE PUBLIC OR PLANT PERSONNEL.

[160] ZION 1 DOCKET 50-295 LER 89-016
FUEL HANDLING SRO DID NOT MEET REQUIREMENTS OF 10CFR55.53 DUE TO LACK OF CONTROLS.
EVENT DATE: 091989 REPORT DATE: 111589 NSSS: WE TYPE: PWR

(NSIC 215851) FROM 9/19/89 TO 9/21/89, UNIT 1 WAS IN MODE 6, REFUELING, WITH CORE OFF-LOAD IN PROGRESS. ON 10/16/89, IN A MEETING WITH THE ASSISTANT SUPERINTENDENT OPERATING (ASO), A SENIOR REACTOR OPERATOR LIMITED TO FUEL HANDLING (SROL) REALIZED THAT HE DID NOT MEET THE CONDITIONS OF 10CFR55.53, WHICH CAUSED HIM TO BE IN VIOLATION OF TECH SPEC 3.13.1.A.6 DURING THE PERIOD 9/19/89 THROUGH 9/21/89. THE ROOT CAUSE OF THE EVENT IS THE LACK OF ADMINISTRATIVE CONTROL REGARDING THE ACTIVITY OF SROLS. THE EXPERIENCE LEVEL OF THE SROL INVOLVED (20 REFUELINGS) WAS SUCH THAT THE HEALTH AND SAFETY OF THE GENERAL PUBLIC WERE NOT COMPROMISED. THE ACTIONS TAKEN TO ADMINISTRATIVELY CONTROL THE SROL PROFICIENCY REQUIREMENTS INCLUDE TWO PROCEDURE CHANGES TO THE FUEL HANDLING INSTRUCTIONS TO EITHER VERIFY PROFICIENCY OR TAKE THE ACTIONS PRESCRIBED IN 10CFR55.53.

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

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