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Licensee Event Report (LER) Compilation

For month of August 1991

Oak Ridge National Laboratory

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

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Abstract

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

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

W. P. Poore
Nuclear Operations Analysis Center
Oak Ridge National Laboratory
P. O. Box 2009, Oak Ridge, TN 37831-8065
Telephone: 615/574-0325, FTS Number 624-0325

[1] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 91-005
AUTOMATIC REACTOR TRIP ON LOSS OF MAIN FEEDWATER CAUSED BY INADEQUATE PROCEDURAL GUIDANCE REGARDING CROSS-TIE OF ELECTRICAL LOAD CENTERS.
EVENT DATE: 052191 REPORT DATE: 061491 NSSS: BW TYPE: PWR

(NSIC 222334) ON MAY 21, 1991, AT APPROXIMATELY 0951, WITH THE PLANT AT 100 PERCENT OF RATED POWER, AN AUTOMATIC ANTICIPATORY REACTOR TRIP ON LOSS OF MAIN FEEDWATER OCCURRED WHEN THE CIRCUIT BREAKER SUPPLYING TWO CROSS-TIED NON-VITAL 480 VOLT LOAD CENTERS TRIPPED ON OVERLOAD AND DE-ENERGIZED THE CONTROL OIL PUMPS FOR BOTH MAIN FEEDWATER PUMPS. PLANT RESPONSE TO THE TRIP WAS AS EXPECTED. WATER LEVEL WAS MAINTAINED IN BOTH STEAM GENERATORS BY THE EMERGENCY FEEDWATER SYSTEM. THE LOAD CENTERS HAD BEEN CROSS-TIED TO PERFORM MAINTENANCE ON ONE OF THE LOAD CENTER FEEDER BREAKERS. THE LOADING FROM BOTH LOAD CENTERS EXCEEDED THE TRIP RATING OF ONE FEEDER BREAKER BECAUSE THE EVOLUTION WAS PERFORMED AT POWER WITHOUT FIRST HAVING SHED THE EXCESS LOADS. THE ROOT CAUSE OF THE EVENT WAS INADEQUATE PROCEDURAL GUIDANCE CONTAINED IN THE ELECTRICAL SYSTEM NORMAL OPERATING PROCEDURE BEING USED FOR THE CROSS-TIE EVOLUTION. THE PROCEDURE HAS BEEN REVISED TO PREVENT RECURRENCE. A CONTRIBUTING CAUSE WAS INSUFFICIENT REVIEW ASSOCIATED WITH THE PLANNING, SCHEDULING, AUTHORIZING AND PERFORMING OF AN INFREQUENT AND UNUSUAL EVOLUTION. THOSE PERSONNEL INVOLVED WITH THIS EVENT HAVE BEEN COUNSELED CONCERNING THEIR RESPONSIBILITIES.

[2] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 91-006
AUTOMATIC ACTUATION DUE TO SPURIOUS TRIPPING OF A RADIATION MONITOR CAUSED BY AN INADEQUATE SETPOINT WHICH RESULTED FROM A PERSONNEL ERROR DURING PROCEDURE DEVELOPMENT.
EVENT DATE: 053091 REPORT DATE: 062891 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 2 (PWR)

(NSIC 222422) ON MAY 30, 1991, AT APPROXIMATELY 1215, AN AUTOMATIC ACTUATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) OCCURRED AS A RESULT OF A SPURIOUS TRIP OF THE ANO-1 CONTROL ROOM AREA RADIATION MONITOR (RE-8001). AT THE TIME OF THE ACTUATION, A NEW PROCEDURE SUPPLEMENT WAS BEING PERFORMED WHICH REQUIRED THE MONITOR'S SETPOINT TO BE ADJUSTED TO TWO TIMES BACKGROUND (0.4 MILLIREM/HOUR). PREVIOUSLY, THE SETPOINT FOR RE-8001 HAD BEEN 10 MILLIREM/HOUR. SHORTLY AFTER THE SETPOINT HAD BEEN ADJUSTED TO 0.4 MILLIREM/HOUR, A RANDOM SPIKE CAUSED THE MONITOR TO TRIP, ACTUATING THE CREVS PERFORMANCE OF THE PROCEDURE WAS TERMINATED AND THE SETPOINT OF RE-8001 WAS RETURNED TO 1.0 MILLIREM/HOUR. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR DURING PROCEDURE DEVELOPMENT A VERIFICATION/VALIDATION OF THE NEW SUPPLEMENT WAS NOT PERFORMED BEFORE IMPLEMENTATION. TRAINING WAS PROVIDED TO APPROPRIATE PERSONNEL REEMPHASIZING THE NEED TO PERFORM VERIFICATION/VALIDATION OF PROCEDURE CHANGES WHICH ADD NEW TESTS OR SURVEILLANCES INTO EXISTING PROCEDURES. A PERMANENT CHANGE WAS MADE TO THE APPLICABLE PROCEDURE SUPPLEMENT TO ESTABLISH A MINIMUM TRIP SETPOINT FOR RE-8001 TO PREVENT RANDOM SPIKES FROM CAUSING CREVS ACTUATIONS.

[3] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 91-013
FIRE WATCHES RELEASED FROM INOPERABLE FIRE BARRIERS DUE TO INADEQUATE DOCUMENTATION CONCERNING POSTING REQUIREMENTS.
EVENT DATE: 051291 REPORT DATE: 060791 NSSS: CE TYPE: PWR

(NSIC 222381) ON MAY 12, 1991, THE OPERATIONS SHIFT SUPERVISOR AUTHORIZED RELEASE OF FIRE WATCHES THAT HAD BEEN POSTED AS REQUIRED BY A TECHNICAL SPECIFICATION ACTION STATEMENT AT CERTAIN INOPERABLE FIRE BARRIERS. THESE FIRE BARRIERS WERE LOCATED IN THE LUBE OIL TANK ROOM, LUBE OIL RESERVOIR ROOM, BOTH BATTERY ROOMS, AND THE CONTROL ROOM. WORK REQUIRED TO RESTORE THE BARRIERS TO AN OPERABLE STATUS HAD NOT BEEN COMPLETED. PREMATURE RELEASE OF THE FIRE WATCHES WAS CAUSED BY INADEQUATE DOCUMENTATION ON THE FIRE WATCH REQUEST FORMS CONCERNING POSTING REQUIREMENTS. DURING THE PERIOD WITHOUT FIRE WATCH COVERAGE, SMOKE OR SPRINKLER PRESSURE SWITCH ALARMS WERE AVAILABLE IN THE CONTROL ROOM FOR THE AFFECTED AREAS. FIRE WATCH POSTING WAS REESTABLISHED AFTER THE ERROR WAS DISCOVERED. THE TOTAL TIME WITHOUT FIRE WATCH COVERAGE WAS APPROXIMATELY THIRTY-FOUR HOURS. THE FIRE WATCH REQUEST FORMS FOR THE AFFECTED BARRIERS WERE REVISED TO PREVENT RECURRENCE OF THIS EVENT BY SPECIFICALLY STATING THAT THE BARRIERS HAVE BEEN INSPECTED AND

ARE INOPERABLE. APPROVAL BY FIRE PROTECTION PERSONNEL WILL BE REQUIRED BEFORE FUTURE RELEASE OF FIRE WATCHES AT THESE BARRIERS.

[4] ARNOLD DOCKET 50-331 LER 91-004
INADVERTENT INBOARD HIGH PRESSURE COOLANT INJECTION SYSTEM STEAM SUPPLY ISOLATION WHILE VENTING A HIGH STEAM FLOW DIFFERENTIAL PRESSURE SWITCH.
EVENT DATE: 050891 REPORT DATE: 060791 NSSS: GE TYPE: BWR
VENDOR: BARTON INSTRUMENT CO., DIV OF ITT

(NSIC 222337) ON MAY 8, 1991, WITH THE PLANT OPERATING AT 100% POWER, A PRIMARY CONTAINMENT ISOLATION SYSTEM HIGH PRESSURE COOLANT INJECTION SYSTEM INBOARD ISOLATION OCCURRED . THE ISOLATION OCCURRED DURING VENTING OF A HPCI TURBINE HIGH FLOW DIFFERENTIAL PRESSURE SWITCH (PDIS2244). VENTING OF PDIS2244 WAS INITIATED WHEN OPERATIONS IDENTIFIED THAT THE INSTRUMENT WAS NOT INDICATING AS EXPECTED APPROXIMATELY THREE HOURS AFTER ITS CALIBRATION. THE ROOT CAUSE OF THIS EVENT WAS A COMBINATION OF TWO FACTORS. THE DISCREPANCY IN INSTRUMENT INDICATION IS SUSPECTED TO HAVE BEEN CAUSED BY AIR BEING TRAPPED IN THE INSTRUMENT OR ITS ASSOCIATED PIPING DURING THE CALIBRATION PROCESS. THE SECOND FACTOR WAS A LACK OF GUIDANCE FOR OPERATIONS PERSONNEL WITH RESPECT TO VENTING INSTRUMENTS TO CORRECT INDICATION DISCREPANCIES. CORRECTIVE ACTIONS FOR THIS EVENT WILL INCLUDE REVIEW OF THE INSTRUMENT CALIBRATION PROCESS/PIPING CONFIGURATION AND ISSUANCE OF FORMAL GUIDANCE FOR OPERATIONS PERSONNEL REGARDING THE PROCESS TO BE FOLLOWED WHEN A DISCREPANCY IN INSTRUMENT INDICATION IS IDENTIFIED. THIS EVENT HAD NO EFFECT ON THE SAFE OPERATION OF THE PLANT. ALTHOUGH HPCI WAS TECHNICALLY INOPERABLE DURING THE THIRTEEN MINUTES THAT THE INBOARD STEAM SUPPLY ISOLATION VALVE WAS CLOSED, HPCI WAS COMPLETELY FUNCTIONAL AND HAVE BEEN PLACED IN SERVICE IMMEDIATELY IF REQUIRED.

[5] BEAVER VALLEY 1 DOCKET 50-334 LER 91-014
TECHNICAL SPECIFICATION TESTING DEFICIENCY.
EVENT DATE: 050291 REPORT DATE: 053191 NSSS: WE TYPE: PWR

(NSIC 222221) ON 5/2/91, IT WAS DISCOVERED THAT OPERATIONS SURVEILLANCE TEST (OST) 1.44C.1, CONTAINMENT PURGE AND EXHAUST ISOLATION TEST, DID NOT DIRECT THE OPERATORS TO VERIFY THAT A CONTAINMENT PURGE AND EXHAUST ISOLATION WOULD OCCUR ON MANUAL INITIATION AS REQUIRED BY TECHNICAL SPECIFICATION SURVEILLANCE 4.9.9. THIS OST IS REQUIRED TO BE PERFORMED WITHIN 150 HOURS PRIOR TO THE START OF AND AT LEAST ONCE PER SEVEN DAYS DURING CORE ALTERATIONS. A REVIEW OF THE PERFORMANCE HISTORY OF THIS OST REVEALED THAT ON 10/14/89, DURING OUR SEVENTH REFUELING OUTAGE, CORE ALTERATIONS WERE PERFORMED WITHOUT PROPER VERIFICATION OF CONTAINMENT PURGE AND EXHAUST ISOLATION ON MANUAL INITIATION. IN THE EIGHTH REFUELING OUTAGE NO CORE ALTERATIONS WERE PERFORMED WITHOUT VERIFICATION OF MANUAL INITIATION. HOWEVER, ON SEVERAL OCCASIONS DURING THE SEVENTH AND EIGHTH REFUELING OUTAGES THE SURVEILLANCE REQUIREMENTS WERE SATISFIED BY MEANS OTHER THAN THE PERFORMANCE OF OST 1.44C.1. ALTHOUGH PREVIOUS REVISIONS OF THIS PROCEDURE REQUIRED THE PROPER VERIFICATIONS, A PROCEDURE REVISION WAS PUT INTO EFFECT ON 8/16/89 WHICH INADVERTENTLY DELETED THE REQUIREMENT TO VERIFY MANUAL INITIATION. FROM 8/16/89 UNTIL 5/3/91, ONLY ONE CORE ALTERATION WAS PERFORMED WITHOUT VERIFYING MANUAL INITIATION OF THE PURGE AND EXHAUST ISOLATION. THE AUTOMATIC ISOLATION FUNCTION WAS VERIFIED OPERABLE ON ALL APPLICABLE OCCASIONS.

[6] BEAVER VALLEY 1 DOCKET 50-334 LER 91-013
PRESSURIZER CODE SAFETY VALVE LIFT SETTING LESS THAN TECHNICAL SPECIFICATION LIMIT.
EVENT DATE: 051091 REPORT DATE: 060791 NSSS: WE TYPE: PWR
VENDOR: TARGET ROCK CORP.

(NSIC 222220) ON 5/2/91, WITH THE UNIT IN REFUELING , THE "C" PRESSURIZER CODE SAFETY VALVE (RV-RC-551C) WAS TESTED BY WYLE LABORATORIES IN ACCORDANCE WITH BEAVER VALLEY TEST PROCEDURE (BVT) 1.60.05 "IST SAFETY AND RELIEF VALVE TRACKING." IN TWO OF THREE LIFT TESTS, THE VALVE ACTUATED AT PRESSURES OUTSIDE THE TECHNICAL SPECIFICATION LIFT SETPOINT OF 2485 PSIG +1/-3 PERCENT (2509.8 PSIG/2410.5 PSIG). THE "AS-FOUND" SETPOINTS WERE: 2431, 2402 AND 2373 PSIG.

AS A RESULT OF THE LIFT TEST ON RV-RC-551C, THE "A" PRESSURIZER CODE SAFETY VALVE (RV-RC-551A) WAS TESTED ON 5/10/91. IN ONE OF THE THREE LIFT TESTS, THIS VALVE ALSO ACTUATED AT A PRESSURE OUTSIDE THE LIFT SETPOINT TOLERANCE OF +1/-3 PERCENT. THE "AS-FOUND" SETPOINTS WERE: 2409, 2429 AND 2425 PSIG. THE VALVES WERE LEFT WITH WYLE LABORATORIES FOR REPAIR AND RESETTING OF VALVE LIFT PRESSURE TO 2485 PSIG +1/-1 PERCENT. THE APPARENT CAUSE FOR THE FAILURE OF THE VALVES TO MEET THE LIFT SETPOINT ON THREE SUCCESSIVE LIFTS WAS SETPOINT DRIFT. THE VALVES WERE RESET TO WITHIN +1/-1 PERCENT OF THE LIFT SETPOINT. THERE WERE NO SAFETY IMPLICATIONS AS A RESULT OF THIS EVENT. THE LIFT POINT FOR RV-RC-551C WAS FOUND WITHIN SPECIFICATION ON THE FIRST AND CONSERVATIVELY LOW ON THE SECOND AND THIRD LIFTS. THE LIFT POINT ON RV-RC-551A WAS FOUND CONSERVATIVELY LOW ON THE FIRST LIFT TEST AND THE TWO SUBSEQUENT LIFTS WERE WITHIN SPECIFICATIONS.

[7] BEAVER VALLEY 1 DOCKET 50-334 LER 91-015
CONTROL ROOM VENTILATION SYSTEM OUTSIDE AIR EXHAUST DAMPERS INADVERTENTLY OPENED.
EVENT DATE: 051791 REPORT DATE: 061791 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BEAVER VALLEY 2 (PWR)
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 222339) ON 5/17/91, AN OPERATOR PERFORMING A WALKDOWN OF THE UNIT CONTROL ROOM VENTILATION SYSTEM FOUND THE TWO OUTSIDE AIR SERIES EXHAUST DAMPERS OPEN. THESE DAMPERS HAD BEEN CLOSED DURING THE UNIT 1 REFUELING OUTAGE, BUT WERE REQUIRED TO BE MAINTAINED CLOSED TO SUPPORT CONTINUED POWER OPERATION AT UNIT 2 SINCE THE TWO UNITS SHARE A COMMON CONTROL ROOM VENTILATION ENVELOPE. THE OPERATOR NOTIFIED THE NUCLEAR SHIFT SUPERVISOR AND IMMEDIATELY CLOSED THE DAMPERS. INVESTIGATION DETERMINED THAT ONE OF THE DAMPERS HAD BEEN OPENED ON 5/16/91 WHEN IT APPARENTLY WAS INADVERTENTLY ENERGIZED MOMENTARILY AFTER ITS MOTOR CONTROL CENTER WAS CLEANED. THE DAMPERS ARE DESIGNED TO OPEN WHEN ENERGIZED WITHOUT A CLOSURE SIGNAL PRESENT. THE OTHER DAMPER WAS APPARENTLY OPENED DUE TO THE SAME CAUSE BETWEEN 5/4/91 AND 5/17/91. THE EXACT DATE FOR THIS DAMPER OPENING COULD NOT BE DETERMINED DUE TO A FAILURE OF THE COMPUTER POINT ASSOCIATED WITH ITS POSITION INDICATING LIMIT SWITCHES. AN INPO HUMAN PERFORMANCE ENHANCEMENT SYSTEM EVALUATION OF THIS EVENT WAS CONDUCTED. STATION PROCEDURES HAVE BEEN REVISED TO REQUIRE LOCKING THE DAMPERS' BREAKERS OFF THEIR BUS. REPAIRS OF THE COMPUTER INPUTS FROM THE DAMPER'S LIMIT SWITCHES HAS BEEN INITIATED. DURING THE PERIOD FROM 5/16 TO 5/17 WHEN BOTH THESE DAMPERS WERE OPEN, THE JOINT UNIT 1/UNIT 2 CONTROL ROOM ENVELOPE COULD NOT HAVE BEEN AUTOMATICALLY ISOLATED.

[8] BEAVER VALLEY 2 DOCKET 50-412 LER 90-011 REV 01
UPDATE ON CONTAINMENT PURGE ISOLATION DUE TO HIGH RADIATION SIGNAL
EVENT DATE: 090690 REPORT DATE: 070591 NSSS: WE TYPE: PWR

(NSIC 222464) ON 9/6/90, THE CONTAINMENT BUILDING WAS BEING PURGED IN PREPARATION FOR REFUELING. AT 1719 HOURS, CONTAINMENT RADIATION MONITOR HVR*RQ104B SPIKED HIGH AND INITIATED AN AUTOMATIC PURGE ISOLATION. SUBSEQUENT SAMPLING BY RADIATION CONTROL TECHNICIANS SHOWED NO MEASURABLE AIRBORNE ACTIVITY IN CONTAINMENT. AFTER THE INITIAL SPIKE, THE MONITOR'S INDICATED ACTIVITY RETURNED TO ITS PRE-EVENT LEVEL. INVESTIGATION DETERMINED THAT THE MONITOR HAD SPIKED HIGH DUE TO AN ELECTRICAL DISTURBANCE RESULTING FROM A LIGHTNING STRIKE. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. NO ACTUAL RADIATION INCREASE OCCURRED. THE SYSTEM ACTUATED IN A CONSERVATIVE DIRECTION. ISOLATION OF CONTAINMENT PURGE IN RESPONSE TO A HIGH RADIATION SIGNAL IS DESCRIBED IN UNIT 2 UFSAR SECTION 9.4.7.3, "CONTAINMENT PURGE AIR SYSTEM".

[9] BIG ROCK POINT DOCKET 50-155 LER 91-003
DAMAGED O-RING SEAL DISCOVERED DURING ROTORK ACTUATOR REFURBISHMENT.
EVENT DATE: 051791 REPORT DATE: 061091 NSSS: GE TYPE: BWR
VENDOR: ROTORK INC.

(NSIC 222232) ON MAY 17, 1991 A DAMAGED O-RING SEAL FOR ROTORK VALVE OPERATOR MODEL 14NA-1 WAS DISCOVERED DURING REFURBISHMENT AT THE VENDOR. THE OPERATOR HAD BEEN REMOVED FROM CORE SPRAY VALVE MOV MO-7071 DURING THE 1990 REFUELING OUTAGE. THIS DAMAGED SEAL PLACED THE OPERATOR IN AN UNTESTED LOCA ENVIRONMENTAL

CONFIGURATION FOR CYCLE 24, WHICH LASTED ABOUT 13 MONTHS. A NEW SEAL WAS REQUIRED TO COMPLETE THE REFURBISHMENT. THE CAUSE OF THE DAMAGE TO THE SEAL HAS BEEN ATTRIBUTED TO THE REPLACEMENT OF THE TERMINAL BLOCK (BECAUSE OF A BROKEN LUG) INTO ITS HOUSING DURING A PREVIOUS REPAIR. APPARENTLY THE INDIVIDUAL PERFORMING THE TASK LACKED THE PROPER KNOWLEDGE OR INSTRUCTION, CAUSING THE SEAL TO BE PINCHED DURING INSTALLATION. TO PREVENT RECURRENCE, THE MAINTENANCE ORDERS AND ACTIVITIES ASSOCIATED WITH THE 1989 REFURBISHMENT WERE EVALUATED. PERSONNEL ASSOCIATED WITH THESE MAINTENANCE ACTIVITIES WERE INTERVIEWED AND MADE AWARE OF THE DEFICIENCY.

[10] BIG ROCK POINT DOCKET 50-155 LER 91-004
PLANT SHUTDOWN DUE TO INOPERABLE REDUNDANT CORE SPRAY VALVE.
EVENT DATE: 052991 REPORT DATE: 062691 NSSS: GE TYPE: BWR
VENDOR: ROTORK INC.

(NSIC 222394) ON 5/28/91 AT 2011 HOURS SURVEILLANCE TEST T30-22 (EMERGENCY CORE COOLING SYSTEM VALVE TESTS) WAS BEING PERFORMED. THE PRIMARY CORE SPRAY SYSTEM VALVES WERE TESTED FIRST AND MET THE ACCEPTANCE CRITERIA REQUIRED BY THE PROCEDURE. THE PROCEDURE THEN DIRECTS THE TESTING OF THE SECONDARY BACKUP CORE SPRAY VALVE, MO-7070. AFTER MO-7070 WAS SUCCESSFULLY STROKED, THE PRIMARY BACKUP CORE SPRAY VALVE, MO-7071 WAS TESTED NEXT. AS MO-7071 WAS BEING OPENED, THE AO HEARD A LOUD "BANG" AND OBSERVED THE BACKUP CORE SPRAY PIPING MOVE IN A NORTH-TO-SOUTH DIRECTION FOR ABOUT 2 SECONDS AND THEN IN A VERTICAL DIRECTION FOR ABOUT 8 SECONDS. IN THE CONTROL ROOM, THE CO HALTED THE OPENING OF MO-7071 (10 SECOND STROKE TIME), OBSERVED BOTH GREEN AND RED POSITION LIMITS ILLUMINATED AND INITIATED CLOSURE (2 SECOND STROKE TIME). AFTER MO-7071 WAS RUN CLOSED, THE BACKUP CORE SPRAY PIPE MOVEMENT TERMINATED. MO-7071 WAS THEN DECLARED INOPERABLE AND IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS 11.3.1.4 A. AND E., THE PLANT ENTERED A LIMITING CONDITION OF OPERATION, WHICH WAS TERMINATED BY BRINGING THE PLANT TO A COLD SHUTDOWN CONDITION. THE CAUSE OF THE EVENT HAS BEEN ATTRIBUTED TO WATER-HAMMER. A WATER-SLUG FORMED ON THE REACTOR SIDE OF MO-7071 WAS ACCELERATED TOWARDS MO-7071, IMPACTING THE VALVE DISC AS IT OPENED. TO ELIMINATE THE FORMATION OF A WATER-SLUG IN THE FUTURE, THE BACKUP CORE SPRAY LINE WAS CUT AND SLOPED TOWARDS THE REACTOR.

[11] BRAIDWOOD 2 DOCKET 50-457 LER 91-002
REACTOR SHUTDOWN CAUSED BY EXCESSIVE CONTAINMENT PURGE VALVE LEAKAGE DISCOVERED DURING LOCAL LEAKAGE RATE TESTING.
EVENT DATE: 051091 REPORT DATE: 060791 NSSS: WE TYPE: PWR
VENDOR: JAMES BURY CORP.

(NSIC 222392) ON 5/10/91 LOCAL LEAKAGE RATE TEST SURVEILLANCE BWVS 6.1.7.3-1 WAS STARTED BY A SYSTEM TEST ENGINEER (STE) TO VERIFY THAT THE MEASURED LEAKAGE RATE OF THE 48 INCH CONTAINMENT PURGE SUPPLY CONTAINMENT PENETRATION (P-97) WAS LESS THAN 23.15 STANDARD CUBIC FEET PER HOUR (SCFH) WHEN PRESSURIZED TO AT LEAST 44.4 PSIG. P-97 CONSISTS OF TWO CONTAINMENT ISOLATION VALVES IN SERIES; 2VQ001A AND 2VQ001B. A LEAK RATE MONITOR (LRM) TEST RIG WAS ATTACHED TO THE P-97 TEST CONNECTION. THE STE STARTED P-97 PRESSURIZATION BY OPENING AN AIR SUPPLY VALVE. AT 2130, IT WAS NOTED THAT THE HIGHEST PRESSURE ACHIEVED FOR P-97 USING THE LRM TEST RIG WAS ONLY 33 PSIG. THE LRM WAS BYPASSED TO DIRECTLY PRESSURIZER P-97 TO 46 PSIG. WHEN THE AIR SUPPLY WAS ISOLATED, THE PRESSURE IN P-97 DECREASED TO 21 PSIG IN 14 MINUTES AND CONFIRMED THAT VALVE LEAKAGE WAS HIGHER THAN THE TECH SPEC LIMIT. AT 2200 ON 5/11/91 A UNIT SHUTDOWN COMMENCED. SINCE THIS SHUTDOWN WAS REQUIRED BY TECH SPECS, AN UNUSUAL EVENT WAS DECLARED. AT 0322 ON 5/12/91 A REACTOR TRIP WAS CAUSED BY A SPIKE ON SOURCE RANGE CHANNEL N-31. AT 0530 ON 5/13/91 THE UNUSUAL EVENT WAS TERMINATED WHEN THE UNIT ENTERED MODE 5. CAUSE OF THE SEAL LEAKAGE WAS ATTRIBUTED TO SEAL DEGRADATION. THE RESILIENT MATERIAL SEALS WERE REPLACED FOR BOTH VALVES. BWVS 6.1.7.3-1 WAS SATISFACTORILY PERFORMED WITH A LEAKAGE RATE OF 5.11 SCFH.

[12] BROWNS FERRY 1 DOCKET 50-259 LER 91-004 REV 01
UPDATE ON UNPLANNED START OF UNIT 1 AND 2 DIESEL GENERATORS RESULTING FROM PERSONNEL ERROR DURING PERFORMANCE OF COMMON ACCIDENT SIGNAL LOGIC SURVEILLANCE INSTRUCTION.
EVENT DATE: 041191 REPORT DATE: 061791 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 222297) ON APRIL 11, 1991 AT 1819 HOURS, EMERGENCY DIESEL GENERATORS (EDGS), 1A, 1B, 1C AND 1D AND THE ASSOCIATED EMERGENCY EQUIPMENT COOLING WATER (EECW) PUMPS UNEXPECTEDLY AUTO-STARTED DURING PERFORMANCE OF SURVEILLANCE INSTRUCTION "COMMON ACCIDENT SIGNAL LOGIC." THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. AN ASSISTANT UNIT OPERATOR (AUO) PLACED THE WRONG KEYLOCK SWITCH IN THE TEST POSITION AND STARTED THE EDGS. CORRECTIVE ACTIONS WERE TO RETURN THE EDGS AND EECW PUMPS TO STANDBY READINESS. THE AUO WAS COUNSELED. LARGER KEYLOCK SWITCH IDENTIFICATION LABELS WERE INSTALLED. THIS EVENT WILL BE REVIEWED BY OPERATIONS PERSONNEL.

[13] BROWNS FERRY 1 DOCKET 50-259 LER 91-006
UNPLANNED ENGINEERED SAFETY FEATURE ACTUATION DURING MAINTENANCE ACTIVITIES DUE TO PROXIMITY OF COMPONENTS.
EVENT DATE: 050791 REPORT DATE: 060691 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 222362) ON MAY 7, 1991 AT 1325 HOURS, AN UNPLANNED ENGINEERED SAFETY FEATURES (ESF) ACTUATION OCCURRED WHEN THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM INITIATED. THE ESF OCCURRED WHEN MAINTENANCE PERSONNEL PERFORMING WORK IN A CONTROL PANEL INADVERTENTLY BUMPED THE CONTACTS OF A RELAY ASSOCIATED WITH A RADIATION MONITOR. THE ROOT CAUSE OF THIS EVENT WAS A DESIGN WHICH MADE IT DIFFICULT TO PERFORM THE WORK. THE LOCATION OF THE COMPONENTS IN THE CONTROL PANEL DOES NOT PROVIDE ADEQUATE SPACE TO PERFORM MAINTENANCE AND/OR CALIBRATION ON THE PRESSURE SWITCH WITHOUT THE POSSIBILITY OF BUMPING THE RELAY CONTACTS. IN ADDITION, THE RELAY DOES NOT HAVE A PROTECTIVE COVER. FACTORS WHICH CONTRIBUTED TO THIS EVENT INCLUDED A LACK OF CAUTIONARY NOTES IN THE PROCEDURE AND AN UNLABELED RELAY. AS A RESULT OF THIS ISOLATED EVENT TVA WILL EVALUATE THE NEED FOR A DESIGN CHANGE TO CORRECT THE INSTRUMENT LAYOUT IN THE AFFECTED CONTROL PANEL AND TO PROVIDE A COVER FOR THE RELAY. TVA WILL ALSO PROVIDE A SPECIFIC PROCEDURE FOR PERFORMING MAINTENANCE/CALIBRATION ON THIS PRESSURE SWITCH. FINALLY, THE AFFECTED RELAY WILL BE APPROPRIATELY LABELED.

[14] BROWNS FERRY 2 DOCKET 50-260 LER 91-004 REV 01
UPDATE ON REACTOR PROTECTION SYSTEM ACTUATION AS A RESULT OF EXCEEDING THE HI-HI SRM CHANNEL SETPOINT DURING TESTING.
EVENT DATE: 032691 REPORT DATE: 062491 NSSS: GE TYPE: BWR

(NSIC 222363) ON MARCH 26, 1991, AT 2024 HOURS, UNIT [illegible] REACTOR PROTECTION SYSTEM ACTUATION OCCURRED WHEN THE COUNT RATE ON THE SOURCE RANGE MONITOR (SRM) CHANNEL C SPIKED WHILE A TIME DOMAIN REFLECTOMETER (TDR) TRACE WAS BEING PERFORMED ON THE UNIT 2 INTERMEDIATE RANGE MONITOR (IRM) CHANNEL C DETECTOR CABLE SHIELD. THE ROOT CAUSE OF THIS EVENT COULD NOT BE DETERMINED. THE SCRAM WAS THE RESULT OF AN UNEXPECTED HI-HI TRIP GENERATED DURING THE PERFORMANCE OF TDR TRACING ON AN IRM CABLE LOCATED IN THE PROXIMITY OF THE SRM CHANNEL C CABLE. HOWEVER, THE CAUSE OF THE HI-HI TRIP SIGNAL COULD NOT BE DETERMINED OR REPEATED. THE GROUND FAULT IN THE IRM CHANNEL C DETECTOR CABLE SHIELD WAS REMOVED. IN ADDITION THE NEUTRON MONITORING SYSTEM, INCLUDING THE INTERACTION BETWEEN CHANNELS, WAS EVALUATED. FROM THE INFORMATION GATHERED DURING THIS EVALUATION, TVA DEVELOPED AND IMPLEMENTED CORRECTIVE ACTIONS DIRECTED TO BOTH THE NEUTRON MONITORING HARDWARE AND TO THE MAINTENANCE PROCEDURES.

[15] BROWNS FERRY 2 DOCKET 50-260 LER 91-006
UNPLANNED REACTOR PROTECTION SYSTEM ACTUATION RESULTING FROM LOCAL POWER RANGE MONITOR LEAKAGE CURRENT.
EVENT DATE: 032991 REPORT DATE: 062491 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 222364) AT 2139 HOURS ON 3/29/91, AN UNPLANNED REACTOR PROTECTION SYSTEM ACTUATION OCCURRED ON UNIT 2 WHEN AVERAGE POWER RANGE MONITOR (APRM) B DRIFTED HIGH AND EXCEEDED THE HI-HI SETPOINT. THE INPUT TO THE APRM IS PROVIDED BY THE LOCAL POWER RANGE MONITORS (LPRMS). DURING THIS EVENT, A SINGLE LPRM DRIFTED HIGH DUE TO A HIGH LEAKAGE CURRENT AND THIS RESULTED IN THE ASSOCIATED APRM EXCEEDING ITS SETPOINT. THIS EVENT WAS TERMINATED AT 2330 HOURS ON 3/29/91 WHEN THE LPRM WAS BYPASSED AND THE REACTOR SCRAM WAS RESET. THE ROOT CAUSE OF THIS EVENT WAS AN UNEXPECTED FAILURE OF THE LPRM DUE TO A HIGH LEAKAGE CURRENT WHICH WAS APPARENTLY CAUSED BY A CRACKED CERAMIC SEAL ON THE LPRM CABLE. THE INITIAL CORRECTIVE ACTION WAS TO TROUBLESHOOT THE LPRM. ADDITIONAL TESTING SHOWED A HIGH LEAKAGE CURRENT IN THE LPRM CABLE. SINCE PRESENT PLANT CONDITIONS PRECLUDE THE REPAIR OR REPLACEMENT OF THIS DETECTOR, THE LPRM HAS BEEN PLACED IN THE BYPASS MODE. IN ADDITION, THE NEUTRON MONITORING SYSTEM, INCLUDING THE INTERACTION BETWEEN CHANNELS, WAS EVALUATED. FROM THE INFORMATION GATHERED DURING THIS EVALUATION, TVA DEVELOPED AND IMPLEMENTED CORRECTIVE ACTIONS DIRECTED TO BOTH THE NEUTRON MONITORING HARDWARE AND TO THE MAINTENANCE PROCEDURES. ADDITIONAL CORRECTIVE ACTIONS INCLUDE THE REPLACEMENT OF THE LPRM DURING THE CYCLE 6 REFUELING OUTAGE.

[16] BROWNS FERRY 2 DOCKET 50-260 LER 91-009 REV 01
UPDATE ON INADVERTENT MAIN STEAM ISOLATION VALVE CLOSURE DURING SURVEILLANCE TESTING.
EVENT DATE: 041291 REPORT DATE: 062691 NSSS: GE TYPE: BWR

(NSIC 222365) ON APRIL 12, 1991, AT 0034 HOURS, THE OUTBOARD MAIN STEAM ISOLATION VALVES (MSIVS) CLOSED DURING THE PERFORMANCE OF A SURVEILLANCE INSTRUCTION (SI). PRIOR TO THE BEGINNING OF THE SURVEILLANCE TEST A RELAY ASSOCIATED WITH THE MSIV ISOLATION LOGIC HAD BEEN REMOVED AND WAS SCHEDULED FOR REPLACEMENT. THE REMOVAL OF THE RELAY PLACED THE MSIVS IN A 1-OUT-OF-2 ISOLATION SIGNAL CONDITION. IN ACCORDANCE WITH THE SI, A REACTOR VESSEL WATER LEVEL TRANSMITTER WAS ISOLATED AT THE MANIFOLD, AND THE TRANSMITTER HIGH SIDE BEGAN TO DEPRESSURIZE THROUGH VALVE PACKING LEAKS. THIS SIMULATED A LOW-LOW-LOW REACTOR WATER LEVEL SIGNAL WHICH COMPLETED THE ISOLATION LOGIC TO CLOSE THE OUTBOARD MSIVS. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THE PERSONNEL INVOLVED WITH THE EVALUATION OF THE IMPACT OF THE MISSING RELAY FAILED TO ADEQUATELY ASSESS ITS EFFECT ON THE MSIV ISOLATION LOGIC. IMMEDIATE CORRECTIVE ACTIONS WERE TAKEN TO TERMINATE THE SURVEILLANCE TEST AND TO RETURN THE REACTOR WATER LEVEL TRANSMITTER TO SERVICE. LICENSED OPERATIONS PERSONNEL, MAINTENANCE FOREMEN, MAINTENANCE PLANNERS, AND SYSTEM ENGINEERING PERSONNEL WILL BE REQUIRED TO REVIEW THIS EVENT. A DESIGN CHANGE WILL BE IMPLEMENTED TO PROVIDE A MEANS TO MONITOR MSIV PILOT SOLENOID CIRCUIT CONTINUITY.

[17] BROWNS FERRY 2 DOCKET 50-260 LER 91-010
PLANT TECHNICAL SPECIFICATION VIOLATION WHEN 24-HOUR LIMITING CONDITION FOR OPERATION EXPIRED WITHOUT DECLARING AFFECTED SYSTEMS INOPERABLE.
EVENT DATE: 051591 REPORT DATE: 061191 NSSS: GE TYPE: BWR

(NSIC 222298) ON MAY 15, 1991, A VIOLATION OF THE PLANT TECHNICAL SPECIFICATIONS (TSS) OCCURRED WHEN A 24-HOUR LIMITING CONDITION FOR OPERATION (LCO) EXPIRED AND THE SYSTEMS AFFECTED BY A REACTOR VESSEL WATER LEVEL TRANSMITTER WERE NOT DECLARED INOPERABLE. THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE PROCEDURES. OPERATIONS IS NOT REQUIRED TO BE NOTIFIED WHEN EQUIPMENT RELATED TO THE TS IS REMOVED FROM SERVICE. INSTEAD, PROCEDURES REQUIRE OPERATIONS TO BE NOTIFIED WHEN IT BECOMES APPARENT THE LCO TIME LIMIT WILL EXPIRE. THE IMMEDIATE CORRECTIVE ACTION WAS TO DECLARE THE AFFECTED SYSTEMS INOPERABLE. ADDITIONAL CORRECTIVE ACTIONS WILL INCLUDE TRAINING OF THE AFFECTED PLANT PERSONNEL AND REVISION OF PLANT PROCEDURES.

[18] BROWNS FERRY 2 DOCKET 50-260 LER 91-012
UNPLANNED ENGINEERED SAFETY FEATURES ACTUATION FOLLOWING REACTOR PROTECTION SYSTEM BUS DE-ENERGIZATION CAUSED BY RANDOM EQUIPMENT FAILURE.
EVENT DATE: 051991 REPORT DATE: 061791 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 1 (BWR)
BROWNS FERRY 3 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 222300) ON MAY 19, 1991 AT 1359 HOURS, AN UNPLANNED EMERGENCY SAFETY FEATURES ACTUATION OCCURRED FOLLOWING DE-ENERGIZATION OF REACTOR PROTECTION SYSTEM (RPS) BUS 2A. THIS RESULTED IN ACTUATION OF THE PRIMARY CONTAINMENT ISOLATION SYSTEM, WHICH INCLUDED CLOSURE OF VALVES IN THE RESIDUAL HEAT REMOVAL AND REACTOR WATER CLEANUP SYSTEMS, WITHDRAWAL OF ANY INSERTED TRAVERSING INCORE PROBES, AND ISOLATION OF PLANT VENTILATION SYSTEMS. IN ADDITION, THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) AND STANDBY GAS TREATMENT SYSTEM (SGTS) RECEIVED ACTUATION SIGNALS. HOWEVER, THE CREVS WAS ALREADY IN SERVICE AND THE SGTS WAS REMOVED FROM SERVICE. THE ROOT CAUSE OF THIS EVENT WAS EQUIPMENT FAILURE. THE RPS BUS DEENERGIZED AFTER THE MOTOR FROM ITS NORMAL POWER SOURCE, MOTOR-GENERATOR SET 2A, FAILED. THE MOTOR FAILURE WAS ATTRIBUTED TO WINDING INSULATION BREAKDOWN BETWEEN PHASES A AND C IN THE OUTBOARD TURN AREA OF THE MOTOR. THE FAILED RPS MG SET MOTOR (GENERAL ELECTRIC MODEL 5K326AK2060; A 460 VOLT, 3-PHASE AC MOTOR) WAS REMOVED AND REPLACED.

[19] BROWNS FERRY 3 DOCKET 50-296 LER 91-003 REV 01
UPDATE ON ON UNPLANNED ENGINEERED SAFETY FEATURE OCCURRED FROM PERSONNEL ERROR WHILE PERFORMING A COMMON ACCIDENT SIGNAL LOGIC SURVEILLANCE INSTRUCTION.
EVENT DATE: 041291 REPORT DATE: 062591 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 1 (BWR)
BROWNS FERRY 2 (BWR)

(NSIC 222417) ON APRIL 12, 1991 AT 1817 HOURS, AN UNPLANNED ENGINEERED SAFETY FEATURE (ESF) ACTUATION OCCURRED WHEN THE FOUR UNIT 3 EMERGENCY DIESEL GENERATORS (EDGS) UNEXPECTEDLY AUTO-STARTED DURING THE PERFORMANCE OF COMMON ACCIDENT SIGNAL (CAS) LOGIC SURVEILLANCE INSTRUCTION (SI). THE AUTOMATIC INITIATION OF AN ESF IS REPORTABLE IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(IV). THE CAUSE OF THIS EVENT IS PERSONNEL ERROR RESULTING FROM A LACK OF ATTENTION TO DETAIL DURING INSTALLATION OF AN INHIBITING BOOT BETWEEN TWO CONTACTS ON THE DIVISION II CORE SPRAY LOGIC B RELAY. CORRECTIVE ACTIONS PERFORMED WERE: OPERATIONS PERSONNEL ENSURED ELECTRICAL MAINTENANCE (EM) PERSONNEL STOPPED THE PERFORMANCE OF THE SI; AN INCIDENT INVESTIGATION WAS CONDUCTED TO DETERMINE THE CAUSE OF THE EVENT; AND EMS RESUMED AND SUCCESSFULLY COMPLETED THE 1/2 CAS SI. MAINTENANCE PERSONNEL WHO INSTALL BOOTS WILL RECEIVE TRAINING ON THE PROPER INSTALLATION OF BOOTS BY JULY 1, 1991.

[20] BRUNSWICK 1 DOCKET 50-325 LER 91-003 REV 01
UPDATE ON FAIL-AS-IS POSITION OF CBEAF SYSTEM INLET AND OUT DAMPERS NOT EVALUATED WITH RESPECT TO A CHLORINE EVENT.
EVENT DATE: 013191 REPORT DATE: 053091 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)

(NSIC 222216) A CONDITION THAT WAS DISCOVERED TO BE OUTSIDE THE DESIGN BASIS WAS REPORTED ON 1/31/91 AT 1602. THIS CONDITION INVOLVES THE INLET AND OUTLET DAMPERS FOR THE CONTROL BUILDING EMERGENCY AIR FILTRATION (CBEAF) TRAINS WHICH FAIL AS-IS ON LOSS OF POWER. THIS WILL RESULT IN AN AIRFLOW PATH TO THE CONTROL ROOM ENVIRONMENT THAT HAS NOT BEEN PREVIOUSLY EVALUATED FOR ITS EFFECT ON CONTROL ROOM HABITABILITY RELATIVE TO CHLORINE (CL) PROTECTION. THIS CONDITION RESULTED FROM A FAILURE TO EVALUATE THE FAIL AS-IS POSITION (IN COMBINATION WITH A SINGLE FAILURE) OF THE CBEAF DAMPERS WITH RESPECT TO A CL EVENT. ON 1/31/91, AT APPROX. 1811, THE CL TANK CAR (CLTC) WAS REMOVED FROM THE SITE PENDING AN ENGINEERING EVALUATION OF CBEAF SINGLE FAILURES. THE EVALUATION DETERMINED THE CLTC COULD BE RETURNED TO THE SITE PROVIDED THE MOST LIMITING CONDITION OF OPERATION (LCO) IS ENTERED WHEN THE CBEAF IS IN OPERATION. A TRACKING LCO WAS ESTABLISHED TO ENSURE IMPLEMENTATION OF THE LCO REQUIREMENTS AND THE CLTC WAS RETURNED TO THE SITE ON 2/10/91. A PROJECT (PCN G0187A) HAS BEEN INITIATED TO MAKE THE INLET AND OUTLET

DAMPERS AUTOMATICALLY CLOSE ON LOSS OF POWER. ADDITIONALLY, THE NUCLEAR ENGINEERING DEPARTMENT (NED) HAS PERFORMED A VALIDATION OF THE CL ISOLATION SYSTEM FUNCTIONAL DESIGN.

[21] BRUNSWICK 1 DOCKET 50-325 LER 91-013
COINCIDENT HPCI/RCIC INOPERABILITY DUE TO RCIC INSTRUMENTATION SENSING LINE LEAKAGE.
EVENT DATE: 050791 REPORT DATE: 060691 NSSS: GE TYPE: BWR

(NSIC 222225) ON 5/7/91 WITH THE UNIT 1 REACTOR AT 100% POWER, TECH SPEC (TS) LIMITING CONDITION FOR OPERATION (LCO) 3.0.3 WAS ENTERED WHEN BOTH THE HIGH PRESSURE COOLANT INJECTION (HPCI) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEMS WERE INOPERABLE. THE COINCIDENCE OF INOPERABLE SYSTEMS OCCURRED AS A RESULT OF IN PROGRESS TESTING OF THE HPCI SYSTEM AND THE INOPERABILITY OF THE RCIC HIGH STEAM FLOW INSTRUMENTATION. THE CAPABILITY OF TERMINATING HPCI TESTING IN SUPPORT OF TIMELY SYSTEM RESTORATION IN CONJUNCTION WITH THE ABILITY TO UTILIZE OPERABLE REACTOR VESSEL DEPRESSURIZATION AND LOW PRESSURE INJECTION SYSTEMS SIGNIFICANTLY REDUCED THE IMPACT TO NUCLEAR SAFETY. THE CAUSE OF THIS EVENT WAS DUE TO THE LOSS OF THE RCIC HIGH STEAM FLOW TRANSMITTER 1-E51-PDT-N017 SENSING LINE LEVEL. THIS CONDITION WAS CAUSED BY A LOOSE 1-E51-PF-N019A ISOLATION VALVE FITTING. THE VOID CREATED FROM THE LOSS OF LEVEL INCREASED THE VOLUME OF TURBULENT STEAM AND GASES WITHIN THE 1-E51-PDT-N017 SENSING LINE AFFECTED BY THE LEAK. THE ADDITIONAL TURBULENCE INCREASED THE TRANSMITTER'S ELECTRONIC NOISE SIGNAL TO A LEVEL WHICH CAUSED THE ASSOCIATED MASTER TRIP UNIT INDICATOR TO FLUCTUATE AND THE CYCLING OF THE MASTER AND SLAVE TRIP UNIT TRIP INDICATING LIGHTS. THE SUBJECT TRANSMITTER AND ANALOG TRIP UNITS RESPONDED PROPERLY TO THE ABNORMAL CONDITION IMPOSED BY THE LOSS OF SENSING LINE LEVEL.

[22] BRUNSWICK 2 DOCKET 50-324 LER 90-004 REV 03
UPDATE ON MANUAL REACTOR SCRAM DUE TO FAILURE OF SAFETY RELIEF VALVE "G" TO CLOSE DURING START-UP TESTING.
EVENT DATE: 031390 REPORT DATE: 070191 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 222423) AT 0536 ON 3/13/90, A MANUAL SCRAM WAS INITIATED DUE TO THE FAILURE OF SAFETY/RELIEF VALVE (SRV) B21-F013G TO CLOSE DURING STARTUP TESTING REACTOR POWER WAS APPROX. 7 AND REACTOR PRESSURE WAS APPROX. 950 PSIG THE 11 UNIT SRV'S WERE BEING CYCLED IN ACCORDANCE WITH PLANT PROCEDURES TO VERIFY OPERABILITY PER TECH SPEC 3.5.2. TEN OF THE ELEVEN SRV'S HAD BEEN SUCCESSFULLY TESTED PRIOR TO THIS FAILURE. A NORMAL SCRAM RECOVERY WAS CONDUCTED PER PLANT PROCEDURES AND NO AUTOMATIC SAFETY ACTUATIONS OR ISOLATIONS OCCURRED. THE INVESTIGATION DETERMINED THAT THE SOLENOID VALVE WHICH ALLOWS REMOTE MANUAL OPERATION OF B21-F013G WAS INOPERABLE. THE SOLENOID WOULD ALLOW THE SRV TO BE OPENED, BUT WOULD NOT ALLOW TIMELY CLOSURE OF THE SRV THE SOLENOID VALVE WAS REPLACED, THE UNIT RETURNED TO THE REQUIRED TESTING CONDITIONS AND THE SRV WAS SUCCESSFULLY TESTED. THE SOLENOID VALVE WAS SENT TO WYLE LABORATORY AND THE ROOT CAUSE WAS DETERMINED TO BE FAILURE OF THE SOLENOID DISC TO PROPERLY REALIGN WITH ITS SEAT AFTER DE-ENERGIZATION. THE REPORT INDICATED THAT A POTENTIAL CAUSE WAS "DIRT" EMBEDDED IN THE RUBBER PAD LOCATED ON THE BACKSEAT. AN INVESTIGATION INTO THE SOURCE OF THE MATERIAL WAS CONDUCTED. A CONTAMINATION SOURCE WAS NOT FOUND AND THE INVESTIGATION CONCLUDED THAT THE INSTRUMENT AIR SYSTEM DOES NOT HAVE A GENERIC PROBLEM WITH CONTAMINATION.

[23] BRUNSWICK 2 DOCKET 50-324 LER 90-018 REV 02
UPDATE ON MANUAL ISOLATION OF THE HPCI SYSTEM WHEN A RELAY COIL FAILED IN THE STEAM LEAK DETECTION SYSTEM.
EVENT DATE: 112390 REPORT DATE: 070191 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 1 (BWR)
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)
GEN ELECTRIC SUPPLY CO

(NSIC 222424) ON NOVEMBER 23, 1990, THE UNIT 2 REACTOR WAS AT 100% POWER. THE ECCS SYSTEMS WERE OPERABLE IN STANDBY READINESS. AT 0501, THE FUSE FOR THE UNIT

2 HPCI STEAM LEAK DETECTION CHANNEL B LOGIC BLEW. THE CHANNEL B LOGIC BUS POWER MONITOR RELAY HAD A BURNED COIL AND WAS THE PROBABLE CAUSE OF THE BROWN FUSE. THE HPCI SYSTEM WAS MANUALLY ISOLATED UNDER TECHNICAL SPECIFICATIONS. THE RELAY AND FUSE WERE REPLACED AND THE HPCI SYSTEM WAS RETURNED TO SERVICE DURING THE INVESTIGATION IT WAS DETERMINED THAT THE SAME EVENT OCCURRED ON UNIT 1 FIVE DAYS BEFORE THIS EVENT. UNIT 1 WAS IN A REFUEL OUTAGE WITH THE REACTOR DEFUELED AND HPCI WAS NOT REQUIRED TO BE OPERABLE. A WORK REQUEST REPLACED THE UNIT 1 RELAY AND FUSE AN INVESTIGATION AND ANALYSIS HAS DETERMINED THAT THE SUBJECT RELAYS WERE FOUND TO HAVE DEGRADED COILS. THE COIL DEGRADATION WAS CAUSEL BY AN ELECTRICAL SHORT. THE SPECIFIC CAUSE OF THE ELECTRICAL SHORT IS INDETERMINATE. IT IS CONCLUDED THAT THE CURRENTLY INSTALLED RELAY GROUP IS NOT EXPERIENCING AN END-OF-LIFE FAILURE; HOWEVER TO PRECLUDE FURTHER SAFETY SYSTEM ACTUATIONS RELAYS WITH THE SAME COIL WILL BE INSPECTED AND/OR REPLACED.

[24] BRUNSWICK 2 DOCKET 50-324 LER 91-003
COMPONENT FAILURE IN RCIC STEAM LINE BREAK ISOLATION ACTUATION TRIP UNIT RESULTS IN UNPLANNED ESF ACTUATION.
EVENT DATE: 052191 REPORT DATE: 062091 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 222336) ON 5/21/91, AT 1500, THE UNIT 2 REACTOR WAS AT 100% POWER. THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM WAS ISOLATED FOR ROUTINE SURVEILLANCE TESTING. AT THIS TIME, A PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) GROUP LOGIC B ISOLATION SIGNAL WAS GENERATED. INVESTIGATION DETERMINED THAT AN ERRONEOUS TRIP IN THE RCIC STEAM LINE BREAK D/P HIGH FLOW, CHANNEL B, TRIP UNIT WAS THE CAUSE OF THIS ESF ACTUATION. PER DESIGN, THE RCIC STEAM SUPPLY OUTBOARD ISOLATION VALVE, 2-E51-F008, RECEIVED AN ISOLATION SIGNAL, BUT SINCE BOTH THE INBOARD AND OUTBOARD ISOLATION VALVES WERE PREVIOUSLY CLOSED PER PROCEDURE, NO VALVE MOVEMENT WAS EXPERIENCED. THIS EVENT WAS DISCOVERED ON 5/21/91, AT APPROXIMATELY 1520 DURING AN ATTEMPT TO RESTORE RCIC TO STANDBY READINESS AFTER COMPLETION OF THE SURVEILLANCE TESTING. AT THIS TIME, THE CONTROL OPERATOR NOTICED THAT THE "RCIC STEAM LINE BREAK D/P HI" ANNUNCIATOR WAS SEALED IN AND WOULD NOT RESET. SUBSEQUENTLY, AT APPROXIMATELY 1545, THE CONDITION CLEARED WITHOUT ANY CORRECTIVE ACTIONS BEING PERFORMED. CAUSE OF THIS EVENT WAS COMPONENT FAILURE. A TRANSISTOR ANOMALY, CHARACTERIZED BY A GRADUAL INCREASE IN THE TRIP OUTPUT VOLTAGE, CAUSED THE OUTPUT OF THE ROSEMOUNT MODEL 51 ODU ANALOG MASTER TRIP CALIBRATION UNIT TRIP CARD TO PRODUCE AN ERRONEOUS TRIP OUTPUT SIGNAL TO THE PCIS LOGIC.

[25] BYRON 1 DOCKET 50-454 LER 91-002
ENGINEERED SAFEGUARD FEATURE ACTUATION DUE TO LIGHTNING STRIKE ON 345KV TRANSMISSION LINES.
EVENT DATE: 050591 REPORT DATE: 052391 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 2 (PWR)

(NSIC 222188) ON MAY 5, 1991 AT 1617 WITH UNIT 1 IN MODE 1 AT 100% REACTOR POWER AND UNIT 2 IN MODE 1 AT 97% REACTOR POWER, A VOLTAGE TRANSIENT OCCURRED WHEN 345KV LINES 0621 AND 0622 MOMENTARILY OPENED DURING A THUNDERSTORM. PROCESS RADIATION MONITORS FOR MAIN CONTROL ROOM OUTSIDE AIR INTAKE OB AND UNIT 1 COMPONENT COOLING HEAT EXCHANGER OUTLET SENSED THE VOLTAGE TRANSIENT WHEN THE VOLTAGE ON THE BUS SUPPLYING THE AFFECTED MONITORS DROPPED BELOW THE MONITOR UNDERVOLTAGE SETPOINT. THIS TRANSFERRED THE MONITOR TO THE INTERLOCK MODE CAUSING AN ENGINEERING SAFEGUARD FEATURES (ESF) ACTUATION. THE MAIN CONTROL ROOM VENTILATION SYSTEM AUTOMATICALLY TRANSFERRED TO EMERGENCY MODE. THE RADIATION MONITORS RETURNED TO NORMAL OPERATING CONDITION IMMEDIATELY AFTER THE VOLTAGE TRANSIENT PASSED. THE COMPONENT COOLING SURGE TANK VENT VALVE AUTOMATICALLY CLOSED. THE SEISMIC MONITORING SYSTEM ALSO EXPERIENCED A TEMPORARY FAILURE. DUE TO A SIMILAR OCCURRENCE, THE UNDERVOLTAGE SETPOINT FOR THE RADIATION MONITORS WAS PREVIOUSLY LOWERED AS MUCH AS THE EQUIPMENT WOULD ALLOW. NO FURTHER ACTION IS PLANNED. THIS EVENT IS REPORTABLE PER 10CFR50.73(A)(2)(IV) ANY EVENT OR CONDITION THAT RESULTS IN A MANUAL OR AUTOMATIC ENGINEERED SAFEGUARDS ACTUATION.

[26] BYRON 2 DOCKET 50-455 LER 90-010 REV 01
UPDATE ON MANUAL REACTOR TRIP AND MAIN STEAM ISOLATION DUE TO SAMPLE PROBE WELD FAILURE.
EVENT DATE: 122090 REPORT DATE: 061791 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 1 (PWR)

(NSIC 222327) ON DECEMBER 20, 1990 AT 0400, A SEVERE STEAM LEAK WAS REPORTED IN THE UNIT 2 MAIN STEAM TUNNEL. AFTER VERIFYING THE SIZE OF THE LEAK, THE REACTOR WAS MANUALLY TRIPPED. BY ELIMINATING STEAM GENERATOR BLOWDOWN AND FEEDWATER AS CAUSES, IT WAS DETERMINED THAT THE LEAK WAS ON THE MAIN STEAM SIDE. THE MAIN STEAM ISOLATION VALVES WERE THEN CLOSED WHICH ISOLATED THE LEAK. THE MAIN STEAM DUMPS WERE OPENED TO DEPRESSURIZE THE MAIN STEAM HEADER. UPON ENTRY INTO THE MAIN STEAM TUNNEL, THE 2C MAIN STEAM SAMPLE PROBE WAS FOUND LYING ON THE FLOOR. THE WELD FOR THE PROBE HAD BEEN IMPROPERLY REPAIRED DURING THE PREVIOUS REFUELING OUTAGE CAUSING THE PROBE AND ITS ISOLATION VALVE TO BE EJECTED LEAVING A ONE INCH HOLE IN THE MAIN STEAM LINE. SINCE THIS PROBE WAS NEEDED ONLY FOR INITIAL START-UP TESTING, THE NOZZLE WAS CAPPED. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV) ANY EVENT THAT RESULTS IN A MANUAL OR AUTOMATIC ACTUATION OF THE ENGINEERED SAFETY FEATURES INCLUDING REACTOR PROTECTION SYSTEM.

[27] CALVERT CLIFFS 2 DOCKET 50-318 LER 91-005
REACTOR PROTECTIVE SYSTEM ACTUATION AND PLANT TRIP DUE TO A LOSS OF FEED CAUSED BY COMPONENT FAILURE.
EVENT DATE: 050291 REPORT DATE: 053191 NSSS: CE TYPE: PWR

(NSIC 222214) AT 1050 ON 5/2/91, A LOW STEAM GENERATOR (SG) LEVEL REACTOR PROTECTIVE SYSTEM TRIP OCCURRED AT CALVERT CLIFFS UNIT 2 WHILE ATTEMPTING TO CONTROL SG LEVEL OSCILLATIONS AT LOW POWER LEVEL. AT THE TIME OF THE TRIP, UNIT 2 WAS OPERATING IN MODE 1 (POWER OPERATION) AT 8% POWER. THE LOW SG LEVEL WAS CAUSED BY A LOSS OF FEEDWATER EVENT WHEN THE OPERATING SG FEED PUMP (SGFP) WAS LOST DUE TO A SPEED CONTROLLER FAILURE. THE INITIATING LOSS OF FEED EVENT WAS CAUSED BY A FAILED CIRCUIT BOARD EDGE CONNECTOR WHICH PREVENTED THE OPERATOR FROM RESETTING THE SGFP SPEED CONTROLLER AND RESTORING FEED. IMMEDIATE CORRECTIVE ACTIONS INCLUDED REPLACING THE CIRCUIT BOARD AND CONNECTOR, AND INSPECTING LIKE SGFP SPEED CONTROLLERS FOR SIMILAR PROBLEMS. A SGFP SPEED CONTROLLER ALIGNMENT PROCEDURE INCLUDING FUNCTIONAL TEST WILL BE WRITTEN, AND THIS EVENT WILL BE INCLUDED IN THE NEXT CYCLE OF INSTRUMENT MAINTENANCE CONTINUING TRAINING.

[28] CALVERT CLIFFS 2 DOCKET 50-318 LER 91-004
INADVERTENT ACTUATION OF AUXILIARY FEEDWATER SYSTEM DURING MAIN TURBINE OVERSPEED TEST.
EVENT DATE: 050491 REPORT DATE: 053191 NSSS: CE TYPE: PWR

(NSIC 222229) AT 2122 ON 5/4/91 AT CALVERT CLIFFS UNIT 2, AN AUX. FEEDWATER ACTUATION SYSTEM (AFAS) ACTUATION SIGNAL WAS RECEIVED DURING A MAIN TURBINE OVERSPEED TEST. A STEAM TRANSIENT RESULTED IN AN ERRONEOUS STEAM GENERATOR (SG) LEVEL SIGNAL WHICH ACTUATED AFAS. AT THE TIME OF THIS EVENT, UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT A POWER LEVEL OF 10%. THE CAUSE OF THIS EVENT WAS A DESIGN DEFICIENCY IN THAT THE WIDE RANGE SG LEVEL INSTRUMENT WAS NOT DESIGNED TO PROVIDE ACCURATE LEVEL INDICATIONS DURING EXTREME PRESSURE AND STEAMING TRANSIENTS. INTERACTION OF HIGH SPEED AFAS CIRCUITRY WITH THE LEVEL INSTRUMENT RESULTED IN AN INADVERTENT AFAS ACTUATION. CORRECTIVE ACTION FOR THIS EVENT INCLUDES IMPLEMENTING A FACILITY MODIFICATION TO PREVENT SG PRESSURE AND STEAMING TRANSIENTS FROM CAUSING UNWANTED AFAS ACTUATIONS. IN THE MEANTIME, A PROCEDURE CHANGE WILL BE UTILIZED TO PREVENT SIMILAR AFAS ACTUATIONS FROM OCCURRING.

[29] CATAWBA 1 DOCKET 50-413 LER 91-012
TECHNICAL SPECIFICATION VIOLATION DUE TO VIOLATION OF CONTAINMENT INTEGRITY DURING CORE ALTERATIONS.
EVENT DATE: 050591 REPORT DATE: 060391 NSSS: WE TYPE: PWR

(NSIC 222389) ON MAY 5, 1991, AT 0758 HOURS, WITH UNIT 1 IN MODE 6, REFUELING, ICE CONDENSER (NF) SYSTEM REPLENISHMENT ACTIVITIES COMMENCED. FUEL RELOADING HAD

BEGUN AT 0535 HOURS. DURING ICE BED REPLENISHMENT, COLD AIR CAN BE DRAWN FROM THE UPPER PORTION OF THE ICE CONDENSER TO SUPPLY THE CHARGING SYSTEM THE AIR REQUIRED TO BLOW BORATED ICE INTO THE BASKETS. AIR IS EXTRACTED THROUGH PENETRATION M371 AND RETURNED WITH BORATED ICE THROUGH PENETRATION M394. HOWEVER, IF CORE ALTERATIONS ARE IN PROGRESS, SUCTION IS NOT TO BE TAKEN FROM THE ICE CONDENSER. ICE REPLENISHMENT ACTIVITIES WERE COMPLETED AT APPROXIMATELY 1206 HOURS. DURING THE PERFORMANCE OF THIS WORK, TECH SPEC 3.9.4 WAS VIOLATED BECAUSE SUCTION WAS BEING TAKEN FROM THE ICE CONDENSER WHILE CORE ALTERATIONS WERE IN PROGRESS. THIS INCIDENT WAS DISCOVERED ON MAY 7, 1991 AT 0800 HOURS BY A MAINTENANCE ENGINEERING SERVICES ENGINEER DURING DISCUSSIONS WITH ICE MACHINE OPERATORS AND VERIFICATION WITH OPERATIONS PERSONNEL. THIS INCIDENT IS ATTRIBUTED TO AN INAPPROPRIATE ACTION, BECAUSE THE ICE MACHINE OPERATOR DID NOT USE THE ICE MAKING AND CHARGING SYSTEM PROCEDURE. THIS INCIDENT IS ATTRIBUTED A CONTRIBUTING CAUSE OF INAPPROPRIATE ACTION DUE TO INADEQUATE COMMUNICATION BETWEEN A CONTROL ROOM OPERATOR (CRO) AND THE ICE MACHINE OPERATOR. A CORRECTIVE ACTION WILL BE TO ENSURE THAT INITIAL CONDITIONS FOR ICE BLOWING ARE SATISFIED PRIOR TO EACH ICE BLOWING.

[30] CATAWBA 1 DOCKET 50-413 LER 91-010
FEEDWATER ISOLATION DUE TO UNKNOWN CAUSE.
EVENT DATE: 052691 REPORT DATE: 062091 NSSS: WE TYPE: PWR

(NSIC 222431) ON 5/26/91, AT 2102 HOURS, WITH UNIT 1 IN MODE 5, COLD SHUTDOWN, A FEEDWATER (CF) SYSTEM ISOLATION OCCURRED DURING RACKING OUT ACTIVITIES OF THE "A" REACTOR TRIP BYPASS BREAKER. OPERATIONS PERSONNEL HAD PREVIOUSLY STARTED THE MANUAL REACTOR TRIP FUNCTIONAL TEST, PT/1/A/4600/15, BUT STOPPED TO PURSUE HIGHER PRIORITY WORK. LATER, AFTER SHIFT CHANGE, OPERATIONS PERSONNEL RESUMED THE PROCEDURE TO ALIGN THE BREAKERS FOR ENGINEERED SAFETY FEATURES (ESF) "A" TRAIN TESTING. DURING THIS ALIGNMENT, THE REACTOR TRIP BYPASS BREAKER "A" WAS RACKED FROM THE CONNECT TO THE DISCONNECT POSITION. DURING BREAKER TRANSIT, THE OPERATOR HEARD RELAYS ACTIVATING AND APPARENT CLOSURE OF CF VALVES WITHIN THE DOGHOUSE. CF ISOLATION WAS IMMEDIATELY RESET AND VALVES REALIGNED FOR ESF TESTING. ALL PROPER PERSONNEL, INCLUDING THE NRC, WERE NOTIFIED AND A WORK REQUEST WAS INITIATED TO INVESTIGATE AND REPAIR THE BREAKER. THE CAUSE FOR THIS EVENT IS UNKNOWN. DURING THE TIME OF THE EVENT, THE EVENTS RECORDER, WHICH COULD HAVE PROVIDED MORE INFORMATION AS TO THE CAUSE OF THIS EVENT, WAS OUT OF SERVICE. NO PROBLEMS WERE FOUND WITH THE BREAKER UPON COMPLETION OF THE WORK REQUEST. OPERATOR ERROR WAS NOT EVIDENT. CATAWBA WILL EVALUATE THE NEED TO REPLACE THE WESTINGHOUSE W-2 CELL SWITCH WHICH HAS HAD SENSITIVITY PROBLEMS IN THE PAST. A DESIGN STUDY IS NOW IN PROGRESS TO FIND AN ACCEPTABLE SUBSTITUTE.

[31] CATAWBA 2 DOCKET 50-414 LER 91-007
TECHNICAL SPECIFICATON 3.0.3 ENTERED AS A RESULT OF BOTH TRAINS OF AUXILIARY BUILDING VENTILATION FILTERED EXHAUST SYSTEM BEING INOPERABLE DUE TO AN INAPPROPRIATE ACTION.
EVENT DATE: 051091 REPORT DATE: 060491 NSSS: WE TYPE: PWR

(NSIC 222390) ON MAY 10, 1991, AT APPROXIMATELY 0338 HOURS, WITH UNIT 2 IN MODE 1, POWER OPERATION, TECHNICAL SPECIFICATION (T/S) 3.0.3 WAS ENTERED DUE TO BOTH TRAINS OF THE AUXILIARY BUILDING FILTERED EXHAUST (VA) SYSTEM BEING INOPERABLE. T/S 3.0.3 WAS EXITED AT 0351 HOURS WHEN THE OPERATOR RETURNED VA 'B' TRAIN TO SERVICE. THE OPERATIONS WORKLIST FOR THE NIGHT SHIFT INCLUDED THE TAGGING OF 'A' TRAIN VA IN PREPARATION FOR SCHEDULED MAINTENANCE ON THE DAY SHIFT. WHILE COMPLETING THE TAGOUTS, THE NON-LICENSED OPERATOR (NLO), WORK CONTROL CENTER SENIOR REACTOR OPERATOR (WCC) (SRO), UNIT SUPERVISOR (US), AND SHIFT SUPERVISOR (SS) MISUNDERSTOOD CONFUSING INFORMATION ON THE WORKLIST. AS A RESULT, THE 'B' TRAIN VA FILTERED EXHAUST FAN MOTOR BREAKER WAS TAGGED IN THE OFF POSITION. THIS INCIDENT IS ATTRIBUTED TO AN INAPPROPRIATE ACTION DUE TO A LACK OF ATTENTION TO DETAIL. AN OPERATIONS COMMITTEE, "REDUCE THE TREND (RTT)," IS REVIEWING ALL PRE-PLANNED TAGOUTS FOR THE VA SYSTEM. THIS COMMITTEE WILL PROVIDE RECOMMENDATIONS ON CHANGES AND IMPROVEMENTS IN THE VA PRE-PLANNED TAGOUTS. IN ADDITION, OPERATORS DURING REQUALIFICATION AND THE HOT LICENSE CLASS WILL RECEIVE OPERATOR PROFICIENCY TRAINING CONCERNING THE INCIDENT.

[32] CONNECTICUT YANKEE DOCKET 50-213 LER 85-003 REV 01
UPDATE ON LOW REACTOR COOLANT FLOW TRIP SETPOINT.
EVENT DATE: 021285 REPORT DATE: 062091 NSSS: WE TYPE: PWR

(NSIC 222290) WHILE AT 100% POWER, TWO CHANNELS OF THE LOW REACTOR COOLANT FLOW TRIP LOGIC OF THE REACTOR PROTECTION SYSTEM WERE DISCOVERED TO HAVE TRIP SETPOINTS THAT WERE NOT IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS. THE TECHNICAL SPECIFICATION STATES THAT THE LOW REACTOR COOLANT FLOW TRIP SHALL BE SET AT GREATER THAN OR EQUAL TO 90% OF NORMAL LOOP FLOW. THE IMMEDIATE CORRECTIVE ACTION TAKEN WAS TO REDUCE POWER BELOW THE P8 PERMISSIVE (84% POWER) TO ALLOW RECALIBRATION OF AFFECTED TRIP LOGIC RELAYS. THE ROOT CAUSE OF THE TRIP SETPOINT ERROR WAS INADEQUATE DEFINITION OF THE ACCEPTANCE CRITERIA IN THE CALIBRATION PROCEDURE. ADDITIONAL CORRECTIVE ACTION WHICH WAS TAKEN INCLUDED (1) CLARIFICATION OF THE ACCEPTANCE CRITERIA IN THE INSTRUMENT AND CONTROL PROCEDURES USED TO SET THE LOW FLOW TRIP SETPOINT BEFORE THESE PROCEDURES WERE USED AGAIN, (2) REVIEW OF ALL SURVEILLANCES TO DETERMINE WHICH PROCEDURES MUST HAVE THEIR ACCEPTANCE CRITERIA UPGRADED AND CLARIFIED, AND (3) TRENDING THE RCS FLOW RATE. THE PURPOSE OF THIS SUPPLEMENTAL REPORT IS TO DOCUMENT THE SIGNIFICANT CHANGES THAT HAVE TAKEN PLACE SINCE 1985 WHICH JUSTIFY ALLEVIATING THE NEED TO CONTINUE RCS FLOW RATE TRENDING.

[33] COOPER DOCKET 50-298 LER 91-002 REV 01
UPDATE ON REACTOR WATER CLEANUP ISOLATIONS DUE TO HIGH SYSTEM TEMPERATURE CAUSED BY INADEQUATE DESIGN.
EVENT DATE: 032491 REPORT DATE: 060391 NSSS: GE TYPE: BWR
VENDOR: MERCOID CORP.
NUCLEAR ENGINEERING COMPANY, INC.

(NSIC 222208) ON 3/24/91, AT 0238 AND, AGAIN, AT 0415, REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATIONS OCCURRED DUE TO HIGH TEMPERATURE CONDITIONS (140F) DOWNSTREAM OF THE NON-REGENERATIVE HEAT EXCHANGERS (NRHX). THE REACTOR WAS SHUTDOWN WITH THE RESIDUAL HEAT REMOVAL SYSTEM PROVIDING SHUTDOWN COOLING. THE RWCU PUMPS WERE SECURED TO FACILITATE FEEDWATER SYSTEM MAINTENANCE. SUBSEQUENT TO THESE EVENTS, FAULTY INSTRUMENTATION WAS REPAIRED AND FURTHER TESTING WAS PLANNED TO VERIFY THE SOURCE OF THE HIGH TEMPERATURE WATER. ON 5/7/91, RWCU WAS REMOVED FROM SERVICE TO ASSIST IN INVESTIGATING THE CAUSE OF A SLIGHTLY INCREASED LEAKAGE INSIDE THE DRYWELL. AT 1036, FOLLOWING CLOSURE OF THE INBOARD RWCU ISOLATION VALVE, ANOTHER RWCU SYSTEM ISOLATION FROM HIGH TEMPERATURE OCCURRED, CAUSING CLOSURE OF THE OUTBOARD RWCU ISOLATION VALVE. DURING THIS OCCURRENCE, THE REACTOR WAS AT APPROX. 100% POWER, 517F AND 998 PSIG. SUBSEQUENT TESTING INDICATED THE HIGH TEMPERATURE CONDITION OCCURRED AS A RESULT OF BACKFLOW THROUGH THE CHECK VALVE IN THE 3/4" SUBCOOLING LINE INSTALLED BETWEEN THE NRHX OUTLET AND THE RWCU SYSTEM INLET PIPING. THE ROOT CAUSE OF THE BACKFLOW IS DESIGN (FAILURE TO ANTICIPATE ALL SYSTEM OPERATING MODES), AS THE DIFFERENTIAL PRESSURE AVAILABLE ACROSS THE CHECK VALVE WHEN THE RWCU PUMPS ARE SECURED IS NOT SUFFICIENT TO OBTAIN A LEAKTIGHT SHUTOFF.

[34] COOPER DOCKET 50-298 LER 91-005
UNPLANNED STARTUP OF DIESEL GENERATOR NUMBER 2 DUE TO AN EQUIPMENT DEFICIENCY THAT OCCURRED DURING PLANT STARTUP WHILE TRANSFERRING LOADS TO THE NORMAL TRANSFORMER.
EVENT DATE: 051191 REPORT DATE: 060791 NSSS: GE TYPE: BWR

(NSIC 222375) ON 5/11/91, AT 3:58 A.M., DURING PLANT STARTUP FOLLOWING AN UNSCHEDULED SHUTDOWN, DIESEL GENERATOR NUMBER 2 (DG 2) AUTOMATICALLY STARTED, BUT WAS NOT REQUIRED TO LOAD, DUE TO A MALFUNCTION OF THE NORMAL TRANSFORMER SUPPLY BREAKER, 1BN. THIS OCCURRED WHEN TRANSFERRING LOADS FROM THE STARTUP TO THE NORMAL TRANSFORMER. UPON CLOSURE OF THE 1BN 4160V BREAKER AND TRIPPING OF THE 1BS BREAKER (FROM THE STARTUP TRANSFORMER), BREAKER 1BG, THE NORMAL SUPPLY BREAKER FOR 4160V 1G CRITICAL BUS, ALSO OPENED. UPON SENSING UNDERVOLTAGE ON THE 1G BUS, DG 2 AUTOMATICALLY STARTED. WITHIN APPROX. ONE SECOND, THE 1G BUS WAS REENERGIZED FROM THE EMERGENCY TRANSFORMER, FUNCTIONING AS DESIGNED. THE DG WAS NOT REQUIRED TO LOAD. THE CAUSE OF THE EVENT WAS DUE TO INSUFFICIENT ACTION OF THE BREAKER AUXILIARY SWITCH UPON CLOSURE OF THE 1BN BREAKER, SUCH THAT SWITCH

CONTACTS IN THE 1B BUS UNDERVOLTAGE CIRCUIT OPENED, ALLOWING RELAY 27 X2/1B TO DEENERGIZE. THE SUBSEQUENT BREAKER OPERATIONS OCCURRED DUE TO THE UNDERVOLTAGE TRIP LOGIC BEING DEENERGIZED. CORRECTIVE ACTION TAKEN INCLUDED INSPECTION OF THE BREAKER AND AUXILIARY SWITCH FOR PROPER OPERATION AND RESHIMMING THE BREAKER TO SLIGHTLY ELEVATE THE BREAKER IN ITS INSTALLED POSITION IN THE CUBICLE. THIS RESULTED IN IMPROVING THE ACTION OF THE AUXILIARY SWITCH TO ACHIEVE CONTACT CLOSURE.

[35]   DIABLO CANYON 1   DOCKET 50-275   LER 91-010
LACK OF PROCEDURAL CONTROL TO MAINTAIN SEISMIC QUALIFICATION OF REFUELING WATER STORAGE TANK DUE TO PROGRAMMATIC FAILURE.
EVENT DATE: 041091   REPORT DATE: 061791   NSSS: WE   TYPE: PWR
OTHER UNITS INVOLVED: DIABLO CANYON 2 (PWR)

(NSIC 222306) ON JUNE 13, 1990, DURING THE PREPARATION OF A DESIGN CHANGE, A CONCERN WITH THE SEISMIC INTEGRITY OF THE REFUELING WATER STORAGE TANK (RWST) WAS IDENTIFIED DUE TO THE DESIGN CLASS II REFUELING WATER PURIFICATION (RWP) SYSTEM BEING ALIGNED TO THE DESIGN CLASS I RWST DURING POWER OPERATION. AN ANALYSIS PERFORMED IN DECEMBER 1990 CONCLUDED THAT THE REDUCTION OF VOLUME IN THE RWST THAT COULD RESULT FROM A BREAK IN THE RWP SYSTEM WOULD NOT RENDER THE RWST INOPERABLE. ON APRIL 10, 1991, THE CONCERN WAS DISCUSSED WITH THE NRC AEOD, AND IT WAS CONCLUDED THAT THE PROBLEM WAS NOT REPORTABLE; HOWEVER, THE AEOD REQUESTED THAT PG&E SUBMIT AN INFORMATIONAL LER DESCRIBING THE PROBLEM. THE ROOT CAUSE WAS DETERMINED TO BE A PROGRAMMATIC FAILURE TO ADEQUATELY IMPLEMENT ENGINEERING OPERATIONAL CONSTRAINTS INTO PLANT PROCEDURES. THE CORRECTIVE ACTIONS ARE: (1) IDENTIFICATION OF VALVE OPERATIONAL CONSTRAINTS FOR THE RWP SYSTEM IN THE DESIGN CRITERIA MEMORANDUM; (2) IDENTIFICATION OF SEALED OR LOCKED VALVES ON P&IDS THAT ARE NOT INCLUDED IN THE SEALED VALVE PROGRAM; (3) PROVIDE THE BASIS AND METHOD FOR CONTROLLING THE VALVES IDENTIFIED ABOVE; AND (4) INCORPORATING THE RESULTS INTO APPROPRIATE OPERATING PROCEDURES.

[36]   DIABLO CANYON 1   DOCKET 50-275   LER 91-009
REACTOR TRIP DUE TO PERSONNEL ERROR AND SAFETY INJECTION DUE TO LEAKING STEAM DUMP VALVES.
EVENT DATE: 051791   REPORT DATE: 061491   NSSS: WE   TYPE: PWR
VENDOR: COPES-VULCAN, INC.

(NSIC 222305) ON MAY 17, 1991, AT 0624 PDT, WITH UNIT 1 OPERATING AT 100% POWER, A REACTOR TRIP OCCURRED DUE TO NUCLEAR INSTRUMENTATION POWER RANGE TWO-OUT-OF-FOUR CHANNELS HIGH FLUX AT HIGH SETPOINT SIGNALS. AT 0625 PDT, A SAFETY INJECTION OCCURRED DUE TO TWO-OUT-OF-FOUR LOW PRESSURIZER PRESSURE SIGNALS. DURING THE EVENT, REACTOR COOLANT SYSTEM COOLDOWN EXCEEDED THE ALLOWABLE RATE OF 100 DEGREES FAHRENHEIT PER HOUR OF TECHNICAL SPECIFICATION 3.4.9.1.B. AN UNUSUAL EVENT WAS DECLARED AT 0625 PDT ON MAY 17, 1991. A ONE-HOUR EMERGENCY REPORT REQUIRED BY 10 CFR 50.72(A)(1)(I) WAS MADE ON MAY 17, 1991, AT 0633 PDT. THE CAUSE OF THE REACTOR TRIP WAS DETERMINED TO BE PERSONNEL ERROR. AN I&C TECHNICIAN INADVERTENTLY DEENERGIZED A SECOND NUCLEAR INSTRUMENTATION POWER RANGE CHANNEL (N42) WHILE PERFORMING SURVEILLANCE TESTING ON ANOTHER POWER RANGE CHANNEL (N41). THE CAUSE OF THE SAFETY INJECTION WAS STEAM DUMP VALVES THAT FAILED OPEN AND OVERCOOLED THE REACTOR COOLANT SYSTEM. CORRECTIVE ACTIONS FOR THE EVENT INCLUDED: (1) TEMPORARY STOPPAGE OF ALL I&C WORK UNTIL I&C PERSONNEL WERE TAILBOARDED ON THE NECESSITY OF SELF-VERIFICATION; AND (2) INSTALLATION OF SWITCH AND FUSE COVERS ON EACH CHANNEL TO ACT AS A PHYSICAL BARRIER TO PREVENT INADVERTENT ACTIONS. CORRECTIVE ACTIONS FOR THE STEAM DUMP VALVES WILL BE DISCUSSED IN LICENSEE EVENT REPORT 1-90-017-01.

[37]   DIABLO CANYON 2   DOCKET 50-323   LER 90-002 REV 02
UPDATE ON FUEL HANDLING BUILDING VENTILATION SYSTEM INOPERABLE DURING FUEL MOVEMENT DUE TO PERSONNEL ERROR.
EVENT DATE: 031290   REPORT DATE: 061891   NSSS: WE   TYPE: PWR

(NSIC 222335) ON 3/12/90, ACTION B OF TECH SPEC (TS) 3.9.12 WAS EXCEEDED WHEN MOVEMENT OF IRRADIATED FUEL WAS CONDUCTED WITH AN INOPERABLE FUEL HANDLING

BUILDING (FHB) VENTILATION SYSTEM DUE TO TEMPORARY HOSES BLOCKING OPEN FHB PERSONNEL DOORS. AN NRC RESIDENT INSPECTOR IDENTIFIED THE TEMPORARY HOSES BLOCKING OPEN THE FHB DOORS, AND QUESTIONED THE OPERABILITY OF THE VENTILATION SYSTEM. ON 3/15/90, A TEST WAS CONDUCTED TO VERIFY THAT ADEQUATE NEGATIVE PRESSURE WAS MAINTAINED. THE TEST RESULTS WERE UNSATISFACTORY. ON 3/15/90, PLANT OPERATORS DECLARED THE FHB INOPERABLE AND VERIFIED THAT ACTION B. OF TS 3.9.12 WAS SATISFIED. ON 3/15/90, AFTER THE FHB SUPPLY FANS WERE TURNED OFF, THE TEST WAS REPERFORMED SATISFACTORILY, AND THE FHB VENTILATION SYSTEM WAS DECLARED OPERABLE. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR IN THAT AN ENGINEER FAILED TO RECOGNIZE PERSONNEL DOORS AS A PART OF THE FHB VENTILATION BOUNDARY. STP M-5 WAS REVISED TO ENSURE THAT FHB DOORS WILL NOT BE BLOCKED OPEN DURING MOVEMENT OF IRRADIATED FUEL. ALSO, SIGNS WERE POSTED ON DOORS THAT ARE VENTILATION BOUNDARIES, A PROCEDURE WAS DEVELOPED TO AID IN IDENTIFYING DOORS THAT ARE VENTILATION BOUNDARIES, AND A MEMORANDUM REGARDING TRAINING WAS ISSUED TO ALL DEPARTMENT MANAGERS REGARDING FHB VENTILATION BOUNDARIES.

[38] DRESDEN 2 DOCKET 50-237 LER 90-008 REV 01
UPDATE ON FAILURE OF HPCI STEAM LINE HIGH FLOW ISOLATION DIFFERENTIAL PRESSURE TRANSMITTER DUE TO UNKNOWN CAUSE.
EVENT DATE: 082090 REPORT DATE: 060491 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 222293) ON AUGUST 20, 1990 AT 0645 HOURS, WITH UNIT 2 AT 82% RATED CORE THERMAL POWER, DURING PERFORMANCE OF ROUTINE SURVEILLANCE TESTING, DIFFERENTIAL PRESSURE TRANSMITTER (DPT) 2352, WHICH MONITORS HIGH PRESSURE COOLANT INJECTION (HPCI) STEAM SUPPLY LINE FLOW, WAS FOUND TO BE OUTSIDE OF THE ESTABLISHED TOLERANCE LIMITS. WITH ONLY ONE OPERABLE INSTRUMENT CHANNEL, WHEN TWO ARE REQUIRED BY THE TECHNICAL SPECIFICATIONS, THE HPCI SYSTEM WAS DECLARED INOPERABLE, AND A SEVEN-DAY LIMITING CONDITION FOR OPERATION (LCO) WAS ENTERED. THE FAILED TRANSMITTER WAS REPLACED AND THE LCO TERMINATED ON AUGUST 22, 1990. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL, AS THE REDUNDANT TRANSMITTER WAS DETERMINED TO BE OPERABLE, AND WOULD HAVE PROVIDED THE PRIMARY CONTAINMENT ISOLATION SIGNAL. FURTHER TESTING OF THE REMOVED TRANSMITTER BY THE VENDOR WAS INCONCLUSIVE. A SIMILAR EVENT OCCURRED WHEN THE SAME TRANSMITTER FAILED IN 1988, DUE TO A LOSS OF TRANSMITTER FILL OIL. THAT EVENT WAS DOCUMENTED IN LER 88-15, DOCKET 050237.

[39] DRESDEN 2 DOCKET 50-237 LER 91-009
FAILURE OF STANDBY GAS TREATMENT SYSTEM CHARCOAL ADSORBER LEAK TEST DUE TO SEAL LEAKAGE.
EVENT DATE: 051791 REPORT DATE: 060791 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)
VENDOR: BARNEBEY-CHENEY

(NSIC 222227) ON 5/17/91 WITH UNIT 2 AT 88% POWER AND UNIT 3 AT 56% POWER WHILE PERFORMING TECH SPEC (TS) 7500-7, STANDBY GAS TREATMENT (SBGT) SYSTEM CHARCOAL LEAK TEST, THE LEAKAGE AROUND THE CHARCOAL TRAYS OF THE B TRAIN OF SBGT WAS FOUND TO BE 1.15%, EXCEEDING THE TS REQUIREMENT OF < 1% LEAKAGE. POST TESTING MAINTENANCE ON THE B TRAIN REVEALED INADEQUATE SEALING BETWEEN THE CHARCOAL TRAYS AND THE B TRAIN HOUSING, SLIGHT SETTLING OF THE CHARCOAL IN THE TRAYS AND BROKEN SEALING MECHANISMS. THE B TRAIN WAS REPAIRED, SUCCESSFULLY RETESTED, AND RETURNED TO SERVICE. AN AS-FOUND TEST PERFORMED ON THE A TRAIN REVEALED FAILURES ON THE A TRAIN SIMILAR TO THE FAILURES FOUND ON THE B TRAIN. THE A TRAIN WAS PROMPTLY REPAIRED AND RETURNED TO SERVICE. THE GASKETS ON ALL 12 TRAYS IN THE A TRAIN AND ON 2 OF THE TRAYS IN THE B TRAIN WERE REPLACED. CHARCOAL WAS ALSO ADDED TO EACH TRAY TO COMPENSATE FOR ANY SETTLING THAT HAS OCCURRED. THE SEALING MECHANISMS WERE ALSO REPAIRED ON BOTH TRAINS. BOTH TRAINS OF SBGT WERE LEFT AT 0.71 AND 0.57% LEAKAGE RESPECTIVELY. A PRELIMINARY ENGINEERING REVIEW INDICATES THAT 10CFR100 SITE BOUNDARY EXPOSURE LIMITS WOULD NOT HAVE BEEN EXCEEDED UNDER DESIGN BASIS ACCIDENT CONDITIONS; A FURTHER ANALYSIS OF THE SAFETY SIGNIFICANCE WILL BE PROVIDED IN A SUPPLEMENTAL REPORT. PREVIOUS OCCURRENCES: LER 82-003 AND LER 75-021.

[40]   DRESDEN 2   DOCKET 50-237   LER 91-010
LIQUID RADWASTE DISCHARGE COMPOSITE SAMPLE DISCREPANCY DUE TO PROCEDURE DEFICIENCY.
EVENT DATE: 053091   REPORT DATE: 062191   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 222398) DURING AN UPGRADE OF DRESDEN CHEMISTRY PROCEDURES (DCP), IT WAS DETERMINED THAT THE METHOD OF PREPARING THE MONTHLY RIVER DISCHARGE COMPOSITE SAMPLE PER DCP 200 0-8, RIVER DISCHARGE CARD, FAILED TO COMPLY WITH TECHNICAL SPECIFICATION 4.8.3. NOTATION 2. THE COMPOSITE SAMPLE HAD PREVIOUSLY BEEN PREPARED BY SAVING A MILLILITER SAMPLE ALIQUOT FROM EACH TANK DISCHARGED REGARDLESS OF THE VOLUME OF THE DISCHARGE. TECH SPEC 4.8.3. NOTATION 2 REQUIRES A PROPORTIONATE SAMPLE FOR EACH TANK DISCHARGED BASED ON THE VOLUME OF THE DISCHARGE. IN THIS WAY THE MONTHLY COMPOSITE SAMPLE WOULD BE MORE REPRESENTATIVE OF THE TOTAL VOLUME DISCHARGED DURING THE MONTH. THE ROOT CAUSE OF THIS DISCREPANCY WAS ATTRIBUTED TO PROCEDURE DEFICIENCY. NO RIVER DISCHARGES WERE PERMITTED UNTIL THE CHEMISTRY DEPARTMENT IMPLEMENTED THE NECESSARY PROCEDURE CHANGES TO ENSURE THAT A PROPORTIONATE SAMPLE BASED ON THE VOLUME OF THE DISCHARGE FROM EACH TANK IS TO PREPARE THE MONTHLY COMPOSITE SAMPLE. THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL IN THAT REVIEW OF SITE SAMPLE ANALYSIS DATA INDICATES SIGNIFICANT MARGIN TO THE MAXIMUM ALLOWABLE LIMITS. THIS IS THE FIRST OCCURRENCE OF THIS TYPE AT DRESDEN STATION.

[41]   DRESDEN 3   DOCKET 50-249   LER 91-003
INOPERABLE TORUS WIDE RANGE LEVEL TRANSMITTERS DUE TO UNKNOWN CAUSE.
EVENT DATE: 060591   REPORT DATE: 070191   NSSS: GE   TYPE: BWR
VENDOR: ITT-BARTON

(NSIC 222400) ON 6/5/91 AT 2300 WITH UNIT 3 AT 83% CORE THERMAL POWER, A NUCLEAR STATION OPERATOR (NSO) DISCOVERED THE PRESSURE SUPPRESSION CHAMBER (TORUS) WIDE RANGE LEVEL INDICATORS 3-1640-10A AND 3-1640-10B WERE READING APPROX. 13 FT. THIS CONFLICTED WITH TORUS NARROW RANGE INDICATOR 3-1602-3. THE TORUS NARROW RANGE INDICATOR HAS A SCALE FROM -20 TO 120 INCHES AND ITS READING ON THIS BAND WAS APPROX. 0 INCHES, WHICH CORRESPONDS TO APPROX. 15 FT. THE TORUS NARROW RANGE LOCAL SIGHTGLASS LEVEL WAS PROMPTLY VERIFIED AT APPROX. 15 FT. THE INSTRUMENT MAINTENANCE DEPARTMENT (IMD) BACKFLUSHED THE ENSUING LINES OF THE TORUS WIDE RANGE LEVEL TRANSMITTERS 3-1641-5A AND 3-1641-5B; NO CHANGE IN INDICATION WAS NOTED. DRESDEN INSTRUMENT SURVEILLANCE (DIS) 1600-17, TORUS WIDE RANGE LEVEL TRANSMITTER CALIBRATION AND MAINTENANCE INSPECTION, WAS THEN PERFORMED TO CALIBRATE THE TRANSMITTERS. WHILE SUBSEQUENTLY INVESTIGATING THE EVENT, THE SYSTEM ENGINEER LEARNED THAT THE TRANSMITTERS HAD APPARENTLY DRIFTED OUT OF CALIBRATION PRIOR TO THE NSO'S DISCOVERY, MEANING THAT THE 7 DAY TECH SPEC TABLE 3.2.6 LIMITING CONDITION FOR OPERATION WITH INOPERABLE TORUS WIDE RANGE LEVEL INSTRUMENTS HAD BEEN EXCEEDED. THE ROOT CAUSE OF THE INSTRUMENT DRIFT IS UNKNOWN; A SUPPLEMENTAL REPORT WILL BE PROVIDED. SAFETY SIGNIFICANCE WAS MINIMAL DUE TO AVAILABILITY OF THE NARROW RANGE INSTRUMENTATION.

[42]   FERMI 2   DOCKET 50-341   LER 91-010
FAILURE OF CONTROL ROOM VENTILATION CHARCOAL FILTER SURVEILLANCE.
EVENT DATE: 051991   REPORT DATE: 061891   NSSS: GE   TYPE: BWR

(NSIC 222341) DURING AN ANALYSIS OF THE CONTROL ROOM EMERGENCY FILTRATION SYSTEM'S EMERGENCY MAKE-UP UNIT CHARCOAL SAMPLE, IT WAS DISCOVERED THAT THE SAMPLE NO LONGER MET THE TECHNICAL SPECIFICATION 4.7.2.C.2 SURVEILLANCE REQUIREMENT CRITERIA FOR METHYL IODIDE PENETRATION OF LESS THAN 1%. AS A RESULT, THE SYSTEM WAS DECLARED INOPERABLE ON MAY 19, 1991 AT 1245 HOURS. THERE ARE POTENTIAL CONTRIBUTING FACTORS TO THIS FAILURE. THIS WAS THE CHARCOAL FILTER INSTALLED AT START UP OF THE UNIT AND, THEREFORE, HAD PROVIDED SEVERAL YEARS OF SERVICE. CHARCOAL EFFICIENCY WILL DEGRADE OVER TIME. FERMI 2 ESTABLISHED A MORE CONSERVATIVE TESTING CONDITION THIS TIME. THE TEST TEMPERATURE WAS LOWERED WHICH WAS REFLECTED IN A REDUCTION IN THE METHYL IODIDE REMOVAL EFFICIENCY. THE CHARCOAL WAS REPLACED AND THE SYSTEM WAS DECLARED OPERABLE. FERMI 2 WILL CONTINUE TO COMPLY WITH TECHNICAL SPECIFICATIONS WHICH ARE INTENDED TO DETECT THIS CONDITION.

[43] FERMI 2 DOCKET 50-341 LER 91-011
CONTROL CENTER HEATING VENTILATION AND AIR CONDITIONING SHIFTS TO RECIRCULATION MODE DUE TO SPURIOUS SPIKE OF RADIATION MONITOR.
EVENT DATE: 052391 REPORT DATE: 062191 NSSS: GE TYPE: BWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 222344) AT 0009 HOURS, ON MAY 23, 1991, THE CONTROL CENTER HEATING VENTILATION AND AIR CONDITIONING SYSTEM (CCHVAC) SHIFTED INTO THE RECIRCULATION MODE AND ANNUNCIATOR 3D45, "CONT. CENTER MAKEUP AIR RADN MONITOR UPSCALE TRIP", WAS RECEIVED. IN ACCORDANCE WITH THE ALARM RESPONSE PROCEDURE (ARP) AN OPERATOR WAS DISPATCHED TO INVESTIGATE THE 3D45 ANNUNCIATOR ALARM. THE OPERATOR OBSERVED THE DIVISION II RADIATION MONITOR (MON) TRIP 2 AND TRIP 1 LIGHTS ON. THE RADIATION MONITOR WAS READING 100 COUNTS PER MINUTE (CPM) AND ITS TRIP POINT IS 281 CPM. THE MONITOR WAS RESET AT 0011 HOURS. APPROXIMATELY 8 SECONDS LATER, THE MONITOR SPIKED TO 1000 CPM THEN SLOWLY DRIFTED BACK DOWN TO A NORMAL READING OF APPROXIMATELY 60 CPM. THE CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE A SPURIOUS SPIKE OF RADIATION MONITOR D11-K813. TROUBLESHOOTING WAS UNABLE TO DETERMINE THE CAUSE OF THE SPIKE. ENGINEERING PERSONNEL WERE CONTACTED FOR CONCURRENCE AND AGREED WITH THE ASSESSMENT MADE BY I&C MAINTENANCE. CCHVAC WAS RESTORED TO NORMAL OPERATION AT 0143 HOURS IN ACCORDANCE WITH THE ARP AND SYSTEM OPERATING PROCEDURE 23.413, "CONTROL CENTER HVAC SYSTEM".

[44] FERMI 2 DOCKET 50-341 LER 91-012
INTERMEDIATE RANGE MONITOR INOPERATIVE TRIP DUE TO DAMAGED FUSE HOLDER.
EVENT DATE: 052591 REPORT DATE: 062191 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 222345) ON MAY 25, 1991, A HALF SCRAM SIGNAL WAS RECEIVED FROM REACTOR PROTECTION SYSTEM (RPS) DIVISION I. SINCE RPS DIVISION II HAD A HALF SCRAM SIGNAL INSERTED FOR RELAY WORK, A FULL SCRAM SIGNAL OCCURRED. THE PLANT WAS IN A REFUELING OUTAGE WITH ALL CONTROL RODS ALREADY FULLY INSERTED. AN INVESTIGATION DETERMINED THE HALF SCRAM SIGNAL FROM RPS DIVISION I WAS DUE TO AN INOPERATIVE TRIP OF AN INTERMEDIATE RANGE MONITOR (IRM). THE CAUSE OF THIS EVENT WAS A DAMAGED FUSE HOLDER WHICH DID NOT ALLOW PROPER SEATING OF THE FUSE. THE DAMAGED FUSE HOLDER WAS REPLACED AND THE FUSE WAS PROPERLY SEATED. AN INSPECTION OF OTHER FUSES IN THE IRM SYSTEM WAS CONDUCTED. REQUIRED READING OF THIS EVENT HAS BEEN ISSUED TO APPROPRIATE PERSONNEL.

[45] FITZPATRICK DOCKET 50-333 LER 90-012 REV 01
UPDATE ON ENGINEERED SAFETY FEATURE ACTUATION DUE TO LOW VOLTAGE TRIP OF POWER SUPPLY TO REACTOR PROTECTION SYSTEM BUS DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 091090 REPORT DATE: 061991 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 222338) THE PLANT WAS OPERATING AT FULL POWER ON 9/10/90. AT 9:38 A.M. A LOW VOLTAGE CONDITION ON THE OUTPUT FROM THE MOTOR GENERATOR (MG) POWER SUPPLY TO THE B REACTOR PROTECTION SYSTEM (RPS) (JC) RESULTED IN TRIPPING OF THE RPS BUS ELECTRICAL PROTECTIVE ASSEMBLY (EPA) UNDERVOLTAGE RELAY AND DEENERGIZING OF THE RPS BUS. AN AUTOMATIC HALF-SCRAM RESULTED, TOGETHER WITH ISOLATION OF THE REACTOR WATER CLEAN-UP SYSTEM (CE), REACTOR BUILDING (NG) VENTILATION (VA), CONTAINMENT AIR DILUTION (CAD) (LK), DRYWELL EQUIPMENT AND FLOOR DRAIN PUMPS (WK), AND CONTAINMENT SAMPLE (KN) SYSTEMS. THE RPS WAS SWITCHED TO ITS ALTERNATE POWER SUPPLY AND THE ISOLATIONS WERE RESET. THE LOW VOLTAGE CONDITION RESULTED FROM DRIFT OF THE MG SET VOLTAGE REGULATOR. THE DOWNWARD DRIFT OF THE OUTPUT VOLTAGE TO THE RANGE OF THE TRIP SETPOINT WAS NOT RECOGNIZED BY THE OPERATORS DURING THEIR SHIFT SURVEILLANCE BECAUSE A DEFICIENCY IN THE DAILY SURVEILLANCE TEST PROCEDURE SPECIFIED AN EXCESSIVELY WIDE RANGE FOR ACCEPTABLE OUTPUT VOLTAGE. CORRECTIVE ACTION INCLUDED REPLACEMENT OF VOLTAGE REGULATOR BOARDS, THE EPA ASSEMBLY CIRCUIT BREAKER, AND IMPROVEMENTS TO THE SURVEILLANCE PROCEDURES WHICH DESCRIBE THE MONITORING OF OUTPUT VOLTAGE. VENDOR ANALYSIS CONCLUDED THAT THE VOLTAGE REGULATOR AND CIRCUIT BREAKER WERE OPERABLE AND DID NOT CONTRIBUTE TO THE EVENT.

[46] FITZPATRICK DOCKET 50-333 LER 91-006
MANUAL REACTOR SHUTDOWN DUE TO INOPERABILITY OF BOTH LOW PRESSURE COOLANT INJECTION SUBSYSTEMS DUE TO MECHANICAL FAILURE OF ONE VALVE IN EACH OF THE TWO SYSTEMS.
EVENT DATE: 050791 REPORT DATE: 060691 NSSS: GE TYPE: BWR
VENDOR: L' .RQUE CORP.
POWELL, WILLIAM COMPANY, THE

(NSIC 222219) BETWEEN 0400 AND 0500 ON 5/7/91 A MONTHLY TECHNICAL SPECIFICATION CHECK FOUND ONE OF THE TWO PRIMARY CONTAINMENT ISOLATION VALVES IN EACH OF THE TWO INDEPENDENT RESIDUAL HEAT REMOVAL (BO) LOW PRESSURE COOLANT INJECTION (LPCI) (BO) SUBSYSTEMS TO BE INOPERABLE. A VALVE OPERATOR TORQUE SWITCH TRIPPED IN BOTH THE OPEN AND CLOSE DIRECTIONS PREVENTING BOTH FULL CLOSURE AND FULL OPENING ONE VALVE. THE OTHER VALVE HAD EXCESSIVE SEAT LEAKAGE. THE INOPERABILITY OF BOTH RHR-LPCI SYSTEMS REQUIRED A REACTOR SHUTDOWN WITHIN 24 HOURS WHICH WAS INITIATED AND AN UNUSUAL EVENT DECLARED AT 1237. THE REACTOR WAS MANUALLY SCRAMMED AT 1820. COLD SHUTDOWN CONDITION WAS ACHIEVED ON 5/8/91 AT 0330. THE UNUSUAL EVENT WAS TERMINATED AT 0400. IN LPCI LOOP B THE THREADS OF THE GATE VALVE STEM NUT IN THE MOTOR OPERATOR WERE WORN AND BROKEN CAUSING THE VALVE TO LOCK IN A PARTIALLY OPEN POSITION. THE STEM NUT WAS REPLACED AND LPCI LOOP B RESTORED TO SERVICE AT 1410 ON 5/12/91. THE STEM OF AN ANGLE GLOBE THROTTLE VALVE IN LOOP A WAS SEVERED INSIDE THE VALVE BODY. THE DISC, DISC GUIDES, AND SEAT WERE SEVERELY DAMAGED. THE VALVE WAS REMOVED FROM THE SYSTEM. THE PLANT REMAINS SHUTDOWN UNTIL THE VALVE IS REPAIRED. ROOT CAUSE FAILURE ANALYSIS IS IN PROGRESS FOR BOTH VALVES.

[47] FT. CALHOUN 1 DOCKET 50-285 LER 91-004 REV 01
UPDATE ON OFFSITE POWER LOW SIGNAL OUTSIDE DESIGN BASIS.
EVENT DATE: 021291 REPORT DATE: 060191 NSSS: CE TYPE: PWR

(NSIC 222317) THE OFFSITE POWER LOW SIGNAL (OPLS) PROVIDES DEGRADED VOLTAGE PROTECTION TO SAFEGUARDS EQUIPMENT; WHEN A DEGRADED VOLTAGE CONDITION EXISTS CONCURRENT WITH A SAFETY INJECTION ACTUATION SIGNAL (SIAS), THE OPLS SIGNAL ISOLATES SAFEGUARDS BUSES 1A3 AND 1A4 FROM THEIR OFFSITE POWER SUPPLY (161 KV) AND INITIATES AUTOMATIC ACTIONS TO LOAD THE SAFEGUARDS EQUIPMENT ONTO THE EMERGENCY DIESEL GENERATORS. ENGINEERING ANALYSIS REVEALED THAT, DURING A POSTULATED ACCIDENT, THE VOLTAGE SUPPLIED TO SOME 480 V SAFEGUARDS LOADS COULD DEGRADE TO AS LOW AS APPROXIMATELY 87.5% OF RATED VOLTAGE WITHOUT OPLS BEING ACTUATED. SINCE THE POSSIBILITY EXISTED FOR VOLTAGE TO BE LOWER THAN THE RECOMMENDED 90% OF RATED VOLTAGE FOR CERTAIN 480 V SAFEGUARDS LOADS WITHOUT AN OPLS ACTUATION, MANAGEMENT DETERMINED ON FEBRUARY 12, 1991 THAT THE PLANT WAS OUTSIDE OF ITS DESIGN BASIS. CORRECTIVE ACTIONS INCLUDE ADMINISTRATIVE CONTROLS OF EQUIPMENT CONFIGURATIONS AND BUS LOADINGS, AS WELL AS RESETTING OF THE OPLS SETPOINTS. A MODIFICATION WILL BE IMPLEMENTED TO ALTER EXISTING LOGIC CIRCUITRY SUCH THAT, UPON RECEIPT OF AN SIAS, LARGE 4160V MOTORS/EQUIPMENT WHICH ARE NOT REQUIRED TO MITIGATE CONSEQUENCES OF AN ACCIDENT WILL BE LOAD SHED.

[48] FT. CALHOUN 1 DOCKET 50-285 LER 91-006
VENTILATION ISOLATION ACTUATION SIGNAL ACTUATION DUE TO INAPPROPRIATE ACTION.
EVENT DATE: 043091 REPORT DATE: 060491 NSSS: CE TYPE: PWR

(NSIC 222206) ON APRIL 30, 1991, AT 0937 HOURS, AN ELECTRICIAN WAS REMOVING A LABEL FROM INSIDE A CONTROL ROOM BREAKER CABINET AND INADVERTENTLY TRIPPED A NORMALLY CLOSED BREAKER. REALIZING WHAT HE HAD DONE, THE ELECTRICIAN IMMEDIATELY RECLOSED THE BREAKER, CAUSING THE VENTILATION ISOLATION ACTUATION SIGNAL (VIAS) TO ACTUATE DUE TO A RESULTANT SPIKE OF RADIATION MONITORS. THE VIAS CAUSED AN UNPLANNED ACTUATION OF SOME ENGINEERED SAFETY FEATURES. THIS EVENT RESULTED FROM INAPPROPRIATE ACTION BY THE ELECTRICIAN AFTER REALIZING THAT HE HAD ACCIDENTALLY TRIPPED THE BREAKER. CORRECTIVE ACTIONS INCLUDE POLICY REVISIONS TO CLEARLY STATE THAT ALL PERSONNEL ARE REQUIRED TO CONTACT THE SHIFT SUPERVISOR OR CONTROL ROOM OPERATORS PRIOR TO CORRECTING ANY UNPLANNED SITUATION.

[49] FT. CALHOUN 1 DOCKET 50-285 LER 91-010
AUXILIARY STEAM PIPING IN ROOM 57 OUTSIDE DESIGN BASIS.
EVENT DATE: 051791 REPORT DATE: 070291 NSSS: CE TYPE: PWR

(NSIC 222495) ON MAY 17, 1991, AT 1420 HOURS, PLANT MANAGEMENT DETERMINED THAT ROOM 57 (THE UPPER ELECTRICAL PENETRATION ROOM), WAS OUTSIDE THE DESIGN BASIS OF THE PLANT AS A RESULT OF DISCOVERING THE PRESENCE OF HIGH ENERGY AUXILIARY STEAM (AS) PIPING IN THE ROOM. THIS PIPING HAD NOT BEEN IDENTIFIED DURING THE INITIAL PLANT HIGH ENERGY LINE BREAK (HELB) EVALUATION. THE MAJOR CONCERN FOR THIS CONDITION IS THE IMPACT ON THE OPERABILITY OF SAFETY RELATED ELECTRICAL EQUIPMENT IN THE HIGH HUMIDITY/TEMPERATURE ENVIRONMENT THAT WOULD OCCUR BASED ON A CRITICAL CRACK IN THE TWO-INCH AUXILIARY STEAM LINE IN ROOM 57. THE CAUSE FOR THIS CONCERN WAS THE LACK OF ATTENTION TO DETAIL DURING ORIGINAL DRAWING REVIEW AND PHYSICAL WALKDOWNS PRIOR TO PERFORMING THE PLANT HELB EVALUATION IN 1973. THIS RESULTED IN FAILURE TO IDENTIFY THE SUBJECT PIPING AND TO TAKE PROPER CORRECTIVE ACTIONS. THE IMMEDIATE ACTION TAKEN WAS TO ISOLATE THE REDUNDANT TRAINS OF EQUIPMENT BY CLOSING THE NORMALLY-OPEN FIRE DAMPERS BETWEEN THOSE AREAS AND ROOM 57, THEN TO ISOLATE THE STEAM SUPPLY TO THE AS PIPING IN THE AFFECTED ROOM. LONG TERM CORRECTIVE ACTIONS WILL INCLUDE AN ENGINEERING ANALYSIS TO EVALUATE THE REMAINDER OF THE AUXILIARY STEAM SYSTEM FOR SIMILAR HELB CONCERNS AND MODIFICATION TO AS SYSTEM LINES IN ROOM 57.

[50] GRAND GULF 1 DOCKET 50-416 LER 91-003
CONTAINMENT PENETRATIONS NOT SURVEILLED.
EVENT DATE: 042591 REPORT DATE: 052891 NSSS: GE TYPE: BWR

(NSIC 222139) A CONTAINMENT PENETRATION WAS DETERMINED TO BE IN NONCOMPLIANCE WITH LOCAL LEAK RATE TESTING REQUIREMENTS. THE HORIZONTAL FUEL TRANSFER TUBE DESIGN INCORPORATES A BELLOWS SEAL CLOSURE ARRANGEMENT. THE TRANSFER TUBE CLOSURE HATCH FEATURES DOUBLE O-RINGS AND TEST CONNECTIONS. ON THE INITIAL REVIEW OF CONTAINMENT PENETRTIONS AND ON SUBSEQUENT REVIEWS, THE CLOSURE HATCH SEAL WAS INTERPRETED TO BE THE ONLY SEAL ARRANGEMENT FOR THIS PENETRATION SUBJECT TO 10CFR 50 APPENDIX J TESTING. DUE TO THE COMPLEX PENETRATION ARRANGEMENT, THE BELLOWS ASSEMBLY WAS NOT IDENTIFIED AS BEING EXPOSED TO CONTAINMENT PRESSURE AND THEREFORE WAS NOT INCLUDED IN THE SURVEILLANCE PROCEDURE FOR LEAK RATE TESTING. PLANT DOCUMENTS AFFECTED BY THIS DISCOVERY WILL BE REVISED TO IDENTIFY THE BELLOWS AS PART OF THE CONTAINMENT PENETRATION BOUNDARY. OTHER CONTAINMENT PENETRATIONS WITH BELLOWS ASSEMBLIES AND OTHER ATYPICAL SEALING ARRANGEMENTS WERE REVIEWED TO DETERMINE IF ANY OTHER SIMILAR CONTAINMENT ISOLATION BOUNDARIES HAD BEEN OVERLOOKED IN DEVELOPMENT OF THE SURVEILLANCE PROCEDURE FOR LEAK RATE TESTING. NO ADDITIONAL DEFICIENCIES WERE FOUND. NO SAFETY RELATED EQUIPMENT WAS AFFECTED ADVERSELY BY THIS CONDITION. THE HEALTH AND SAFETY OF THE GENERAL PUBLIC WAS NOT AFFECTED.

[51] GRAND GULF 1 DOCKET 50-416 LER 91-004
REACTOR SCRAM DUE TO LOSS OF FEEDWATER.
EVENT DATE: 061191 REPORT DATE: 071091 NSSS: GE TYPE: BWR
VENDOR: BAILEY METER COMPANY

(NSIC 222465) ON JUNE 11, 1991, A CONTROLLED SHUTDOWN WAS IN PROGRESS FOR MAINTENANCE ON THE MECHANICAL SEALS FOR BOTH REACTOR RECIRCULATION WATER PUMPS. IN ACCORDANCE WITH APPROVED PLANT PROCEDURES, THE REACTOR RECIRCULATION WATER PUMPS WERE SHIFTED FROM FAST TO SLOW SPEED. THIS CAUSED A SWELL IN REACTOR WATER LEVEL WHICH RESULTED IN A DECREASE IN FEEDWATER AND CONDENSATE FLOW. THE CONDENSATE SYSTEM FLOW RATE DECREASED TO THE MINIMUM FLOW SETPOINT, BUT THE MINIMUM FLOW VALVE FAILED TO RESPOND TO THE LOW FLOW CONDITION. THE FAILURE RESULTED IN A LOSS OF FEEDWATER AND A SUBSEQUENT AUTOMATIC REACTOR SCRAM. THE FAILURE IS ATTRIBUTED TO A FAILED PNEUMATIC RELAY INSIDE THE VALVE POSITIONER. THE MOST PROBABLE CAUSE FOR THE RELAY FAILURE IS EXCESSIVE VIBRATION EXPERIENCED DURING MINIMUM FLOW CONDITIONS. THE POSITIONER WAS REPLACED AND RETESTED SATISFACTORILY. AN EVALUATION WILL BE PERFORMED TO DETERMINE METHODS OR MODIFICATIONS TO PRECLUDE VALVE FAILURE DUE TO VIBRATION. IN THE INTERIM, A TASK WILL BE IMPLEMENTED TO PERFORM A FUNCTIONAL TEST AND A VISUAL INSPECTION OF THE POSITIONER PRIOR TO CONTROLLED SHUTDOWNS AND START-UPS. ADDITIONALLY, A TASK

WILL BE GENERATED TO REPLACE THE VALVE POSITIONER EACH REFUELING OUTAGE UNTIL A MORE SUITABLE RESOLUTION IS DETERMINED. BASED ON REVIEWS OF THE EVENT, ALL SAFETY SYSTEMS FUNCTIONED AS DESIGNED.

[52] HATCH 2 DOCKET 50-366 LER 91-001 REV 01
UPDATE ON INSTRUMENT TRIP SETPOINTS DETERMINED TO BE OUTSIDE TECHNICAL SPECIFICATIONS.
EVENT DATE: 012991 REPORT DATE: 061491 NSSS: GE TYPE: BWR

(NSIC 222356) ON 1/29/91 AT APPROX. 0915, WHILE IN THE RUN MODE AT AN APPROX. POWER LEVEL OF 3436, IT WAS DETERMINED THE SETPOINTS FOR UNIT 2 CONDENSATE STORAGE TANK (CST, EIIS CODE KA) LEVEL SWITCHES 2E41-N002 AND 2E41-N 003 WERE NOT IN COMPLIANCE WITH THE REQUIREMENTS OF UNIT 2 TECH SPECS TABLE 3.3.3-2, ITEM 3.C. THE SWITCHES, WHICH CAUSE HIGH PRESSURE COOLANT INJECTION (HPCI, EIIS CODE BJ) SYSTEM SUCTION SOURCE TRANSFER FROM THE CST TO THE SUPPRESSION POOL (EIIS CODE BT) ON LOW CST WATER LEVEL, WERE NOT SET TO INITIATE THE TRANSFER WHEN 10,000 USABLE GALLONS OF WATER WERE AVAILABLE TO THE HPCI SYSTEM AS INTENDED BY THE TS. AT THE TIME OF EVENT DISCOVERY, THE HPCI SYSTEM WAS ALIGNED TO TAKE SUCTION FROM THE SUPPRESSION POOL AND REMAINED IN THIS ALIGNMENT UNTIL THE CST WATER LEVEL SWITCH SETPOINTS COULD BE RAISED. THE CAUSE OF THIS EVENT WAS LESS-THAN-ADEQUATE DESIGN DOCUMENTATION. ALTHOUGH THE LEVEL SWITCH SETPOINTS WERE DESIGNED SUCH THAT 10,000 GALLONS OF WATER REMAINED IN THE CST AT THE TIME OF THE SUCTION SOURCE TRANSFER, DESIGN DOCUMENTS DID NOT REQUIRE 10,000 GALLONS OF WATER TO BE AVAILABLE TO THE HPCI SYSTEM. CORRECTIVE ACTIONS INCLUDE RAISING THE CST LEVEL SWITCH SETPOINTS AND INITIATING REVISIONS TO APPROPRIATE PORTIONS OF THE TECH SPECS AND FSAR. ALL SETPOINT CHANGES COVERED UNDER DCR 84-138 HAVE BEEN REVIEWED.

[53] HATCH 2 DOCKET 50-366 LER 91-013
SPURIOUS ELECTRICAL SPIKING IN NEUTRON MONITORING SYSTEM RESULTS IN REACTOR PROTECTION SYSTEM ACTUATION.
EVENT DATE: 050691 REPORT DATE: 060391 NSSS: GE TYPE: BWR

(NSIC 222170) ON 5/6/91, AT 0333 CDT, UNIT 2 WAS IN A REFUELING OUTRAGE WITH THE VESSEL FLOODED AND THE CORE PARTIALLY RELOADED WITH FUEL. AT THAT TIME, A SPURIOUS ELECTRICAL SPIKE IN THE NEUTRON MONITORING SYSTEM (NMS, EIIS CODE IG) RESULTED IN AN ACTUATION OF THE REACTOR PROTECTION SYSTEM (RPS, EIIS CODE JE). ALL CONTROL RODS WERE FULLY INSERTED INTO THE CORE AT THE TIME THE SIGNAL WAS RECEIVED. THEREFORE, NO CONTROL ROD MOTION OCCURRED AS A RESULT OF THE SIGNAL. NO OTHER NUCLEAR INSTRUMENTATION SHOWED UNUSUAL ACTIVITY. THEREFORE, THE AFFECTED INSTRUMENT WAS BYPASSED AND THE SCRAM WAS RESET BY LICENSED PLANT OPERATIONS PERSONNEL. NO OTHER AUTOMATIC ACTUATIONS RESULTED FROM THIS EVENT. THE CAUSE OF THE EVENT WAS SPURIOUS ELECTRICAL SPIKING OF A NEUTRON MONITORING INSTRUMENT. GIVEN THE PLANT CONFIGURATION AT THE TIME OF THE EVENT, ANY NMS TRIP SIGNAL WOULD HAVE CAUSED, PER DESIGN, AN ACTUATION OF THE RPS. THE CAUSE OF THE SPURIOUS ELECTRICAL SPIKING COULD NOT BE DETERMINED. NO TESTING ON THIS INSTRUMENT WAS IN PROGRESS. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED BYPASSING THE AFFECTED NEUTRON MONITORING INSTRUMENT, RESETTING THE SCRAM, AND RETURNING THE AFFECTED INSTRUMENT TO SERVICE AFTER ITS ERRATIC OPERATION HAD SUBSIDED.

[54] HATCH 2 DOCKET 50-366 LER 91-014
LESS THAN ADEQUATE STORAGE RACK DESIGN EVALUATION RESULTS IN TECHNICAL SPECIFICATIONS NON-COMPLIANCE.
EVENT DATE: 050891 REPORT DATE: 060591 NSSS: GE TYPE: BWR

(NSIC 222171) ON 5/8/91 AT 0930 CDT, UNIT 2 WAS IN THE REFUELING MODE WITH THE CORE COMPLETELY LOADED WHEN A DISCREPANCY BETWEEN UNIT 2 TECHNICAL SPECIFICATION (TS) 3.9.10 AND THE DESIGN OF THE HIGH DENSITY FUEL STORAGE SYSTEM (HDFSS) (EIIS CODE DB) RACKS IN THE UNIT 2 SPENT FUEL POOL WAS IDENTIFIED. TS 3.9.10 REQUIRES AT LEAST 23 FT OF WATER BE MAINTAINED OVER THE TOP OF IRRADIATED FUEL ASSEMBLIES SEATED IN THE STORAGE RACKS. HOWEVER, RACK DESIGN IS SUCH THAT 23 FT OF WATER CAN BE MAINTAINED ONLY ABOVE THE TOP OF THE ACTIVE FUEL REGION OF ASSEMBLIES SEATED IN THE RACKS. THE TOP OF EACH FUEL ASSEMBLY, I.E., TOP OF THE UPPER TIE PLATE, IS APPROXIMATELY 12.125 IN. ABOVE THE TOP OF THE ACTIVE FUEL (TAF). GE

WAS CONTACTED TO ASSESS THE SAFETY SIGNIFICANCE OF THE DISCREPANCY AND PROVIDE A BASIS FOR THE 23-FT REQUIREMENT. GE INDICATED THE DISCREPANCY WAS NOT A SAFETY CONCERN, BUT WAS UNABLE TO DOCUMENT THE 23-FT DEPTH WAS INTENDED TO BE MEASURED WITH RESPECT TO TAF. PLANT HATCH CORPORATE SUPPORT PERSONNEL ANALYZED THE IMPACT OF THIS DISCREPANCY AND CONCLUDED THAT A SAFETY CONCERN DID NOT EXIST. THE CAUSE OF THIS EVENT IS A LESS-THAN-ADEQUATE DESIGN CHANGE SAFETY EVALUATION (SE) IN THAT THE SE FOR THE HDFSS DESIGN CHANGE DEVELOPED IN THE EARLY 1980S DID NOT ADDRESS COMPLIANCE WITH TS 3.9.10. ON 5/20/91, LCO 2-91-424 WAS INITIATED TO ENSURE COMPLIANCE WITH ACTION STATEMENT 3.9.10.

[55] HATCH 2 DOCKET 50-366 LER 91-015
COMPONENT FAILURE RESULTS IN AN UNPLANNED ESF ACTUATION.
EVENT DATE: 052691 REPORT DATE: 062491 NSSS: GE TYPE: BWR
VENDOR: ABEX CORPORATION

(NSIC 222495) ON 5/26/91 AT 0742 CDT, UNIT 2 WAS IN THE COLD SHUTDOWN MODE WITH PREPARATIONS IN PROGRESS FOR STARTUP FOLLOWING A REFUELING OUTAGE. AT THAT TIME, A FULL GROUP 1 PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS, EIIS CODE JM) ISOLATION SIGNAL WAS RECEIVED WHEN THE FOUR MAIN TURBINE STOP VALVES UNEXPECTEDLY OPENED WHEN THE CONTROL INPUT LOGIC BOARD FOR STOP VALVE NO. 2 WAS PULLED DURING SET-UP AND CHECK-OUT OF THE ELECTROHYDRAULIC CONTROL SYSTEM (EHC, EIIS CODE TG). THE STOP VALVES IN THE OPEN POSITION, IN CONJUNCTION WITH LOW CONDENSER VACUUM, RESULT IN A GROUP 1 PCIS ISOLATION SIGNAL. BECAUSE THE UNIT WAS IN A REFUELING OUTAGE, THERE WAS NO VACUUM IN THE CONDENSER; THEREFORE, WHEN THE STOP VALVES OPENED, THE LOGIC FOR A GROUP 1 ISOLATION WAS SATISFIED AND A FULL GROUP 1 ISOLATION SIGNAL WAS RECEIVED. THE PCIS FUNCTIONED AS DESIGNED. GROUP 1 PRIMARY CONTAINMENT ISOLATION VALVE (PCIV) 2B31-F020, REACTOR WATER SAMPLE LINE ISOLATION VALVE, CLOSED. OTHER GROUP 1 PCIVS WERE ALREADY CLOSED. THE CAUSE OF THIS EVENT WAS COMPONENT FAILURE. THE SERVO VALVE FOR TURBINE STOP VALVE NO. 2 FAILED IN SUCH A MANNER AS TO CAUSE THE STOP VALVE TO OPEN. PER DESIGN, THE OTHER THREE TURBINE STOP VALVES OPENED WHEN TURBINE STOP VALVE NO. 2 OPENED. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED REPLACING THE FAILED SERVO VALVE AND VERIFYING THE SERVO VALVES FOR THE MAIN TURBINE CONTROL VALVES WERE FUNCTIONING PROPERLY.

[56] HATCH 2 DOCKET 50-366 LER 91-016
PERSONNEL ERROR RESULTS IN MISSED TECHNICAL SPECIFICATIONS SURVEILLANCE.
EVENT DATE: 053091 REPORT DATE: 062891 NSSS: GE TYPE: BWR

(NSIC 222496) ON 5/30/91 AT 0800, WHILE IN THE COLD SHUTDOWN CONDITION WITH REACTOR COOLANT TEMPERATURE AT 170F AND REACTOR PRESSURE. A PLANT ENGINEER REVIEWING PRIMARY CONTAINMENT (PC, EIIS CODE NH) ISOLATION RESPONSE TIME TESTING DATA DISCOVERED THAT NOT ALL DATA HAD BEEN COLLECTED AS REQUIRED BY TECH SPEC (TS) 4.3.2 3 SPECIFICALLY, RESPONSE TIME DATA WERE OMITTED FOR THE PC ISOLATION FUNCTION INITIATED FROM A HIGH PC PRESSURE SIGNAL. THIS SIGNAL RESULTS IN CLOSURE OF THE VACUUM BREAKER ISOLATION VALVES IN THE HPCI (EIIS CODE BJ) SYSTEM AND THE REACTOR CORE ISOLATION COOLING (RCIC, EIIS CODE BN) SYSTEM. THE DATA SHOULD HAVE BEEN COLLECTED DURING THE 1989 AND 1991 REFUELING OUTAGES BUT WAS MISSED DUE TO A PROCEDURAL ERROR. ALL INSTRUMENTS AND LOGIC THAT WERE REQUIRED TO BE TESTED WERE SUCCESSFULLY SURVEILLED PRIOR TO UNIT STARTUP. THEREFORE, ONLY ONE SURVEILLANCE WAS PERFORMED AFTER THE REQUIRED TIME INTERVAL HAD EXPIRED. THIS SURVEILLANCE PERTAINED TO PRESSURE SENSOR 2E11-N094D WHICH SHOULD HAVE BEEN TESTED IN THE 1989 REFUELING OUTAGE. THE ROOT CAUSE OF THIS EVENT IS PERSONNEL ERROR. THE PROCEDURE WRITERS NEGLECTED TO ACCOUNT FOR DIFFERENCES IN THE NUMBER OF LOGIC CHANNELS WHICH COULD AFFECT THE TESTING FREQUENCY. THE TECHNICAL REVIEWERS DID NOT IDENTIFY THE OVERSIGHT; THEREFORE, THE ERROR WAS NOT CORRECTED.

[57] HATCH 2 DOCKET 50-366 LER 91-018
ERROR IN THE FSAR RESULTS IN MISSED TECHNICAL SPECIFICATIONS SURVEILLANCES.
EVENT DATE: 060491 REPORT DATE: 070391 NSSS: GE TYPE: BWR

(NSIC 222498) ON 6/4/91 AT APPROX. 1330, WHILE IN THE RUN MODE AT A POWER LEVEL OF 1218 CMWT (50% RATED THERMAL POWER). IT WAS DETERMINED A LOCAL LEAK RATE TEST OF PRIMARY CONTAINMENT PENETRATION X-228B HAD NOT BEEN PERFORMED AS REQUIRED BY

TECH SPECS SECTION 4.6.1.2.D. LIMITING CONDITION FOR OPERATION (LCO) 2-91-468 WAS INITIATED AT 1330 PER THE REQUIREMENTS OF UNIT 2 TECH SPECS SECTION 3.6.1.1. A LOCAL LEAK RATE TEST OF PENETRATION X-228B WAS PERFORMED SATISFACTORILY ON 6/4/91. LCO 2-91-468 WAS TERMINATED AT 1745 FOLLOWING SUCCESSFUL PERFORMANCE OF THE TEST. ON 6/10/91 AT APPROX. 0930, DURING A FOLLOWUP INVESTIGATION OF THE ABOVE EVENT, IT WAS DETERMINED PENETRATION X-228B HAD NOT BEEN VERIFIED TO BE CLOSED AT LEAST ONCE EVERY 31 DAYS AS REQUIRED BY TS 4.6.1.1. SINCE THE PENETRATION HAD BEEN VERIFIED TO BE CLOSED ON 6/4/91 WHEN THE LEAK RATE TEST HAD BEEN PERFORMED, THE REQUIRED SURVEILLANCE WAS CURRENT AND AN LCO DID NOT HAVE TO BE ENTERED. THE CAUSE OF THESE EVENTS IS AN ERROR IN THE FINAL SAFETY ANALYSIS REPORT (FSAR). THE FSAR INCORRECTLY IDENTIFIED PENETRATION X-228B AS NOT REQUIRING A LOCAL LEAK RATE TEST. THIS ERROR ALSO LED PERSONNEL TO BELIEVE THE REQUIREMENTS OF TS SECTION 4.6.1.1 DID NOT APPLY.

[58] HOPE CREEK 1 DOCKET 50-354 LER 91-008
REACTOR SCRAM ON LOW WATER LEVEL DUE TO PERSONNEL ERROR.
EVENT DATE: 050791 REPORT DATE: 060691 NSSS: GE TYPE: BWR

(NSIC 222380) ON 5/7/91, AT 2102, A REACTOR SCRAM OCCURRED DUE TO A LOW REACTOR WATER LEVEL. CONDITIONS LEADING TO THE SCRAM WERE THE 'A' FEEDWATER LEVEL CONTROL INDICATION ALARMING WITH A HIGH LEVEL, CAUSING THE FEEDPUMPS TO BACK DOWN. THIS WAS FOLLOWED BY THE 'C' CHANNEL ALARMING AT 30" AND A SUBSEQUENT REACTOR SCRAM. ALL CONTROL RODS WERE VERIFIED TO BE INSERTED AND PLANT SYSTEMS RESPONDED AS EXPECTEL, WITH MINOR EXCEPTIONS AS NOTED IN THE TEXT OF THIS REPORT. FOLLOW-UP INVESTIGATION DETERMINED THAT THE MOST PROBABLE CAUSE OF THE EVENT WAS PERSONNEL ERROR WHEN A CONTROLS TECHNICIAN INADVERTENTLY CONNECTED A CURRENT SOURCE TO THE WRONG TRANSMITTER WHILE PERFORMING A SURVEILLANCE TEST. ALTHOUGH THE MOST PROBABLE ROOT CAUSE WAS PERSONNEL ERROR, A LESS THAN ADEQUATE CABINET DESIGN ALSO CONTRIBUTED TO THE EVENT. IMMEDIATE CORRECTIVE ACTIONS INCLUDED COUNSELING THE CONTROLS TECHNICIAN WITH RESPECT TO SELF-VERIFICATION AND ATTENTION TO DETAIL AND IMPLEMENTING A DESIGN CHANGE TO INSTALL TEST SWITCHES WITH INPUT JACKS AND TO LABEL THE INPUT JACKS WITH A CHANNEL DESIGNATOR. THE SURVEILLANCE PROCEDURE WAS REVISED IN ACCORDANCE WITH THE DESIGN CHANGE AND IS BEING REVIEWED FOR ADDITIONAL ENHANCEMENTS.

[59] HOPE CREEK 1 DOCKET 50-354 LER 91-009
FULL INITIATION OF THE REACTOR PROTECTION SYSTEM DUE TO SPIKING OF THE "G" INTERMEDIATE RANGE MONITOR DUE TO EQUIPMENT MALFUNCTION.
EVENT DATE: 050891 REPORT DATE: 060691 NSSS: GE TYPE: BWR

(NSIC 222169) ON 5/8/91 AT 0758, WITH THE REACTOR IN HOT SHUTDOWN (OPERATIONAL CONDITION 3), A FULL REACTOR PROTECTION SYSTEM (RPS) ACTUATION OCCURRED DUE TO ELECTRONIC SPIKING OF THE "G" INTERMEDIATE RANGE MONITOR (IRM) (RPS CHANNEL "A"). A 1/2 SCRAM SIGNAL HAD BEEN MANUALLY INSERTED TO THE "B" RPS CHANNEL ON THE PREVIOUS SHIFT DUE TO THE INOPERABILITY OF VARIOUS IRMS DURING MAINTENANCE AND TESTING ACTIVITIES . WHEN THE "G" IRM SPIKED, BOTH HALVES OF THE RPS LOGIC WERE SATISFIED, AND THE FULL RPS INITIATION OCCURRED. THE SPIKING OF THE "G" IRM HAS BEEN ATTRIBUTED TO EQUIPMENT MALFUNCTION, AS A WORK REQUEST HAD BEEN PREVIOUSLY WRITTEN (4/26/91) AGAINST IRM "G" DUE TO INTERMITTENT ELECTRONIC SPIKING (THE IRM WAS SCHEDULED TO BE REPAIRED AT THE NEXT FORCED OR PLANNED OUTAGE). CORRECTIVE ACTIONS CONSISTED OF PERFORMING REPAIRS TO THE "G" IRM, AS PLANNED, PRIOR TO RETURNING TO POWER OPERATION, AND CONTINUING AN INTERNAL STUDY REGARDING IRM RELIABILITY TO IDENTIFY POTENTIAL FUTURE SYSTEM ENHANCEMENTS.

[60] HOPE CREEK 1 DOCKET 50-354 LER 91-010
AUTOMATIC CLOSURE OF REACTOR CORE ISOLATION COOLING TORUS VACUUM BREAKER INBOARD ISOLATION VALVE DURING FUNCTIONAL TESTING DUE TO UNDETERMINED CAUSES.
EVENT DATE: 051591 REPORT DATE: 061391 NSSS: GE TYPE: BWR

(NSIC 222354) ON MAY 15, 1990, DURING THE PERFORMANCE OF A MONTHLY FUNCTION TEST ON DIVISION 2 DRYWELL PRESSURE INSTRUMENTATION, AN AUTOMATIC CLOSURE OF THE INBOARD REACTOR CORE ISOLATION COOLING (RCIC) TORUS VACUUM BREAKER ISOLATION VALVE (FC-HV-F062) OCCURRED. AFTER VERIFYING THAT THE ISOLATION HAD NOT OCCURRED

FROM A VALID CONDITION, THE SENIOR NUCLEAR SHIFT SUPERVISOR (STNSS, SRO LICENSED) DIRECTED THAT THE ISOLATION BE RESET AND THE VALVE REOPENED. THE MONTHLY FUNCTIONAL TEST WAS THEN PERFORMED AGAIN SUCCESSFULLY. THE RESULTS OF INVESTIGATION OF THE EVENT HAVE BEEN INCONCLUSIVE IN DETERMINING A CAUSE OF THE ISOLATION; FURTHER TESTING AND INSPECTION OF THE VALVE LOGIC CONTROL CIRCUITRY WILL BE CONDUCTED AT THE NEXT FORCED OR PLANNED UNIT OUTAGE. A SUPPLEMENT TO THIS REPORT WILL BE SUBMITTED NO LATER THAN 11/15/92 DETAILING THE RESULTS OF THIS TESTING AND INSPECTION.

[61] HOPE CREEK 1 DOCKET 50-354 LER 91-011
BOTH CONTROL ROOM VENTILATION SYSTEM TRAINS INOPERABLE DUE TO SEPARATE AND UNRELATED EQUIPMENT MALFUNCTIONS RESULTING IN ENTERING TECH SPEC 3.0.3.
EVENT DATE: 052291 REPORT DATE: 061991 NSSS: GE TYPE: BWR

(NSIC 222355) ON 5/22/91 AT 131 , TECH SPEC 3.0.3 WAS ENTERED DUE TO THE INOPERABILITY OF BOTH CONTROL ROOM VENTILATION (CRV) SYSTEM UNITS. AT 1252, THE "B" CRV CHILLED WATER SYSTEM TRIPPED. THE "A" CRV CHILLED WATER SYSTEM AUTOMATICALLY STARTED, AS DESIGNED, BUT TRIPPED 22 SECONDS LATER. EFFORTS TO RESTART THE "A" CRV CHILLED WATER SYSTEM WERE UNSUCCESSFUL AND AN ATTEMPT WAS MADE TO RESTART THE "B" CRV CHILLED WATER SYSTEM, WITH THE COMPRESSOR VANE CONTROL IN "AUTOMATIC" (NORMAL STARTUP ALIGNMENT). THE "B" CRV CHILLED WATER SYSTEM COMPRESSOR STARTED, HOWEVER, FAILED TO LOAD AND WAS ORDERED STOPPED. AT 1320, BOTH UNITS WERE DECLARED INOPERABLE, AND AFTER ALLOWING THE "B" SYSTEM TO COOL, THE "B" CRV CHILLED WATER UNIT WAS SUCCESSFULLY RESTARTED WITH THE COMPRESSOR VANE CONTROL IN "MANUAL". AT 1348, COMPRESSOR VANE CONTROL WAS PLACED IN AUTOMATIC, PROPER OPERATION WAS VERIFIED, AND THE UNIT WAS DECLARED OPERABLE. TECH SPEC 3.0.3 WAS EXITED AT THIS TIME. THE CAUSE OF THIS EVENT WAS UNRELATED EQUIPMENT MALFUNCTIONS - THE "B" CRV CHILLED WATER SYSTEM TRIPPING DUE TO AN APPARENTLY SPURIOUS TRIP SIGNAL, AND THE "A" SYSTEM TRIPPING DUE TO A BROKEN TERMINAL ON THE COMPRESSOR IMPELLER SHAFT DISPLACEMENT TRIP CIRCUIT. CORRECTIVE ACTIONS CONSISTED OF PERFORMING A TEMPORARY REPAIR TO THE BROKEN TERMINAL ON "A" CRV COMPRESSOR, AND TROUBLESHOOTING.

[62] HOPE CREEK 1 DOCKET 50-354 LER 91-012
ENTRY INTO TECHNICAL SPECIFICATION 3.0.3 DUE TO CONCURRENT INOPERABILITY OF HIGH PRESSURE COOLANT INJECTION SYSTEM FOR SURVEILLANCE AND "B" LOW PRESSURE COOLANT INJECTION SYSTEM FOR MAINTENANCE.
EVENT DATE: 060491 REPORT DATE: 070291 NSSS: GE TYPE: BWR

(NSIC 222457) ON 6/4/91 AT 1914, THE SENIOR NUCLEAR SHIFT SUPERVISOR (SNSS, SRO LICENSED) DETERMINED THAT HOPE CREEK HAD BEEN IN A CONDITION REQUIRING ENTRY INTO TECH SPEC 3.0.3 FOR A PERIOD OF 6 MINUTES BETWEEN 1901 AND 1907. THE CONDITION REQUIRING ENTRY WAS DUE TO THE CONCURRENT INOPERABILITY OF THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM AND "B" LOOP OF THE LOW PRESSURE COOLANT INJECTION (LPCI) MODE OF THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM. A SURVEILLANCE WAS BEING CONDUCTED ON HPCI AT THE SAME TIME THAT RHR WAS BEING RETURNED TO SERVICE FOLLOWING A SCHEDULED SYSTEM OUTAGE, HOWEVER, OVERLAP BETWEEN THE TIME THE SURVEILLANCE WAS ACTUALLY STARTED AND LPCI WAS DECLARED OPERABLE EXCEEDED ONE HOUR. THE PRIMARY CAUSES OF THIS OCCURRENCE WERE PERSONNEL ERRORS ON THE PART OF THE NUCLEAR SHIFT SUPERVISOR (NSS, SRO LICENSED) AND CONTROLS DEPARTMENT SENIOR SUPERVISOR. THE NSS FAILED TO RECOGNIZE THE TECH SPEC CONFLICT BETWEEN THE HPCI SURVEILLANCE AND THE RHR SYSTEM OUTAGE. THE CONTROLS DEPARTMENT SENIOR SUPERVISOR DID NOT GIVE AN ADEQUATE RHR SYSTEM STATUS TURNOVER BECAUSE HE ERRONEOUSLY ASSUMED THE RHR SYSTEM OUTAGE WAS COMPLETED. CORRECTIVE ACTIONS INCLUDED REVIEWING THIS EVENT WITH THE NSS AND CONTROLS DEPARTMENT SENIOR SUPERVISORS, AND REVIEWING HPCI SYSTEM SURVEILLANCE.

[63] HOPE CREEK 1 DOCKET 50-354 LER 91-013
BOTH TRAINS OF STANDBY LIQUID CONTROL INOPERABLE DURING PERFORMANCE OF INSERVICE TESTING - PROCEDURAL DEFICIENCY - VOLUNTARY REPORT.
EVENT DATE: 060591 REPORT DATE: 070291 NSSS: GE TYPE: BWR

(NSIC 222458) ON 6/5/91, DURING A REVIEW OF AN INDUSTRY OPERATING EXPERIENCE

DOCUMENT FOR APPLICABILITY TO HOPE CREEK, A SAFETY REVIEW ENGINEER DETERMINED THAT, DURING QUARTERLY INSERVICE TESTING (IST) OF THE STANDBY LIQUID CONTROL (SLC) SYSTEM PUMPS, BOTH SLC LOOPS WOULD BE INOPERABLE FOR A SHORT PERIOD OF TIME DURING THE TESTING. THE SYSTEM VALVE ALIGNMENTS REQUIRED BY THE QUARTERLY SLC IST PROCEDURES COULD, FOR A BRIEF PERIOD, ESTABLISH A SYSTEM FLOWPATH SUCH THAT THE OPERABLE SLC LOOP WOULD NOT INJECT TO THE REACTOR VESSEL IF A VALID INJECTION SIGNAL OCCURRED CONCURRENT WITH PERFORMING THE FIRST PORTION OF THE IST PROCEDURE. THESE FINDINGS WERE COMMUNICATED TO THE SENIOR NUCLEAR SHIFT SUPERVISOR (SNSS, SRO LICENSED), AND A CHANGE REQUEST WAS IMMEDIATELY INITIATED TO MODIFY THE AFFECTED QUARTERLY SLC IST PROCEDURES.

[64] INDIAN POINT 3 DOCKET 50-286 LER 88-007 REV 01
UPDATE ON MANUAL REACTOR SHUTDOWN DUE TO 2 GALLON PER MINUTE STEAM GENERATOR PRIMARY TO SECONDARY LEAK IN NUMBER 31 STEAM GENERATOR CAUSED BY DEGRADED TUBE.
EVENT DATE: 101988 REPORT DATE: 061991 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 222318) ON OCTOBER 19, 1988 WITH THE REACTOR AT 100 PERCENT POWER, A PRIMARY TO SECONDARY LEAK OF APPROXIMATELY TWO GALLONS PER MINUTE DEVELOPED IN THE NO. 31 STEAM GENERATOR. AN ORDERLY PLANT SHUTDOWN COMMENCED AT 2325 HOURS ON OCTOBER 19, 1988 AND THE PLANT ACHIEVED COLD SHUTDOWN AT 1100 HOURS ON OCTOBER 20, 1988. ALL PLANT SYSTEMS FUNCTIONED PROPERLY DURING THE EVENT. IT WAS SUBSEQUENTLY DETERMINED BY INSPECTION THAT A 250 DEGREE CIRCUMFERENTIAL THRUWALL CRACK OCCURRED IN A PERIPHERAL TUBE IN THE 31 STEAM GENERATOR (SG) AT THE SIXTH SUPPORT PLATE. INSPECTIONS WERE CONDUCTED ON THE DAMAGED TUBE IN THE NO.31 STEAM GENERATOR BY VISUAL AND EDDY CURRENT METHODS WITH THE ASSISTANCE OF WESTINGHOUSE ELECTRIC CORPORATION. THE ROOT CAUSE FOR THIS EVENT HAS BEEN IDENTIFIED AS HIGH CYCLIC FATIGUE. THE DAMAGED TUBE WAS PLUGGED AND STABILIZED BY THE INSERTION OF A CABLE STABILIZER, AND AN ARRAY OF TUBES SURROUNDING THE DAMAGED TUBE WERE PLUGGED WITH CONTROLLED LEAKAGE SENTINEL PLUGS.

[65] INDIAN POINT 3 DOCKET 50-286 LER 91-007
INOPERABILITY OF REACTOR PROTECTION SYSTEM TRAIN "A".
EVENT DATE: 051091 REPORT DATE: 061391 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 222319) ON MAY 10, 1991, WITH THE REACTOR OPERATING AT 100 PERCENT POWER, INSTRUMENTATION AND CONTROLS TECHNICIANS PERFORMING A SURVEILLANCE PROCEDURE ON REACTOR PROTECTION SYSTEM TRAIN A DISCOVERED THAT THE TURBINE TRIP - REACTOR TRIP LOGIC IN THAT TRAIN WOULD NOT INITIATE A REACTOR TRIP. THE TECHNICIANS FOUND THAT A RELAY ASSOCIATED WITH THE P-7 PERMISSIVE WAS NOT FULLY DROPPED OUT AND THEY MANUALLY OPERATED THE RELAY TO FREE IT. THEY THEN COMPLETED THE SURVEILLANCE PROCEDURE. ON MAY 12, 1991 THE REACTOR WAS SHUT DOWN FOR A PLANNED MAINTENANCE OUTAGE. ON MAY 13, 1991, WITH THE REACTOR AT COLD SHUTDOWN, INSTRUMENTATION AND CONTROLS TECHNICIANS REPLACED THE FAULTY RELAY IN REACTOR PROTECTION SYSTEM TRAIN A. POWER OPERATION WAS RESUMED ON MAY 23, 1991.

[66] KEWAUNEE DOCKET 50-305 LER 91-006
IMPROPER USE OF CONTROL ROOM SUPPLY SHEET RESULTS IN AN INADVERTENT ESF ACTUATION.
EVENT DATE: 052991 REPORT DATE: 062891 NSSS: WE TYPE: PWR

(NSIC 222419) ON MAY 29, 1991, AT 1027 CDT, WITH THE PLANT AT 100 PERCENT POWER, THE CONDENSER AIR EJECTOR RADIATION MONITOR, R-15, FAILED DOWNSCALE. AS DESIGNED, THE FAILURE CAUSED THE STEAM GENERATOR BLOWDOWN ISOLATION VALVES AND THE BLOWDOWN SAMPLE ISOLATION VALVES TO CLOSE. THE DOWNSCALE FAILURE OF R-15 AND SUBSEQUENT ASSOCIATED ENGINEERED SAFEGUARDS FEATURE ACTUATION WAS CAUSED BY THE MISINTERPRETATION OF A NOTE ON THE CONTROL ROOM SUPPLY SHEET. THE NUCLEAR CONTROL ROOM OPERATOR USED THE CONTROL ROOM SUPPLY SHEET AS GUIDANCE TO DETERMINE THE APPLICATIONS OF REPLACEMENT BULBS. A HANDWRITTEN NOTE ON THE CONTROL ROOM SUPPLY SHEET WAS USED TO DETERMINE A REPLACEMENT BULB. AS A RESULT, AN INCORRECT REPLACEMENT BULB WAS CHOSEN AND WHEN IT WAS INSTALLED IT CAUSED A FUSE TO OPEN IN THE R-15 CONTROL ROOM DRAWER, WHICH IN TURN CAUSED R-15 TO FAIL DOWNSCALE. THE INTENT OF THE CONTROL ROOM SUPPLY SHEET IS TO HELP LOCATE REPLACEMENT BULBS FROM

STORAGE AND TO ASSIST IN REORDERING WHEN SUPPLIES ARE LOW. THE SUPPLY SHEET WILL BE UPDATED AND THIS LER WILL BE PLACED IN REQUIRED READING FOR OPERATIONS PERSONNEL.

[67] LA SALLE 1 DOCKET 50-373 LER 91-006
REACTOR SCRAM ON LOW REACTOR VESSEL WATER LEVEL DUE TO LOSS OF 'A' TURBINE DRIVEN REACTOR FEEDWATER PUMP CAUSED BY CONTROL VALVE CLOSURE.
EVENT DATE: 051991 REPORT DATE: 061791 NSSS: GE TYPE: BWR

(NSIC 222358) ON MAY 19, 1991 AT 0423 HOURS, WITH UNIT 1 IN OPERATIONAL CONDITION 1 (RUN) AT 80% POWER (STEADY-STATE) AND BOTH TURBINE DRIVEN REACTOR FEED PUMPS (TDRFP) IN THREE-ELEMENT AUTOMATIC CONTROL, THE "A" TDRFP DISCHARGE FLOW DECREASED FROM 5.5 MLB/HR TO 0.1 MLB/HR INITIATING A DECREASE IN REACTOR WATER LEVEL. APPROXIMATELY THREE SECONDS AFTER THE LOSS OF THE "A" TDRFP FLOW, A REACTOR LOW LEVEL ALARM ANNUNCIATED. THE LICENSED UNIT REACTOR OPERATOR (RO) IMMEDIATELY RESPONDED TO THE ALARM AND DIAGNOSED THE PROBLEM AS LOSS OF "A" TDRFP FLOW. THE OPERATOR PLACED THE "A" TDRFP CONTROLLER UNDER MANUAL CONTROL AND ATTEMPTED TO MANUALLY INCREASE FLOW. THE "A" TDRFP FAILED TO RESPOND AND SINCE THE "B" TDRFP COULD NOT TURN THE LEVEL DECREASE BY ITSELF THE REACTOR SCRAMMED AT LEVEL 3 (12.5" REACTOR WATER LEVEL). ALL REACTOR PROTECTION ACTUATIONS OCCURRED AS EXPECTED. THERE WERE NO EMERGENCY CORE COOLING SYSTEM (ECCS) OR PRIMARY CONTAINMENT ISOLATION SYSTEM ACTUATIONS. THE UNIT WAS PLACED IN HOT SHUTDOWN FOLLOWING THE SCRAM. THE CAUSE OF THIS EVENT WAS THE "A" TDRFP CONTROL VALVE SPONTANEOUSLY CLOSING, EFFECTIVELY DISABLING THE "A" FEEDWATER TURBINE AND PUMP FLOW CONTROL. THE ROOT CAUSE FOR THIS CLOSURE IS NOT KNOWN. AFTER EXTENSIVE "A" TDRFP TESTING, UNIT 1 WAS RESTARTED ON MAY 21, 1991 AND NO PROBLEMS HAVE OCCURRED SINCE THAT TIME.

[68] LA SALLE 2 DOCKET 50-374 LER 91-001 REV 01
UPDATE ON HIGH PRESSURE CORE SPRAY PUMP ROOM AND TURBINE BUILDING FIRE RATED BARRIER FOUND DEGRADED DURING INSPECTION.
EVENT DATE: 011091 REPORT DATE: 060491 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 1 (BWR)

(NSIC 222382) STATION TECHNICAL STAFF WAS PERFORMING TECHNICAL SURVEILLANCE LTS-1000-42 AND FOUND OPEN PENETRATION IN TECH SPEC RELATED FIRE RATED WALLS. AT 0830 ON 1/10/91 WHILE IN OPERATIONAL CONDITION 1 (RUN) AT 100% POWER, 3 OPEN PENETRATIONS WERE FOUND IN THE HIGH PRESSURE CORE SPRAY ROOM (HPCS). A 1 HOUR FIRE WATCH WAS INITIATED IN ACCORDANCE WITH LASALLE TECH SPEC 3.7.6 ACTION REQUIREMENT A. A WORK REQUEST WAS INITIATED TO SEAL THE PENETRATION AND WAS COMPLETED ON 1/15/91. AT 1400 ON 4/10/91 WHILE IN OPERATIONAL CONDITION (RUN) AT 100% POWER AND UNIT 1 IN THE REFUEL MODE, OPENINGS WERE FOUND BETWEEN THE METAL DECKING AND STEEL BEAM SEPARATING THE TURBINE BUILDING AND THE AUXILIARY BUILDING. A 1 HOUR FIRE WATCH WAS ESTABLISHED BETWEEN COLUMNS 13 -15 AND A CONTINUOUS FIRE WATCH WAS ESTABLISHED FOR COLUMNS 15-17 IN ACCORDANCE WITH LASALLE TECH SPEC 3.7.6 ACTION REQUIREMENT A. A WORK REQUEST WAS INITIATED TO SEAL THE PENETRATION AND WAS COMPLETED ON 4/16/91. BECAUSE THE DEGRADATION OF THE FIRE BARRIER WOULD NOT HAVE IMPAIRED SAFE SHUTDOWN OF UNIT 2, THE SAFETY SIGNIFICANCE OF THIS IS CONSIDERED TO BE MINIMAL. THE ROOT CAUSE OF THIS EVENT IS THE FAILURE TO INSTALL THESE REQUIRED FIRE BARRIERS DURING INITIAL CONSTRUCTION AND ANNOTATE THEM ON THE DESIGN DRAWINGS. AN ADDITIONAL CAUSE IS THE LOCATION OF THE OPENINGS (17-20' ABOVE THE FLOOR).

[69] LIMERICK 1 DOCKET 50-352 LER 91-011
CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS WHERE THE 'B' LOOP OF EMERGENCY SERVICE WATER WAS INOPERABLE DUE TO PERSONNEL ERRORS.
EVENT DATE: 042591 REPORT DATE: 053091 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 222165) ON APRIL 30, 1991, PLANT PERSONNEL DISCOVERED THAT A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS (TS) HAD EXISTED FOR APPROXIMATELY 10 HOURS ON APRIL 25, 1991. A TS SURVEILLANCE REQUIREMENT (SR) HAD NOT BEEN MET FOR TWO SAFEGUARD BUS UNDERVOLTAGE RELAYS COINCIDENT WITH THE "B" EMERGENCY SERVICE

WATER (ESW) SYSTEM PUMP BEING OUT OF SERVICE. THIS RESULTED IN THE "B" LOOP OF ESW BEING INOPERABLE AND THE TS ACTION TO SHUTDOWN THE PLANT WAS NOT TAKEN WITHIN THE REQUIRED TIME. THE ACTUAL CONSEQUENCES OF THIS EVENT WERE MINIMAL IN THAT THERE WAS NO ACCIDENT REQUIRING ESW TO PERFORM ITS DESIGN FUNCTION AND NO RADIOACTIVE MATERIAL WAS RELEASED TO THE ENVIRONMENT. THIS EVENT WAS THE RESULT OF PERSONNEL ERRORS RESULTING FROM DEFICIENT WRITTEN INSTRUCTIONS FOR NOTIFICATIONS WHEN TS SR ARE NOT MET AND FOR THE PLANNING OF SYSTEM OUTAGES. AN IMMEDIATE CORRECTIVE ACTION WAS PLACED INTO EFFECT TO PROVIDE WRITTEN NOTIFICATION TO SHIFT OPERATIONS PERSONNEL OF TS REQUIRED SURVEILLANCE TEST PROCEDURES THAT ARE APPROACHING OR PAST THEIR OVERDUE DATES. THE GUIDELINE FOR COMMUNICATING IMPENDING OUT-OF-SURVEILLANCE CONDITIONS DUE TO EXPIRATION OF THE SR TIME INTERVAL WILL BE REVISED TO CLARIFY THE EXPECTATION OF THE COMMUNICATIONS BETWEEN THE WORK GROUPS AND SHIFT SUPERVISION. THE GUIDELINE FOR THE SYSTEM OUTAGES WILL ALSO BE ENHANCED.

[70]   LIMERICK 1   DOCKET 50-352   LER 91-012
LOSS OF REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY DUE TO A REACTOR ENCLOSURE OVERPRESSURIZATION TRANSIENT CAUSING A BLOWOUT PANEL TO OPEN.
EVENT DATE: 050891   REPORT DATE: 060791   NSSS: GE   TYPE: BWR

(NSIC 222166) ON 5/8/91, AT 1220 HRS, THE NORMAL REACTOR ENCLOSURE (RE) HEATING, VENTILATION, AND AIR CONDITIONING (HVAC) SYSTEM WAS SECURED FOR PERFORMANCE OF THE STANDBY GAS TREATMENT SYSTEM (SGTS) AND REACTOR ENCLOSURE RECIRCULATION SYSTEM (RERS) FLOW VERIFICATION SURVEILLANCE TEST (ST), AND THE RE HIGH DIFFERENTIAL PRESSURE (DP) ANNUNCIATOR ALARMED. THE STATUS LIGHTS FOR THE RE HVAC SUPPLY AIR FANS (AF) WERE CHECKED AND INDICATED THAT NO SUPPLY AF WERE OPERATING. PERFORMANCE OF THE ST CONTINUED, BUT THE SGTS EXPERIENCED DIFFICULTY IN MAINTAINING THE REQUIRED RE NEGATIVE DP. AT 1249 HRS, A RE BLOWOUT PANEL WAS DISCOVERED OPEN CAUSING A LOSS OF RE SECONDARY CONTAINMENT INTEGRITY (SCI). THE OPENING OF THE BLOW OUT PANEL WAS CAUSED BY THE "C" RE HVAC SUPPLY AIR FAN OVER-PRESSURIZING THE RE, AFTER THE RE HVAC SYSTEM WAS SECURED. THE NECESSARY ACTIONS AND REPAIRS WERE IMPLEMENTED, AND THE RE SCI WAS RESTORED AT 1326 HRS ON 5/8/91. THE RADIOACTIVE RELEASE AS A RESULT OF THIS EVENT WAS LIMITED TO LESS THAN 0.01% OF THE OFFSITE DOSE CALCULATION MANUAL LIMITS. THE CAUSE OF THE MALFUNCTIONING "C" RE HVAC SUPPLY AIR FAN WAS DUE TO A BURNED OUT POWER SUPPLY BREAKER TRIP COIL. AN INVESTIGATION OF THE TRIP COIL FAILURE IDENTIFIED NO APPARENT CAUSE. THE PREVENTIVE MAINTENANCE PROCEDURES FOR THE 480V POWER SUPPLY BREAKERS WILL BE REVISED EVERY 6 MONTHS.

[71]   LIMERICK 1   DOCKET 50-352   LER 91-013
BOTH TRAINS OF THE STANDBY GAS TREATMENT SYSTEM WERE INOPERABLE AS A RESULT OF COGNITIVE PERSONNEL ERROR.
EVENT DATE: 050891   REPORT DATE: 060691   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 222167) ON MAY 8, 1991, DURING THE PERFORMANCE OF SURVEILLANCE TEST (ST) PROCEDURE ST-6-076-250-1, "SGTS AND RERS FLOW TEST," THE "A" TRAINS OF THE STANDBY GAS TREATMENT SYSTEM (SGTS) AND REACTOR ENCLOSURE RECIRCULATION SYSTEM INADVERTENTLY SHUTDOWN AND THE MANUAL UNIT 1 REACTOR ENCLOSURE (RE) SECONDARY CONTAINMENT ISOLATION SIGNAL, WHICH WAS PRESENT FOR THE TEST, CLEARED. LICENSED MAIN CONTROL ROOM (MCR) PERSONNEL MANUALLY INITIATED THE REDUNDANT UNIT 1 "B" RE SECONDARY CONTAINMENT ISOLATION SIGNAL IN AN ATTEMPT TO START THE "B" TRAIN OF THE SGTS. HOWEVER, THE "B" TRAIN OF SGTS DID NOT INITIATE. WITH BOTH THE "A" AND "B" TRAINS OF SGTS INOPERABLE, TECHNICAL SPECIFICATIONS (TS) ACTION 3.0.3 WAS ENTERED. AN OPERATOR FOUND SLIDE GATE DAMPER SGD-076-206-1 IN THE CLOSED POSITION WHICH HAD BEEN CLOSED DURING APPLICATION OF A BLOCKING PERMIT. THE CLOSED SGD-076-206-1 WHICH CAUSED THE UNIT 1 RE SECONDARY CONTAINMENT ISOLATION SIGNAL TO BE BYPASSED WAS THEN OPENED. BOTH TRAINS OF THE SGTS WERE DECLARED OPERABLE AND TS ACTION 3.0.3 WAS EXITED. THE ACTUAL CONSEQUENCES OF THE EVENT WERE MINIMAL, HOWEVER, SGTS COULD NOT HAVE CONTROLLED THE RELEASE OF RADIOACTIVE MATERIAL IN THE EVENT OF AN ACCIDENT. THE CAUSE OF THIS EVENT IS COGNITIVE PERSONNEL ERROR, LESS THAN ADEQUATE COMMUNICATION, AND A FAILURE TO COMPLY WITH A PROCEDURE.

[72] LIMERICK 1 DOCKET 50-352 LER 91-015
PRESSURE SETPOINT DRIFT OF THE MAIN STEAM SAFETY RELIEF VALVES DUE TO CORROSION INDUCED BONDING WITHIN THE VALVES.
EVENT DATE: 052891 REPORT DATE: 070591 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 222494) ON MAY 28, 1991, LIMERICK GENERATING STATION PERSONNEL IDENTIFIED THAT PRESSURE SETPOINT TESTING OF THE FOURTEEN REACTOR MAIN STEAM SYSTEM TARGET ROCK CORP., MODEL 7567 F, PILOT OPERATED TWO-STAGE SAFETY RELIEF VALVES (SRVS) REVEALED THAT ONLY THREE SRVS LIFTED WITHIN THE TECHNICAL SPECIFICATIONS (TS) REQUIRED LIMIT OF +1% OF THE NAMEPLATE SETPOINT AS SPECIFIED IN TS SECTION 3.4.2. THE ROOT CAUSE FOR THE SETPOINT DRIFT OF THE ELEVEN SRVS WAS PRIMARILY CORROSION INDUCED BONDING BETWEEN THE PILOT DISC MADE OF EITHER STELLITE OR STAINLESS STEEL (SS) AND THE SATELLITE SEAT. THE FOURTEEN SRVS WERE ALL REFURBISHED USING SATELLITE PILOT DISCS, PRESSURE TESTED, AND RECERTIFIED PRIOR TO BEING REINSTALLED DURING THE FIRST UNIT 2 REFUEL OUTAGE. THERE WERE NO ACTUAL ADVERSE CONSEQUENCES OR RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS CONDITION. THIS CONDITION WAS DETERMINED AS REPORTABLE ON JUNE 6, 1991, SINCE THIS CONDITION RESULTED IN MORE THAN TWO INDEPENDENT TRAINS BECOMING INOPERABLE IN A SINGLE SAFETY SYSTEM DUE TO A SINGLE CAUSE. THEREFORE, THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH THE REQUIREMEN TS OF 10CFR50.73(A)(2)(VII).

[73] LIMERICK 2 DOCKET 50-353 LER 89-010 REV 01
UPDATE ON HPCI SYSTEM IN A DEGRADED CONDITION DUE TO A PERSONNEL ERROR WHICH RESULTED IN CONDENSATE ACCUMULATING INSIDE THE TURBINE EXHAUST PIPE.
EVENT DATE: 101389 REPORT DATE: 062091 NSSS: GE TYPE: BWR

(NSIC 222351) ON 10/13/89, HPCI SYSTEM WAS DECLARED INOPERABLE DUE TO A CONDENSATE ACCUMULATION INSIDE THE HPCI TURBINE EXHAUST PIPE RESULTING FROM A DRAIN PIPE FLOW ORIFICE (FO) BLOCKAGE. THIS EVENT RESULTED IN THE HPCI SYSTEM BEING IN A DEGRADED CONDITION WHICH ALONE COULD HAVE PREVENTED THE FULFILLMENT OF ITS SAFETY FUNCTION NEEDED TO SHUTDOWN THE REACTOR AND MITIGATE THE CONSEQUENCES OF AN ACCIDENT. AN ACCIDENT CONDITION DID NOT OCCUR DURING THE TIME IN WHICH THE HPCI SYSTEM WAS INOPERABLE OR COULD HAVE BEEN DEGRADED, AND HPCI WAS NOT CALLED UPON TO PERFORM ITS INTENDED SAFETY FUNCTION. IF HPCI HAD BECOME UNAVAILABLE DUE TO THE ACCUMULATION OF CONDENSATE IN THE TURBINE EXHAUST PIPE, SUFFICIENT ECCS AND THE REACTOR CORE ISOLATION COOLING SYSTEM WERE AVAILABLE TO ENSURE SAFE SHUTDOWN OF THE REACTOR. A PERSONNEL ERROR LEAD TO THE INADVERTENT OVERFILLING OF THE SUPPRESSION POOL AND FLOODING OF THE HPCI TURBINE EXHAUST PIPE DURING START-UP ACTIVITIES. THIS RESULTED IN THE ADDITION OF DEBRIS INTO THE TURBINE EXHAUST PIPE WHICH LATER RESULTED IN THE FO BLOCKAGE. COMPRESSED AIR WAS UTILIZED TO UNBLOCK THE FO AND THE ACCUMULATED CONDENSATE THEN DRAINED TO THE BAROMETRIC CONDENSER. QUARTERLY ROUTINE TESTS ARE BEING PERFORMED TO VERIFY FLOW THROUGH BOTH UNIT 1 AND 2 HPCI SYSTEM FOS. UNIT 2 DRAIN POT WAS FLUSHED DURING HPCI SYSTEM OUTAGE.

[74] LIMERICK 2 DOCKET 50-353 LER 91-008
VOLUNTARY REPORT FOR POTENTIAL CONDITION OF DIESEL GENERATOR.
EVENT DATE: 040191 REPORT DATE: 061291 NSSS: GE TYPE: BWR

(NSIC 222352) ON 4/1/91, WHILE IN A REFUELING CONDITION, PLANT PERSONNEL WERE PERFORMING SURVEILLANCE TEST PROCEDURE ST-1-092-111-2, "D21 DIESEL GENERATOR 4KV SF GD LOSS OF POWER LSF/SM AND OUTAGE TESTING." THE D21 EMERGENCY DIESEL GENERATOR (EDG) SUCCESSFULLY REJECTED THE 2A RESIDUAL HEAT REMOVAL (RHR) SYSTEM PUMP MOTOR LOAD OF 992 KW. IN ACCORDANCE WITH THE PROCEDURE, THE 2A RHR PUMP WAS RESTARTED; HOWEVER, THE D21 EDG OUTPUT VOLTAGE INCREASED ABOVE THE ACCEPTANCE CRITERIA VALUE TO APPROXIMATELY 5200 VOLTS. THE CAUSE OF THIS EVENT WAS AN IMPROPERLY INSTALLED POTENTIAL TRANSFORMER PRIMARY FUSE THAT MAY HAVE BEEN DUE TO PERSONNEL ERROR. THIS RESULTED IN A LOSS OF GENERATOR OUTPUT VOLTAGE FEEDBACK TO THE AUTOMATIC VOLTAGE REGULATOR. THE D21 EDG FUSE MAY HAVE BEEN IMPROPERLY INSTALLED SINCE 2/2/91. IN THIS CONDITION THE OPERABILITY OF THE D21 EDG WAS NOT ASSURED, IF A DESIGN BASIS EVENT HAD OCCURRED, SUCH AS A SAFE SHUTDOWN EARTHQUAKE. IF THE D21 EDG WAS INOPERABLE ON 2/2/91, A TECH SPEC VIOLATION OF TS SECTION 3.8.1.1B OCCURRED BECAUSE THE ASSOCIATED TS ACTIONS WERE NOT TAKEN WITHIN

THE REQUIRED TIME PERIOD. AS A RESULT, THIS VOLUNTARY LER IS BEING SUBMITTED TO REPORT A CONDITION THAT MAY HAVE BEEN A CONDITION PROHIBITED BY TS. ON 5/31/91, AN OPERATOR AID WAS ADDED ONTO EACH OF THE 16 EDG CONTROL PANELS. THE OPERATOR AID REFERENCES A PROCEDURE REVISED TO PROVIDE CAUTION & DIRECTION FOR FUSE INSTALLATION.

[75] LIMERICK 2 DOCKET 50-353 LER 91-009
VOLUNTARY LER AND SPECIAL REPORT FOR A LOOSE WIRE IN THE POTENTIAL TRANSFORMER SENSING NETWORK OF AN EMERGENCY DIESEL GENERATOR.
EVENT DATE: 052191 REPORT DATE: 062091 NSSS: GE TYPE: BWR

(NSIC 222353) ON 5/21/91, WHILE IN A REFUELING CONDITION, PLANT PERSONNEL WERE PERFORMING SURVEILLANCE TEST PROCEDURE ST-1-092-114-2. WHILE PERFORMING THIS PROCEDURE, D24 EMERGENCY DIESEL GENERATOR OUTPUT VOLTAGE INCREASED ABOVE THE ACCEPTANCE CRITERION VALUE. THE CAUSE OF THIS EVENT WAS A LOOSE WIRE IN THE POTENTIAL TRANSFORMER SENSING NETWORK WHICH RESULTED IN A LOSS OF GENERATOR OUTPUT VOLTAGE FEEDBACK TO THE AUTOMATIC VOLTAGE REGULATOR. THE CAUSE OR TIME THAT THE WIRE BECAME LOOSE CANNOT BE DETERMINED. IN THIS CONDITION THE OPERABILITY OF THE D24 ED G WAS NOT ASSURED IF A DESIGN BASIS EVENT HAD OCCURRED, SUCH AS A SAFE SHUTDOWN EARTHQUAKE. THIS SITUATION REPRESENTS A POTENTIAL CONDITION PROHIBITED BY TECH SPECS SINCE THE D24 EDG MAY HAVE BEEN INOPERABLE SINCE 7/05/89, AND THE ASSOCIATED TS ACTIONS WERE NOT TAKEN WITHIN THE REQUIRED TIME PERIOD. THIS VOLUNTARY LER IS BEING SUBMITTED TO REPORT A CONDITION THAT MAY HAVE BEEN PROHIBITED BY TS IN ADDITION TO THE REQUIREMENTS OF TS SECTION 6.9.2 AS REQUIRED BY TS SURVEILLANCE REQUIREMENT 4.8.1.1.3 DUE TO AN EDG TEST FAILURE. ON 5/22/91, THE LOOSE WIRE WAS CORRECTED. THE REMAINING CONNECTIONS ON THE D24 EDG AS WELL AS ALL OTHER UNIT 2 EDGS WERE INSPECTED AND NO ABNORMALITIES IN THE WIRING OF THE POTENTIAL TRANSFORMER SENSING NETWORK WERE FOUND.

[76] LIMERICK 2 DOCKET 50-353 LER 91-010
SPECIAL REPORT FOR A NONVALID DIESEL GENERATOR START FAILURE DURING A VALID TEST.
EVENT DATE: 052591 REPORT DATE: 062491 NSSS: GE TYPE: BWR

(NSIC 222379) ON MAY 25, 1991, PLANT PERSONNEL WERE PERFORMING SURVEILLANCE TEST (ST) PROCEDURE ST-6-092-311-2, "D21 DIESEL GENERATOR OPERABILITY TEST RUN." AT 2016 HOURS, AFTER THE D21 EMERGENCY DIESEL GENERATOR (EDG) HAD BEEN RUNNING FOR APPROXIMATELY 3 HOURS ITS OUTPUT INCREASED TO 3500 KW. THE D21 EDG WAS DECLARED INOPERABLE AT 2017 HOURS. THE CAUSE OF THIS EVENT WAS AN IMPROPERLY OPERATING AUXILIARY TEST START RELAY WHICH CONVERTED THE D21 EDG CONTROL CIRCUIT TO ISOCHRONOUS MODE. THE D21 EDG AUXILIARY TEST START RELAY WAS REPLACED AND THE TEST START CIRCUIT WAS INSTRUMENTED. THE D21 EDG WAS STARTED AND LOADED TO 2000 KW FOR 1 HOUR. THE SPECIAL TEST START CIRCUIT INSTRUMENTATION DETECTED NO ABNORMALITIES. PROCEDURE ST-6-092-311-2 WAS SUCCESSFULLY PERFORMED AND ON MAY 29, 1991, THE D21 EDG WAS DECLARED OPERABLE AT 2030 HOURS. THE D21 EDG FAILURE WAS CLASSIFIED AS A NONVALID FAILURE USING THE GUIDANCE OF REGULATORY GUIDE 1.108, "PERIODIC TESTING OF DIESEL GENERATOR UNITS USED AS ONSITE ELECTRIC POWER SYSTEM AT NUCLEAR POWER PLANTS". IN THE EVENT OF AN ACTUAL LOSS OF OFFSITE POWER, THE TWO OPERABLE UNIT 2 EDGS WOULD HAVE PROVIDED ADEQUATE POWER TO MAINTAIN THE COLD SHUTDOWN CONDITION OF THE REACTOR.

[77] MAINE YANKEE DOCKET 50-309 LER 91-005
PLANT TRIP ON MAIN TRANSFORMER FAILURE.
EVENT DATE: 042991 REPORT DATE: 052991 NSSS: CE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 222159) ON 4/29/91, MAINE YANKEE TRIPPED FROM 100% POWER WHEN ONE OF THE TWO MAIN OUTPUT TRANSFORMERS DEVELOPED AN INTERNAL FAULT. THE ELECTRICAL FAULT IN THE MAIN TRANSFORMER X-1A RESULTED IN A GENERATOR PRIMARY RELAY TRIP OF THE MAIN TURBINE. THE REACTOR PROTECTIVE SYSTEM TRIPPED THE REACTOR ON LOSS OF LOAD. PROTECTIVE RELAYING AND THE AUTOMATIC TRANSFER TO RESERVE POWER PERFORMED AS DESIGNED. THE TRANSFORMER FAULT CAUSED ARCING AT THE NEUTRAL BUS BENEATH THE MAIN GENERATOR. THE ARCING DAMAGED HYDROGEN LINES AND THE NEUTRAL LEADS BUSHINGS AND RESULTED IN A HYDROGEN FIRE BENEATH THE MAIN GENERATOR. THE PLANT DECLARED

AN UNUSUAL EVENT DUE TO THE FIRE, WHICH WAS ALLOWED TO BURN ITSELF OUT OVER A 3 HOUR PERIOD. THE DAMAGED TRANSFORMER HAS BEEN REPLACED WITH AN ON-SITE SPARE (X-1S). THE MAIN GENERATOR REPAIRS ARE EXPECTED TO BE COMPLETED BY THE END OF MAY, 1991, AT WHICH TIME THE PLANT EXPECTS TO RESUME OPERATION. TWO TRANSFORMER FAILURES (WITHOUT FIRES) OCCURRED PREVIOUSLY AT MAINE YANKEE ON 8/13/88 (X-1A)(LER-88-006), AND 8/31/78 (X-1B). THE ROOT CAUSE OF THE TRANSFORMER FAULT IS INDETERMINATE PENDING FURTHER INSPECTION. THE TRANSFORMER (X-1A) HAD BEEN REBUILT FOLLOWING THE FAULT IN 1988 AND HAD BEEN IN SERVICE FOR APPROX. 9 MONTHS.

[78] MAINE YANKEE DOCKET 50-309 LER 91-006
PLANT T[illegible] DUE TO LOW STEAM GENERATOR WATER LEVEL.
EVEN[illegible] [illegible]53091 REPORT DATE: 070191 NSSS: CE TYPE: PWR
VEN[illegible] [illegible] VALVE COMPANY
[illegible] [illegible]LECTRIC CO.

(NSI[illegible] [illegible]/84 ON MAY 30, 1991, WITH THE REACTOR AT 12% POWER, AN AUTOMA[illegible] [illegible] OCCURRED DUE TO ACTUATION OF THE REACTOR PROTECTIVE SYSTEM. [illegible] [illegible]ENERATOR NUMBER ONE DROPPED BELOW THE LOW STEAM GENERATOR WATER [illegible] WHEN MAIN FEEDWATER PUMP (P-2A) TRIPPED ON LOW SUCTION HEADER [illegible] [illegible]ATER PUMP TRIPPED WHEN THE MAIN FEEDWATER REGULATING VALVES HAD [illegible] [illegible]XCESSIVE AMOUNT ALLOWING A LARGE INFLOW OF FEEDWATER. DURING THIS [illegible] MAIN FEEDWATER PUMP P2B DID NOT RUN AND STANDBY CONDENSATE P[illegible] P27C [illegible] AUTOMATICALLY START WHEN THEY SHOULD HAVE. SUBSEQUENT TESTING AN[illegible] [illegible]ALUATION FOUND PRESSURE SWITCH PS 1305B, WHICH PROVIDES AN AUTO TRIP SIGNAL TO P2B, TO BE DEFECTIVE. A REPLACEMENT PRESSURE SWITCH WAS INSTALLED AND TESTED. PRESSURE SENSING SWITCHES ASSOCIATED WITH MAIN FEEDWATER PUMP AUTO START AND AUTO TRIP FUNCTIONS WERE TESTED AND CALIBRATED. PRESSURE SENSING LINES WERE BLOWN DOWN. PRESSURE SENSING EQUIPMENT AND CIRCUITRY ASSOCIATED WITH P27C AUTO START FUNCTIONS WERE INSPECTED AND FUNCTIONALLY CHECKED. AN ATTACHMENT PROVIDING ENHANCED GUIDANCE FOR OPERATING THE FEEDWATER STATION DURING PLANT STARTUP AT LOW POWER WILL BE ADDED TO THE PLANT STARTUP PROCEDURE. DURING THE NEXT REFUELING OUTAGE, FEBRUARY 1992, A MODIFICATION TO THE MAIN FEEDWATER REGULATING SYSTEM WILL INCLUDE INSTALLING NEW MAIN FEEDWATER REGULATING VALVE CONTROLLERS.

[79] MCGUIRE 1 DOCKET 50-369 LER 91-008
UNDER CERTAIN POSTULATED CONDITIONS, THE DIESEL GENERATORS COULD BE RENDERED INOPERABLE BECAUSE OF A DESIGN DEFICIENCY.
EVENT DATE: 051491 REPORT DATE: 062791 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 222499) ON 5/14/91, DESIGN ENGINEERING (DE) PERSONNEL ISSUED PROBLEM INVESTIGATION REPORT (PIR) 0-M91-0088. THE PIR ADDRESSED A CONCERN REGARDING NON-SEISMICALLY QUALIFIED PIPING IN THE VICINITY OF THE DIESEL GENERATOR (D/G) AIR INTAKES. THE CONCERN WAS RAISED DURING AN NRC AUDIT. THE PIPING IN QUESTION WAS THE 2 INCH MAIN STEAM DRAIN LINES AND THE 4 INCH CONDENSATE FEEDWATER TEMPERING FLOW LINES. ON 5/17/91, DE PERSONNEL ISSUED AN OPERABILITY EVALUATION STATING THE D/GS ON UNITS 1 AND 2 WERE CONDITIONALLY OPERABLE. ON 5/22/91, PIR 0-M91-0100 WAS ISSUED TO ADDRESS THE CONCERN REGARDING THE SAME MAIN STEAM AND CONDENSATE FEEDWATER LINES WHEN SUBJECTED TO TORNADO WIND LOADINGS AND TORNADO GENERATED MISSILES. AFTER EVALUATING THE PIRS, DE PERSONNEL DETERMINED IT WOULD BE NECESSARY TO UPGRADE THE MAIN STEAM DRAIN LINE PIPING TO WITHSTAND A SEISMIC EVENT AND WIND LOADS OF 180 MILES PER HOUR (MPH). THE UPGRADES WERE IMPLEMENTED UNDER NUCLEAR STATION MODIFICATIONS (NSMS) MG-12391 AND MG-22391. DE PERSONNEL DETERMINED THE CONDENSATE FEEDWATER TEMPERING FLOW LINES WERE SEISMICALLY RUGGED AND WOULD WITHSTAND WIND LOADS OF 300 MPH. UNIT 1 WAS IN MODE 2 (STARTUP) AND UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT 100% POWER AT THE TIME THE INITIAL CONCERN WAS ADDRESSED. THE EVENT HAS BEEN ASSIGNED A CAUSE OF DESIGN DEFICIENCY.

[80] MCGUIRE 1 DOCKET 50-369 LER 91-005
BOTH TRAINS OF THE ANNULUS VENTILATION SYSTEM WERE INOPERABLE DUE TO AN INAPPROPRIATE ACTION CAUSED BY MANAGEMENT DEFICIENCIES AND DEFICIENT COMMUNICATION.

EVENT DATE: 051591   REPORT DATE: 061491   NSSS: WE   TYPE: PWR

(NSIC 222172) ON 5/15/91, AT 1130, INTEGRATED SCHEDULING PERSONNEL WERE INFORMED BY SECURITY PERSONNEL THAT BOTH UNIT 1 LOWER ANNULUS DOORS WERE OPEN. THE INTEGRATED SCHEDULING PERSON THEN CHECKED WITH OPERATIONS (OPS) PERSONNEL TO DETERMINE IF THEY WERE AWARE OF THE DOORS BEING OPEN. CONSEQUENTLY, SINCE IT COULD NOT BE IMMEDIATELY VERIFIED WHETHER THE DOORS WERE STILL HELD OPEN OR NOT, OPS PERSONNEL ENTERED TECH SPEC (TS) ACTION STATEMENT 3.0.3 FOR UNIT 1 DUE TO BOTH TRAINS OF THE ANNULUS VENTILATION (VE) SYSTEM BEING INOPERABLE. UPON INVESTIGATION BY OPS PERSONNEL, IT WAS FOUND THAT THE DOORS HAD BEEN HELD OPEN FOR PAINTING OF THE DOORS AND DOOR FRAMES ON 5/14/91, FROM 1500 TO 1630, AND ON 5/15/91, FROM 0800 TO 0930 AND 1020 TO 1140. OPS PERSONNEL HAD NOT BEEN AWARE THAT THE DOORS WERE HELD OPEN DURING THESE TIMES AND APPROPRIATE COMPENSATORY ACTIONS HAD NOT BEEN ESTABLISHED. THEREFORE, BOTH TRAINS OF THE VE SYSTEM WERE TECHNICALLY INOPERABLE DURING THE TIMES WHEN THE DOORS WERE HELD OPEN. THE DOORS WERE VERIFIED TO BE CLOSED AND OPS PERSONNEL EXITED TS ACTION STATEMENT 3.0.3 FOR UNIT 1. UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT 30 PERCENT POWER AT THE TIME OF THE EVENT DISCOVERY. THIS INCIDENT IS ASSIGNED CAUSES OF MANAGEMENT DEFICIENCY RESULTING FROM INADEQUATE WORK CONTROL AND MANAGEMENT INTERFACE, AND INAPPROPRIATE ACTION.

[81]   MCGUIRE 1   DOCKET 50-369   LER 91-007
TRAIN "B" OF THE RESIDUAL HEAT REMOVAL SYSTEM WAS INOPERABLE DUE TO INAPPROPRIATE ACTIONS AND AN UNKNOWN POSSIBLE INSTALLATION DEFICIENCY.
EVENT DATE: 052091   REPORT DATE: 061891   NSSS: WE   TYPE: PWR

(NSIC 222357) ON MAY 19, 1991 AT APPROXIMATELY 1100, WHILE UNIT 1 WAS IN MODE 1 (100 PERCENT POWER), THE ACTION STATEMENT OF TECHNICAL SPECIFICATION 3/4.5.2 WAS EXCEEDED WITHOUT APPROPRIATE COMPENSATORY ACTION. VALVE 1ND-67B (RESIDUAL HEAT REMOVAL SYSTEM PUMP 1B AND HEAT EXCHANGER 1B MINIFLOW) HAD BEEN INOPERABLE FOR MORE THAN 72 HOURS WITHOUT THE KNOWLEDGE OF OPERATIONS (OPS) CONTROL ROOM PERSONNEL. A WORK REQUEST WAS WRITTEN ON MAY 16, 1991 DUE TO WIRES BEING PULLED OUT OF PRESSURE SWITCH 1NDPS5050 (RESIDUAL HEAT REMOVAL PUMP 1B MINIFLOW) WHILE TEST EQUIPMENT WAS REMOVED AT THE CONCLUSION OF PERIODIC TEST PT/1/A/4204/01B (RESIDUAL HEAT REMOVAL PUMP 1B PERFORMANCE TEST). IT WAS NOT RECOGNIZED BY PERFORMANCE (PRF) PERSONNEL AT THE TIME THAT THIS AFFECTED THE OPERATION OF VALVE 1ND-67B. VALVE 1ND-67B WOULD NOT AUTOMATICALLY OPEN AS REQUIRED WHEN ND PUMP 1B WAS MANUALLY STARTED ON MAY 21, 1991. THE SEPARATED WIRES WERE SUBSEQUENTLY REPAIRED UNDER WORK REQUEST 144977OPS. THE RESIDUAL HEAT REMOVAL (ND) SYSTEM WAS RETURNED TO OPERABLE STATUS ON MAY 21, 1991 AT 0550. THIS EVENT IS ASSIGNED CAUSES OF INAPPROPRIATE ACTIONS, AND AN UNKNOWN.

[82]   MCGUIRE 2   DOCKET 50-370   LER 91-002
TURBINE DRIVEN AUXILIARY FEEDWATER PUMP WAS INOPERABLE BECAUSE OF A MISPOSITIONED SLIDING LINK DUE TO A POSSIBLE INAPPROPRIATE ACTION.
EVENT DATE: 051591   REPORT DATE: 061491   NSSS: WE   TYPE: PWR

(NSIC 222173) ON MAY 15, 1991, WHILE PREPARING TO PERFORM PREVENTIVE MAINTENANCE ON UNIT 2 TURBINE DRIVEN AUXILIARY FEEDWATER PUMP SUCTION PRESSURE SWITCH 2MCAPSE390, INSTRUMENT AND ELECTRICAL PERSONNEL DISCOVERED SLIDING LINK C-8 IN CABINET 2AFP2B WAS OPEN. THIS OPEN LINK WOULD HAVE PREVENTED VALVE 2CA-116B (TURBINE DRIVEN AUXILIARY FEEDWATER PUMP SUPPLY FROM TRAIN B NUCLEAR SERVICE WATER), FROM OPENING IF A LOW SUCTION PRESSURE SIGNAL HAD BEEN RECEIVED. THIS RENDERED THE PUMP INOPERABLE AND LEFT THE STANDBY SHUTDOWN FACILITY (SSF) IN A DEGRADED CONDITION. THE LINK WAS SUBSEQUENTLY RESTORED TO THE CLOSED POSITION BY INSTRUMENT AND ELECTRICAL PERSONNEL AND VALVE 2CA-116B WAS STROKED BY OPERATIONS CONTROL ROOM PERSONNEL. SLIDING LINKS FOR UNITS 1 AND 2 AUXILIARY FEEDWATER SUCTION PRESSURE SWITCHES WERE SUBSEQUENTLY INSPECTED PER WORK REQUESTS 601113 AND 601114 WITH NO FURTHER PROBLEMS NOTED. THE EVENT IS ASSIGNED A CAUSE OF POSSIBLE INAPPROPRIATE ACTION BECAUSE NO DEFINITE CAUSE COULD BE DETERMINED. UNIT 2 WAS IN MODE 1 (POWER OPERATION), OPERATING BETWEEN 95 AND 100 PERCENT POWER DURING THE TIME PERIOD FROM APRIL 22, 1991 THROUGH MAY 15, 1991. THIS EVENT WILL BE REVIEWED WITH INSTRUMENT AND ELECTRICAL PERSONNEL AND QUALITY ASSURANCE PERSONNEL AS APPLICABLE.

[83] MILLSTONE 1 DOCKET 50-245 LER 91-011
TECHNICAL SPECIFICATION VIOLATION FIRE PENETRATION.
EVENT DATE: 041791 REPORT DATE: 051591 NSSS: GE TYPE: BWR

(NSIC 222056) ON 4/17/91, AT 2000 HOURS, WITH THE PLANT IN COLD SHUTDOWN AND THE CORE OFF-LOADED, (LESS THAN 100F AND 0 PSIG), IT WAS DISCOVERED THAT THE FIRE WATCH REQUIREMENTS SPECIFIED BY WORK ORDER PROCEDURES WERE NOT CORRECTLY IMPLEMENTED AND DID NOT MEET THE REQUIREMENTS OF TECH SPEC 3.12.F.2. UPON DISCOVERY OF THE OPEN FIRE PENETRATION, A TEMPORARY FIRE PENETRATION SEAL WAS INSTALLED UNTIL A FIRE WATCH WAS ESTABLISHED. THE AUTOMATIC FIRE DETECTION SYSTEM AND THE FIRE SUPPRESSION SYSTEMS IN THE TURBINE BUILDING AREA AFFECTED BY THE PENETRATION WERE OPERABLE THROUGHOUT THE EVENT. THE AVAILABLE FIRE PROTECTION FEATURES PROVIDED IN THE AREAS, AND THE LOW COMBUSTIBLE LOADING, MINIMIZED ANY POTENTIAL ADVERSE IMPACT ON MAINTAINING THE SAFE SHUTDOWN CONDITION. NO SAFETY SYSTEMS WERE REQUIRED TO FUNCTION AS A RESULT OF THIS EVENT AND NO SAFETY CONSEQUENCES RESULTED FROM THIS EVENT.

[84] MILLSTONE 1 DOCKET 50-245 LER 91-014
HIGH ENERGY LINE BREAK INTERACTION WITH CLOSED COOLING WATER.
EVENT DATE: 042691 REPORT DATE: 052891 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 222231) ON 4/26/91, AT 1500 HRS, WITH THE PLANT IN COLD SHUTDOWN AND THE CORE OFF-LOADED, IT WAS DETERMINED THAT A POSTULATED FEEDWATER LINE BREAK COULD RESULT IN A COMPLETE LOSS OF AC POWER. THE POSTULATED HIGH ENERGY LINE BREAK (HELB) COULD HAVE DISRUPTED POWER LINES IN OVERHEAD CABLE TRAYS ASSOCIATED WITH THE GAS TURBINE GENERATOR. THIS HELB INTERACTION WAS PREVIOUSLY IDENTIFIED. HOWEVER, POTENTIAL HELB DAMAGE TO THE TURBINE BUILDING SECONDARY CLOSED COOLING WATER (TBSCCW) LINES USED TO SUPPLY COOLING TO THE DIESEL GENERATOR ROOM COOLERS WAS NOT EVALUATED. DIESEL GENERATOR ROOM COOLING IS REQUIRED FOR DIESEL GENERATOR (DG) OPERABILITY. THIS CONSEQUENCE, ALONG WITH THE ASSUMED LOSS OF NORMAL POWER COINCIDENT WITH A FEEDWATER BREAK WOULD RESULT IN THE TOTAL LOSS OF AC POWER. FOR THIS HELB, A REDUNDANT SHUTDOWN METHOD CANNOT BE ENSUED. SUBSEQUENT EVALUATIONS REVEALED PIPING INTERACITONS FROM THE CONDENSATE PIPING, REACTOR FEED PUMP SEAL WATER INJECTION SYSTEM, REACTOR WATER CLEANUP (RWCU) SYSTEM AND THE ISOLATION CONDENSER SYSTEM THAT HAD THE POTENTIAL TO ADVERSELY IMPACT TBSCCW SYSTEM PIPING AND, THEREFORE, AFFECT THE OPERABILITY OF THE DG. FOR THESE BREAK LOCATIONS, REDUNDANT SAFE SHUTDOWN METHODS CANNOT BE ENSURED. THE FINAL RESULTS OF THE HELB REVIEW ASSOCIATED WITH THESE SYSTEMS ARE STILL IN PROGRESS. NO SAFETY SYSTEMS WERE REQUIRED TO FUNCTION AS A RESULT AND NO SAFETY CONSEQUENCES RESULTED FROM THIS EVENT.

[85] MILLSTONE 1 DOCKET 50-245 LER 91-015
STANDBY GAS TREATMENT INITIATION WHEN SRM/IRM DRY TUBE BROKE THE SURFACE OF THE FUEL POOL.
EVENT DATE: 050191 REPORT DATE: 053191 NSSS: GE TYPE: BWR

(NSIC 222228) ON 5/1/91, AT 1030 HOURS, WITH THE PLANT IN COLD SHUTDOWN FOR THE SCHEDULED REFUELING OUTAGE (0% POWER, 90F), SOURCE RANGE MONITOR AND INTERMEDIATE RANGE MONITOR DRY TUBE REPLACEMENT WAS IN PROGRESS WHEN AN IRRADIATED DRY TUBE ASSEMBLY MOMENTARILY BROKE THE SURFACE OF THE FUEL POOL DURING TRANSFER FROM THE REACTOR VESSEL. THE EVENT RESULTED IN THE AUTOMATIC INITIATION OF STANDBY GAS TREATMENT SYSTEM DUE TO REFUEL FLOOR HIGH RADIATION. ALL SYSTEMS FUNCTIONED AS DESIGNED AND THERE WERE NO SIGNIFICANT SAFETY CONSEQUENCES FROM THIS EVENT. THE DRY TUBE WAS SUBSEQUENTLY DISPOSED OF USING A SPECIAL PROCEDURE.

[86] MILLSTONE 1 DOCKET 50-245 LER 91-019
LPCI SYSTEM NOT DEMONSTRATED TO BE OPERABLE DURING POSTULATED ACCIDENT CONDITIONS.
EVENT DATE: 061091 REPORT DATE: 070391 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 222399) ON JUNE 10, 1991, AT 1530 HOURS, WITH THE PLANT IN COLD SHUTDOWN (82 DEGREES FAHRENHEIT AND 0 PSIG), AN ONGOING ENGINEERING ANALYSIS CONCLUDED

THAT THE LOW PRESSURE COOLANT INJECTION (LPCI) SYSTEM AND THE CORE SPRAY SYSTEM MOTORS COULD NOT BE DEMONSTRATED TO BE OPERABLE DURING POSTULATED ACCIDENT CONDITIONS. THIS CONCLUSION WAS BASED UPON HIGHER CALCULATED POST ACCIDENT TORUS TEMPERATURES COMBINED WITH HIGHER THAN PREVIOUSLY ASSUMED AMBIENT ROOM TEMPERATURES. THE COOLING WATER SUPPLY FOR THE LPCI AND CORE SPRAY MOLAR BEARINGS IS SUPPLIED FROM THE CONTAINMENT TORUS WATER VIA THE PUMP DISCHARGE PIPING. THE POSTULATED INCREASE IN THE TORUS AND REACTOR BUILDING TEMPERATURES ADVERSELY AFFECT THE OPERATION OF THE UPPER MOTOR BEARINGS. ON JUNE 10, 1991, AT 1510, WITH THE PLANT IN COLD SHUTDOWN, IT WAS ALSO DETERMINED AS A RESULT OF THE POSTULATED INCREASE IN REACTOR BUILDING TEMPERATURES NEAR THE TORUS AREA THAT THE WIDE RANGE TORUS LEVEL INSTRUMENTS DID NOT MEET THE ENVIRONMENTAL TEMPERATURE CONDITIONS POSTULATED DURING ACCIDENT CONDITIONS. NO SAFETY SYSTEMS WERE REQUIRED TO FUNCTION AS A RESULT OF THIS EVENT AND NO SAFETY CONSEQUENCES RESULTED FROM THIS EVENT.

[87] MILLSTONE 2 DOCKET 50-336 LER 90-022 REV 01
UPDATE ON SERVICE WATER HEADERS CROSS-TIED.
EVENT DATE: 111590 REPORT DATE: 070291 NSSS: CE TYPE: PWR
VENDOR: FISHER CONTROLS CO.

(NSIC 222487) ON 11/15/90, AT 1330, WITH THE PLANT IN MODE 1 (75% POWER, 565F, 2270 PSIG), SERVICE WATER HEADER CROSS-TIE VALVE, 2-SW-97A, WAS FOUND OPEN BY A PLANT ENGINEERING TECHNICIAN PERFORMING A ROUTINE INTAKE STRUCTURE INSPECTION. THE CONTROL ROOM WAS NOTIFIED AND AN OPERATOR MANUALLY CLOSED THE VALVE. NO EMERGENCY OPERATIONS WERE PERFORMED. NO EQUIPMENT WAS CYCLED TO ITS ACCIDENT POSITION. THE CAUSE OF THE EVENT WAS THE 2-SW-97A VALVE OPERATOR SHIFTING POSITION UPON RESTORATION OF ITS INSTRUMENT AIR SUPPLY, FOLLOWING MAINTENANCE ON AN UNRELATED INTAKE STRUCTURE AIR OPERATED VALVE. THIS EVENT IS BEING REPORTED PURSUANT TO THE REQUIREMENTS OF PARAGRAPH 50.73(A)(2)(I), REPORTING ANY OPERATION OR CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[88] MILLSTONE 2 DOCKET 50-336 LER 91-001 REV 01
UPDATE ON ELECTRO-HYDRAULIC CONTROL SYSTEM FAILURE CAUSED REACTOR TRIP.
EVENT DATE: 011091 REPORT DATE: 052491 NSSS: CE TYPE: PWR
VENDOR: DENISON DIVISION

(NSIC 222160) ON 1/10/91 AT 1612 HOURS, WITH THE UNIT OPERATING AT 92% POWER, THE MAIN TURBINE TRIPPED ON LOW ELECTRO-HYDRAULIC CONTROL (EHC) PRESSURE. THE RPS INITIATED A REACTOR TRIP FOLLOWING THE TURBINE TRIP. THE 'A' PUMP DISCHARGE FILTER PRESSURE ALARM ANNUNCIATED, INDICATING THAT THE DISCHARGE FILTER WAS PLUGGING. AT 1430 THE 'A' PUMP DISCHARGE PRESSURE DROPPED TO 1350 PSIG CAUSING THE 'B' PUMP TO AUTO-START. WHEN STARTED, THE 'B' PUMP COULD NOT MAINTAIN STABLE DISCHARGE PRESSURE DUE TO SEVERE VIBRATION. AFTER SEVERAL ADJUSTMENTS WERE MADE TO THE COMPENSATOR CONTROL ON THE 'B' PUMP, SYSTEM PRESSURE WAS STABILIZED AT 1475 PSIG. AT 1500 A DECISION WAS MADE TO LEAVE THE SYSTEM OPERATING IN ITS EXISTING CONFIGURATION ('B' PUMP RUNNING, 'A' PUMP IN AUTO-START). THE SUPERVISOR COMPLETED HIS PRE-JOB BRIEFING. THE MECHANICS MISTAKENLY BELIEVED THAT THEY HAD AUTHORIZATION TO BEGIN WORK ON THE PUMP AND BEGAN DISASSEMBLING THE 'A' PUMP FILTER. AT 1605 THE 'B' PUMP EXPERIENCED A VIBRATIONAL TRANSIENT AND SYSTEM MANIFOLD PRESSURE DECREASED TO 1350 PSIG. THE EHC SYSTEM PRESSURE CONTINED TO DROP CAUSING A TURBINE TRIP AND CORRESPONDING REACTOR TRIP, BEFORE THE 'A' PUMP IN AUTO-START).

[89] MILLSTONE 2 DOCKET 50-336 LER 91-008
REACTOR TRIP DUE TO LOW STEAM GENERATOR PRESSURE.
EVENT DATE: 052591 REPORT DATE: 062491 NSSS: CE TYPE: PWR

(NSIC 222488) ON 5/25/91, AT 1732, WITH THE REACTOR SUBCRITICAL IN MODE 3 (0% POWER, 511F, 2250 PSIG), THE REACTOR AUTOMATICALLY TRIPPED WHILE IT WAS BEING MANUALLY SHUT-DOWN. THE REACTOR TRIPPED ON LOW STEAM GENERATOR PRESSURE WHICH WAS CAUSED BY EXCESSIVE COOLING OF THE REACTOR COOLANT SYSTEM (RCS). THE CONTROL ROOM OPERATING SHIFT COMPLETED APPLICABLE PORTIONS OF EOP 2525, "STANDARD POST TRIP ACTIONS" AND EOP 2526, "REACTOR TRIP RECOVERY" WITHOUT FURTHER INCIDENT.

ALL EQUIPMENT RESPONDED AS EXPECTED AND THE REACTOR REMAINED IN A STABLE CONDITION. THE CAUSE OF THE EVENT WAS INEFFECTIVE COORDINATION BETWEEN THE REACTOR AND TURBINE SHUTDOWN. ACTIONS TO ADDRESS THE COORDINATION SHORTCOMINGS HAVE BEEN IMPLEMENTED AND PROCEDURE ENHANCEMENTS HAVE BEEN INCORPORATED TO ELIMINATE THE POSSIBILITY OF THIS SPECIFIC SCENARIO BEING REPEATED. THIS EVENT IS BEING REPORTED PURSUANT TO THE REQUIREMENTS OF PARAGRAPH 50.73(A)92)(IV), REPORTING ANY EVENT OR CONDITION THAT RESULTED IN MANUAL OR AUTOMATIC ACTUATION OF ANY ENGINEERED SAFETY FEATURE SYSTEM.

[90] MILLSTONE 3 DOCKET 50-423 LER 91-012
SPURIOUS CONTROL BUILDING ISOLATION SIGNALS DUE TO RADIATION MONITOR POWER SUPPLY FAILURE.
EVENT DATE: 050391 REPORT DATE: 060391 NSSS: WE TYPE: PWR
VENDOR: KAMAN SCIENCES CORP.

(NSIC 222185) AT 1430 HOURS, ON MAY 3, 1991, WITH THE PLANT IN MODE 1 AT 90% REACTOR POWER, 584F, AND 2250 PSIA, A SPURIOUS CONTROL BUILDING ISOLATION (CBI) SIGNAL WAS INITIATED BY THE B TRAIN CONTROL BUILDING VENTILATION INLET RADIATION MONITOR DURING POWER ASCENSION FOLLOWING THE THIRD REFUELING OUTAGE. THE REACTOR OPERATORS DETERMINED THAT THE B TRAIN RADIATION MONITOR WAS MALFUNCTIONING AND PROCEEDED TO BLOCK THE CBI SIGNAL PRIOR TO THE ACTUATION OF THE AUTOMATIC CONTROL BUILDING PRESSURIZATION. THE REACTOR OPERATORS THEN MANUALLY ALIGNED AND OPERATED THE VENTILATION SYSTEM WITH FULL FILTERED RECIRCULATED AIR FOR THE CONTROL ROOM PRESSURE ENVELOPE. A SUBSEQUENT INVESTIGATION REVEALED THAT THE SPURIOUS CBI SIGNAL WAS CAUSED BY A HIGH VOLTAGE POWER SUPPLY SPIKE. THE ROOT CAUSE OF THE MONITOR FAILURE WAS DETERMINED TO BE A FAILED SINGLE CHANNEL AREA MONITOR (SCAM) BOARD HIGH VOLTAGE POWER SUPPLY. THE SCAM BOARD WAS REPLACED AND SATISFACTORILY RETESTED. THE MONITOR WAS RETURNED TO SERVICE AT 1809 HOURS ON MAY 8, 1991.

[91] MILLSTONE 3 DOCKET 50-423 LER 91-013
INOPERABLE "B" TRAIN EMERGENCY DIESEL GENERATOR DUE TO INADEQUATE REQUIREMENTS REVIEW.
EVENT DATE: 051591 REPORT DATE: 061491 NSSS: WE TYPE: PWR
VENDOR: COLT INDUSTRIES, INC.

(NSIC 222328) AT 1630 HOURS ON 5/15/91, WITH THE PLANT AT 100% POWER IN MODE 1 AT A PRESSURE OF 2250 PSIA AND A TEMPERATURE OF 587F, A SENIOR LICENSED OPERATOR DISCOVERED THE TRAIN "B" EMERGENCY DIESEL GENERATOR (EDG) AIR RECEIVER 1B DEPRESSURIZED AND ISOLATED FOR MAINTENANCE WITH A SYSTEM CROSS-CONNECT VALVE CLOSED. THIS RESULTED IN ONLY AIR RECEIVER 2B BEING AVAILABLE TO START THE EDG ON ONE BANK OF CYLINDERS. THE EDG WILL START IN THIS LINEUP, BUT A START IN THE REQUIRED 11-SECOND TIME INTERVAL IS NOT ASSURED. CONSEQUENTLY, TRAIN "B" EDG WAS DECLARED INOPERABLE. THE ROOT CAUSE OF THIS EVENT WAS IMPROPER WORK PRACTICES IN THAT A SENIOR LICENSED OPERATOR FAILED TO VERIFY THE TECH SPEC (TS) REQUIREMENTS. TESTING HAS DEMONSTRATED THAT THE B EDG WILL START WITHIN 11 SECONDS IF THE BANK CROSS-CONNECT VALVE IS OPENED. A REVIEW OF THESE REQUIREMENTS FOR THE TRAIN "B" EDG WOULD HAVE INDICATED THAT IT WAS NECESSARY TO OPEN THE VALVE WITHIN ONLY ONE RECEIVER AVAILABLE TO START THE DIESEL. UPON DISCOVERY, THE 1B AIR RECEIVER WAS RETURNED TO SERVICE AND THE CROSS-CONNECT VALVE WAS CLOSED AS REQUIRED BY THE NORMAL LINEUP. THE SENIOR LICENSED OPERATOR INVOLVED WITH PLACING THE RECEIVER OUT OF SERVICE WAS COUNSELED ON THE IMPORTANCE OF REVIEWING THE TS REQUIREMENTS ASSOCIATED WITH REMOVING SAFETY SYSTEM COMPONENTS FROM SERVICE.

[92] MONTICELLO DOCKET 50-263 LER 91-011
FAILURE TO VOLTAGE REGULATOR CHIPS IN CONTROL CIRCUITRY CAUSES CONTAINMENT COMBUSTIBLE GAS CONTROL SYSTEM INOPERABILITY.
EVENT DATE: 051391 REPORT DATE: 061291 NSSS: GE TYPE: BWR
VENDOR: NATIONAL SEMI CONDUCTOR
ROCKWELL-INTERNATIONAL

(NSIC 222301) WHILE TESTING CONTAINMENT COMBUSTIBLE GAS CONTROL SYSTEM DURING THE 1991 REFUELING OUTAGE BOTH DIVISIONS OF THE SYSTEM WERE DETERMINED TO BE

INOPERABLE. THE COMBUSTIBLE GAS CONTROL SYSTEM WAS NOT REQUIRED TO BE OPERABLE WHILE THE PLANT WAS IN COLD SHUTDOWN, BUT THE CONDITION COULD HAVE PREVENTED THE SYSTEM FROM PERFORMING ITS FUNCTION WHILE AT POWER. SET POINTS FOR SEVERAL INSTRUMENT CHANNELS HAD DRIFTED, WHICH WOULD HAVE PREVENTED BOTH DIVISIONS OF THE COMBUSTIBLE GAS CONTROL SYSTEM FROM REACHING THE TEMPERATURE REQUIRED FOR THE RECOMBINATION OF OXYGEN AND HYDROGEN PRESENT IN THE PRIMARY CONTAINMENT ATMOSPHERE AFTER A LOSS OF COOLANT ACCIDENT. THE ROOT CAUSE OF THE PROBLEM WAS THE FAILURE OF A VOLTAGE REGULATOR CHIP FOR CIRCUIT BOARDS WHICH COMPRISE EACH INSTRUMENT CHANNEL. THE PROBLEM WAS CORRECTED BY REPAIRING THE AFFECTED INSTRUMENT CHANNELS. AFTER THE REPAIRS, ALL OF THE AFFECTED INSTRUMENT CHANNELS WERE TESTED WITH SATISFACTORY RESULTS. TO PREVENT RECURRENCE, PROCEDURES USED TO CALIBRATE AND OPERATE THE SYSTEM WILL BE REVISED TO INCLUDE CHECKS TO DETECT SIMILAR FAILURES.

[93] MONTICELLO DOCKET 50-263 LER 91-012
MOMENTARY LOSS OF POWER TO CONTROL LOGIC RESULTS IN UNPLANNED EMERGENCY FILTRATION TRAIN ACTUATION.
EVENT DATE: 052491 REPORT DATE: 062191 NSSS: GE TYPE: BWR

(NSIC 222367) DURING THE 1991 REFUELING OUTAGE, BOTH TRAINS OF THE CONTROL ROOM VENTILATION-EMERGENCY FILTRATION TRAIN SYSTEM WERE TRIPPED INTO THE EMERGENCY MODE OF OPERATION. THE CAUSE OF THE EVENT WAS A MOMENTARY LOSS OF POWER TO CONTROL LOGIC DURING AN AUTOMATIC TRANSFER OF STATION POWER FROM THE 2R TRANSFORMER TO THE 1R TRANSFORMER. ONCE THE CAUSE OF THE EVENT WAS DETERMINED, CONTROL ROOM PERSONNEL RESTORED BOTH TRAINS OF THE CONTROL ROOM VENTILATION-EMERGENCY FILTRATION TRAIN SYSTEM TO NORMAL OPERATING STATUS. CORRECTIVE ACTIONS INCLUDE REVISING SETPOINTS FOR THE CURRENT LIMITING FUSES AND GROUND DETECTION RELAY THAT MONITOR THE SOURCE TO 2R TRANSFORMER AND DETERMINING THE FEASIBILITY OF ELIMINATING THE AUTOMATIC ACTUATION OF THE CONTROL ROOM VENTILATION-EMERGENCY FILTRATION TRAIN SYSTEM.

[94] MONTICELLO DOCKET 50-263 LER 91-013
INSUFFICIENT MAINTENANCE OF REACTOR PROTECTION SYSTEM MONITOR GENERATOR SET OVERVOLTAGE RELAY CAUSES STANDBY GAS INITIATION AND REACTOR BUILDING ISOLATION.
EVENT DATE: 060191 REPORT DATE: 070191 NSSS: GE TYPE: BWR

(NSIC 222403) ON 6/1/91 WITH THE PLANT OPERATING AT 90% POWER, A PARTIAL GROUP II ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION OCCURRED. THIS WAS THE RESULT OF LOSS OF POWER TO THE CHANNEL "A" SPENT FUEL POOL AND REACTOR BUILDING EXHAUST PLENUM RADIATION MONITORS DUE TO THE INADVERTENT TRIP OF THE "A" REACTOR PROTECTION SYSTEM MOTOR GENERATOR SET. THE ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION WERE UNPLANNED ACTUATIONS OF ENGINEERED SAFETY FEATURES. POWER WAS IMMEDIATELY RESTORED TO THE "A" REACTOR PROTECTION SYSTEM BUS FROM AN ALTERNATE SOURCE. THE PARTIAL GROUP II ISOLATION WAS RESET AND THE STANDBY GAS TREATMENT SYSTEM WAS PLACED BACK IN STANDBY. THE CAUSE OF THE MOTOR GENERATOR SET TRIP WAS DETERMINED TO BE INADEQUATE MAINTENANCE OF THE GENERATOR SET'S OVERVOLTAGE RELAY. THE OVERVOLTAGE RELAY SETPOINT WAS LATER RE-CALIBRATED AND THE GENERATOR SET RETURNED TO SERVICE. THE OVERVOLTAGE RELAY FOR THE "B" MOTOR GENERATOR SET WILL ALSO BE RE-CALIBRATED. THE NEED FOR FURTHER MOTOR GENERATOR SET COMPONENT MAINTENANCE WILL BE EVALUATED. TRANSFER OF THE RADIATION MONITORS TO A MORE RELIABLE POWER SUPPLY WILL BE INVESTIGATED.

[95] MONTICELLO DOCKET 50-263 LER 91-014
SPURIOUS SIGNAL CAUSES MAIN STEAM ISOLATION AND REACTOR SCRAM.
EVENT DATE: 060591 REPORT DATE: 070591 NSSS: GE TYPE: BWR

(NSIC 222404) A REACTOR SCRAM OCCURRED FROM 100% POWER DUE TO A SPURIOUS MAIN STEAM ISOLATION SIGNAL OCCURRING IN THE A CHANNEL WHILE THE B CHANNEL WAS TRIPPED IN ACCORDANCE WITH AN APPROVED SURVEILLANCE PROCEDURE WHICH WAS IN PROGRESS AT THE TIME. NORMAL POST-SCRAM RECOVERY PROCEDURES WERE FOLLOWED. INVESTIGATIVE EFFORTS TO DETERMINE THE SOURCE OF THE SIGNAL DID NOT IDENTIFY THE ROOT CAUSE. AFTER THE SAFETY ASPECTS OF THE OCCURRENCE WERE EXAMINED AND FOUND TO BE ACCEPTABLE, THE PLANT WAS RESTARTED WITH DATA GATHERING INSTRUMENTATION IN PLACE

TO GIVE BETTER RESOLUTION IN THE EVENT OF A RECURRENCE. PLANT INSTRUMENT AND CONTROL PERSONNEL WERE REMINDED THAT PASSIVE PREPARATION WORK SHOULD NEVER GO BEYOND THE CHANNEL THAT IS CURRENTLY TRIPPED. A TECHNICAL SPECIFICATION REVISION IS BEING PURSUED WHICH WILL MINIMIZE THE TIME CHANNELS ARE TRIPPED DURING SURVEILLANCE PROCEDURES. IMPROVEMENTS ARE BEING CONSIDERED FOR THE SENSOR RELAY INPUTS TO THE PLANT COMPUTER.

[96] NINE MILE POINT 1 DOCKET 50-220 LER 91-006
CONTAINMENT H2/O2 MONITORING INOPERABLE DUE TO NON-SAFETY RELATED GAS.
EVENT DATE: 050891 REPORT DATE: 060591 NSSS: GE TYPE: BWR.

(NSIC 222197) DURING THE PERFORMANCE OF N1-1SP-201-M020, ON DECEMBER 11, 1990, AT APPROXIMATELY 2025, 11 AND 12 CONTAINMENT HYDROGEN AND OXYGEN MONITORS WERE FOUND TO CONTAIN NON-SAFETY RELATED CALIBRATION GASES. THE MODE SWITCH WAS IN THE RUN POSITION AND REACTOR POWER WAS AT 1811 MW. AN EVALUATION WAS PERFORMED BY NIAGARA MOHAWK LICENSING DEPARTMENT FOR PAST OPERABILITY. IT HAS DETERMINED THAT THE CALIBRATION GAS SHOULD HAVE BEEN SAFETY RELATED SINCE DECEMBER 10, 1987. THE CAUSE OF THIS CONDITION WAS A PROGRAMMATIC DEFICIENCY. THE SPECIFIC CAUSE RESULTED FROM A LACK OF GUIDELINES OR A PROGRAM THAT ADDRESSES NECESSARY ACTIONS AND REVIEWS WHICH SHOULD TAKE PLACE WHEN COMPONENTS OR SYSTEMS SAFETY CLASSIFICATIONS ARE CHANGED FROM NON-SAFETY TO SAFETY RELATED. CORRECTIVE ACTIONS TAKEN WERE TO INITIATE THE PRE-PLANNED ALTERNATE SAMPLING METHOD, TO INSTALL SAFETY RELATED CALIBRATION GAS BOTTLES AND REVISE PROCUREMENT REQUIREMENTS TO ENSURE INSTALLATION OF SAFETY RELATED GASES IN THE FUTURE. SPECIAL REPORT NMP73977 DATED DECEMBER 22, 1990, WAS WRITTEN AS REQUIRED BY TECH SPECS .

[97] NINE MILE POINT 2 DOCKET 50-410 LER 90-005 REV 02
UPDATE ON REACTOR WATER CLEANUP SYSTEM ISOLATION (ENGINEERED SAFETY FEATURE) DUE TO HIGH DIFFERENTIAL FLOW SIGNAL CAUSED BY EQUIPMENT FAILURE.
EVENT DATE: 020690 REPORT DATE: 053091 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 222180) THIS SUPPLEMENT IS A COMPLETE REVISION OF LER 90-05 AND ADDRESSES ONE OF THE TWO EVENTS REPORTED IN LER 90-05. ON FEBRUARY 6, 1990, AT 2025 HOURS, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED THE ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF), SPECIFICALLY, ISOLATION OF THE REACTOR WATER CLEANUP SYSTEM (WCS). AT THE TIME OF THE EVENT, THE PLANT WAS AT TWELVE (12) PERCENT POWER WITH THE MODE SWITCH IN THE "RUN" POSITION (OPERATIONAL CONDITION 1). THE SYSTEM ISOLATION, INITIATED BY A HIGH DIFFERENTIAL FLOW SIGNAL, OCCURRED AS THE PLANT OPERATORS WERE ATTEMPTING TO PUT WCS IN SERVICE. THE ROOT CAUSE FOR THIS EVENT HAS BEEN DETERMINED TO BE EQUIPMENT FAILURE. IMMEDIATE CORRECTIVE ACTIONS WERE FOR THE LICENSED OPERATORS TO VERIFY THE AUTOMATIC RESPONSE OF WCS, VERIFY THE PLANT STATUS AS NORMAL, RESET THE ISOLATION AND RESTORE WCS TO SERVICE. ADDITIONAL CORRECTIVE ACTION INCLUDED ISSUING A WORK REQUEST TO REMOVE FLOW TRANSMITTER 2WCS*FT67X FROM SERVICE AND RETURN IT TO MANUFACTURER FOR A FAILURE ANALYSIS.

[98] NINE MILE POINT 2 DOCKET 50-410 LER 91-009
ENGINEERED SAFETY FEATURE ACTUATION DUE TO A SPURIOUS HIGH RADIATION LEVEL SIGNAL.
EVENT DATE: 050391 REPORT DATE: 060391 NSSS: GE TYPE: BWR
VENDOR: KAMAN SCIENCES CORP.

(NSIC 222388) ON MAY 3, 1991, AT 0831 HOURS, NINE MILE POINT UNIT 2 EXPERIENCED AN ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF). SPECIFICALLY, THE SECONDARY CONTAINMENT (REACTOR BUILDING) ISOLATED AND THE REACTOR BUILDING EMERGENCY RECIRCULATION UNIT COOLER AND STANDBY GAS TREATMENT SYSTEM (GTS) STARTED AUTOMATICALLY. THE ESF ACTUATION WAS INITIATED BY A HIGH RADIATION LEVEL SIGNAL IN THE REACTOR BUILDING VENTILATION SYSTEM (HVR). AT THE TIME OF THE EVENT, THE REACTOR MODE SWITCH WAS IN THE POSITION (MODE 1) WITH THE REACTOR OPERATING AT 100% RATED THERMAL POWER. THE ROOT CAUSE OF THE EVENT IS STILL UNDER INVESTIGATION. THE CONTROL ROOM OPERATORS IMPLEMENTED THE EMERGENCY OPERATING PROCEDURE (EOP) FOR SECONDARY CONTAINMENT CONTROL UNTIL REACTOR BUILDING RADIATION WAS VERIFIED AT NORMAL OPERATING LEVELS AND THE CAUSE FOR THE ESF

ACTUATION WAS DETERMINED TO BE A SPURIOUS HIGH RADIATION LEVEL TRIP. OTHER CORRECTIVE ACTIONS INCLUDED RETURNING THE HVR SYSTEM TO A NORMAL LINE UP AFTER WELDING WAS COMPLETE, REPLACING A FAILED COMPONENT IN THE RADIATION MONITORING CABINET, AND AS AN INTERIM MEASURE, ISSUING GUIDANCE TO WELDERS TO ENSURE THEIR WELDING CABLES ARE NOT IN CONTACT WITH ANY INSTRUMENTATION CABLES.

[99]   NINE MILE POINT 2   DOCKET 50-410   LER 91-011
TECHNICAL SPECIFICATION VIOLATION, MISSED CHEMISTRY SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 050691   REPORT DATE: 060591   NSSS: GE   TYPE: BWR

(NSIC 222183) AT APPROXIMATELY 1200 HOURS ON MAY 7, 1991, IT WAS DISCOVERED THAT THE REACTOR WATER GROSS BETA AND GAMMA ACTIVITY MEASUREMENTS HAD NOT BEEN PERFORMED WITHIN THE REQUIRED 72 HOURS. THESE MEASUREMENTS WERE REQUIRED TO BE COMPLETED BY 0040 HOURS ON MAY 6, 1991, BUT WERE NOT COMPLETED UNTIL 0130 HOURS ON MAY 7, 1991. THIS IS A VIOLATION OF STATION TECHNICAL SPECIFICATION SECTION 4.4.5, TABLE 4.4.5.1(1). AT THE TIME OF THE EVENT, THE REACTOR MODE SWITCH WAS IN THE "RUN" POSITION (MODE 1) AND THE REACTOR WAS OPERATING AT 100% OF RATED THERMAL POWER. THE ROOT CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE PERSONNEL ERROR. CORRECTIVE ACTIONS INCLUDE: 1) GROSS BETA AND GAMMA ACTIVITY ANALYSES ARE BEING PERFORMED DAILY; 2) THE METHODS USED TO TRACK AND REVIEW THE COMPLETION OF FREQUENTLY PERFORMED TECHNICAL SPECIFICATION TASKS IS BEING REVIEWED FOR IMPROVEMENTS; 3) ALL CHEMISTRY DEPARTMENT PERSONNEL HAVE BEEN INFORMED OF THE MISSED SURVEILLANCE; 4) AN ACCOUNTABILITY MEETING WAS HELD TO ASCERTAIN CORRECTIVE ACTIONS AND LESSONS LEARNED; 5) A LESSONS LEARNED TRANSMITTAL WILL BE WRITTEN CONCERNING TRACKING OF FREQUENTLY PERFORMED SURVEILLANCES; AND 6) A DEVIATION/EVENT REPORT HAS BEEN WRITTEN TO EVALUATE THE EFFECTIVENESS OF THE PRESENT LESSONS LEARNED PROGRAM.

[100]   NINE MILE POINT 2   DOCKET 50-410   LER 91-010
REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION DUE TO A SPURIOUS REACTOR BUILDING HIGH AREA TEMPERATURE SIGNAL.
EVENT DATE: 050891   REPORT DATE: 060791   NSSS: GE   TYPE: BWR
VENDOR: RILEY COMPANY, THE - PANALARM DIVISION

(NSIC 222182) ON MAY 8, 1991, AT 0757 HOURS, WITH THE REACTOR AT 99 PERCENT RATED THERMAL POWER, AND THE MODE SWITCH IN THE "RUN" POSITION (OPERATIONAL CONDITION 1), NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED AN ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF). SPECIFICALLY, THE PRIMARY CONTAINMENT INBOARD ISOLATION VALVES FOR THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM ISOLATED ON A SPURIOUS DIVISION II REACTOR BUILDING GENERAL AREA HIGH TEMPERATURE SIGNAL. THE MOST PROBABLE SOURCE OF THE SPURIOUS HIGH TEMPERATURE SIGNAL HAS BEEN IDENTIFIED AS A PANALARM (RILEY) MODEL 86 THERMOCOUPLE MONITOR. THE SPECIFIC CAUSE OF THE SIGNAL, HOWEVER, COULD NOT BE DETERMINED. THE IMMEDIATE ACTIONS TAKEN WERE TO: VERIFY THAT A HIGH REACTOR BUILDING AREA TEMPERATURE DID NOT EXIST; AND INITIATE AN INVESTIGATION INTO THE CAUSE FOR THE HIGH TEMPERATURE SIGNAL. ADDITIONAL ACTIONS INCLUDED: THE SATISFACTORY COMPLETION OF TEMPERATURE CIRCUIT FUNCTIONAL TESTS; RETURNING THE RCIC SYSTEM TO ITS NORMAL STANDBY CONDITION; ISSUING A LESSONS LEARNED TRANSMITTAL; AND CONDUCTING AN ACCOUNTABILITY MEETING.

[101]   NINE MILE POINT 2   DOCKET 50-410   LER 91-012
ACTUATION OF SEVERAL ENGINEERED SAFETY FEATURES CAUSED BY A LOSS OF ONE OFFSITE POWER FEED, DUE TO EXCAVATION IN SCRIBA STATION.
EVENT DATE: 052191   REPORT DATE: 062091   NSSS: GE   TYPE: BWR

(NSIC 222325) ON MAY 21, 1991, AT 0954 HOURS, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED ACTUATIONS OF SEVERAL ENGINEERED SAFETY FEATURES (ESF) DUE TO A LOSS OF ONE OF THE TWO OFFSITE POWER FEED LINES. AT THE TIME OF THE EVENT, THE REACTOR MODE SWITCH WAS IN THE "RUN" POSITION (MODE 1) AND THE REACTOR WAS OPERATING AT 100% RATED THERMAL POWER. THE ROOT CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE INADEQUATE MANAGERIAL METHODS. CORRECTIVE ACTIONS INCLUDE: (1) STABILIZATION OF THE PLANT IN RESPONSE TO THE LOSS OF ONE 115 KV FEEDER; (2) ALIGNMENT OF DIVISION I AND III SWITCHGEAR TO THE REMAINING NUMBER 6 LINE 115 KV FEEDER AND ENTRY INTO

A 72 HOUR LIMITING CONDITION OF OPERATION (LCO) PER TECHNICAL SPECIFICATION 3.8.1.1.A; (3) BRINGING OF WORK ACTIVITIES RELATED TO THE SCRIBA STATION INTO THE ENVELOPE OF CONTROL OF THE UNIT 2 PLANT MANAGER; (4) EXPLICIT DIRECTIONS FROM NIAGARA MOHAWK POWER DELIVERY SUPERVISION WITH REGARDS TO SCRIBA STATION ACCESS FOR ANY TYPE OF ACTIVITY.

[102] NINE MILE POINT 2 DOCKET 50-410 LER 91-013
REACTOR WATER CLEANUP SYSTEM ISOLATION CAUSED BY HIGH DIFFERENTIAL FLOW SIGNAL DUE TO INADEQUATE SYSTEM OPERATION.
EVENT DATE: 060391 REPORT DATE: 062891 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 222463) ON 6/3/91, AT 1717 HOURS, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF). THIS EVENT CONSISTED OF AN ISOLATION OF THE REACTOR WATER CLEANUP SYSTEM (WCS). CLOSURE OF THE WCS OUTBOARD ISOLATION VALVE WAS INITIATED BY CONTROL ROOM OPERATORS IMMEDIATELY BEFORE A HIGH DIFFERENTIAL FLOW ISOLATION SIGNAL WAS RECEIVED. THE ISOLATION SIGNAL OCCURRED AS PLANT OPERATORS WERE MANIPULATING WCS FILTER/DEMINERALIZER UNITS. AT THE TIME OF THE EVENT, THE REACTOR MODE SWITCH WAS IN THE "RUN" POSITION (MODE 1) AND THE REACTOR WAS OPERATING AT 100% RATED THERMAL POWER. THE ROOT CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE INADEQUATE SYSTEM OPERATION DUE TO UNRELIABLE FILTER/DEMINERALIZER LEVEL INDICATION. THE IMMEDIATE CORRECTIVE ACTIONS WERE FOR OPERATORS TO VERIFY AUTOMATIC WCS SYSTEM RESPONSE AND DETERMINE THERE WAS NO ACTUAL WCS BREACH. OTHER CORRECTIVE ACTIONS INCLUDE: A CHANGE TO OPERATING PROCEDURE N2-OP-37; A REQUEST TO HAVE MAINTENANCE AND OPERATION TRAINING PROGRAMS REVIEW THIS EVENT; A PLANT CHANGE REQUEST TO IMPROVE LEVEL ELEMENT RELIABILITY; ISSUANCE OF A TEMPORARY MODIFICATION TO SIMPLIFY OPERATION; AND IMPLEMENTATION OF A COMPREHENSIVE SYSTEM OPERATIONAL STUDY.

[103] NORTH ANNA 1 DOCKET 50-338 LER 91-010
AUTO START OF EDG 1J DUE TO LOSS OF POWER FROM "A" RESERVE STATION SERVICE TRANSFORMER AND SUBSEQUENT START DUE TO INADVERTENT BYPASS OF AIR START SOLENOID OPERATED VALVE.
EVENT DATE: 042391 REPORT DATE: 051691 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 222036) AT 0920 HOURS ON 4/23/91, WITH UNIT 1 OPERATING AT 100 PERCENT POWER (MODE 1), THE "A" RESERVE STATION SERVICE TRANSFORMER (RSST) ISOLATED BECAUSE OF AN INADVERTENT OVERCURRENT AUXILIARY RELAY TRIP. THE CAUSE OF THE "A" RSST ISOLATION WAS PERSONNEL ERROR ASSOCIATED WITH IMPROPER INSTALLATION OF AN OVERCURRENT TRIP AUXILIARY RELAY. THE UNIT 1 "J" EMERGENCY DIESEL GENERATOR (EDG) AUTO-STARTED AND RE-ENERGIZED THE UNIT 1 "J" EMERGENCY BUS. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV) AS AN AUTOMATIC ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF). A FOUR-HOUR REPORT WAS MADE IN ACCORDANCE WITH 10CFR50.72(B)(2)(II) AT 1106 HOURS. AT 0726 HOURS ON APRIL 26, 1991, WITH UNIT 1 AT 100 PERCENT POWER, THE 1J EDG WAS INADVERTENTLY STARTED DUE TO PLANT PERSONNEL ACCIDENTLY BYPASSING THE AIR START SOV VIA THE MANUAL OVERRIDE WHILE PERFORMING A WALKDOWN TO PLAN A WORK ORDER. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV) AS AN AUTOMATIC ESF ACTUATION. A FOUR-HOUR REPORT WAS MADE IN ACCORDANCE WITH 10CFR50.72(B)(2)(II) AT 0908 HOURS. THE 1J EDG FUNCTIONED PROPERLY TO RESTORE POWER TO THE UNIT 1 "J" EMERGENCY BUS. ALL OTHER AUTOMATIC ACTUATIONS ALSO FUNCTIONED PROPERLY.

[104] NORTH ANNA 1 DOCKET 50-338 LER 91-011
UNIT SHUTDOWN DUE TO EXCEEDING TECH SPEC LIMIT FOR REACTOR COOLANT SYSTEM PRESSURE BOUNDARY LEAKAGE.
EVENT DATE: 051191 REPORT DATE: 060691 NSSS: WE TYPE: PWR

(NSIC 222376) AT 0400 HOURS ON MAY 11, 1991, WITH UNIT 1 AT 30 PERCENT POWER, A CONTAINMENT ENTRY TEAM IDENTIFIED A SOURCE OF LEAKAGE BETWEEN THE "B" COLD LEG LOOP STOP VALVE AND THE ISOLATION VALVE FOR THE 3/4 INCH (3/8 INCH INSIDE DIAMETER) UPPER DISC PRESSURIZATION LINE, AND THE ACTION STATEMENT OF TECHNICAL SPECIFICATION 3.4.6.2 WAS ENTERED. A UNIT SHUTDOWN COMMENCED AT 0500 HOURS DUE TO

AN UNISOLABLE REACTOR COOLANT SYSTEM (RCS) PRESSURE BOUNDARY LEAK. A NOTIFICATION OF UNUSUAL EVENT (NOUE) WAS DECLARED IN ACCORDANCE WITH THE EMERGENCY PLAN, AND ALL APPROPRIATE NOTIFICATIONS WERE MADE IN A TIMELY MANNER. THE UNIT WAS PLACED IN COLD SHUTDOWN (MODE 5) AT 1640 HOURS. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(A) AS A COMPLETION OF A PLANT SHUTDOWN REQUIRED BY TS. A ONE HOUR REPORT WAS MADE PURSUANT TO 10CFR50.72(A)(I). A VISUAL INSPECTION OF THE PIPE CRACK INDICATED A PROBABLE HIGH CYCLE FATIGUE FAILURE MECHANISM. NO SIGNIFICANT SAFETY CONSEQUENCES RESULTED FROM THE PIPE CRACK BECAUSE THE LEAK, APPROXIMATELY 0.7 GPM, WAS MUCH LESS THAN THE NORMAL CHARGING SYSTEM MAKE UP CAPACITY. IN ADDITION, THERE ARE NO PIPE WHIP OR IMPINGEMENT CONCERNS ASSOCIATED WITH THE DISC PRESSURIZATION PIPING. THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED AT ANY TIME DURING THIS EVENT BECAUSE THE LEAK WAS CONFINED TO THE CONTAINMENT STRUCTURE AND NO RELEASE WAS MADE TO THE ENVIRONMENT.

[105] NORTH ANNA 1 DOCKET 50-338 LER 91-012
AUXILIARY FEEDWATER PUMP AUTO-START SIGNAL RECEIVED.
EVENT DATE: 052091 REPORT DATE: 061191 NSSS: WE TYPE: PWR

(NSIC 222161) AT 0603 HOURS ON MAY 20, 1991, WITH UNIT 1 IN MODE 4 (HOT SHUTDOWN) AND HEATING UP IN PREPARATION FOR UNIT START-UP, THE AUXILIARY FEEDWATER PUMPS (AFWP) RECEIVED AN AUTO START SIGNAL WHEN THE CIRCUIT BREAKERS FOR THE "C" MAIN FEEDWATER PUMP (FWP) OPENED DUE TO A HI-HI LEVEL IN THE "C" STEAM GENERATOR (SG). THE "C" SG HI-HI LEVEL WAS CAUSED BY OPENING THE "C" MAIN STEAM (MS) NON-RETURN VALVE (NRV) WITH A SUFFICIENT DIFFERENTIAL PRESSURE TO INDUCE A SWELL IN SG WATER LEVEL. A FOUR HOUR REPORT WAS MADE IN ACCORDANCE WITH 10CFR50.72(B)(2)(II) AT 0929 HOURS. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV) AS AN AUTOMATIC ACTUATION OF AN ENGINEERED SAFETY FEATURE. THE EVENT WAS CAUSED BY PERSONNEL ERROR. STEAM GENERATOR LEVEL RETURNED TO NORMAL, FEEDWATER(FW) ISOLATION WAS RESET, AND THE MAIN FEEDWATER PUMP BREAKERS WERE RE-CLOSED TO RESTORE FEED CAPABILITY FROM THE CONDENSATE SYSTEM. THE AFWP'S ARE NOT REQUIRED UNTIL MODE 3 AND WERE IN "PULL-TO-LOCK". THEREFORE, AN ACTUAL AUTO-START DID NOT OCCUR. SUFFICIENT FEEDWATER WAS AVAILABLE AND ALL SYSTEMS FUNCTIONED AS DESIGNED. THE PLANT REMAINED STABLE DURING THE EVENT. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED AT ANY TIME DURING THIS EVENT.

[106] NORTH ANNA 1 DOCKET 50-338 LER 91-013
MISSED SURVEILLANCE ON QUENCH SPRAY PUMP DISCHARGE MOTOR OPERATED VALVES.
EVENT DATE: 052291 REPORT DATE: 061391 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)

(NSIC 222196) ON MAY 22, 1991 WITH UNIT 1 IN MODE 2 AT 3 PERCENT POWER AND UNIT 2 IN MODE 1 AT 100 PERCENT POWER IT WAS DETERMINED DURING A REVIEW OF THE INSERVICE TESTING (IST) PROGRAM IMPLEMENTATION PROCEDURES THAT STROKE TIME SURVEILLANCE REQUIREMENTS FOR CLOSED DIRECTION TESTING OF FOUR QUENCH SPRAY PUMP DISCHARGE MOTOR OPERATED VALVES HAD NOT BEEN PERFORMED. THIS IS A VIOLATION OF TECHNICAL SPECIFICATION 4.0.5 AND THEREFORE REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B). THE CAUSE OF THE EVENT WAS PERSONNEL ERROR RESULTING IN THE REQUIREMENTS OF THE IST PROGRAM NOT BEING ADEQUATELY IMPLEMENTED. THE CURRENT TEST PROCEDURE REQUIRES TESTING OF THE SUBJECT VALVES IN THE OPEN DIRECTION ONLY. UPON DETERMINATION THAT THE SURVEILLANCES WERE NOT MET, THE AFFECTED VALVES WERE SUCCESSFULLY TESTED. THESE INCIDENTS POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE THE VALVES WERE CAPABLE OF PERFORMING THEIR INTENDED SAFETY FUNCTION. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED AT ANY TIME DURING THIS EVENTS.

[107] NORTH ANNA 2 DOCKET 50-339 LER 91-002
AUTOMATIC START OF THE 2H EMERGENCY DIESEL GENERATOR CAUSED BY DE-ENERGIZATION OF THE 2H EMERGENCY BUS DUE TO PERSONNEL ERROR.
EVENT DATE: 051491 REPORT DATE: 060391 NSSS: WE TYPE: PWR

(NSIC 222162) AT 0000 HOURS ON MAY 14, 1991, WITH UNIT 2 AT 97% POWER (MODE 1) AND UNIT 1 IN COLD SHUTDOWN (MODE 5); THE UNIT 2 EMERGENCY BUS 2H WAS INADVERTENTLY DE-ENERGIZED DUE TO ISOLATION FROM THE RESERVE STATION SERVICE

TRANSFORMER B. THIS CAUSED AN AUTOMATIC ACTUATION OF THE 2H EDG DUE TO LOSS OF VOLTAGE ON THE 2H BUS. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV) AS AN AUTOMATIC ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF). A FOUR-HOUR REPORT WAS MADE IN ACCORDANCE WITH 10CFR50.72(B)(2)(II) AT 0131 HOURS ON MAY 14, 1991. THIS EVENT WAS CAUSED BY PERSONNEL ERROR. THE CONTROL ROOM OPERATOR DID NOT PROPERLY FOLLOW THE OPERATING PROCEDURE AND ACCIDENTLY MANIPULATED THE WRONG CIRCUIT BREAKER WHILE TRANSFERRING THE UNIT 1 B STATION SERVICE BUS TO THE B RESERVE STATION SERVICE TRANSFORMER. NO SIGNIFICANT SAFETY CONSEQUENCES OCCURRED BECAUSE THE REDUNDANT EMERGENCY BUS 2J REMAINED AVAILABLE TO SUPPLY POWER TO REQUIRED PLANT EQUIPMENT. THE 2H EDG FUNCTIONED PROPERLY AND RE-ENERGIZED THE 2H EMERGENCY BUS. ALL OTHER AUTOMATICALLY ACTUATED EQUIPMENT FUNCTIONED AS DESIGNED. THE HEALTH AND SAFETY THE PUBLIC WERE NOT AFFECTED AT ANY TIME DURING THIS EVENT.

[108] NORTH ANNA 2 DOCKET 50-339 LER 91-003
MISSED SURVEILLANCE ON SAFETY INJECTION SYSTEM ACCUMULATOR NITROGEN SUPPLY AND VENT VALVES DUE TO PERSONNEL ERROR.
EVENT DATE: 052291 REPORT DATE: 060791 NSSS: WE TYPE: PWR

(NSIC 222377) ON MAY 22, 1991 WITH UNIT 2 IN MODE 1 (100 PERCENT POWER) IT WAS DETERMINED, THROUGH AN INSERVICE TESTING PROGRAM IMPLEMENTATION ASSESSMENT, THAT A SURVEILLANCE TEST ON FOUR SAFETY INJECTION SYSTEM ACCUMULATOR NITROGEN SUPPLY AND VENT VALVES WAS MISSED. THIS IS A VIOLATION OF TECHNICAL SPECIFICATION 4.0.5 AND THEREFORE REPORTABLE PURSUANT TO 10CFR50.73 (A)(2)(I)(B). THE CAUSE OF THE EVENT WAS PERSONNEL ERROR. THE PTSS COORDINATOR, WHO IS RESPONSIBLE FOR TEST SCHEDULING, DID NOT REVISE THE PERFORMANCE MODES FOR 2-PT-212.18 THAT WERE APPROVED AND DOCUMENTED ON A PROCEDURE ACTION REQUEST. UPON DETERMINATION THAT THE SURVEILLANCE WAS NOT MET, TECHNICAL SPECIFICATION 4.0.2 WAS ENTERED, AND THE AFFECTED VALVES WERE SUCCESSFULLY TESTED. THIS INCIDENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE THE VALVES WERE CAPABLE OF PERFORMING THEIR INTENDED SAFETY FUNCTION AS DEMONSTRATED DURING TESTING THAT WAS CONDUCTED ON MAY [illegible], 1991. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED AT ANY TIME DURING THESE EVENTS.

[109] OCONEE 1 DOCKET 50-269 LER 91-004
TECHNICAL INOPERABILITY OF AN EMERGENCY ELECTRICAL POWER PATH DUE TO INCORRECT RELAY TRIP SETPOINTS RESULTS FROM INAPPROPRIATE ACTION.
EVENT DATE: 050691 REPORT DATE: 060591 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 222368) ON 5/6/91, AT APPROX. 1500 HOURS WITH ALL 3 OCONEE UNITS OPERATING AT 100% FULL POWER, IT WAS DISCOVERED DURING A REVIEW OF RELAY SETPOINTS THAT ZONE PROTECTION DIFFERENTIAL RELAYS IN THE 230 KV SWITCHYARD WERE INCORRECT. THE REVIEW HAD BEEN INITIATED AS AN ACTION ITEM ASSOCIATED WITH A DESIGN BASIS DOCUMENT PREPARATION. AS A RESULT OF SUBSEQUENT INVESTIGATIONS, THE UNIT 1 STARTUP TRANSFORMER (CT-1) AND THE ELECTRICAL BUS WHICH PROVIDES AN EMERGENCY POWER PATH TO ALL THREE UNITS (YELLOW BUS), BOTH OF WHICH ARE REQUIRED BY TECH SPECS, WERE FOUND TO BE SUBJECT TO INADVERTENT LOCKOUTS IF A FAULT OCCURRED ON CERTAIN NON-SAFETY RELATED EQUIPMENT. AT 1800 HOURS ON 5/6/91, CT-1 WAS DECLARED TECHNICALLY INOPERABLE. THE YELLOW BUS WAS DECLARED TECHNICALLY INOPERABLE AT 1030 HOURS ON 5/7/91. AFTER CORRECTING THE RELAY SETPOINTS, ALL EMERGENCY POWER EQUIPMENT WAS DECLARED OPERABLE ON 5/7/91 AT 1740 HOURS. THE ROOT CAUSE OF THIS EVENT WAS INAPPROPRIATE ACTION, INATTENTION TO DETAIL, ON THE PART OF THE PERSONNEL WHO CALCULATED THE TRIP SETPOINTS FOR RELAY REPLACEMENT IN AUGUST, 1987. THE REVIEW OF SAFETY RELATED RELAY TRIP SETPOINTS WILL CONTINUE.

[110] OCONEE 1 DOCKET 50-269 LER 91-005
INCORRECT FLOW ORIFICE ORIENTATION DUE TO INSTALLATION DEFICIENCY CAUSES REACTOR BUILDING SPRAY TRAINS TO BE INOPERABLE DURING CERTAIN ACCIDENT SCENARIOS.
EVENT DATE: 050991 REPORT DATE: 061391 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)

(NSIC 222302) DURING POST MODIFICATION TESTING, IN 3/91, A SAFETY RELATED FLOW TRANSMITTER ON UNIT 3 HPI SYSTEM WAS FOUND TO BE INOPERABLE. DURING CORRECTIVE ACTION TO INSPECT OTHER SAFETY RELATED INSTRUMENTATION, A FLOW TRANSMITTER ORIFICE WAS FOUND TO BE INSTALLED BACKWARDS IN THE HPI SYSTEM. INSPECTION OF ALL OTHER ACCESSIBLE SAFETY RELATED FLOW INSTRUMENTATION HAS REVEALED 2 OTHER OCCURRENCES OF INCORRECTLY INSTALLED ORIFICE PLATES. ON 5/9/91, WITH UNIT 1 AT 100% FULL POWER, A FLOW ORIFICE ON THE '1B' TRAIN OF THE REACTOR BUILDING SPRAY (RBS) SYSTEM WAS FOUND TO BE INSTALLED IMPROPERLY, CAUSING '1B' TRAIN TO BE INOPERABLE. ON 5/16/91, WITH UNIT 2 AT 100% FULL POWER, A FLOW ORIFICE ON THE '2B' TRAIN OF THE RBS SYSTEM WAS FOUND TO BE INSTALLED IMPROPERLY, CAUSING '2B' TRAIN TO BE INOPERABLE. DURING CERTAIN ACCIDENT SCENARIOS THAT REQUIRE RBS TO BE RECIRCULATED FROM THE REACTOR BLDG. EMERGENCY SUMP, EACH RESPECTIVE TRAIN WOULD BE INOPERABLE BECAUSE THE INCORRECTLY INSTALLED ORIFICE COULD RESULT IN PARTICLE BUILDUP AND THEREFORE GIVE INACCURATE FLOW MEASUREMENTS. ACTUAL FLOW RATE WOULD BE HIGHER THAN INDICATED AND THEREFORE COULD CAUSE RBS PUMP RUNOUT. BOTH ORIFICES WERE REMOVED AND INSTALLED CORRECTLY. ALL OTHER ACCESSIBLE SAFETY RELATED ORIFICES HAVE BEEN INSPECTED AND VERIFIED TO BE CORRECTLY INSTALLED. ROOT CAUSE IS IMPROPER INSTALLATION.

[111] OCONEE 1 DOCKET 50-269 LER 91-006
UNIT TRIP FOR UNKNOWN REASON, POSSIBLE INAPPROPRIATE ACTION.
EVENT DATE: 051691 REPORT DATE: 061791 NSSS: BW TYPE: PWR

(NSIC 222303) ON MAY 16, 1991, AT 1507 HOURS, UNIT 1 TRIPPED FROM 100% FULL POWER, DURING A THUNDERSTORM. THE UNIT EVENTS RECORDER FAILED IMMEDIATELY PRIOR TO THE TRIP, SO IT IS UNKNOWN WHICH TRIP SIGNAL WAS RECEIVED FIRST. COMPUTER AND CHART RECORDER OUTPUTS INDICATED THAT AN INSTRUMENT AND ELECTRICAL (I&E) TECHNICIAN MAY HAVE ERRONEOUSLY OPENED THE HIGH PRESSURE IMPULSE LINE ISOLATION VALVE INSTEAD OF THE LOW PRESSURE VALVE ON A NONSAFETY RELATED REACTOR COOLANT SYSTEM FLOW TRANSMITTER FOLLOWING CALIBRATION. THIS MAY HAVE CAUSED A MOMENTARY DIP IN THE PRESSURE SIGNAL TO FOUR CHANNELS OF REACTOR PROTECTIVE SYSTEM FLOW INSTRUMENTS WHICH SHARE THE IMPULSE LINES FROM THE FLOW ELEMENT. THE PRESSURE DIP WOULD HAVE CAUSED A TRIP ON FLUX/FLOW/IMBALANCE. ALTERNATIVELY, THE TRIP COULD HAVE OCCURRED DUE TO A SWITCHYARD TRANSIENT. OPERATORS TOOK IMMEDIATE ACTION TO STABILIZE THE UNIT FOLLOWING THE TRIP. THE ROOT CAUSE IS UNKNOWN INAPPROPRIATE ACTION. ADDITIONAL INVESTIGATION IS SCHEDULED FOR THE NEXT UNIT 1 REFUELING OUTAGE.

[112] OCONEE 1 DOCKET 50-269 LER 91-007
BREAKER COORDINATION PROBLEM DUE TO DESIGN DEFICIENCY RESULTS IN TECHNICAL INOPERABILITY OF SAFETY RELATED EQUIPMENT.
EVENT DATE: 052991 REPORT DATE: 062791 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 222406) ON 5/29/91 AT 1015 HOURS WITH ALL THREE OCONEE UNITS AT 100% FULL POWER, AN ONGOING REVIEW OF BREAKER AND RELAY TRIP SETPOINTS BY DUKE DESIGN ENGINEERING FOUND THAT A HEATER COORDINATION PROBLEM EXISTED WITH MOTOR CONTROL CENTERS 1, 2, AND 3XS2. ON EACH UNIT, IF AN OVERCURRENT CONDITION OCCURRED ON A NON-SAFETY RELATED PANELBOARD (1, 2, OR 3KM) DURING AN EMERGENCY SITUATION, THE FEEDPANEL BREAKER TO 1, 2, OR 3XS2 COULD TRIP OPEN, ISOLATING SAFETY-RELATED LOADS BEFORE THE BREAKER SUPPLYING THE PANELBOARD WOULD TRIP. SUCH FAILURE HAS ACTUALLY OCCURRED. PANELBOARDS 1, 2, AND 3KM WERE REWIRED TO BE SUPPLIED FROM A NON-SAFETY RELATED MOTOR CONTROL CENTER. THE ROOT CAUSE WAS DESIGN DEFICIENCY, UNANTICIPATED INTERACTION OF COMPONENTS. DESIGN ENGINEERING IS CONTINUING THEIR REVIEW OF SAFETY RELATED BREAKER AND RELAY SETPOINTS.

[113] OYSTER CREEK DOCKET 50-219 LER 88-021 REV 01
UPDATE ON PLANT SHUTDOWN REQUIRED DUE TO BOTH ISOLATION CONDENSERS BEING IN AN UNANALYZED CONDITION DUE TO THERMO-HYDRAULIC OPERATION OUTSIDE NORMAL SYSTEM DESIGN.
EVENT DATE: 092988 REPORT DATE: 061791 NSSS: GE TYPE: BWR

(NSIC 222291) ON 9/29/88, WHILE OPERATING AT 100% REACTOR POWER, IT WAS DETERMINED THAT BOTH THE "A" AND "B" ISOLATION CONDENSER WERE OPERATING IN A MODE OUTSIDE OF THE SYSTEM DESIGN. TEMPERATURE DATA SUGGESTED THAT THE STEAM LINES TO THE ISOLATION CONDENSERS WERE AT LEAST PARTIALLY FILLED WITH WATER, AND A THERMO-HYDRAULIC MECHANISM HAD BEEN ESTABLISHED THAT RESULTED IN BOTH ISOLATION CONDENSING STEAMING AT A COMBINED RATE OF ABOUT 4 MWTH. THE TEMPERATURE DATA ALSO SUGGESTED THAT CONDENSATE FLOW WAS REVERSED THROUGH ONE-HALF OF EACH CONDENSER. DUE TO CONCERNS ABOUT POTENTIAL WATER HAMMER IN THE CONDENSER STEAM LINES AND THE POTENTIAL SYSTEM DAMAGE WHICH COULD RESULT IN A LOSS OF SAFETY FUNCTION, A DECISION WAS MADE TO ISOLATE THE CONDENSERS FROM THE REACTOR. AT 1315 ON 9/29/88, ALL ISOLATION VALVES TO BOTH ISOLATION CONDENSERS WERE CLOSED AND THE CONDENSERS WERE DECLARED INOPERABLE. SINCE THE TECH SPECS REQUIRE A REACTOR SHUTDOWN (COLD SHUTDOWN) IN THE EVENT BOTH ISOLATION CONDENSERS BECOME INOPERABLE, AN ORDERLY REACTOR SHUTDOWN WAS COMMENCED. THE PLANT REACHED COLD SHUTDOWN ON 9/30/88. THE SAFETY SIGNIFICANCE AND CORRECTIVE ACTIONS HAVE BEEN EVALUATED AND WERE PROVIDED IN CORRESPONDENCE DATED 12/15/88 (TDR 950), 12/28/88 (TR056), AND 1/9/89 (TDR 950, REV. 1).

[114] OYSTER CREEK DOCKET 50-219 LER 90-012 REV 01
UPDATE ON AN ERROR IN A FEEDWATER FLOW CALCULATION EQUATION RESULTED IN OPERATION OF THE REACTOR IN EXCESS OF THE LICENSE LIMIT.
EVENT DATE: 080190 REPORT DATE: 062591 NSSS: GE TYPE: BWR

(NSIC 222397) DURING THE LAST OPERATING CYCLE, A DECREASE IN PLANT PERFORMANCE HAD BEEN NOTED. A LEAK IN THE HIGH PRESSURE FEEDWATER REHEATERS WAS INITIALLY POSTULATED AND INVESTIGATED. VISUAL INSPECTIONS REVEALED NO LEAKS. A DETAILED REVIEW OF PLANT DATA WAS INITIATED ON JULY 11, 1990 TO DETERMINE THE SOURCE OF THE PERFORMANCE DECREASE. ON AUGUST 1, 1990, IT WAS NOTED THAT A REVISION TO THE FEEDWATER FLOW CALIBRATION CALCULATION PROCEDURE, WHICH HAD BEEN APPROVED ON FEBRUARY 9, 1987, HAD REGULATED IN A 2% CORRECTION TO THE INDICATED FEEDWATER FLOW. THIS CAUSED A DECREASE IN THE ALLOWED REACTOR PLANT POWER. THEREFORE, PRIOR TO FEBRUARY 9, 1987, THE REACTOR MAY HAVE BEEN OPERATED IN EXCESS OF THE LICENSE LIMIT OF 1930 MW THERMAL. A SUBSEQUENT EVALUATION DETERMINED THAT THE 1930 MWT LIMIT HAD BEEN EXCEEDED, BUT THAT THE IMPACT ON REACTOR SAFETY WAS NOT SIGNIFICANT. THE CURRENT LOSS OF COOLANT ACCIDENT STUDIES HAVE REVEALED THAT THE PREVIOUS ANALYSES WERE EXTREMELY CONSERVATIVE IN TERMS OF FUEL BUNDLE HEATUP AND PLANT RESPONSE. THEREFORE, A LARGER SAFETY MARGIN TO THE MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE LIMIT WAS AVAILABLE THAN WAS PREVIOUSLY CALCULATED. ADDITIONALLY, NO MINIMUM CRITICAL POWER RATIO LIMITS WERE CHALLENGED. THEREFORE, THIS EVENT HAD MINIMAL SAFETY SIGNIFICANCE.

[115] OYSTER CREEK DOCKET 50-219 LER 90-017 REV 01
UPDATE ON BOTH STANDBY GAS TREATMENT SYSTEMS DECLARED INOPERABLE DUE TO COMMON DUCT FAILURE.
EVENT DATE: 122090 REPORT DATE: 061991 NSSS: GE TYPE: BWR

(NSIC 222292) ON DECEMBER 20, 1990 AT APPROXIMATELY 1415 HOURS A DEGRADATION IN DUCTWORK WAS DISCOVERED THAT CAUSED BOTH STANDBY GAS TREATMENT SYSTEMS TO BECOME INOPERABLE. THIS CONDITION IS CONSIDERED REPORTABLE IN ACCORDANCE WITH 10CFR50.73(A)(2)(V). THE DUCT IS CONSTRUCTED OF 1/8 INCH SHEET ALUMINUM AND HAS A CROSS SECTIONAL MEASUREMENT OF 14 INCHES BY 14 INCHES. THE DEGRADATION CONSISTED OF A SIDE PANEL SEPARATING FROM THE TOP AND BOTTOM CORNERS FOR A SPAN OF APPROXIMATELY THREE FEET. THE CAUSE OF THE DUCT FAILURE IS STILL UNDER INVESTIGATION. THE DEGRADATION OF THE DUCT IS A POTENTIALLY SIGNIFICANT CONDITION AS IT COULD HAVE AFFECTED THE OPERATION OF BOTH TRAINS OF THE SGTS. IMMEDIATE CORRECTIVE ACTION CONSISTED OF DECLARING BOTH STANDBY GAS TREATMENT SYSTEMS INOPERABLE AND COMMENCING AN ORDERLY SHUTDOWN IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS. CONCURRENT WITH THE PLANT SHUTDOWN, REPAIRS WERE MADE TO RESTORE THE INTEGRITY OF THE DUCTWORK. SUBSEQUENT INSPECTIONS AND AN EVALUATION OF THE REPAIR HAS DETERMINED THAT NO ADDITIONAL ACTIONS ARE REQUIRED. THE REPAIR HAS BEEN RECLASSIFIED AS A MODIFICATION AND APPROPRIATE DOCUMENT CHANGES ARE IN PREPARATION.

[116] PALISADES DOCKET 50-255 LER 91-010
EMERGENCY DIESEL GENERATOR UNANTICIPATED START DURING THE PERFORMANCE OF SPECIAL TEST T-297, "D/G 1-1 LOAD REJECT".
EVENT DATE: 022491 REPORT DATE: 061491 NSSS: CE TYPE: PWR

(NSIC 222296) ON FEBRUARY 24, 1991, AT 0618 HOURS, WITH THE PLANT IN COLD SHUTDOWN, AN UNANTICIPATED START OF THE 1-2 DIESEL GENERATOR OCCURRED DURING THE PERFORMANCE OF SPECIAL TEST T-297, "D/G 1-1 LOAD REJECT". THIS TEST WAS BEING PERFORMED AS A RESULT OF COMMITMENTS MADE TO NUCLEAR MUTUAL LIMITED (NML) REGARDING COMPLIANCE WITH IEEE 387, REGULATORY GUIDE 1.108 AND REGULATORY GUIDE 1.9 FOR DIESEL GENERATOR LOAD REJECTION. THE CAUSE OF THE EVENT WAS AN INADEQUATE PROCEDURE. NO PRECAUTIONARY STATEMENT EXISTED IN THE PROCEDURE TO ALERT THE OPERATOR THAT A DIESEL GENERATOR START MAY OCCUR. THE FOLLOWING CORRECTIVE ACTIONS WERE ASSIGNED AS A RESULT OF THE ANALYSIS OF THIS EVENT: REVIEW ELECTRICAL SPECIAL TEST PROCEDURES AND SURVEILLANCE PROCEDURES TO IDENTIFY VOLTAGE TRANSIENTS WHICH WILL CAUSE DIESEL GENERATOR STARTS. REVISE PROCEDURES AS NECESSARY. REPLACE RELAY 63X/LS-0204 WITH A RELAY HAVING A TIME DELAY SUCH THAT IT WILL NOT ACTUATE DURING A TRANSFER OF POWER SOURCES TO Y-01. REPLACE FCX-0218B. REVIEW APPLICATION OF NORMALLY ENERGIZED HFA RELAYS ON Y-01 WITH RESPECT TO Y-01 POWER TRANSFER. THIS EVENT DID NOT INVOLVE THE FAILURE OF ANY SYSTEMS OR COMPONENTS.

[117] PALISADES DOCKET 50-255 LER 91-009
QUALIFIED CORE EXIT THERMOCOUPLE INOPERABLE AND CANNOT BE REPAIRED WHILE THE PLANT IS IN POWER.
EVENT DATE: 051991 REPORT DATE: 061491 NSSS: CE TYPE: PWR
VENDOR: REUTER-STOKES ELECTRIC COMPANY

(NSIC 222295) ON MAY 1[illegible], 1991, AT 1903 HOURS, WITH THE PLANT OPERATING AT 100% POWER, QUALIFIED CORE EXIT THERMOCOUPLE (CET) NO. 16, WAS DECLARED INOPERABLE DUE TO INACCURATE READINGS. THE METHOD OF DISCOVERY WAS ERRATIC READINGS ON THE PRIMARY INFORMATION PROCESSOR (PIP). A WORK ORDER WAS WRITTEN TO TROUBLESHOOT AND REPAIR CET 16 AND THE CET WAS LISTED ON THE LIMITING CONDITION OF OPERATION (LCO) BOARD. THE WORK ORDER FOR CET 16 WAS RELEASED FOR REPAIR ON MAY 20, 1991, AT 1419 HOURS, WITH REPAIRS BEING INITIATED AT THAT TIME. REPAIRS WERE ATTEMPTED BY THE INSTRUMENT & CONTROL DEPARTMENT (I&C), HOWEVER, NONE OF THE REPAIR TECHNIQUES THAT HAD BEEN SUCCESSFUL IN THE PAST WERE SUCCESSFUL ON THIS OCCASION. ON MAY 21, 1991 IT WAS DETERMINED THAT CET 16 COULD NOT BE REPAIRED WHILE THE PLANT WAS AT POWER, THUS CET 16 WAS DECLARED INOPERABLE. THE CAUSE OF THE EVENT WAS A FAILURE OF THE INSTRUMENT. A CORRECTIVE ACTION DOCUMENT WAS INITIATED TO DOCUMENT THE EVENT AND TO ENSURE THAT THE 30 DAY SPECIAL REPORT REQUIRED BY THE PROPOSED TECHNICAL SPECIFICATIONS VIA ADMINISTRATIVE PROCEDURES WAS SUBMITTED.

[118] PALO VERDE 1 DOCKET 50-528 LER 89-025
MISSING RADIANT ENERGY BARRIER.
EVENT DATE: 110389 REPORT DATE: 051791 NSSS: CE TYPE: PWR

(NSIC 222194) ON NOVEMBER 3, 1989, AT APPROXIMATELY 1400 MST, PALO VERDE UNIT 1 WAS IN MODE 6 (REFUELING) WHEN APS ENGINEERING PERSONNEL DISCOVERED THAT A RADIANT ENERGY FIRE BARRIER WAS MISSING FROM ONE OF THE TWO PRESSURIZER AUXILIARY SPRAY VALVES IN THE PRESSURIZER ROOM. PRIOR TO RESTART FROM THE REFUELING OUTAGE, THE MISSING RADIANT ENERGY BARRIER WAS REPLACED ON MARCH 26, 1990. ON APRIL 17, 1991, APS COMPLETED AN EVALUATION OF THE SAFETY SIGNIFICANCE OF THE MISSING RADIANT ENERGY BARRIER AND DETERMINED THAT THE MISSING RADIANT ENERGY BARRIER WOULD HAVE ADVERSELY AFFECTED THE ABILITY TO ACHIEVE AND MAINTAIN SAFE SHUTDOWN IN THE EVENT OF A FIRE. INVESTIGATION HAD DETERMINED THAT THE RADIANT ENERGY BARRIER HAD BEEN MISSING FOR APPROXIMATELY 32 MONTHS. THE BARRIER HAD NOT BEEN REPLACED FOLLOWING MAINTENANCE ON THE PRESSURIZER AUXILIARY SPRAY VALVE IN JULY OF 1987. AN INSPECTION OF THE PRESSURIZER AUXILIARY SPRAY VALVES IN UNITS 2 AND 3 FOUND THE RADIANT ENERGY BARRIERS PROPERLY INSTALLED ON MARCH 14, 1990. THE CAUSE OF THIS EVENT WAS THAT THE DRAWINGS FOR THE PRESSURIZER AUXILIARY SPRAY VALVE DID NOT INCLUDE A PLANT DESIGN CHANGE WHICH ADDED THE RADIANT ENERGY BARRIER. THIS OMISSION ALLOWED PERSONNEL GENERATING WORK DOCUMENTS TO OMIT THE

REQUIREMENT IN THE WORK DOCUMENT TO REINSTALL THE RADIANT ENERGY BARRIER. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED PURSUANT TO T.S. 6.9.3.

[119] PALO VERDE 1 DOCKET 50-528 LER 91-006
ESF ACTUATION DUE TO RADIATION MONITOR FAILURE.
EVENT DATE: 051791 REPORT DATE: 061191 NSSS: CE TYPE: PWR
VENDOR: KAMAN SCIENCES CORP.

(NSIC 222333) ON MAY 17, 1991, AT APPROXIMATELY 0629 MST, PALO VERDE UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 100 PERCENT POWER, WHEN A SPURIOUS TRAIN "B" CONTAINMENT PURGE ISOLATION ACTUATION SIGNAL (CPIAS) WAS INITIATED ON THE BALANCE OF PLANT ENGINEERED SAFETY FEATURES ACTUATION SYSTEM. THE TRAIN "B" CPIAS RESULTED IN THE DESIGNED CROSS-TRIPS OF TRAIN "A" CPIAS AND TRAIN "A" AND "B" CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNALS (CREFAS). THE ACTUATIONS OCCURRED WHEN THE TRAIN "B" POWER ACCESS PURGE AREA RADIATION MONITOR (RU-38) SPIKED, WENT OFF LINE, AND WAS UNREACHABLE FROM THE RADIATION MONITORING SYSTEM (RMS) DATA CONTROL UNIT (DCU). AT THE TIME OF THE EVENT NO CONTAINMENT PURGE WAS IN PROGRESS AND ALL CONTAINMENT PURGE SYSTEM ISOLATION VALVES WERE CLOSED. FOLLOWING THE SPURIOUS TRAIN "B" CPIAS ALL COMPONENTS OPERATED AS DESIGNED. CONTROL ROOM AND RADIATION PROTECTION PERSONNEL VERIFIED THAT NO ABNORMAL RADIATION LEVELS EXISTED IN THE VICINITY OF RU-38. THE CAUSE OF THE EVENT WAS A MALFUNCTIONING CENTRAL PROCESSING UNIT AND RANDOM ACCESS MEMORY BOARD ON THE TRAIN "B" POWER ACCESS PURGE AREA RADIATION MONITOR (RU-38). AS CORRECTIVE ACTION, THE BOARDS HAVE BEEN REPLACED. A PREVIOUS SIMILAR EVENT WAS REPORTED.

[120] PALO VERDE 3 DOCKET 50-530 LER 91-001 REV 01
UPDATE ON SAFETY VALVE SETPOINTS OUT OF TOLERANCE.
EVENT DATE: 031591 REPORT DATE: 060991 NSSS: CE TYPE: PWR
VENDOR: DRESSER INDUSTRIES, INC.

(NSIC 222195) ON MARCH 15, 1991 WHILE UNIT 3 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 70 PERCENT POWER, AN ENGINEERING EVALUATION OF ASME SURVEILLANCE TESTING RESULTS DETERMINED THAT 10 OF THE 20 MAIN STEAM SAFETY VALVE (MSSV) AS-FOUND RELIEF SETTINGS WERE OUT OF THE TOLERANCE LIMITS SPECIFIED IN TECHNICAL SPECIFICATION (TS) 3.7.1.1 AND IN THE TESTING REQUIREMENTS ESTABLISHED BY APS. THE TESTING AND ADJUSTMENTS WERE PERFORMED DURING THE PERIOD OF MARCH 13 THROUGH MARCH 15, 1991, WHILE UNIT 3 WAS IN MODE 1, TO VERIFY THE MSSV RELIEF SETTINGS. AFTER SHUTDOWN FOR A SCHEDULED REFUELING OUTAGE, THE UNIT 3 PRESSURIZER CODE SAFETY VALVES (PSV) WERE REMOVED AND SENT TO AN OFFSITE TESTING LAB. THE UNIT 3 PSVS WERE TESTED APRIL 1 AND 2, 1991 AND THREE (3) OF THE FOUR (4) PSV AS-FOUND RELIEF SETTINGS WERE OUT OF THE TOLERANCE LIMITS SPECIFIED IN TS 3.4.2.2 AND IN THE TESTING REQUIREMENTS ESTABLISHED BY APS. THE CAUSE OF THE EVENT IS SETPOINT DRIFT. AS IMMEDIATE CORRECTIVE ACTION THE MSSVS AND PSVS HAVE BEEN ADJUSTED AND TESTED SATISFACTORILY. A SAFETY ANALYSIS DETERMINED THAT THE AS-FOUND SETPOINTS WOULD NOT HAVE RESULTED IN ANY SAFETY LIMITS BEING VIOLATED. PREVIOUS SIMILAR EVENTS WERE REPORTED IN MSSV LERS 528/88-014-01, 528/89-010-00, 529/89-002-00, 529/89-007-00 AND PSV LERS 528/89-007-01 AND 529/90-004-01.

[121] PEACH BOTTOM 2 DOCKET 50-277 LER 90-004 REV 01
UPDATE ON POTENTIALLY INOPERABLE SAFETY SYSTEMS DUE TO INADEQUATE EMERGENCY SERVICE WATER COOLING FLOW THROUGH SAFETY SYSTEM COMPARTMENT ROOM COOLERS.
EVENT DATE: 032190 REPORT DATE: 061391 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 222307) ON 3/21/90, DURING A REVIEW OF A COMPLETED SPECIAL PROCEDURE IT WAS CALCULATED THAT UNDER DESIGN BASIS ACCIDENT CONDITIONS SOME OF THE UNIT 2 ECCS AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM ROOM COOLERS WOULD NOT RECEIVE MINIMUM ACCEPTABLE EMERGENCY SERVICE WATER (ESW) FLOW. THIS RESULTED IN THE POTENTIAL OF ECCS, AND RCIC PUMPS BEING INOPERABLE UNDER CERTAIN PLANT ENVIRONMENTAL AND CLIMATIC CONDITIONS. CAUSE OF THIS EVENT WAS GRADUAL BUILDUP OF CORROSION PRODUCTS AND SILT ON THE INTERIOR WALL OF THE ESW PIPING. A ROOT CAUSE ANALYSIS REVEALED NUMEROUS CONTRIBUTING CAUSES INCLUDING LESS THAN ADEQUATE (LTA) CHEMICAL TREATMENT OF THE ESW SYSTEM, LACK OF UNDERSTANDING OF ESW DESIGN

BASIS, LTA ADMINISTRATIVE CONTROLS INVOLVING MODIFICATIONS AND TESTING AND LTA ORIGINAL DESIGN. AS CORRECTIVE ACTIONS, ONE ECCS AND RCIC SYSTEM PUMP ROOM COOLER INLET HAND VALVE IN EACH UNIT 2 PUMP ROOM WAS CLOSED TO PROVIDE SATISFACTORY FLOW TO THE ALTERNATE ROOM COOLER. SECTIONS OF THE UNIT 2 ESW SYSTEM WERE CLEANED, AND A MODIFICATION WAS COMPLETED ON UNIT 2 WHICH REPLACED MAJOR PORTIONS OF THE ESW SYSTEM PIPING. A SIMILAR MODIFICATION WAS ALSO COMPLETED ON UNIT 3. CHEMICAL TREATMENT AND DESIGN BASIS DOCUMENTATION PROGRAMS HAVE BEEN INITIATED. PROGRAMMATIC ENHANCEMENTS HAVE BEEN MADE TO THE MODIFICATIONS AND TESTING PROGRAMS.

[122] PEACH BOTTOM 2 DOCKET 50-277 LER 91-013
UNQUALIFIED FIRE SEALS DUE TO ORIGINAL DESIGN RESULT IN PLANT OUTSIDE DESIGN BASIS.
EVENT DATE: 031991 REPORT DATE: 061391 NSSS: GE TYPE: BWR

(NSIC 222309) ON 5/30/91 AN ENGINEERING EVALUATION INCLUDING PHYSICAL TESTING DETERMINED THAT TWO PENETRATION FIRE SEALS WERE NOT QUALIFIABLE AS REQUIRED BY 10 CFR 50 APPENDIX R. THE TWO SEALS HAD RECENTLY BEEN DISCOVERED TO CONTAIN VOIDS AND UNCURED POLYURETHANE SEALANT MATERIAL. THIS COULD HAVE RESULTED IN NOT HAVING THE CAPABILITY TO ACHIEVE SHUTDOWN PER 10 CFR 50 APPENDIX R HAD A FIRE OCCURRED IN THE AFFECTED FIRE AREAS. THE CAUSE OF THE EVENT IS IMPROPER ORIGINAL SEAL INSTALLATION DURING PLANT CONSTRUCTION. THERE WERE NO ACTUAL SAFETY CONSEQUENCES. THE AFFECTED SEALS HAVE BEEN REPLACED. COMPENSATORY FIREWATCHES HAVE BEEN PUT IN PLACE ON SIMILAR SEAL CONFIGURATIONS. THE SIMILAR SEALS WILL BE INSPECTED AND REPAIRED AS NECESSARY. THERE WERE THREE PREVIOUS SIMILAR EVENTS.

[123] PEACH BOTTOM 2 DOCKET 50-277 LER 91-012
TECHNICAL SPECIFICATION VIOLATION DUE TO FAILURE TO PERFORM TESTING AS A RESULT OF IMPROPER DEVELOPMENT OF ORIGINAL PROCEDURE.
EVENT DATE: 051491 REPORT DATE: 061091 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 222308) ON 5/14/91 DURING THE PERFORMANCE OF A LOCAL LEAK RATE TEST (LLRT), IT WAS DISCOVERED THAT A PRIMARY CONTAINMENT ISOLATION VALVE ON BOTH THE UNIT 2 AND 3 TRAVERSING INCORE (TIP) SYSTEM PURGE LINES WAS NOT BEING LEAK RATE TESTED. THIS IS A VIOLATION OF TECHNICAL SPECIFICATION 4.7.A.2.F, WHICH REQUIRES THAT LLRT'S BE PERFORMED ON PRIMARY CONTAINMENT ISOLATION VALVES ONCE EACH OPERATING CYCLE. ALTHOUGH LLRT'S HAD BEEN PERFORMED EACH OPERATING CYCLE ON THE TIP SYSTEM PURGE LINE ISOLATION VALVES, OUTBOARD ISOLATION VALVE (SV-109) WAS NOT ACTUALLY BEING LEAK RATE TESTED DUE TO ROTOMETER TYPE FLOW INDICATOR (FI-110) INSTALLED BETWEEN THE TEST INPUT TAP AND SV-109. THE CAUSE OF THE EVENT WAS AN INADEQUATE REVIEW OF THE TIP SYSTEM PURGE PIPING CONFIGURATION DURING THE DEVELOPMENT OF THE LLRT PROCEDURE. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. THE SV-109'S ON BOTH UNIT 2 AND 3 SUBSEQUENTLY TESTED AND LEAKAGE WAS VERIFIED TO BE WITHIN ACCEPTABLE LIMITS. THE LLRT PROCEDURE FOR SV-109 WILL BE REVISED TO ALLOW LEAK RATE TESTING DOWNSTREAM OF FI-110. A MODIFICATION WILL BE CONSIDERED TO EITHER RELOCATE THE TEST TAP OR FI-110 TO PROVIDE A MORE SUITABLE MEANS OF ADEQUATELY LEAK RATE TESTING SV-109. NO PREVIOUS SIMILAR EVENTS WERE IDENTIFIED.

[124] PEACH BOTTOM 2 DOCKET 50-277 LER 91-015
PRIMARY CONTAINMENT ISOLATION SYSTEM ACTUATION DURING SURVEILLANCE TESTING DUE TO PERSONNEL ERROR.
EVENT DATE: 051591 REPORT DATE: 061391 NSSS: GE TYPE: BWR

(NSIC 222311) ON 5/15/91 AT 1805 HOURS DURING A PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) (EIIS:JM) LOGIC SYSTEM FUNCTIONAL SURVEILLANCE TEST (ST), THE REACTOR BUILDING AND REFUELING FLOOR VENTILATION SYSTEMS TRIPPED AND THE STANDBY GAS TREATMENT SYSTEM (SBGTS) (EIIS:BH) AUTOMATICALLY INITIATED WHEN AN INCORRECT FUSE WAS PULLED. THE ENGINEER INADVERTENTLY PULLED FUSE 16A-F34A AT LOCATION F-29 INSTEAD OF 16A- F33A AT LOCATION F-28 AS SPECIFIED IN THE ST. PULLING THE INCORRECT FUSE RESULTED IN A ONE HALF GROUP 3 ISOLATION. THE CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO THE ENGINEER'S FAILURE TO PROPERLY IDENTIFY THE FUSE TO BE PULLED. SYSTEM ISOLATION WAS RESET FOLLOWING THE EVENT. THE EVENT HAS BEEN

DISCUSSED WITH THE INVOLVED INDIVIDUALS. THE LEARNING EXPERIENCE FROM THIS EVENT HAS BEEN PRESENTED TO OTHER SYSTEM ENGINEERS BY THE INDIVIDUAL RESPONSIBLE FOR THE EVENT. A PROJECT IS NOW IN PLACE TO LABEL THESE FUSES ALONG WITH OTHER FUSES IN THE MAIN CONTROL ROOM AND THE CABLE SPREADING ROOM. THE EXISTING ST REWRITE PROJECT WILL PERFORM A HUMAN FACTORS REVIEW ON THIS ST AND OTHER SIMILAR LONG STS. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT.

[125] PEACH BOTTOM 2 DOCKET 50-277 LER 91-016
TECHNICAL SPECIFICATION VIOLATION AS A RESULT OF A MISSED SURVEILLANCE TEST DUE TO PROGRAMMATIC WEAKNESSES.
EVENT DATE: 051791 REPORT DATE: 061891 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 222312) ON 5/17/91 FOR UNIT 2 AND 5/24/91 FOR UNIT 3, A TECHNICAL SPECIFICATION VIOLATION OCCURRED WHEN THE MODE SWITCHES WERE PLACED IN THE STARTUP MODE WITH THE REACTOR WATER LEVEL INSTRUMENT PERTURBATION SURVEILLANCE TESTS (ST) BEING OUT OF SURVEILLANCE. WHEN IDENTIFIED THAT THE STS WERE OUT OF SURVEILLANCE, THE TESTS WERE IMMEDIATELY COMPLETED SATISFACTORILY. THE CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO THE INADEQUACY OF THE STS AND PROCESSES WHICH ENSURE THAT THESE TESTS ARE COMPLETED PRIOR TO PLACING THE MODE SWITCH IN STARTUP. SEVERAL PROGRAMMATIC AND PROCEDURAL ENHANCEMENTS WILL BE INCORPORATED. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. THERE HAVE BEEN SEVERAL PREVIOUS SIMILAR LERS INVOLVING MISSED STS. THE CORRECTIVE ACTIONS TAKEN AS A RESULT OF THESE LERS ADDRESSED THE SPECIFIC CAUSES ONLY. ALTHOUGH THIS EVENT IS ALSO SPECIFIC IN NATURE, A TASK FORCE IS PRESENTLY EVALUATING THE ISSUE OF MISSED STS AND WILL PROVIDE RECOMMENDATIONS REGARDING ST PERFORMANCE, SCHEDULING, AND TRACKING.

[126] PEACH BOTTOM 2 DOCKET 50-277 LER 91-017
POTENTIAL FOR THE INOPERABILITY OF THE HIGH PRESSURE COOLANT INJECTION SYSTEM DUE TO THE USE OF UNQUALIFIED RELAYS IN THE CARDOX LOGIC.
EVENT DATE: 051891 REPORT DATE: 062091 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 222313) ON 5/21/91, IT WAS DISCOVERED THAT THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM HAD BEEN UNKNOWINGLY RENDERED INOPERABLE ON 5/18/91 DURING THE PERFORMANCE OF A SURVEILLANCE TEST (ST). AS A RESULT OF ADDITIONAL INVESTIGATION, IT WAS DETERMINED ON 5/22/91 THAT THE UNIT 2 & 3 HPCI SYSTEMS WERE POSSIBLY INOPERABLE DUE TO NON-QUALIFIED (Q) AND NON-ENVIRONMENTALLY QUALIFIED (EQ) RELAYS IN THE HPCI R OOM COOLER FAN LOGIC. THE CAUSE OF THESE EVENTS HAS BEEN ATTRIBUTED TO A PROCEDURAL INADEQUACY AND A LESS THAN ADEQUATE DESIGN REVIEW AND SUBSEQUENT SYSTEM REVIEWS. TEMPORARY PLANT ALTERATIONS HAVE BEEN INSTALLED TO MAINTAIN HPCI OPERABLE AND A MODIFICATION WILL BE INITIATED TO UPGRADE THE RELAYS OR REMOVE THE TRIP SIGNAL. THE ST INVOLVED WILL BE ENHANCED. A REVIEW HAS BEEN COMPLETED ON OTHER FIRE SUPPRESSION SYSTEM INTERLOCKS AND NO ADDITIONAL NON-Q INTERLOCKS WERE IDENTIFIED. FINALLY, INVESTIGATIONS WILL BE PERFORMED TO DETERMINE WHY THE EXISTING PROCESSES FAILED TO IDENTIFY THESE RELAYS. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. THERE WERE THREE PREVIOUS SIMILAR LERS.

[127] PEACH BOTTOM 2 DOCKET 50-277 LER 91-014
ENGINEERED SAFETY FEATURE ACTUATION DURING SURVEILLANCE TESTING DUE TO A LEAKING ISOLATION VALVE.
EVENT DATE: 052191 REPORT DATE: 061391 NSSS: GE TYPE: BWR
VENDOR: DRAGON VALVE, INC.

(NSIC 222310) ON 5/21/91, AT 1815 HOURS, WITH UNIT 2 OPERATING AT 25% POWER, AN ISOLATION OF THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM OCCURRED UNEXPECTEDLY DURING THE PERFORMANCE OF A SURVEILLANCE TEST (ST). THIS IS AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION. THE CAUSE OF THE EVENT WAS A LEAKING ISOLATION VALVE WHICH ALLOWED THE SWITCH BEING TESTED TO BECOME PRESSURIZED, CAUSING THE ISOLATION SIGNAL TO OCCUR SOONER THAN EXPECTED. THERE WERE NO ACTUAL SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT. THE ISOLATION VALVE WAS REPLACED

ON 6/8/9' AND THE ST WAS PERFORMED SATISFACTORILY. THERE WERE FOUR PREVIOUS SIMILAR EVENTS.

[128] PEACH BOTTOM 2 DOCKET 50-277 LER 91-018
VOLUNTARY REPORT CONCERNING ENTRY INTO TECH SPEC 3.0.C DUE TO BELIEF THAT DIESEL GENERATORS INOPERABLE DUE TO NON-SAFETY CLASSIFIED SWITCHES.
EVENT DATE: 052391 REPORT DATE: 062191 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 222370) ON 5/23/91 AT 1545 HOURS, TECHNICAL SPECIFICATION 3.0.C WAS ENTERED AND A PLANT SHUTDOWN INITIATED DUE TO A BELIEF THAT THE 4 STATION EMERGENCY DIESEL GENERATORS (EDG'S) COULD POTENTIALLY BE RENDERED INOPERABLE DURING DESIGN BASIS EVENTS. THIS COULD RESULT IN THE INABILITY OF UNIT 2 AND 3 SAFETY SYSTEMS TO PERFORM THEIR FUNCTION DURING DESIGN BASIS EVENTS COINCIDENT WITH LOSS OF OFF-SITE POWER EVENT. THE CAUSE OF THE EVENT WAS DUE TO MISCLASSIFICATION OF THE EDG DAY TANK TEMPERATURE SWITCHES (TS'S) AS NOT SAFETY RELATED DURING THE SAFETY RELATED COMPONENT LIST GENERATION IN 1987-1988. IT WAS BELIEVED THAT THIS COULD RESULT IN INADVERTENT TRIPPING OF THE DAY TANK TRANSFER PUMP THEREBY NOT ALLOWING FUEL OIL TO BE AUTOMATICALLY TRANSFERRED FOR LONGER TERM EDG OPERABILITY. A TEMPORARY PLANT ALTERATION WAS COMPLETED BY 2200 HOURS TO REMOVE THE TS FROM THE CIRCUIT. FURTHER ENGINEERING REVIEW CONCLUDED ON 6/ 7/91 THAT THE EXISTING TS'S WERE OPERABLE AND COULD FUNCTION PROPERLY IN DESIGN BASIS EVENTS. THEREFORE THE EDG'S WERE OPERABLE. THE EDG DAY TANK TS'S WILL BE RECLASSIFIED AS SAFETY-RELATED. OTHER COMPONENTS WILL BE REVIEWED. THERE WERE NO SAFETY CONSEQUENCES OR PREVIOUS EVENTS.

[129] PEACH BOTTOM 2 DOCKET 50-277 LER 91-019
PRIMARY CONTAINMENT ISOLATION VALVE LOGGING NOT BEING PERFORMED AS REQUIRED BY TECHNICAL SPECIFICATIONS DUE TO PERSONNEL ERROR.
EVENT DATE: 060391 REPORT DATE: 070391 NSSS: GE TYPE: BWR

(NSIC 222410) ON 6/3/91, A TECHNICAL SPECIFICATION (TECH SPEC) VIOLATION OCCURRED WHEN THE UNIT 2 REACTOR OPERATOR (RO) FAILED TO INITIAL SURVEILLANCE TEST (ST) 5.3, INOPERABLE ISOLATION VALVE POSITION DAILY LOGIC, WHICH SIGNIFIES THAT CONTAINMENT PENETRATIONS WITH INOPERABLE ISOLATION VALVES ARE STILL ISOLATED; REQUIRED BY TECH SPECS. THE CAUSE OF THIS EVENT IS PERSONNEL ERROR DUE TO FAILURE TO FOLLOW PROCEDURE. THE ST REQUIRES THE RO TO INITIAL DAILY TO VERIFY THAT THE PENETRATION IS ISOLATED. THE INDIVIDUALS INVOLVED IN THIS EVENT WERE COUNSELED AND COACHED FOLLOWING THE INCIDENT ON THE PERFORMANCE OF ADMINISTRATIVE TASKS ASSOCIATED WITH THE RO'S RESPONSIBILITIES. ADDITIONAL CORRECTIVE ACTION INCLUDED DEVELOPMENT OF A REACTOR OPERATOR TASK LIST TO BE POSTED AT THE REACTOR OPERATOR'S CONSOLE. PERTINENT INFORMATION FROM THIS LER WILL ALSO BE ROUTED TO THE APPROPRIATE OPERATIONS PERSONNEL. THERE WERE NO SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT.

[130] PEACH BOTTOM 3 DOCKET 50-278 LER 91-008
ENGINEERED SAFETY FEATURE ACTUATION DURING SURVEILLANCE TESTING DUE TO PERSONNEL ERROR.
EVENT DATE: 052091 REPORT DATE: 061491 NSSS: GE TYPE: BWR

(NSIC 222314) ON 5/20/91 AT 0730 HOURS DURING THE PERFORMANCE OF A RESIDUAL HEAT REMOVAL "B" CHANNEL LOGIC SYSTEM FUNCTIONAL SURVEILLANCE TEST (ST), THE 3C RHR PUMP UNEXPECTEDLY STARTED WHEN A TEST SWITCH WAS INADVERTENTLY LEFT IN THE WRONG POSITION DURING TESTING. THE SYSTEM WAS SECURED AND THE ST WAS COMPLETED SATISFACTORILY. THE CAUSE HAS BEEN ATTRIBUTED TO PERSONNEL ERROR CAUSED BY A REPETITIVE PATTERN SET UP IN THE ST WHICH WAS THEN SUDDENLY CHANGED. THE EVENT HAS BEEN DISCUSSED WITH THE INVOLVED INDIVIDUALS AND THE PERTINENT INFORMATION HAS BEEN PRESENTED TO OTHER I&C PERSONNEL. ADDITIONALLY, THE ST WILL BE ENHANCED. NO SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. THERE WAS ONE OTHER PREVIOUS SIMILAR EVENT THAT INVOLVED THE MISPOSITIONING OF A TEST SWITCH.

[131] PERRY 1 DOCKET 50-440 LER 91-012
PROGRAM DEFICIENCIES RESULTED IN AN INOPERABLE EFFLUENT FLOW MONITOR WITHOUT THE REQUIRED COMPENSATORY ACTIONS TAKEN, RESULTING IN A TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 060591 REPORT DATE: 070391 NSSS: GE TYPE: BWR

(NSIC 222508) ON JUNE 5, 1991, IT WAS DETERMINED THAT THE EFFLUENT SYSTEM FLOW RATE MONITOR FOR THE TURBINE BUILDING/HEATER BAY BUILDING (TB/HB) VENT RADIATION MONITOR HAD BEEN INOPERABLE DURING TWO FAN OPERATION SINCE THE SUMMER OF 1988. DURING THIS TIME, THE EFFLUENT FLOW RATE WAS NOT ESTIMATED EVERY FOUR HOURS IN VIOLATION OF TECHNICAL SPECIFICATION 3.3.7.10. THE CAUSES OF THIS EVENT WERE PROGRAM DEFICIENCIES. THE PREOPERATIONAL TESTING PROGRAM DID NOT FULLY TEST THE ACCURACY OF THE VENT EFFLUENT FLOW MONITORS. THE CURRENT PROGRAM DOES NOT VERIFY THE ACCURACY OF THIS FLOW SENSING DEVICE, OR ADEQUATELY COMPENSATE FOR DEVIATIONS IN AIR TEMPERATURE. ALL OTHER VENT EFFLUENT FLOW MONITORS WERE EVALUATED FOR SIMILAR FLOW DEGRADATION WITH NO PROBLEM FOUND. THIS PILOT TUBE ARRAY WILL BE CALIBRATED BY CAPPING INDIVIDUAL PILOT TUBES AS REQUIRED AND POSSIBLE DEGRADATION WILL BE INVESTIGATED. THE CALIBRATION INSTRUCTIONS FOR THE TB/HB VENT AND SIMILAR EFFLUENT FLOW MONITORS WILL BE REVISED TO EVALUATE THE PERFORMANCE OF THE FLOW SENSING DEVICE. PRIOR TO RESTORING THE FLOW MONITOR TO OPERABLE STATUS DURING TWO FAN OPERATION, A MECHANISM FOR TEMPERATURE COMPENSATION WILL BE ESTABLISHED. THIS EVENT WILL BE DISCUSSED AS PART OF THE LICENSED OPERATOR REQUALIFICATION PROGRAM.

[132] PILGRIM 1 DOCKET 50-293 LER 89-024 REV 01
UPDATE ON AUTOMATIC CLOSING OF THE OUTBOARD PRIMARY CONTAINMENT SYSTEM GROUP 6 ISOLATION VALVES.
EVENT DATE: 072689 REPORT DATE: 061891 NSSS: GE TYPE: BWR

(NSIC 222320) ON 7/26/89, AT 0123 HOURS, AN AUTOMATIC ACTUATION OF THE OUTBOARD REACTOR WATER CLEANUP (RWCU) SYSTEM PORTION OF THE PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM (PCIS) OCCURRED. THE ACTUATION RESULTED IN AUTOMATIC CLOSING OF THE OUTBOARD PRIMARY CONTAINMENT SYSTEM GROUP 6 RWCU SYSTEM ISOLATION VALVES AND A TEMPORARY INTERRUPTION IN RWCU SYSTEM OPERATION. PCIS LOGIC CIRCUITRY WAS RESET & RWCU SYSTEM WAS RETURNED TO SERVICE ON 7/26/89, AT APPROX. 0200 HRS. THE DIRECT CAUSE OF THE ACTUATION WAS A TRIP SIGNAL FROM THE OUTBOARD RWCU SYSTEM FLOW SENSOR. A MULTI-DISCIPLINARY INVESTIGATION TEAM DETERMINED THAT THE CAUSE OF THE TRIP SIGNAL WAS ENTRAPPED AIR IN THE INSTRUMENT SENSING LINES. IT WAS DETERMINED THAT A PORTION OF THE SENSING LINES WAS NOT SLOPED PROPERLY, CREATING A POTENTIAL AIR TRAP. IN ADDITION, THE SNUBBERS CONTAINED WITH IN THE INSTRUMENT SENSING LINES WERE IMPEDING THE EFFECTIVENESS OF THE BACKFILLS BEING PERFORMED. CORRECTIVE ACTIONS INCLUDED REDESIGNING THE INCORRECTLY SLOPED SENSING LINE TO PREVENT AIR FROM BEING TRAPPED AND ADDING VENT CONNECTIONS UPSTREAM OF THE SNUBBERS TO ALLOW FOR MORE EFFECTIVE BACKFILLS. THIS EVENT OCCURRED DURING A STARTUP WITH THE REACTOR MODE SELECTOR SWITCH IN THE STARTUP POSITION. THE CONTROL RODS WERE IN A PARTIALLY WITHDRAWN POSITION. THE REACTOR VESSEL PRESSURE WAS APPROX. 10 PSIG AND RV WATER TEMP. WAS APPROX. 221F.

[133] PILGRIM 1 DOCKET 50-293 LER 91-008
THREE AUTOMATIC GROUP 1 ISOLATIONS DUE TO FALSE HIGH REACTOR WATER LEVEL SIGNALS WHILE SHUTDOWN.
EVENT DATE: 043091 REPORT DATE: 053091 NSSS: GE TYPE: BWR

(NSIC 222207) ON 4/30/91, 3 AUTOMATIC PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM (PCIS) GROUP 1 ISOLATIONS OCCURRED WHILE SHUTDOWN AT 0116 HRS, 0933 HRS, AND 1037 HRS, RESPECTIVELY, DUE TO A FALSE HIGH REACTOR VESSEL (RV) WATER LEVEL SIGNAL. THE ACTUATIONS RESULTED IN AUTOMATIC CLOSING OF THE RELATED PRIMARY CONTAINMENT SYSTEM ISOLATION VALVES. THE GROUP 1 ISOLATIONS WERE INITIATED BY THE REACTOR WATER LEVEL TRIP UNITS DOWNSTREAM OF REFERENCE LEG CONDENSING CHAMBER 12B. THE PLANT INFORMATION COMPUTER TRACES SHOWED THE REACTOR WATER LEVEL RAPIDLY INCREASING ABOVE THE TRIP SETPOINT AND THEN RETURNING TO THE ORIGINAL LEVEL. THE REACTOR WATER LEVEL TRIP UNITS DOWNSTREAM OF REFERENCE LEG CONDENSING CHAMBER 12A ALSO EXHIBITED LEVEL SPIKING. AN INVESTIGATION TO DETERMINE THE CAUSE OF THE FALSE HIGH REACTOR WATER LEVEL INDICATIONS WAS ONGOING WHEN THIS REPORT

WAS SUBMITTED. THIS REPORT WILL BE UPDATED TO PROVIDE THE RESULTS OF THE INVESTIGATION. THESE EVENTS OCCURRED WHEN IN THE HOT SHUTDOWN MODE OF OPERATION WITH THE REACTOR MODE SELECTOR SWITCH IN THE SHUTDOWN POSITION. THE REACTOR POWER LEVEL WAS 0%. THE RV PRESSURES AND RV WATER TEMPERATURES FOR THE 3 EVENTS WERE AS FOLLOWS: FIRST EVENT, 60 PSIG; SECOND EVENT, 12 PSIG; THIRD EVENT, 3 PSIG. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV).

[134] PILGRIM 1 DOCKET 50-293 LER 91-009
FIRE BARRIER FOUND BREACHED IN INTAKE STRUCTURE.
EVENT DATE: 051891 REPORT DATE: 061491 NSSS: GE TYPE: BWR

(NSIC 222321) ON MAY 18, 1991 AT 1000 HOURS DURING A REFUELING OUTAGE, THE EAST WALL OF THE "B " TRAIN SALT SERVICE WATER (SSW) PUMP ROOM IN THE INTAKE STRUCTURE WAS FOUND TO BE BREACHED. THE BREACH CONSISTED OF A 4 INCH DRAIN CHECK VALVE THAT WAS FOUND IN THE OPEN POSITION. THE WALL IS A TECHNICAL SPECIFICATION APPENDIX R FIRE BARRI ER THAT SEPARATES THE SAFETY RELATED SSW PUMP ROOM FROM THE NON-SAFETY RELATED " A" CIRCULATING WATER PUMP ENCLOSURE. THE BARRIER WAS BREACHED WHEN AN AIR HOSE WAS RUN THROUGH THE DRAIN CHECK VALVE PENETRATION IN ORDER TO SUPPORT AN ONGOING WORK ACTIVITY. THE PERSONNEL PERFORMING THE WORK DID NOT RECOGNIZE THE WALL WAS A TECHNICAL SPECIFICATION BARRIER. CORRECTIVE ACTIONS TAKEN INCLUDE LABELING THE BARRIER AND PENETRATION TO IDENTIFY THEY ARE TECHNICAL SPECIFICATION COMPONENTS AND REQUIRING THE NUCLEAR WATCH ENGINEER BE NOTIFIED PRIOR TO ALTERING THE BARRIER. LONG TERM CORRECTIVE ACTION BEING CONSIDERED INCLUDES ELIMINATING THE PENETRATION AND GROUTING THE OPENING. THE CONDITION WAS IDENTIFIED WITH THE REACTOR MODE SELECTOR SWITCH IN THE REFUEL POSITION. THE REACTOR POWER LEVEL WAS 0 PERCENT. THE REACTOR VESSEL (RV) WATER TEMPERATURE WAS 84 DEGREES FAHRENHEIT AND THE RV PRESSURE WAS 0 PSIG. THE REPORT IS SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73 (A)(2)(I)(B) AND THE CONDITION POSED NO THREAT TO THE PUBLIC HEALTH AND SAFETY.

[135] PILGRIM 1 DOCKET 50-293 LER 91-010
INADVERTENT SECONDARY CONTAINMENT SYSTEM ISOLATION WHILE BACKING OUT OF SURVEILLANCE TESTING DUE TO PERSONNEL ERROR.
EVENT DATE: 052791 REPORT DATE: 062691 NSSS: GE TYPE: BWR

(NSIC 222456) ON MAY 27, 1991 AT 0034 HOURS, AN INADVERTENT ACTUATION OF THE CHANNEL "A" REACTOR BUILDING ISOLATION CONTROL SYSTEM (RBIS) OCCURRED WHILE BACKING OUT OF A SURVEILLANCE TEST. THE ACTUATION RESULTED IN THE AUTOMATIC CLOSING OF THE TRAIN "A" SECONDARY CONTAINMENT SYSTEM/REACTOR BUILDING VENTILATION DAMPERS, AND THE AUTOMATIC START OF THE STANDBY GAS TREATMENT SYSTEM TRAIN "A" FAN. THE RBIS CIRCUITRY WAS RESET AND THE AFFECTED COMPONENTS WERE RETURNED TO NORMAL SERVICE. THE CAUSE OF THE ACTUATION WAS A MOMENTARY LOSS OF POWER TO THE "A" 125 VDC BATTERY BUS . THE LOSS OF POWER RESULTED FROM AN IMPROPER SEQUENCE USED TO RESTORE THE 125 VDC SYSTEM TO NORMAL ALIGNMENT. THIS OCCURRED WHILE BACKING OUT OF A BATTERY CAPACITY TEST PRIOR TO THE COMPLETION OF THE TEST. SPECIFICALLY, THE "A" BATTERY CHARGER WAS CONNECTED TO THE "A" BUS PRIOR TO CONNECTING THE 125 VDC BATTERY TO THE BUS. THIS CAUSED ERRATIC BATTERY CHARGER OPERATION AND RESULTED IN THE MOMENTARY LOSS OF POWER. CORRECTIVE ACTIONS PLANNED INCLUDE ADDING A CAUTION TO THE BATTERY CAPACITY TEST PROCEDURE ALERTING PERSONNEL NOT TO CONNECT THE BATTERY CHARGER TO ITS ASSOCIATED BUS WITHOUT THE ASSOCIATED BATTERY CONNECTED TO THE BUS. THE ACTUATION OCCURRED DURING A REFUELING OUTAGE. THE REACTOR MODE SELECTOR SWITCH WAS IN THE REFUEL POSITION WITH THE REACTOR VESSEL (RV) COMPLETELY DEFUELED AND NO FUEL MOVEMENT IN PROGRESS.

[136] POINT BEACH 1 DOCKET 50-266 LER 91-005
LOSS OF INSTRUMENT BUS RESULTING IN REACTOR TRIP.
EVENT DATE: 053091 REPORT DATE: 062891 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 222405) ON MAY 30, 1991, THE POINT BEACH NUCLEAR PLANT UNIT 1 WAS OPERATING AT 0% POWER. AT 1146, POWER TO THE UNIT 1 RED INSTRUMENT BUS WAS LOST DUE TO THE FAILURE OF THE INVERTER (1DY01) SUPPLYING THE BUS. THIS RESULTED IN A REACTOR

TRIP. HOWEVER, LOSS OF POWER TO THE BUS DOES NOT DIRECTLY CAUSE A TRIP. THE PLANT PROCESS COMPUTER SYSTEM (PPCS) SEQUENCE OF EVENT RECORDER DID NOT INDICATE THE EXACT CAUSE OF THE TRIP. START-UP TESTS WERE PERFORMED AND ALL RESULTS WERE SATISFACTORY. THE INSTRUMENT BUS WAS TRANSFERRED TO THE SPARE INVERTER AND UNIT 1 WAS STARTED UP AND RETURNED ON LINE AT 0047 ON MAY 31, 1991. 1DY01 WAS SUBSEQUENTLY REPAIRED, TESTED, AND RETURNED TO SERVICE SUPPLYING THE UNIT 1 RED INSTRUMENT BUS.

[137] PRAIRIE ISLAND 1 DOCKET 50-282 LER 90-012 REV 01
UPDATE ON DISCOVERY THAT SEVERAL INSTRUMENTS USED FOR SURVEILLANCE ACCEPTANCE ARE NOT ROUTINELY CALIBRATED.
EVENT DATE: 081590 REPORT DATE: 060691 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)

(NSIC 222371) DURING ROUTINE INDEPENDENT AUDITS, IT WAS DETERMINED THAT SEVERAL INSTRUMENTS USED FOR JUDGING THE ACCEPTABILITY OF CERTAIN SURVEILLANCE TESTS WERE NOT ON A ROUTINE CALIBRATION SCHEDULE. THESE INSTRUMENTS INCLUDE PRESSURE GAUGES USED FOR MEASURING SUCTION AND DISCHARGE PRESSURE FOR ASME SECTION XI TESTING OF CERTAIN PUMPS, AND PRESSURE GAUGES AND FLOWMETERS USED TO PERFORM AIRLOCK DOOR SEAL TESTS. ALL THE INSTRUMENTS IN QUESTION WERE CALIBRATED. AS-FOUND CALIBRATION DATA WERE USED TO JUDGE ACCEPTABILITY OF PREVIOUS TEST DATA. ALL PREVIOUS TEST RESULTS WERE FOUND TO BE ACCEPTABLE. ALL THE INSTRUMENTS WERE PLACED ON A ROUTINE CALIBRATION SCHEDULE.

[138] PRAIRIE ISLAND 1 DOCKET 50-282 LER 91-005
ONE PRESSURIZER SAFETY VALVE LIFT SETPOINT FOUND 2.5% LOW DURING TEST.
EVENT DATE: 051691 REPORT DATE: 061791 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)
VENDOR: CROSBY VALVE

(NSIC 222316) HISTORICALLY, PRESSURIZER SAFETY VALVES USED AT PRAIRIE ISLAND HAVE BEEN SET USING NITROGEN. AS A RESULT OF NRC INFORMATION NOTICE 89-90 AND GUIDANCE IN WESTINGHOUSE WCAP-12910, IT WAS DECIDED TO TEST THE SPARE PRESSURIZER SAFETY VALVES USING STEAM. THE TWO SPARE VALVES WERE SENT TO A CONTRACT FACILITY FOR THE TESTING IN PREPARATION FOR INSTALLATION ON THE UNIT 1 PRESSURIZER AT THE UPCOMING REFUELING OUTAGE. ON MAY 16, 1991 THE CONTRACT FACILITY NOTIFIED PRAIRIE ISLAND OF THE RESULTS OF THE TESTING. ONE VALVE'S LIFT SETPOINT WAS FOUND WITHIN THE 1% TOLERANCE, BUT THE OTHER VALVE WAS FOUND TO LIFT AT 2421 PSIG, ABOUT 2.5% BELOW ITS NOMINAL SETPOINT OF 2485 PSIG. THE SUBJECT VALVE HAD BEEN ORIGINALLY INSTALLED ON UNIT 1 FROM NOVEMBER 18, 1982 TILL SEPTEMBER 15, 1988.

[139] PRAIRIE ISLAND 1 DOCKET 50-282 LER 91-007
AUTO-START OF AUXILIARY BUILDING SPECIAL VENTILATION SYSTEM CAUSED BY TESTING THE WRONG RADIATION MONITOR.
EVENT DATE: 052891 REPORT DATE: 062691 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)

(NSIC 222414) ON MAY 28 1991 UNIT 1 WAS AT 64% POWER COASTING DOWN TO A JUNE REFUELING AND UNIT 2 WAS AT 100% POWER. AN INSTRUMENT AND CONTROL SPECIALIST HAD JUST COMPLETED MAINTENANCE ON RADIATION MONITOR 1R-37, AN AUXILIARY BUILDING VENTILATION GAS MONITOR. THE FUNCTIONAL RESPONSE OF MONITOR 1R-37, UPON RECEIPT OF A HIGH RADIATION SIGNAL, IS TO DEACTIVATE THE AUXILIARY BUILDING NORMAL VENTILATION SYSTEM AND ACTUATE THE NO. 121 AUXILIARY BUILDING SPECIAL VENTILATION SYSTEM. THE FUNCTIONAL RESPONSE OF THE REDUNDANT MONITOR, 1R-30, IS TO DEACTIVATE THE AUXILIARY BUILDING NORMAL VENTILATION SYSTEM AND ACTUATE THE NO. 122 AUXILIARY BUILDING SPECIAL VENTILATION SYSTEM. TO COMPLETE FUNCTIONAL TESTING OF MONITOR 1R-37, THE INSTRUMENT AND CONTROL SPECIALIST ARRANGED FOR THE ASSISTANCE OF A RADIATION PROTECTION SPECIALIST. THE RADIATION PROTECTION SPECIALIST APPROACHED MONITOR 1R-30, THE REDUNDANT MONITOR, WITH A RADIATION SOURCE AND REPORTED VIA RADIO THAT HE WAS STANDING BY MONITOR 1R-37 WHEN THE INSTRUMENT AND CONTROL SPECIALIST ASKED THE RADIATION PROTECTION SPECIALIST TO SIMULATE HIGH RADIATION ON THE MONITOR NO. 122 AUXILIARY BUILDING SPECIAL VENTILATION SYSTEM STARTED, AN UNPLANNED AUTOSTART.

[140] QUAD CITIES 1 DOCKET 50-254 LER 91-009
REACTOR CORE ISOLATION COOLING INOPERABLE FROM NOT MEETING TECHNICAL SPECIFICATION REQUIREMENTS DUE TO CONTROLLER AND GOVERNOR PROBLEMS.
EVENT DATE: 042791 REPORT DATE: 052891 NSSS: GE TYPE: BWR
VENDOR: WOODWARD GOVERNOR COMPANY

(NSIC 222198) AT 0100 HOURS ON APRIL 26, 1991, UNIT ONE REACTOR WAS OPERATING IN THE RUN MODE AT 22 PERCENT RATED CORE THERMAL POWER. AT THIS TIME, THE UNIT ONE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM WAS DECLARED INOPERABLE DUE TO FAILURE TO MEET PUMP REQUIREMENTS. TROUBLESHOOTING OF THE SYSTEM CONTINUED THROUGHOUT THIS EVENT DUE TO REPEATED FAILURES OF THE RCIC CONTROLLER AND GOVERNOR. THE SYSTEM WAS SUCCESSFULLY TESTED AND DECLARED OPERABLE ON MAY 3, 1991 AT 0345 HOURS. THE MOST SIGNIFICANT CAUSE OF THIS EVENT WAS DUE TO AN INADEQUATE CALIBRATION PROCEDURE WHICH RESULTED IN UNSTABLE RESPONSES FROM THE CONTROLLER AND GOVERNOR. ALSO, BECAUSE THE SYSTEM CONTROL PARAMETERS CHANGED, THE RESULT AFFECTED THE OPERATION OF THE FLOW CONTROLLER. THIS EVENT IS BEING REPORTED ACCORDING TO 10CFR50.73(A)(2)(V)(D) AND 10CFR50.73(A)(2)(I)(A).

[141] QUAD CITIES 1 DOCKET 50-254 LER 91-012
HPCI INOPERABLE DUE TO FAILED BUSHING IN STOP VALVE.
EVENT DATE: 050791 REPORT DATE: 060691 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)
VENDOR: ATWOOD & MORRILL CO., INC.

(NSIC 222294) AT 1015 HOURS ON MAY 7, 1991, UNIT ONE WAS IN THE RUN MODE AT 93 PERCENT RATED CORE THERMAL POWER. UNIT ONE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM WAS DECLARED INOPERABLE TO PERFORM QCOS 2300-13, HPCI SYSTEM MANUAL INITIATION TEST. DURING THIS TEST THE HPCI TURBINE STOP VALVE FAILED TO STROKE FULLY OPEN. UPON INVESTIGATING THE PROBLEM, IT WAS IDENTIFIED THAT THE LONG BUSHING IN THE STUFFING BOX OF THE STOP VALVE HAD SLID DOWN THE VALVE STEM AND THE BUSHING PIN HAD SHEARED. UNIT TWO HPCI STOP VALVE WAS CHECKED TO VERIFY THAT THE SAME PROBLEM DID NOT EXIST. IT WAS DISCOVERED THAT THE LONG BUSHING WAS ALSO PARTIALLY OUT OF THE STUFFING BOX WITH NO BUSHING PIN IN THE BUSHING. UNIT TWO HPCI WAS DECLARED INOPERABLE AS OF 1000 HOURS, MAY 8, 1991. THE FAILURE OF BOTH HPCI SYSTEMS WAS DUE TO INADEQUATE VENDOR INSTRUCTIONS WHICH RESULTED IN THE STUFFING BOXES NOT BEING OPENED. UNIT ONE HPCI SYSTEM WAS SUCCESSFULLY REPAIRED, TESTED, AND DECLARED OPERABLE ON MAY 17, 1991, AT 1745 HOURS. UNIT TWO HPCI WAS REPAIRED AND DECLARED OPERABLE ON MAY 19, 1991, AT 1320 HOURS. THIS EVENT WAS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(V)(D).

[142] RIVERBEND 1 DOCKET 50-458 LER 90-005 REV 03
UPDATE ON CONDITIONS AFFECTING THE OFFGAS PRETREATMENT RADIATION MONITOR.
EVENT DATE: 030290 REPORT DATE: 053191 NSSS: GE TYPE: BWR
VENDOR: STONE & WEBSTER ENGINEERING CORP.

(NSIC 222189) ON 03/02/90 AND ON 03/06/90 WITH THE REACTOR OPERATING AT 100% POWER (OPERATIONAL CONDITION 1), CONDITIONS AFFECTING THE OFFGAS PRETREATMENT RADIATION MONITOR (PTRM) WERE IDENTIFIED BY ENGINEERING PERSONNEL. THESE CONDITIONS WERE: 1) A NON-CONSERVATIVE VALUE FOR THE HIGH ALARM SETPOINT, AND (2) PERIODS OF INOPERABILITY IN THE PAST IN WHICH THE MONITOR HAS BEEN INOPERABLE DUE TO INADEQUATE SAMPLE FLOW. THESE ARE REPORTABLE AS (1) A CONDITION OUTSIDE OF THE DESIGN BASIS, AND 2) OPERATION PROHIBITED BY THE TECHNICAL SPECIFICATIONS, RESPECTIVELY. THE ERRONEOUS HIGH ALARM SETPOINT HAS BEEN CORRECTED. GSU HAS ALSO IDENTIFIED SHORT AND LONG TERM CORRECTIVE ACTIONS TO ASSURE ADEQUATE SAMPLE FLOW IS MAINTAINED TO THE MONITOR AND THUS ASSURE OPERABILITY. SUBSEQUENT OPERATIONS HAVE DEMONSTRATED THAT THE CORRECTED HIGH ALARM SETPOINT IS OVERLY CONSERVATIVE. THEREFORE, GSU PLANS TO REVISE THE SETPOINT UPWARD BASED ON A MORE REALISTIC CALCULATION. DUE TO THE PRESENCE OF REDUNDANT INDEPENDENT MONITORS AND ALARMS, ADEQUATE ASSURANCE EXISTS THAT RBS HAS NOT EXCEEDED THE REVISED ALARM SETPOINT. FOR THE SAME REASON, ALTERNATIVE MEANS OF MONITORING RADIOACTIVITY IS PROVIDED IN THE RBS DESIGN EVEN WITH THE PTRM INOPERABLE. THEREFORE, THESE CONDITIONS HAVE NOT ADVERSELY AFFECTED THE HEALTH AND SAFETY OF THE PUBLIC.

[143] RIVERBEND 1 DOCKET 50-458 LER 90-034 REV 01
UPDATE ON REACTOR PROTECTION SYSTEM ACTUATION DUE TO SHORTING IN THE CONNECTORS DURING UNDER VESSEL DECONTAMINATION ACTIVITIES.
EVENT DATE: 102790 REPORT DATE: 061391 NSSS: GE TYPE: BWR

(NSIC 222329) AT APPROXIMATELY 1138 ON 10/27/90 WITH THE UNIT IN OPERATIONAL CONDITION 5 (REFUELING), THE REACTOR PROTECTION SYSTEM (RPS) ACTUATED ON HIGH NEUTRON FLUX SIGNALS FROM INTERMEDIATE RANGE MONITORS (IRMS) F AND G. NO ROD MOVEMENT RESULTED FROM THIS RPS ACTUATION. THE ROOT CAUSE WAS SHORTING IN TWO IRM DETECTOR CONNECTORS DUE TO THE INTRODUCTION OF WATER INTO THE CONNECTORS. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV) AS AN ENGINEERED SAFETY (ESF) ACTUATION. THE MAINTENANCE PROCEDURE GOVERNING CONTROL ROD DRIVE REMOVAL WILL BE REVISED. A PRECAUTION WILL BE ADDED TO ALERT PERSONNEL TO THE NEED FOR CARE WHEN PERFORMING DECONTAMINATION WHILE IN PROXIMITY TO NUCLEAR INSTRUMENTS. IN ADDITION, PREREQUISITES WILL BE ADDED TO VERIFY THE INTEGRITY OF THE CABLE GUARDS WHICH PROTECT THE IRM CONNECTORS, PRIOR TO CRD REMOVAL WORK. ALL INSTALLED CONTROL RODS WERE INSERTED PRIOR TO THIS EVENT. NOTE THAT SOME CONTROL RODS HAD BEEN REMOVED FOR REPLACEMENT. ADDITIONALLY, NO ROD MOTION OCCURRED AS A RESULT OF THE UNPLANNED RPS ACTUATION AND THE RPS SYSTEM RESPONDED AS DESIGNED. THEREFORE, THIS EVENT DID NOT ADVERSELY AFFECT THE HEALTH AND SAFETY OF THE PUBLIC.

[144] RIVERBEND 1 DOCKET 50-458 LER 90-033 REV 03
UPDATE ON RWCU ISOLATION DURING MODIFICATIONS TO POWER SUPPLY WIRING IN A CONTROL ROOM PANEL.
EVENT DATE: 110490 REPORT DATE: 070191 NSSS: GE TYPE: BWR

(NSIC 222512) ON 11/04/90 AT 1140 WITH THE UNIT IN OPERATIONAL CONDITION 5 (REFUELING), AN ISOLATION OF THE REACTOR WATER CLEANUP SYSTEM (RWCU) OCCURRED. THE ISOLATION OCCURRED WHILE PERFORMING PLANT MODIFICATIONS TO POWER SUPPLY WIRING IN CONTROL ROOM PANEL 1H13-P642. THE POWER SUPPLY CIRCUIT IN THIS PANEL PROVIDES POWER TO SEVERAL SYSTEMS INCLUDING THE LEAK DETECTION SYSTEM (LDS) AND THE RWCU SYSTEM AND IS DAISY-CHAINED FROM ONE COMPONENT TO ANOTHER. THE MODIFICATION INVOLVED DE-TERMINATING AND RE-TERMINATING POWER SUPPLY LEADS ON SEVERAL LDS COMPONENTS, AND WHEN THE FIRST DE-TERMINATION WAS MADE, POWER WAS LOST TO THE RWCU SYSTEM THROUGH THE DAISY-CHAIN. SINCE THE SAFETY RELATED COMPONENTS OF THE RWCU SYSTEM FAIL TO A SAFE POSITION UPON A LOSS OF POWER, AN ISOLATION RESULTED. THIS ISOLATION CONSTITUTES AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION; THEREFORE, THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73A)(2)(IV). THE RWCU ISOLATION OCCURRED AS DESIGNED AND WAS RESTORED FOLLOWING RE-TERMINATION OF THE LEAD. THEREFORE, THIS EVENT DID NOT ADVERSELY AFFECT THE HEALTH AND SAFETY OF THE PUBLIC. AT THE DIRECTION OF THE PLANT MANAGER, A TASK FORCE HAS BEEN FORMED TO ESTABLISH A PROGRAM TO RESOLVE ENGINEERING/MAINTENANCE PLANNING INTERFACE ISSUES. THE TASK FORCE HAS COMPLETED ITS RECOMMENDATIONS.

[145] RIVERBEND 1 DOCKET 50-458 LER 91-009
PERSONNEL FAILURE TO PROPERLY MOVE HIGH RADIATION AREA BARRIER.
EVENT DATE: 043091 REPORT DATE: 053091 NSSS: GE TYPE: BWR

(NSIC 222190) ON 04/30/91 AT APPROXIMATELY 1745 WITH THE UNIT IN OPERATIONAL CONDITION 1 (POWER OPERATION) TWO ELECTRICAL MAINTENANCE WORKERS MOVED A HIGH RADIATION AREA (HRA) BARRIER, THEREBY CREATING AN OPENING IN THE BARRIER. THE ROPE BARRIER WAS LOCATED ON THE 95' ELEVATION OF THE FUEL BUILDING (*ND*) IN THE CASK WASH DOWN AREA. TECHNICAL SPECIFICATION 6.12.1 REQUIRES THAT HIGH RADIATION AREAS BE BARRICADED. THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(I)(B) AS OPERATION PROHIBITED BY TECHNICAL SPECIFICATIONS. ALL NON-ESSENTIAL PERSONNEL IN THE RADIOLOGICALLY CONTROLLED AREA (RCA) WERE REQUIRED TO LEAVE THE RCA TO ATTEND MEETINGS WITH MANAGEMENT. THEY WERE MADE AWARE THAT THE PLANT MANAGER WOULD ENFORCE PROCEDURAL COMPLIANCE IN HRAS AND DISCIPLINARY ACTION WOULD BE TAKEN AGAINST VIOLATORS. DISCIPLINARY ACTION HAS BEEN TAKEN AGAINST THE TWO ELECTRICAL MAINTENANCE WORKERS, THE ELECTRICAL FOREMAN AND THE ACTING ELECTRICAL SUPERVISOR. NO UNAUTHORIZED INDIVIDUAL ENTERED THE CASK WASH DOWN AREA. THIS EVENT DID NOT AFFECT PLANT SYSTEMS AND HAD NO OPERATIONAL IMPACT. THEREFORE THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT ADVERSELY AFFECTED.

[146] RIVERBEND 1 DOCKET 50-458 LER 91-010
CONTAINMENT ISOLATION VALVES NOT VERIFIED CLOSED EVERY 31 DAYS AS REQUIRED BY TECHNICAL SPECIFICATIONS.
EVENT DATE: 051691 REPORT DATE: 061791 NSSS: GE TYPE: BWR

(NSIC 222330) ON MAY 16, 1991, IT WAS DISCOVERED THAT CONTAINMENT ISOLATION VALVES 1CPP*MOV104, 1CPP*MOV105 AND 1CPP*SOV140 WERE NOT BEING VERIFIED CLOSED AND SECURED EVERY 31 DAYS AS REQUIRED BY UPDATED SAFETY ANALYSIS REPORT (USAR) SECTION 6.2.4.3.7.2. TECHNICAL SPECIFICATION 4.6.1.1.B, "PRIMARY CONTAINMENT INTEGRITY - OPERATING" ALSO REQUIRES THAT THESE VALVES BE VERIFIED CLOSED EVERY 31 DAYS. THEREFORE, THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(I)(B) AS OPERATION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. SURVEILLANCE PROCEDURE STP-000-0201 WAS REVISED TO REQUIRE VERIFICATION THAT VALVES 1CPP*SOV140, 1CPP*MOV104, AND 1CPP*MOV105 ARE CLOSED AND SECURED EVERY 31 DAYS. PROCEDURE GOP-0001 WAS ALSO REVISED TO ADD THE SAME REQUIREMENTS WHEN CONTAINMENT IS SECURED. IN ADDITION, A REVIEW OF SELECTED COMMITMENTS WILL BE PERFORMED TO IDENTIFY THOSE WITH ACTIONS WHICH REQUIRE PROCEDURAL CONTROL. THE SUBJECT CONTAINMENT ISOLATION VALVES WERE FOUND TO BE CLOSED AND SECURED WHEN THE CONDITION WAS IDENTIFIED. NO EVIDENCE THAT THESE VALVES HAD BEEN INCORRECTLY POSITIONED WAS FOUND. THEREFORE, THE VALVES WOULD HAVE PERFORMED THEIR CONTAINMENT ISOLATION FUNCTION AND THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED BY THIS CONDITION.

[147] ROBINSON 2 DOCKET 50-261 LER 91-007
FAILURE TO PERFORM SURVEILLANCE TEST.
EVENT DATE: 051791 REPORT DATE: 060691 NSSS: WE TYPE: PWR

(NSIC 222366) AT ABOUT 2000 ON MAY 19, 1991, WITH THE H. B. ROBINSON UNIT NO. 2 OPERATING AT ABOUT 92 POWER, IT WAS IDENTIFIED THAT THE MAINTENANCE SURVEILLANCE TEST, MST-902, BATTERY TEST - DAILY 5 DAYS PER WEEK, HAD NOT BEEN PERFORMED ON MAY 17, 1991, TO MEET THE 5 DAYS PER WEEK REQUIREMENT FOR THE WEEK OF MAY 11 THROUGH 17, 1991. THIS SURVEILLANCE TEST IS REQUIRED BY TECHNICAL SPECIFICATION 4.6.3. WHEN IT WAS IDENTIFIED THAT THE SCHEDULED SURVEILLANCE TEST WAS MISSED, LICENSEE TECHNICIANS PERFORMED THE TEST PROVING STATION BATTERY OPERABILITY BY 2015 MAY 19, 1991. THE LICENSEE INSTRUMENTATION AND CONTROL (I&C) SUPERVISOR AND THE SHIFT SUPERVISOR WERE NOTIFIED OF THE MISSED SURVEILLANCE AT 2030 MAY 19, 1991, AND THE SHIFT SUPERVISOR DOCUMENTED THAT THE STATION BATTERIES WERE ADMINISTRATIVELY OUT OF SERVICE FROM 2000 TO 2015. THIS LER IS THE RESULT OF PERSONNEL ERROR AND IS BEING REPORTED UNDER 10CFR50.73(A)(2)(I) AND 50.73(A)(2)(VII).

[148] SALEM 1 DOCKET 50-272 LER 91-020
CONTROL ROOM VENTILATION SWITCH TO ACCIDENT MODE DUE TO EQUIPMENT FAILURE.
EVENT DATE: 051091 REPORT DATE: 060691 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)
VENDOR: TRACER LAB

(NSIC 222369) ON MAY 10, 1991 AT 1400 HOURS, DURING NORMAL PLANT OPERATION, THE CONTROL ROOM GENERAL AREA RADIATION MONITORING SYSTEM (RMS) MONITOR (1R1A) SPIKED HIGH. THIS RESULTED IN THE AUTOMATIC SWITCHING OF THE CONTROL ROOM VENTILATION FROM NORMAL OPERATION TO ITS ACCIDENT MODE OF OPERATION (100% RECIRCULATION) FOR BOTH SALEM UNIT 1 AND UNIT 2 (BY DESIGN). THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO EQUIPMENT FAILURE. AT THE TIME OF THE CHANNEL SPIKE, A MAINTENANCE-I&C TECHNICIAN WAS RESTORING POWER TO THE 1R33 RMS CHANNEL (ION EXCHANGE FILTER MONITOR). THE 1R33 CHANNEL CONTROL DRAWER IS LOCATED IN THE SAME CABINET, PHYSICALLY NEXT TO, THE 1R1A CHANNEL. A SIMILAR EVENT OCCURRED ON SEPTEMBER 17, 1990 (REFERENCE LER 272 /90-031-00); HOWEVER, THE CAUSE OF THAT EVENT WAS DIFFERENT. THE 1R33 CHANNEL WAS INSPECTED. A FAILED LIGHT SOCKET, FOR "POWER ON/FAIL" INDICATION WAS FOUND. THE SOCKET HAD DEVELOPED A SHORT. THIS IS BELIEVED TO BE THE CAUSE OF THE SPIKE. WHEN THE 1R33 CHANNEL WAS TURNED ON, THE LIGHT SOCKET SHORT RESULTED IN AN EM/RFI (ELECTRO-MAGNETIC/RADIO FREQUENCY INTERFERENCE) SIGNAL CAUSING THE 1R1A CHANNEL SPIKE. THE FAILED 1R33 CHANNEL LIGHT SOCKET WAS REPLACED.

[149] SALEM 1 DOCKET 50-272 LER 91-021
INADEQUATE AMENDMENT IMPLEMENTATION & SURVEILLANCE NOT DONE.
EVENT DATE: 052291 REPORT DATE: 062791 NSSS: WE TYPE: PWR

(NSIC 222409) THIS LER ADDRESSES 3 TECH. SPEC. 3/4.8.3.1 NONCOMPLIANCE EVENTS. IN ALL EVENTS, TECH. SPEC. SURV. 4.8.3.1 REQUIREMENTS WERE NOT COMPLETELY SATISFIED. THE FIRST EVENT WAS DISCOVERED ON 5/22/91 WHEN IT WAS DETERMINED THAT A CONTAINMENT PENETRATION OVERCURRENT PROTECTION DEVICE WAS NOT INCLUDED ON THE UFSAR LISTING. THE SECOND EVENT WAS DISCOVERED ON 6/12/91 WHEN IT WAS DETERMINED THAT THIS DEVICE HAD NOT BEEN SURVEILLED IN ACCORDANCE WITH SPEC. 4.8.3.1.B REQUIREMENTS NOR HAD THE ASSOCIATED ACTION STATEMENTS BEEN APPLIED ON 5/22/91. THE THIRD TECH. SPEC. NONCOMPLIANCE EVENT WAS DISCOVERED ON 6/3/91 WHEN IT WAS IDENTIFIED THAT IMPLEMENTATION OF TECH. SPEC. AMENDMENT 105 IN APRIL 1991 (I.E., COMPLETION OF THE RECENT NINTH REFUELING OUTAGE) WAS INADEQUATE. SURV. 4.8.3.1.B. REQUIREMENTS WERE NOT MET FOR ALL MOLDED CASE CIRCUIT BREAKERS. THE ROOT CAUSE OF THE FIRST EVENT IS ATTRIBUTED AN INCORRECT SCHEMATIC AND THE SECOND AND THIRD EVENTS ARE DUE TO PERSONNEL ERROR. THE TECH. SPEC. ADMINISTRATOR DID NOT ENSURE THAT SURVEILLANCE REQUIREMENTS WERE MET DUE TO POOR WORK PRACTICES (SECOND EVENT ONLY) AND INADEQUATE COMMUNICATIONS (SECOND & THIRD EVENTS). THESE EVENTS HAVE BEEN REVIEWED WITH THE INDIVIDUALS INVOLVED. AS PREVIOUSLY COMMITTED TO BY UNIT 1 LER 272/91-019-00 (DATED MAY 30, 1991) INPO GOOD PRACTICE OE-906, IS BEING PRESENTED TO STATION PERSONNEL.

[150] SALEM 2 DOCKET 50-311 LER 91-007
ESF ACTUATION SIGNALS FOR CVI: 2R12B AND 1R41C CHANNEL FAILURES.
EVENT DATE: 052291 REPORT DATE: 062091 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 1 (PWR)
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 222323) THIS LER ADDRESSES CONTAINMENT PURGE/PRESSURE-VACUUM RELIEF SYSTEM (CP/P-VRS) ISOLATION SIGNALS (3 EVENTS) (AN ENGINEERED SAFETY FEATURE (ESF)), ONCE FROM THE SALEM UNIT 2 2R12B CONTAINMENT IODINE RADIATION MONITOR RADIATION MONITORING SYSTEM (RMS) CHANNEL AND TWICE FROM THE UNIT 1 LR41C PLANT VENT RADIOACTIVE NOBLE GAS MONITOR RMS CHANNEL. BOTH RMS CHANNELS ARE MANUFACTURED BY VICTOREEN. ON 5/22/91, THE 2R12B RMS CHANNEL FAILED LOW. ON 5/29/91, THE LR41C RMS CHANNEL FAILED LOW. TECH. SPEC. TABLE 3.3-13 ACTION 31 WAS ENTERED. INVESTIGATION OF THIS ACTUATION DID NOT IDENTIFY THE SPECIFIC CAUSE OF THE EVENT. UPON SUCCESSFUL COMPLETION OF FUNCTIONAL TESTING, THE CHANNEL WAS RETURNED TO SERVICE (AND TECH. SPEC. TABLE 3.3-13 ACTION 31) ON 6/3/91 AT 1735 HOURS. ON 6/3/91 AT 2100 HOURS, DURING NORMAL POWER OPERATIONS, THE LR41C RMS CHANNEL AGAIN FAILED. THE CHANNEL WAS DECLARED INOPERABLE AND TECH. SPEC. TABLE 3.3-13 ACTION 31 EXITED WAS RE-ENTERED. THE ROOT CAUSE OF THE ACTUATION OF THE CP/P-VRS ISOLATION SIGNALS IS ATTRIBUTED TO EQUIPMENT DESIGN CONCERNS. PERIODIC PROBLEMS WITH THE VICTOREEN SYSTEM HAVE BEEN EXPERIENCED, AS INDICATED IN PRIOR LERS (E.G., 311/90-040-00). INVESTIGATION OF THE SECOND LR41C CHANNEL FAILURE IDENTIFIED THE SPECIFIC CAUSE OF THE TWO CHANNEL FAILURES. THE CONTROL MODULE HAD FAILED. IT WAS SUBSEQUENTLY REPLACED. SEVERAL SYSTEM DESIGN MODIFICATIONS WILL BE IMPLEMENTED

[151] SALEM 2 DOCKET 50-311 LER 91-008
4KV VITAL BUS UV RELAY SETPOINTS FOUND BELOW MINIMUM TECH SPEC ALLOWABLE VALUE.
EVENT DATE: 060391 REPORT DATE: 070391 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 1 (PWR)
VENDOR: ROCHESTER INSTRUMENT SYSTEMS, INC.

(NSIC 222421) ON JUNE 3, 1991, THE 91.6% SUSTAINED UNDERVOLTAGE RELAY'S MINIMUM DROPOUT TRIP SETPOINT VOLTAGE, FOR TWO (2) OF THE THREE (3) OF THE 2A 4KV VITAL BUS UNDERVOLTAGE RELAYS AND FOR ALL THREE (3) 2B 4KV VITAL BUS UNDERVOLTAGE RELAYS, WERE FOUND TO BE BELOW THE TECHNICAL SPECIFICATION MINIMUM ALLOWABLE VALUE OF 91%. THIS WAS DISCOVERED DURING TECHNICAL SPECIFICATION SURVEILLANCE 4.3.2.1.1 TESTING WHICH REQUIRES MONTHLY TESTING OF UNDERVOLTAGE RELAY SETPOINTS. PROCEDURES S2 .MD-FT.4KV-0001(Q) (0002), "ESFAS INSTRUMENTATION MONTHLY FUNCTIONAL TEST-2A (2B) 4KV VITAL BUS UNDER VOLTAGE") WERE BEING USED SUPPORT THE SURVEILLANCE TESTING. THE LOWE ST AS-FOUND TRIP POINT OF THE FIVE SUBJECT RELAYS

WAS 90.5%. THE ROOT CAUSE INVESTIGATION, TO DETERMINE THE REASON FOR THE AS FOUND SETPOINT VARIANCE, IS CONTINUING. WEEKLY TESTING OF THE INSTALLED RELAYS IS BEING PERFORMED UNDER THE DIRECTION OF SYSTEM ENGINEERING. IN ADDITION, CONTROLLED BENCH TESTS ARE BEING PERFORMED ON IDENTICAL RELAYS (I.E., SPARE RELAYS PURCHASED WHEN THE INSTALLED RELAYS WERE CHASED). IN PARALLEL WITH THE ON-GOING ENGINEERING INVESTIGATION, DESIGN CHANGE TO REPLACE THE SUBJECT RELAYS HAS BEEN INITIATED.

[152] SAN ONOFRE 1 DOCKET 50-206 LER 91-007 REV 01
UPDATE ON LARGE BREAK LOSS OF COOLANT ACCIDENT ANALYSES NON-CONSERVATISMS DUE TO INCORRECT REACTOR COOLANT SYSTEM VOLUMES UTILIZED IN THE ANALYSES.
EVENT DATE: 032891 REPORT DATE: 061491 NSSS: WE TYPE: PWR

(NSIC 222289) ON 3/28/91, WITH UNIT 1 OPERATING AT 20% REACTOR POWER, IT WAS DETERMINED THAT THE VALUE FOR THE REACTOR VESSEL REFILL VOLUME USED IN THE LARGE BREAK LOSS OF COOLANT ACCIDENT (LBLOCA) ANALYSIS PERFORMED BY WESTINGHOUSE WAS UNDERESTIMATED BY APPROX. 182 CUBIC FEET. AN INITIAL EVALUATION OF THE EFFECTS OF THIS VOLUME DISCREPANCY USING ACTUAL PLANT OPERATING PARAMETERS IN THE EVENT OF LBLOCA INDICATE THAT THE PEAK CLAD TEMPERATURE (PCT) WOULD NOT HAVE EXCEEDED THE SONGS 1 ACCEPTANCE CRITERIA OF 2300 DEGREES F. ALTHOUGH ACTUAL OPERATION COULD NOT HAVE RESULTED IN EXCEEDING THE LOCA/PCT LIMIT, UNDER HYPOTHETICAL OPERATING CONDITIONS ASSUMED IN THE DESIGN BASIS ACCIDENT ANALYSIS, IT IS POSSIBLE THAT LOCA/PCT LIMIT EXCEEDENCE COULD BE PREDICTED TO OCCUR. CORRECTIVE ACTIONS INCLUDED: 1) IMMEDIATELY RESTRICTING REACTOR POWER LEVEL TO 75% (THIS REDUCED POWER LEVEL ENSURED THAT THE SONGS 1 PCT ACCEPTANCE CRITERIA FOLLOWING A LBLOCA WOULD HAVE BEEN SATISFIED), AND 2) ADMINISTRATIVELY RESTRICTING INCORE AXIAL OFFSET (IAO) TO ALLOW FOR FULL POWER OPERATION. SUBSEQUENTLY, ADDITIONAL VOLUME DIFFERENCES WERE IDENTIFIED IN OTHER ANALYSES PERFORMED BY WESTINGHOUSE. THESE ADDITIONAL DIFFERENCES COULD AFFECT THE LBLOCA/PCT AND THE LBLOCA CONTAINMENT MASS AND ENERGY ANALYSES, CONSERVATIVELY ASSUMING THESE DIFFERENCES TO BE ERRORS.

[153] SAN ONOFRE 1 DOCKET 50-206 LER 91-010
MANUAL REACTOR TRIP FOLLOWING DROPPED CONTROL RODS DUE TO AN OPEN CIRCUIT IN THE CONTROL ROD DRIVE CIRCUITRY.
EVENT DATE: 052891 REPORT DATE: 062791 NSSS: WE TYPE: PWR
VENDOR: AMPHENOL
WESTINGHOUSE ELECTRIC CORP.

(NSIC 222395) AT 2343 ON 5/28/91, WITH UNIT 1 IN MODE 1 AT 91% POWER, TWO OF THE 12 CONTROL BANK 1 CONTROL RODS (CRS) (H-6 AND H-10) DROPPED INTO THE CORE. APPROPRIATE CONTROL ROOM ALARMS WERE RECEIVED AND OPERATORS CONFIRMED NEGATIVE REACTIVITY INSERTION PER PROCEDURES. OPERATORS PROPERLY RESPONDED (WITHIN 20 SECONDS) BY MANUALLY TRIPPING THE REACTOR. AT 2344, THE AUXILIARY FEEDWATER (AFW) SYSTEM ACTUATED ON LOW STEAM GENERATOR (SG) WATER LEVEL WHICH OCCURRED DUE TO EXPECTED SG WATER LEVEL SHRINKAGE. AT 0044 ON 5/29/91, THE REACTOR TRIP RESPONSE PROCEDURE WAS EXITED WITH THE PLANT STABILIZED IN MODE 3. THE CRS DROPPED AS A RESULT OF AN OPEN CONNECTION IN THE "HALF-VOLTAGE" RESISTOR NETWORK ASSOCIATED WITH THIS ROD SUBGROUP. THE OPEN CONNECTION DE-ENERGIZED THE MOVEABLE GRIPPER COILS ASSOCIATED WITH THE AFFECTED RODS, THUS PERMITTING THEM TO DROP INTO THE CORE. THE OPEN CONNECTION AS CAUSED BY A CONDUCTOR/LUG (JUMPER) CONNECTION FAILURE AT ONE TERMINATION POINT. DESTRUCTIVE PHYSICAL ANALYSIS HAS REVEALED THAT THE OPEN CONNECTION RESULTED FROM LONG-TERM LOCALIZED HEATING RESULTING IN OXIDATION OF THE COPPER WIRE IN THE VICINITY OF THE FAILURE SITE. CONTINUED HEATING LED TO ARCING MELTING, AND EVENTUAL FAILURE. THE FAILED JUMPER WAS REPLACED WITH IN-KIND PARTS. IN ADDITION, OTHER JUMPER WIRES/LUGS WITHIN THE CR "HALF POWER" RESISTOR BANK WERE REPLACED IN-KIND.

[154] SEABROOK 1 DOCKET 50-443 LER 91-006
REACTOR TRIP DUE TO AN INADVERTENT ACTUATION OF THE TURBINE MECHANICAL OVERSPEED PROTECTION SYSTEM.
EVENT DATE: 060291 REPORT DATE: 070291 NSSS: WE TYPE: PWR
VENDOR: SPERRY VICKERS (SPERRY RAND CORP)

(NSIC 222510) ON 6/2/91, AT 6:47 A.M., A TURBINE GENERATOR TRIP WITH A SUBSEQUENT REACTOR TRIP OCCURRED. RHE TRIP WAS INITIATED BY AN INADVERTENT ACTUATION OF THE TURBINE YESHANICAL OVERSPEED PROTECTION SYSTEM. A MAIN FEEDWATER ISOLATION ALSO OCCURRED SUBSEQUENT TO THE TRIP. DURING A WEEKLY TURBINE MECHANICAL OVERSPEED TRIP TEST, THE OIL TRIP SOLENOID VALVE (OTSV) DID NOT RETURN TO ITS ORIGINAL POSITION. MAINTENANCE PERSONNEL REMOVED ONE OF THE SOLENOID VALVE HOUSING COVERS TO CONDUCT A VISUAL VERIFICATION OF THE ACTUAL POSITION OF THE OTSV AND ITS LIMIT SWITCH. WHILE THE HOUSING COVER WAS REMOVED THE LIMIT SWITCH ASSEMBLY BASE PLATE AND BRACKET DROPPED AWAY FROM THE HOUSING. CONSEQUENTLY, THE LIMIT SWITCH CHANGED STATE CAUSING THE MECHANICAL LOCKOUT SOLENOID VALVE TO RESET. A TURBINE TRIP IMMEDIATELY RESULTED SINCE THE OTSV WAS STILL SUPPLYING BEARING OIL TO ACTUATE THE MECHANICAL OVERSPEED TRIP LEVER ASSEMBLY. IN RESPONSE TO THE TURBINE TRIP AND REACTOR TRIP, THE CONDENSER STEAM DUMP VALVES ACTUATED AS DESIGNED. HOWEVER, TWO STEAM DUMP VALVES REMAINED OPEN LONGER THAN EXPECTED. THE CAUSE OF THE OIL TRIP SOLENOID VALVE TO REMAIN IN THE TEST POSITION WAS THE PRESENCE OF CORROSION PRODUCTS INSIDE THE VALVE BODY. THE OIL TRIP SOLENOID VALVE WAS REPLACED AND SUBSEQUENTLY EXERCISED TO ENSURE FREE MOVEMENT.

[155] SEQUOYAH 1 DOCKET 50-327 LER 88-007 REV 05
UPDATE ON OPENING OF CONTAINMENT RESULTS IN CONTAINMENT ENVELOPE OUTSIDE THE BOUNDARY SET FOR SURVEILLANCE TESTING OF AUXILIARY BUILDING GAS TREATMENT SYSTEM.
EVENT DATE: 012488 REPORT DATE: 053091 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 222226) THIS LER HAS BEEN REVISED TO ENSURE THAT THE AUX. BLDG. GAS TREATMENT SYSTEM REMAINS OPERABLE DURING 2-UNIT OPERATION. ON 1/24/88, WITH BOTH UNITS IN MODE 5, IT WAS DISCOVERED THAT THE AUX. BLDG. SECONDARY CONTAINMENT ENCLOSURE WAS NOT BEING MAINTAINED WITHIN THE CONFIGURATION SET DURING THE TECH SPEC (TS) SURVEILLANCE TESTING USED TO VERIFY ABGTS OPERABILITY. ON 8/24/88, WITH UNIT 1 IN MODE 5 AND UNIT 2 IN MODE 1, IT WAS DETERMINED THAT THE UNIT 1 CONTAINMENT PURGE SYSTEM WAS IN OPERATION WITHOUT THE REQUIRED COMPENSATORY MEASURES BEING PROPERLY DOCUMENTED. THESE CONDITIONS WERE CAUSED BY (1) THE LACK OF ADEQUATE CONTROLS TO ENSURE THE ABSCE BOUNDARY WAS MAINTAINED WITHIN THE CONDITION SET BY SURVEILLANCE TESTING, (2) AN INAPPROPRIATE DESIGN ASSUMPTION MADE DURING PLANT CONSTRUCTION ON HOW ABSCE BREACHES WOULD BE CONTROLLED, AND (3) AN INCOMPLETE COMPENSATORY MEASURES (CM) PROGRAM. CORRECTIVE ACTIONS INCLUDED CLOSING THE UNIT 1 BLAST DOOR AND TAGGING THE UNIT 1 CONTAINMENT PURGE SYSTEM OUT OF SERVICE, REVISING THE PROCEDURE GOVERNING ABSCE BREACHES, AND UPGRADING THE CM PROGRAM. FOLLOWING SUBSEQUENT LEAK TESTING OF THE UNIT 1 ANNULUS, THE UNIT 1 BLAST DOOR WAS REOPENED.

[156] SEQUOYAH 1 DOCKET 50-327 LER 91-010 REV 01
UPDATE ON FAILURE TO CONDUCT VISUAL INSPECTION OF EXPANSION JOINT SEALS PREVIOUSLY NOT CONSIDERED TO BE FIRE BARRIERS AND DISCOVERY THAT SEAL MATERIAL DOES NOT MEET FIRE BARRIER REQUIREMENTS.
EVENT DATE: 051591 REPORT DATE: 070591 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 222453) ON 5/15/91, AT 1800 EASTERN DAYLIGHT TIME (EDT), WITH UNITS 1 AND 2 OPERATING IN MODE 1 IT WAS DETERMINED THAT THE EXPANSION JOINT MATERIAL BETWEEN THE REACTOR BUILDING SHIELD WALLS AND THE AUXILIARY BUILDING DID NOT MEET UNDERWRITERS LABORATORY (UL) OR FACTORY MUTUAL STANDARDS FOR A FLAME SPREAD OF LESS THAN 25 FEET. LIMITING CONDITION FOR OPERATION (LCO) 3.7.12 WAS ENTERED ON BOTH UNITS AND APPROPRIATE FIRE WATCHES ESTABLISHED. THE EXPANSION JOINTS BETWEEN THE AUXILIARY BUILDING AND SHIELD BUILDINGS AT ELEVATIONS 669.0, 690.0, 706.0, 714.0, 734.0, AND 759.0 CONTAIN A MATERIAL FOR WHICH SPECIFIC DOCUMENTATION SUPPORTING FIRE RESISTIVENESS TO AN ACCEPTED UL OR FACTORY MUTUAL STANDARD IS NOT AVAILABLE AND THEREFORE, THE CONFIGURATION CANNOT BE CONSIDERED A CREDITED FIRE BARRIER. INVESTIGATION DETERMINED THAT THESE SEALS HAVE BEEN IN PLACE SINCE INITIAL CONSTRUCTION, THE INITIAL DESIGN DID NOT CONSIDER THESE SEALS TO BE A FIRE BARRIER; AND ACCORDINGLY, THEY WERE NOT INCLUDED IN THE SURVEILLANCE AND SUBSEQUENT APPENDIX R ANALYSIS. A PRELIMINARY TEST OF THE FIRE RESISTANCE OF THE EXPANSION POINT MATERIAL PROVIDED REASONABLE ASSURANCE THAT THE MATERIAL IS CERTIFIABLE AS A FIRE BARRIER AND QUALIFICATION TESTING IS BEING PURSUED.

[157] SEQUOYAH 1 DOCKET 50-327 LER 91-011
LIMITING CONDITION FOR OPERATION 3.0.3 WAS ENTERED WHEN THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM WAS DECLARED INOPERABLE BECAUSE A PRESSURE BOUNDARY DOOR WAS IMPROPERLY BREACHED OPEN.
EVENT DATE: 060291 REPORT DATE: 070591 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 222427) ON 6/2/91, AT 1705 EASTERN DAYLIGHT TIME (EDT) WITH UNITS 1 AND 2 OPERATING IN MODE 1 AT 100% POWER, LIMITING CONDITION FOR OPERATION (LCO) 3.0.3 WAS ENTERED WHEN THE CONTROL ROOM EMERGENCY VENTILATION SYSTEMS (CREVS) WAS DETERMINED TO BE INOPERABLE. THE CREVS WAS DETERMINED TO BE INOPERABLE BECAUSE A VENTILATION BOUNDARY DOOR WAS IMPROPERLY BREACHED. MODIFICATIONS GROUP WAS IMPLEMENTING A DESIGN CHANGE INVOLVING AN ASBESTOS ABATEMENT IN THE RELAY ROOM ON ELEVATION 732 OF THE CONTROL BUILDING AND HAD BREACHED A BOUNDARY DOOR TO FACILITATE TRAFFIC AND REMOVAL OF ASBESTOS. THE BREACH WAS IMPROPERLY CONTROLLED AS A RESULT OF INAPPROPRIATE PERSONNEL ACTIONS. AFTER DETERMINING THE BREACH WAS INAPPROPRIATE, THE STRUCTURE WAS DISMANTLED IN AN ORDERLY MANNER. LCO 3.0.3 WAS EXITED AT 1735 EDT WHEN THE STRUCTURE WAS REMOVED AND THE DOOR CLOSED. CORRECTIVE ACTIONS INCLUDE ADDITIONAL TRAINING, PROCEDURE AND PROCESS REVISIONS, AND COMMUNICATION OF EXPECTATIONS REGARDING PERFORMANCE.

[158] SHEARON HARRIS 1 DOCKET 50-400 LER 91-011
HIGH BORON CONCENTRATION IN REFUELING WATER STORAGE.
EVENT DATE: 051291 REPORT DATE: 061091 NSSS: WE TYPE: PWR

(NSIC 222387) TECHNICAL SPECIFICATION 3.1.2.1 REQUIRES ONE BORATION FLOWPATH BE AVAILABLE WHILE IN MODE 5 AND 6. IT WAS IDENTIFIED THAT THE REQUIRED BORATION FLOWPATH WAS FUNCTIONAL BUT NOT OPERABLE FOR SIX DAYS BECAUSE REFUELING WATER STORAGE TANK BORON CONCENTRATION WAS 2217 PPM. THIS EXCEEDED THE REQUIRED BORON CONCENTRATION RANGE OF 2000 - 2200 PPM. DURING THIS TIME A FLOWPATH FROM THE BORIC ACID TANK WAS AVAILABLE, IF REQUIRED. ALTHOUGH THIS FLOWPATH WAS AVAILABLE, THE SURVEILLANCE ON THIS FLOWPATH WAS NOT CURRENT, THEREFORE, THIS FLOWPATH COULD NOT BE CONSIDERED OPERABLE. THIS EVENT IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(I)(B) AS A TECHNICAL SPECIFICATION VIOLATION.

[159] SHEARON HARRIS 1 DOCKET 50-400 LER 91-012
UNPLANNED ACTUATION OF AUXILIARY FEEDWATER DUE TO FAILURE OF "A" MAIN FEED PUMP TO START.
EVENT DATE: 051591 REPORT DATE: 061491 NSSS: WE TYPE: PWR

(NSIC 222342) ON 5/15/91, DURING PLANT HEAT-UP, AN ATTEMPT TO START THE "A" MAIN FEEDWATER PUMP (MFP) WAS MADE TO PROVIDE MAKE-UP TO THE STEAM GENERATOR. THIS PUMP START WAS UNSUCCESSFUL. AT THE SAME TIME, THE "B" MFP WAS UNDER CLEARANCE AND THE "LOSS OF BOTH MFP'S" LOGIC WAS SATISFIED TO ACTUATE AN AUTOMATIC AUXILIARY FEEDWATER (AFW) PUMP START SIGNAL. THE "A" AFW PUMP STARTED AS REQUIRED. CONTROL ROOM PERSONNEL PROMPTLY SECURED THE "A" AFW PUMP TO STABILIZE STEAM GENERATOR LEVEL. INVESTIGATION WAS UNABLE TO DETERMINE THE CAUSE OF THE "A" MFP'S FAILURE TO START. THIS IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73 (A)(2)(IV) AS AN UNPLANNED ACTUATION OF AN ENGINEERED SAFETY FEATURE COMPONENT.

[160] SHEARON HARRIS 1 DOCKET 50-400 LER 91-013
REACTOR TRIP DUE TO SPURIOUS OVERTEMPERATURE DELTA TEMPERATURE SIGNAL.
EVENT DATE: 051591 REPORT DATE: 061491 NSSS: WE TYPE: PWR

(NSIC 222343) IN PREPARATION FOR COOLDOWN TO REPAIR A STEAM LEAK, CONTROL ROD SHUTDOWN BANK "C" WAS WITHDRAWN FULLY AND ALL OTHER CONTROL RODS WERE WITHDRAWN SIX STEPS. NUCLEAR INSTRUMENT N-43 WAS IN THE TRIPPED CONDITION FOR INSTALLATION OF MONITORING INSTRUMENTATION DURING POST-REFUELING START-UP TESTING. WHEN SURVEILLANCE TEST INSTRUMENTS WERE DISCONNECTED FOLLOWING REACTOR COOLANT SYSTEM RESISTANCE TEMPERATURE DETECTOR TESTING, A SPURIOUS OVERTEMPERATURE DELTA TEMPERATURE SIGNAL WAS GENERATED CAUSING A REACTOR TRIP. THE POTENTIAL FOR A REACTOR TRIP SIGNAL DURING RESTORATION FROM THE SURVEILLANCE TEST HAD BEEN PREVIOUSLY DISCUSSED BY OPERATIONS AND MAINTENANCE PERSONNEL, BUT THE INFORMATION

WAS NOT COMMUNICATED TO THE SHIFT OPERATORS. THE PROCEDURE GOVERNING THE RTD SURVEILLANCE TEST PROVIDED PRECAUTIONS DURING DATA ACQUISITION, BUT FAILED TO IDENTIFY THE RISK OF GENERATING A TRIP SIGNAL DURING RESTORATION. THIS EVENT WILL BE DISCUSSED WITH MAINTENANCE AND OPERATIONS PERSONNEL AND THE SURVEILLANCE TEST PROCEDURE WILL BE REVISED TO INCLUDE APPROPRIATE PRECAUTIONS DURING RESTORATION.

[161] SHEARON HARRIS 1 DOCKET 50-400 LER 91-015
AFW ACTUATION DUE TO IMPROPERLY SET FEED FLOW TRANSMITTER.
EVENT DATE: 051991 REPORT DATE: 061891 NSSS: WE TYPE: PWR
VENDOR: ITT-BARTON

(NSIC 222324) ON 5/19/91, A SIGNAL PROVIDED BY A FEED FLOW TRANSMITTER CAUSED THE TRIP OF THE ONLY RUNNING MAIN FEED WATER PUMP. THIS CONDITION SATISFIED THE LOGIC FOR AN AUTOMATIC ACTUATION OF THE AUXILIARY FEED WATER SYSTEM. THE CONTROL ROOM STAFF IMMEDIATELY VERIFIED PROPER SYSTEM RESPONSE BY ENSURING THAT BOTH MOTOR DRIVEN PUMPS HAD STARTED AS REQUIRED. THIS EVENT WAS CAUSED BY AN IMPROPERLY SET SIGNAL DAMPENING DEVICE IN THE MAIN FEED WATER PUMP SUCTION FLOW TRANSMITTER. TROUBLE SHOOTING AND SUBSEQUENT ADJUSTMENTS TO THIS DEVICE RESULTED IN SUCCESSFUL TESTING AND OPERATION OF THE FLOW TRANSMITTER AND SHOULD PREVENT RECURRENCE OF THIS CONDITION.

[162] SHEARON HARRIS 1 DOCKET 50-400 LER 91-009
AFW ACTUATION DUE TO STEAM GENERATOR HI-HI LEVEL ALARM THAT WAS INADVERTENTLY GENERATED DURING TROUBLESHOOTING.
EVENT DATE: 052191 REPORT DATE: 062091 NSSS: WE TYPE: PWR

(NSIC 222361) ON 5/21/91, TROUBLESHOOTING OF THE STEAM GENERATOR (SG) LEVEL INSTRUMENTATION RESULTED IN THE GENERATION OF A HI-HI SG LEVEL SIGNAL. THIS CAUSED A FEEDWATER ISOLATION SIGNAL WHICH BY DESIGN TRIPPED THE RUNNING MAIN FEED PUMP AND STARTED BOTH MOTOR DRIVEN AUXILIARY FEED WATER PUMPS. THE CONTROL ROOM STAFF IMMEDIATELY TOOK ACTIONS TO VERIFY PROPER SYSTEM RESPONSE AND STABILIZE SG LEVELS. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR ON THE PART OF THE MAINTENANCE PERSONNEL PERFORMING THE TROUBLESHOOTING. 222361TIVE ACTIONS WILL CONSIST OF COUNSELING THE INVOLVED TECHNICIANS, REVIEWING THE INCIDENT WITH OTHER APPROPRIATE PERSONNEL AND DEVELOPING A PROCEDURE THAT PROVIDES SPECIFIC GUIDANCE FOR TROUBLESHOOTING ACTIVITIES. THIS CONDITION IS BEING REPORTED AS AN UNPLANNED ENGINEERED SAFETY FEATURE ACTUATION IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV) .

[163] SHEARON HARRIS 1 DOCKET 50-400 LER 91-010
REACTOR TRIP DURING SURVEILLANCE TESTING AND ONE REACTOR TRIP BREAKER FAILED TO OPEN.
EVENT DATE: 060391 REPORT DATE: 070391 NSSS: WE TYPE: PWR

(NSIC 222503) THE PLANT RECEIVED AN AUTOMATIC REACTOR TRIP ON RCS LOW FLOW DURING THE PERFORMANCE OF A MAINTENANCE CALIBRATION PROCEDURE ON AN RCS FLOW TRANSMITTER. ALL RODS FULLY INSERTED ON THE TRIP SIGNAL. THE "A" REACTOR TRIP BREAKER DID NOT OPEN ON THE AUTOMATIC REACTOR TRIP SIGNAL BUT DID OPEN ON A SUBSEQUENT MANUAL TRIP SIGNAL . THE FAILURE OF THE "A" REACTOR TRIP BREAKER TO OPEN WAS DUE TO A FAILED UNDERVOLTAGE OUTPUT DRIVER CARD IN THE "A" SOLID STATE PROTECTION SYSTEM. THE FAILURE OF THE UNDERVOLTAGE OUTPUT DRIVER CARD WAS THE SAME AS DESCRIBED IN IE NOTICE 85-13 AND HAD APPARENTLY OCCURRED DURING MAINTENANCE ON MAY 18, 1991. THE FAILED UNDERVOLTAGE OUTPUT DRIVER CARD WAS REPLACED WITH A MODIFIED CARD WHICH PREVENTS THIS FAILURE MECHANISM. THIS EVENT IS BEING REPORTED AS A TECHNICAL SPECIFICATION VIOLATION AND ENGINEERED SAFETY FEATURES ACTUATION PER 10CFR50.73(A)(2)(I)(B) AND 10CFR50.73(A)(2)(IV).

[164] SOUTH TEXAS 1 DOCKET 50-498 LER 91-016
FAILURE TO PERFORM A SURVEILLANCE TEST DUE TO PERSONNEL ERROR.
EVENT DATE: 051391 REPORT DATE: 061191 NSSS: WE TYPE: PWR

(NSIC 222192) ON MAY 13, 1991, AT APPROXIMATELY 2230 HOURS, UNIT 1 WAS IN MODE 1

AT 100 PERCENT POWER. IT WAS DISCOVERED THAT THE TECHNICAL SPECIFICATION 3/4.7.1.4 REQUIREMENTS FOR DETERMINING THE SPECIFIC ACTIVITY OF THE SECONDARY COOLANT SYSTEM HAD NOT BEEN PERFORMED WITHIN THE REQUIRED SURVEILLANCE INTERVAL. THIS IS A VIOLATION OF TECHNICAL SPECIFICATION 3/4.7.1.4 AND IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B). STEAM GENERATOR BLOWDOWN RADIATION MONITOR DATA WAS CHECKED, AND IT WAS VERIFIED THAT SECONDARY ACTIVITY HAD NOT EXCEEDED NORMAL VALUES OR THE TECHNICAL SPECIFICATION LIMIT DURING THIS PERIOD. THE CAUSE OF THIS EVENT WAS FAILURE TO ENSURE TESTING WAS PERFORMED BEFORE EXCEEDING THE SURVEILLANCE INTERVAL. CORRECTIVE ACTIONS INCLUDED ISSUANCE OF SPECIAL ORDERS, AND CHANGING PROCEDURE AND LABORATORY SCHEDULES TO IMPROVE VISIBILITY AND INCREASE AWARENESS OF SURVEILLANCE TIMES.

[165] SOUTH TEXAS 2 DOCKET 50-499 LER 91-006
CONTROL ROOM VENTILATION ACTUATIONS OF RECIRCULATION MODE DUE TO TWO SPURIOUS SIGNALS FROM A TOXIC GAS ANALYZER.
EVENT DATE: 051691 REPORT DATE: 061491 NSSS: WE TYPE: PWR
VENDOR: FOXBORO CO., THE

(NSIC 222193) ON MAY 16, 1991 UNIT 2 WAS IN MODE 1 AT 100 PERCENT POWER. AT 0558 HOURS, THE CONTROL ROOM VENTILATION SYSTEM ACTUATED TO THE RECIRCULATION MODE AS A RESULT OF A SPURIOUS TRIP FROM A TOXIC GAS ANALYZER. THE SPURIOUS ACTUATION SIGNAL SELF-RESET AT 0559 HOURS. ALSO AT 0042 HOURS ON MAY 21, 1991 ANOTHER SIMILAR ACTUATION OCCURRED FROM THE SAME ANALYZER AS THE FIRST EVENT. THE EXACT CAUSE OF BOTH EVENTS COULD NOT BE DETERMINED BUT HAS BEEN ATTRIBUTED TO POOR ELECTRICAL CONNECTION ON ONE OR MORE PLUG-IN INTEGRATED CIRCUIT CHIPS IN THE ANALYZER. CORRECTIVE ACTIONS INCLUDE TROUBLESHOOTING OF THE FAILED ANALYZER, FURTHER DESIGN IMPROVEMENTS TO MINIMIZE FALSE ACTUATION SIGNALS, AND DEVELOPMENT OF PREVENTIVE MAINTENANCE TASKS TO PERIODICALLY RESEAT INTEGRATED CIRCUIT CHIPS IN THE TOXIC GAS ANALYZERS.

[166] SOUTH TEXAS 2 DOCKET 50-499 LER 91-007
REACTOR TRIP CAUSED BY INADVERTENT ACTUATION OF GENERATOR BREAKER EMERGENCY TRIP.
EVENT DATE: 052291 REPORT DATE: 062191 NSSS: WE TYPE: PWR
VENDOR: PAUL-MUNROE HYDAULICS INC.

(NSIC 222332) ON MAY 22, 1991, UNIT 2 WAS IN MODE 1 AT 100% POWER. AT APPROXIMATELY 2220 HRS, WHILE WAITING IN THE AREA OF THE MAIN GENERATOR BREAKER TO UNLOCK A LOCAL CABINET FOR AN ELECTRICAL MAINTENANCE INDIVIDUAL, A NON-LICENSED OPERATOR INADVERTENTLY ACTUATED THE LOCAL GENERATOR BREAKER EMERGENCY TRIP PUSHBUTTON. THE SUDDEN LOSS OF SECONDARY LOAD CAUSED AN AUTOMATIC OVER TEMPERATURE DELTA TEMPERATURE REACTOR TRIP. FOLLOWING THE REACTOR TRIP, PRESSURIZER SPRAY WAS UNABLE TO REDUCE THE PRESSURE BEFORE THE PRESSURIZER PORVS OPENED AT APPROXIMATELY 2335 PSIG BECAUSE ONE OF THE SPRAY VALVES WAS IN MANUAL CONTROL. STEAM GENERATOR 2C POWER-OPERATED RELIEF VALVE (PORV) FAILED TO OPEN EVEN THOUGH THE PRESSURE EXCEEDED THE LIFT SETPOINT. THE NON-LICENSED OPERATOR RESPONSIBLE FOR THE TRIP WAS COUNSELLED WITH REGARDS TO PAYING STRICT ATTENTION TO PERFORMANCE OF OPERATIONS ACTIVITIES. THE STEAM GENERATOR 2C PORV HAS BEEN REPAIRED. OTHER SWITCH DESIGNS WILL BE REVIEWED TO IDENTIFY CHANGES THAT CAN PREVENT SIMILAR INADVERTENT ACTUATIONS.

[167] SOUTH TEXAS 2 DOCKET 50-499 LER 91-009
TWO CONTAINMENT VENTILATION ISOLATION ACTUATION DUE TO A FAILURE IN THE RADIATION MONITORING SYSTEMS.
EVENT DATE: 052591 REPORT DATE: 062491 NSSS: WE TYPE: PWR
VENDOR: GA ELECTRONIC SYSTEMS DIV

(NSIC 222392) ON MAY 25, 1991, UNIT 2 WAS IN MODE 3 AT 2235 PSIG AND 567 DEGREES. AT 0107 A CONTAINMENT VENTILATION ISOLATION (CVI) ACTUATION OCCURRED. ON MAY 26, UNIT 2 WAS IN MODE 1 AT 75% POWER WHEN AT 0558 A SECOND CVI ACTUATION OCCURRED. TROUBLESHOOTING FOLLOWING THE ACTUATIONS INDICATED THAT A FAULTY RM-23 MODULE ASSOCIATED WITH ONE OF THE TWO PURGE EXHAUST RADIATION MONITORS (RT-8012) CAUSED THE TWO SPURIOUS ACTUATIONS. THE FAULTY MODULE HAS BEEN REPLACED. AN ANALYSIS IS BEING PERFORMED TO DETERMINE THE FAILURE MODE.

[168] ST. LUCIE 1 DOCKET 50-335 LER 91-004
LATE TECHNICAL SPECIFICATION REQUIRED FIRE WATCH PATROL DUE TO PERSONNEL ERROR.
EVENT DATE: 051691 REPORT DATE: 061491 NSSS: CE TYPE: PWR

(NSIC 222340) ON 5/16/91, WHILE OPERATING AT 100% POWER, UNIT 1 EXPERIENCED A LATE FIRE WATCH PATROL OF THE SHIELD BUILDING ANNULUS AREA. DUE TO MODIFICATIONS BEING PERFORMED TO THE UNIT 1 FIRE DETECTION SYSTEM, THE "B" SIDE FIRE DETECTION SYSTEM WAS OUT OF SERVICE, FOR WHICH TECH SPEC 3.3.3.7 REQUIRES AN 8 HOUR FIRE WATCH INSPECTION OF THE SHIELD BUILDING ANNULUS. A CONCURRENT 10 HOUR SURVEILLANCE RUN OF A SHIELD BUILDING EXHAUST FAN CREATED A DIFFERENTIAL PRESSURE ACROSS THE ANNULUS DOOR THAT MADE IT IMPOSSIBLE TO OPEN AND GAIN ACCESS TO PERFORM THE INSPECTION. UPON NOTIFICATION OF THIS CONDITION, THE CONTROL ROOM SENIOR REACTOR OPERATOR ASSUMED THAT THE FIRE WATCH WAS REQUIRED SHIFTLY RATHER THAN EVERY 8 HOURS, AND CONTINUED THE EXHAUST FAN SURVEILLANCE. WHEN THE SITE FIRE PROTECTION DEPARTMENT LATER NOTIFIED THE CONTROL ROOM THAT THE FIRE WATCH WAS LATE, OPERATORS SECURED THE 6B SHIELD BUILDING EXHAUST FAN TO ALLOW THE FIRE WATCH TO BE PERFORMED. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THE SURVEILLANCE RUN OF THE SHIELD BUILDING EXHAUST FAN INTERFERED WITH THE SCHEDULED FIRE WATCH, RESULTING IN THE FIRE WATCH TIME INTERVAL BEING EXCEEDED. CORRECTIVE ACTIONS FOR THIS EVENT WERE TO SECURE THE SHIELD BUILDING EXHAUST FAN TO ALLOW ACCESS TO THE ANNULUS, PERFORM THE REQUIRED FIRE PATROL, AND DEFER THE EXHAUST FAN SURVEILLANCE UNTIL THE B SIDE FIRE DETECTION SYSTEM WAS PLACED BACK IN SERVICE.

[169] ST. LUCIE 2 DOCKET 50-389 LER 91-004
TWO CONTAINMENT FAN COOLERS INOPERABLE BASED ON LOW COMPONENT COOLING WATER FLOW DUE TO A DESIGN ERROR CAUSING LOW FLOW ALARM LIMIT LESS THAN TECHNICAL SPECIFICATIONS.
EVENT DATE: 050391 REPORT DATE: 052491 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)
VENDOR: PRATT, HENRY COMPANY

(NSIC 222138) ON 4/26/91, WITH UNIT 2 AT 100% POWER, A NON-LICENSED UTILITY OPERATOR PHYSICALLY VERIFIED THE LOCKED THROTTLED POSITION OF SB-14530, THE COMMON HEADER ISOLATION VALVE FOR COMPONENT COOLING WATER FROM THE 2A AND 2B CONTAINMENT FAN COOLERS, IN ACCORDANCE WITH ADMINISTRATIVE PROCEDURE #2-0010125A, DATA SHEET #36, WEEKLY VALVE STATUS CHECK. A FUNCTIONAL CHECK OF THE SYSTEM FLOWS WAS NOT CONDUCTED AFTER PERFORMANCE OF DATA SHEET #36, AND IS SURMISED SLIGHTLY LESS THAN NORMAL CCW FLOW TO THE CONTAINMENT FAN COOLERS WAS THE RESULTANT EFFECT. AT 0800 ON 5/3/91, WITH UNIT 2 AT 100% POWER, A NON-LICENSED UTILITY OPERATOR DISCOVERED THE COMPONENT COOLING WATER FLOWS FROM THE 2A AND 2B CONTAINMENT FAN COOLERS WERE READING APPROX. 1170 GPM, BELOW THE TECH SPEC REQUIRED MINIMUM OF 1200 GPM. CCW RETURN FROM THE 2A AND 2B CONTAINMENT FAN COOLERS, VALVE SB-14530, WAS FOUND TO BE IN THE LOCKED THROTTLED POSITION, AS REQUIRED. PROPER FLOWS WERE THEN RE-ESTABLISHED ON THE 2A AND 2B CONTAINMENT FAN COOLERS AND THEY WERE DECLARED OPERABLE AGAIN. CAUSE OF THIS EVENT WAS A DESIGN ERROR IN THAT THE CCW LOW FLOW TO EACH CONTAINMENT FAN COOLER ALARM LIMITS WERE FOUND TO BE SET NON-CONSERVATIVELY WITH RESPECT TO THE TECH SPECS.

[170] SURRY 1 DOCKET 50-280 LER 91-009
TECHNICAL SPECIFICATIONS SURVEILLANCE REQUIREMENTS VIOLATED FOR IN-SERVICE INSPECTION OF REACTOR VESSEL DUE TO A COGNITIVE PERSONNEL ERROR.
EVENT DATE: 050991 REPORT DATE: 060591 NSSS: WE TYPE: PWR

(NSIC 222204) ON MAY 9, 1991, WITH UNIT 1 AT 100% POWER AND UNIT 2 IN COLD SHUTDOWN, A VIOLATION OF THE TECHNICAL SPECIFICATIONS (TS) SURVEILLANCE REQUIREMENTS FOR THE UNIT 1 IN-SERVICE INSPECTION (ISI) AND TESTING OF AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) CODE CLASS 1, 2, AND 3 COMPONENTS WAS IDENTIFIED DURING AN INTERNAL AUDIT OF THE UNIT 2 ISI PROGRAM. THE VISUAL (VT-2) EXAMINATIONS OF THE UNIT 1 REACTOR VESSEL PARTIAL PENETRATION WELDS AND THE BOTTOM OF THE REACTOR VESSEL WERE NOT PERFORMED DURING THE 1990 REFUELING OUTAGE. THE CAUSE OF THIS EVENT WAS AN ADMINISTRATIVE OVERSIGHT ATTRIBUTED TO A COGNITIVE ERROR ON THE PART OF UTILITY ENGINEERING PERSONNEL RESPONSIBLE FOR THE ISI PROGRAM. THE SUBJECT EXAMINATIONS WILL BE PERFORMED AT THE NEXT OUTAGE OF

SUFFICIENT DURATION. LEAKAGE MONITORING OF THE REACTOR COOLANT SYSTEM HAS BEEN ENHANCED AND THE NORMAL WEEKLY FREQUENCY OF UNIT 1 CONTAINMENT AIR SAMPLING AND ANALYSIS HAS BEEN INCREASED TO DAILY UNTIL THE EXAMINATIONS ARE PERFORMED. THE ISI PLAN WILL BE REVISED TO INCLUDE THE SUBJECT EXAMINATIONS. THE PROCEDURE GOVERNING THE RCS PRESSURE TEST WILL ALSO BE REVISED TO CLARIFY THE SUBJECT SECTION XI ASME BOILER AND PRESSURE VESSEL CODE REQUIRED EXAMINATIONS. THE EVENT IS BEING REPORTED, PURSUANT TO 10CFR 50.73(A)(2)(I)(B), SINCE THIS CONDITION IS PROHIBITED BY TS 4.0.3.A.

[171] SURRY 1 DOCKET 50-280 LER 91-010
INSERVICE TESTING INSTRUMENTATION UNCERTAINTY EXCEEDED +/-2% DUE TO PERSONNEL ERROR.
EVENT DATE: 053091 REPORT DATE: 070191 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 222411) ON MAY 30, 1991, WITH UNIT 1 AT 100% POWER AND UNIT 2 AT COLD SHUTDOWN, IT WAS DISCOVERED THAT FLOW INSTRUMENTATION USED FOR TESTING PUMPS IN ACCORDANCE WITH SURRY UNITS 1 AND 2 ASME SECTION XL INSERVICE TESTING PROGRAMS MAY NOT HAVE INSTRUMENT ACCURACIES WITHIN THE CODE REQUIRED +/-2%. FOLLOWING THIS DISCOVERY, AN EVALUATION OF THE ACCURACY OF INSTRUMENTS USED FOR INSERVICE TESTING WAS PERFORMED WHICH CONCLUDED THAT ACCURACIES OF CERTAIN FLOW INSTRUMENTS RANGED FROM 2.5% TO 3.19%, THUS EXCEEDING CODE REQUIREMENTS. NO SIGNIFICANT SAFETY IMPLICATIONS WERE POSED BY THIS EVENT BECAUSE, AFTER ADJUSTING FOR THE DECREASED INSTRUMENT ACCURACIES, PUMPS WHICH WERE PREVIOUSLY CONSIDERED OPERABLE REMAINED OPERABLE. THIS EVENT OCCURRED AS THE RESULT OF A COGNITIVE PERSONNEL ERROR. A UTILITY PROJECT ENGINEER FAILED TO ENSURE ACTIONS NECESSARY TO IMPLEMENT CALIBRATION REQUIREMENTS IDENTIFIED DURING THE DEVELOPMENT OF DESIGN CHANGE PACKAGES WERE EFFECTIVELY COMMUNICATED TO THE ORGANIZATIONS RESPONSIBLE FOR IMPLEMENTING THE REQUIREMENTS. THIS OCCURRENCE WAS CONSIDERED TO BE A VIOLATION OF TECHNICAL SPECIFICATION 4.0.3 AND THEREFORE IS BEING REPORTED PURSUANT TO 10CFR50.73(A)(2)(I)(B) .

[172] SURRY 1 DOCKET 50-280 LER 91-011
SERVICE WATER RADIATION MONITORING PUMPS INOPERABLE DUE TO AIR BINDING AS A RESULT OF INADEQUATE DESIGN.
EVENT DATE: 053091 REPORT DATE: 062791 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 222412) ON APRIL 6, 1991, WITH UNIT 1 AT 100% POWER AND UNIT 2 IN COLD SHUTDOWN, A FUNCTIONAL TEST OF THE UNIT 2 SERVICE WATER (SW) RADIATION MONITORING PUMPS UNDER SIMULATED DESIGN BASIS ACCIDENT CONDITIONS RESULTED IN A FAILURE OF THE PUMPS TO DEVELOP DESIGN FLOW AND DISCHARGE PRESSURE. THIS CONDITION IMPAIRS THE CAPABILITY OF THE RADIATION MONITORS TO DETECT RADIOACTIVE LEAKAGE INTO THE SW SYSTEM FROM THE RECIRCULATION SPRAY HEAT EXCHANGERS; HOWEVER, ALTERNATE MEANS OF LEAKAGE DETECTION WOULD HAVE BEEN AVAILABLE IF A DESIGN BASIS ACCIDENT HAD OCCURRED. THIS CONDITION WAS CAUSED BY A TRANSIENT VACUUM CONDITION AT THE PUMP SUCTION WHICH RESULTED IN AIR BINDING OF THE PUMPS. THE TRANSIENT VACUUM CONDITION IN THE CIRCULATING WATER (CW) DISCHARGE TUNNEL RESULTED FROM ISOLATION OF THE MAIN CONDENSER AND PREVENTED NORMAL SELF-PRIMING OF THE RADIATION MONITOR PUMPS. THE AUTOMATIC STARTING OF THESE PUMPS HAS BEEN DEFEATED. EMERGENCY OPERATING PROCEDURES HAVE BEEN REVISED TO REQUIRE MANUALLY BREAKING CW DISCHARGE TUNNEL VACUUM TO PERMIT NORMAL SELF-PRIMING OF THESE PUMPS AND TO PROVIDE FOR MANUALLY STARTING THE PUMPS . PERMANENT MODIFICATION OF THE RADIATION MONITORING SYSTEM IS BEING EVALUATED. THIS REPORT IS REQUIRED BY 10CFR50.73(A)(2)(VII)(C).

[173] SURRY 2 DOCKET 50-281 LER 91-004
INADVERTENT OVERFILLING OF REFUELING WATER STORAGE TANK.
EVENT DATE: 051491 REPORT DATE: 061391 NSSS: WE TYPE: PWR

(NSIC 222315) ON MAY 14, 1991, WITH UNIT I AT 100% POWER AND UNIT 2 AT COLD SHUTDOWN, DURING A TECHNICAL REVIEW OF AN INADVERTENT OVERFILLING OF THE UNIT 2 REFUELING WATER STORAGE TANK (RWST), THE AS-BUILT CONFIGURATIONS OF BOTH UNITS' RWSTS WERE REEVALUATED. THE REEVALUATION DETERMINED THAT THE POSITIONING OF THE

UNIT 2 RWST OVERFLOW LINE WOULD PERMIT THE TANK TO BE FILLED IN EXCESS OF THE TECHNICAL SPECIFICATION CAPACITY LIMIT. THROUGH CONSULTATION WITH THE ARCHITECT/ENGINEER, IT WAS DETERMINED THAT NO STRUCTURAL LIMITS WERE EXCEEDED. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B).

[174] SURRY 2 DOCKET 50-281 LER 91-005
REACTOR COOLANT SYSTEM LEAKAGE EXCEEDED TECHNICAL SPECIFICATION LIMITS DUE TO THE MECHANICAL FAILURE OF ISOLATION VALVE.
EVENT DATE: 060291 REPORT DATE: 062791 NSSS: WE TYPE: PWR
VENDOR: VELAN VALVE CORP.

(NSIC 222813) ON JUNE 2, 1991, AT 1431 HOURS, WITH UNIT 1 AT 100% POWER AND UNIT 2 AT HOT SHUTDOWN, LEAKAGE FROM THE UNIT 2 REACTOR COOLANT SYSTEM (RCS) EXCEEDED THAT ALLOWED BY TECHNICAL SPECIFICATIONS (TS). ON JUNE 2, 1991, AT 0659 HOURS, THE THREE INCH DIAMETER RCS LOOP "C" RESISTANCE TEMPERATURE DEVICE MANIFOLD ISOLATION VALVE (2-RC-95) WAS IDENTIFIED AS THE SOURCE. AT 1432 HOURS, A 30 HOUR LIMITING CONDITION FOR OPERATION TO COLD SHUTDOWN WAS ENTERED IN ACCORDANCE WITH TS 3.0.1 SINCE THE LEAKAGE EXCEEDED THE TS ALLOWED RCS TOTAL LEAKAGE CRITERIA (10 GALLONS PER M INUTE) AND A COOLDOWN FROM HOT SHUTDOWN TO COLD SHUTDOWN WAS INITIATED. A NOTIFICATION OF UNUSUAL EVENT WAS DECLARED ON JUNE 2, 1991, AT 1431 HOURS. THE EVENT WAS TERMINATED ON JUNE 3, 1991, AT 0506 HOURS, WHEN UNIT 2 WAS PLACED IN COLD SHUTDOWN. ON JUNE 3, 1991, AT APPROXIMATELY 1300 HOURS, VALVE 2-RC-95 WAS CLOSED AND LEAKAGE FROM THE VALVE WAS REDUCED TO 0.5 TO 1.0 GALLONS PER MINUTE. ON JUNE 4 , 1991, FOLLOWING AN ENGINEERING REVIEW, THE VALVE INTERNALS WERE REMOVED AND A BLANK CAP WAS INSTALLED ON THE VALVE BODY. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO A MECHANICAL FAILURE OF VALVE 2-RC-95. THE EVENT IS BEING REPORTED, PURSUANT TO 10CFR 50.73(A)(2)(I)(B), SINCE THIS CONDITION IS PROHIBITED BY TS 3.1.C.5.

[175] SUSQUEHANNA 2 DOCKET 50-388 LER 91-008
ESF ACTUATIONS DUE TO RPS EPA BREAKERS SPURIOUS TRIP.
EVENT DATE: 053091 REPORT DATE: 062791 NSSS: GE TYPE: BWR

(NSIC 222501) AT 0925 HOURS ON 5/30/91 WITH UNIT 2 OPERATING AT 100% POWER, THE PRIMARY POWER SUPPLY TO THE "A" REACTOR PROTECTION SYSTEM (RPS) POWER DISTRIBUTION PANEL WAS LOST WHEN ITS ELECTRICAL PROTECTION ASSEMBLY (EPA) BREAKERS TRIPPED. RPS AS WELL AS OTHER PLANT SYSTEMS AND COMPONENTS FUNCTIONED PROPERLY AND AS EXPECTED IN RESPONSE TO THE EVENT. NO REACTOR PARAMETERS WERE AFFECTED AND NO EMERGENCY CORE COOLING SYSTEMS WERE ACTUATED. THE 'A' DISTRIBUTION PANEL WAS SWAPPED TO ALTERNATE POWER UNTIL THE PRIMARY POWER SUPPLY WAS RESTORED. THERE WAS NO INDICATION OF ABNORMALITIES AND ALL ISOLATION SIGNALS WERE RESET BY 1015 HOURS. THE PRIMARY POWER SOURCE EPA BREAKERS WERE RESET AT 2150 HOURS ON 6/1/91. FULL POWER OPERATION OF THE UNIT CONTINUED WITHOUT INTERRUPTION. WHILE THE EXACT CAUSE OF THE TRIP WAS NOT DEFINITIVELY DETERMINED, THE MOST PROBABLE CAUSE FOR THE TRIP WAS ATTRIBUTED TO OSCILLATOR OUTPUT SHIFT ON THE EPA LOGIC CARD. A TASK TEAM HAS BEEN ORGANIZED TO ENHANCE THE RELIABILITY OF THE REACTOR PROTECTION SYSTEM. IN ADDITION, AN EPA STUDY PREPARED BY GE - NUCLEAR ENERGY FOR THE BWR OWNERS GROUP IS ALSO UNDER PP&L REVIEW TO ASSESS ANY OPERATIONAL OR DESIGN CHANGES THAT MAY BE WARRANTED. LER 90-007 (DOCKET NO. 50-388/LICENSE NO. NPF-22) WILL BE UPDATED TO PROVIDE THE RESULTS OF THE EFFORTS TO ENHANCE RPS RELIABILITY.

[176] THREE MILE ISLAND 1 DOCKET 50-289 LER 89-002 REV 01
UPDATE ON POTENTIAL FOR INOPERABILITY OF BOTH EMERGENCY DIESELS DUE TO SLUDGE FORMATION.
EVENT DATE: 111489 REPORT DATE: 062191 NSSS: BW TYPE: PWR
VENDOR: FAIRBANKS MORSE

(NSIC 222372) AT APPROX. 1230 HOURS ON 11/14/89, DURING PERFORMANCE OF THE EMERGENCY DIESEL GENERATOR (EDG) 1B ANNUAL INSPECTION, WHICH COMMENCED ON 11/13/89, IT WAS DISCOVERED THAT THE RADIATOR FAN RIGHT ANGLE GEAR DRIVE BEARING LUBRICATION WAS INADEQUATE DUE TO FORMATION OF SLUDGE IN THE LUBRICATION LINES.

A SIMILAR CONDITION RESULTED IN THE UPPER BEARING FAILURE OF THE EDG 1A RADIATOR FAN RIGHT ANGLE DRIVE BEARING AT APPROX. 0514 HOURS ON 11/2/89. ALTHOUGH BOTH EDGS WERE NOT INOPERABLE AT THE SAME TIME, THE POTENTIAL EXISTED FOR SIMULTANEOUS FAILURE OF BOTH EDGS. THUS, THIS EVENT IS REPORTABLE PURSUANT TO 10 CFR50.73(A)(2)(V). EDG 1B WAS RESTORED TO OPERABLE STATUS AT APPROX. 0844 HOURS ON 11/18/89 WITHIN THE 7 DAY TIMECLOCK PERMITTED BY TECH SPEC 3.7.2.C. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO A COMBINATION OF OVERHEATING OF THE LUBRICATING OIL AND INADEQUATE/INCOMPLETE PROCEDURES FOR THE INSPECTION AND MAINTENANCE OF THE EDG LUBRICATING SYSTEM. APPLICABLE PLANT PROCEDURES HAVE BEEN REVISED TO PROVIDE INCREASED MAINTENANCE ON THE EDG LUBRICATION SYSTEM AND TO DE-ENERGIZE THE IMMERSION HEATER BEFORE DRAINING THE OIL FROM THE FAN DRIVE GEAR BOX.

[177] TROJAN DOCKET 50-344 LER 91-009 REV 01
UPDATE ON ELECTRICAL COMPONENTS WHICH LACK SUFFICENT DOCUMENTATION TO DEMONSTRATE OPERABILITY UNDER LOW TEMPERATURE SERVICE CONDITIONS DUE TO DESIGN WEAKNESSES.
EVENT DATE: 031591 REPORT DATE: 062191 NSSS: WE TYPE: PWR
VENDOR: AGASTAT RELAY CO.
AUTOMATIC SWITCH COMPANY (ASCO)
GENERAL ELECTRIC CO.
GOULD INC.
MICRO SWITCH
NAMCO CONTROLS

(NSIC 222346) ON 3/15/91, WHILE IN MODE 5 (COLD SHUTDOWN) DURING A REVIEW TO ADDRESS EQUIPMENT OPERABILITY CONCERNS IDENTIFIED ON 3/7/91, IT WAS DETERMINED THAT ASCO CATALOG NO. NP-1 SOLENOID VALVES LACKED SUFFICIENT DOCUMENTATION TO DEMONSTRATE OPERABILITY FOR LOW-TEMPERATURE (-5F) SERVICE CONDITIONS. THE CAUSE OF THESE DEFICIENCILS HAS BEEN DETERMINED TO BE WEAKNESSES IN THE ESTABLISHMENT OF LOW-TEMPERATURE MINIMUM TEMPERATURE REQUIREMENTS IN THE TROJAN DESIGN BASIS, IN EQUIPMENT PROCUREMENT AND INSTALLATION SPECIFICATIONS AND A FAILURE TO DOCUMENT THESE WEAKNESSES IN THE CORRECTIVE ACTION PROCESSES IN USE AT TROJAN. THE ASSESSMENT OF THESE DEFICIENCIES, INCLUDING SAFETY SIGNIFICANCE DURING PREVIOUS POWER OPERATION, AND THE DETERMINATION OF THE SCOPE OF THESE DOCUMENTATION WEAKNESSES ARE COMPLETE. HOWEVER, OF THE OVER 5,500 COMPONENTS REVIEWED, ONLY 6 COMPONENT TYPES/LOCATIONS HAVE BEEN IDENTIFIED AS REQUIRING FURTHER SAFETY SIGNIFICANCE ASSESSMENT. THESE DEVICES WOULD HAVE MOST PROBABLY PERFORMED THEIR REQUIRED SAFETY FUNCTION(S), EITHER BECAUSE THE DEVICE WAS NOT SERIOUSLY AFFECTED BY COLD WEATHER OR ADMINISTRATIVE CONTROLS FOR COLD WEATHER WOULD HAVE ASSURED AN ADEQUATE ENVIRONMENT; OR SUFFICIENT TIME AND INSTRUCTIONS FOR OPERATOR ACTIONS EXISTED TO COMPENSATE FOR THE DEFICIENCY OR REDUNDANT EQUIPMENT WAS AVAILABLE.

[178] TROJAN DOCKET 50-344 LER 91-013
RAPID COOLDOWN OF THE PRESSURIZER DURING MODE 5 OPERATION DUE TO PROCEDURE WEAKNESS.
EVENT DATE: 032591 REPORT DATE: 062591 NSSS: WE TYPE: PWR

(NSIC 222493) ON MARCH 25, 1991, WHILE IN MODE 5, DURING BORATION OF THE REACTOR COOLANT SYSTEM (RCS) TO REFUELING BORON CONCENTRATION, PRESSURIZER LIQUID SPACE TEMPERATURE DECREASED FROM APPROXIMATELY 425F TO 145F IN LESS THAN ONE HOUR. THIS 280F COOLDOWN OF THE PRESSURIZER LIQUID EXCEEDED THE TROJAN TECHNICAL SPECIFICATION (TTS) SECTION 3.4.9.2.B, LIMITING CONDITION FOR OPERATION (LCO) OF 200F IN ANY ONE-HOUR PERIOD AND PLACED THE PLANT IN THE ASSOCIATED TTS ACTION STATEMENT. THE ACTION STATEMENT REQUIREMENT WAS ALREADY PARTIALLY SATISFIED BECAUSE THE PLANT WAS IN COLD SHUTDOWN, HOWEVER, PRESSURIZER LIQUID SPACE TEMPERATURE WAS NOT RESTORED WITHIN THE 200F IN ANY ONE-HOUR PERIOD LIMIT WITHIN 30 MINUTES. THEREFORE, THIS LICENSEE EVENT REPORT (LER) IS BEING SUBMITTED AS A CONDITION PROHIBITED BY THE PLANT TECHNICAL SPECIFICATIONS. AN ANALYSIS OF THIS EVENT HAS BEEN PERFORMED BY THE NUCLEAR STEAM SUPPLY SYSTEM (NSSS) VENDOR. THE INTEGRITY OF THE PRESSURIZER VESSEL WAS NOT COMPROMISED BY THIS EVENT AND AMPLE MARGIN EXISTS FOR FUTURE OPERATION; THEREFORE, THIS EVENT HAS MINIMAL SAFETY SIGNIFICANCE.

[179] TROJAN DOCKET 50-344 LER 91-019
SPECIAL REPORT NOT SUBMITTED WHEN A METEOROLOGICAL MONITORING INSTRUMENT WAS INOPERABLE FOR GREATER THAN SEVEN DAYS.
EVENT DATE: 040891 REPORT DATE: 061591 NSSS: WE TYPE: PWR

(NSIC 222348) TROJAN TECHNICAL SPECIFICATION 3.3.3.4 "METEOROLOGICAL INSTRUMENTATION" REQUIRES THE PREPARATION AND SUBMITTAL OF A SPECIAL REPORT IF EITHER OR BOTH OF THE WIND DIRECTION INSTRUMENTS IS INOPERABLE FOR MORE THAN SEVEN DAYS. IT HAS BEEN DETERMINED THAT THE CHANNEL "A" WIND DIRECTION SENSOR ON THE 33 FOOT METEOROLOGICAL TOWER WAS INOPERABLE FOR A PERIOD IN EXCESS OF SEVEN DAYS IN 1989, AND A SPECIAL REPORT WAS NOT SUBMITTED. FAILURE TO SUBMIT THE SPECIAL REPORT WAS THE RESULT OF AN INADEQUATE OPERABILITY DETERMINATION AND MISCOMMUNICATION. THE WIND DIRECTION INSTRUMENT WAS INOPERABLE BECAUSE IT WAS LOOSE ON ITS MOUNT AND DID NOT MAINTAIN ALIGNMENT WITH ITS REFERENCE DIRECTION (NORTH). THE WIND DIRECTION SENSORS WERE REALIGNED ON MAY 2, 1989. AN ACTION PLAN TO RESOLVE OUTSTANDING CONCERNS WITH THE METEOROLOGICAL MONITORING SYSTEM AND EVALUATE UPGRADING THE SYSTEM HAS BEEN DEVELOPED. OPERABILITY DETERMINATIONS FOR PLANT EQUIPMENT HAVE BEEN STRENGTHENED THROUGH THE DEVELOPMENT AND IMPLEMENTATION OF A PLANT PROCEDURE. THIS PROCEDURE ESTABLISHES THE RESPONSIBILITY AND PROVIDES TIME LIMITS FOR PERFORMING EQUIPMENT OPERABILITY DETERMINATIONS. AN ENGINEER HAS BEEN ASSIGNED TO THE METEOROLOGICAL MONITORING SYSTEM TO ESTABLISH OWNERSHIP AND PROVIDE A POINT OF CONTACT AS A SYSTEM EXPERT TO RESOLVE SYSTEM CONCERNS.

[180] TROJAN DOCKET 50-344 LER 91-011
POTENTIAL DEGRADATION OF ELECTRICAL PENETRATION ASSEMBLY MODULE SEALS.
EVENT DATE: 051091 REPORT DATE: 061091 NSSS: WE TYPE: PWR
VENDOR: AMPHENOL
PARKER PACKING COMPANY

(NSIC 222164) ON MAY 10, 1991, THE PLANT WAS IN MODE 6, FOLLOWING THE RELOADING OF FUEL IN THE REACTOR VESSEL, DURING THE 1991 REFUELING OUTAGE. NUCLEAR PLANT ENGINEERING (NPE) PERFORMED AN EVALUATION OF THE POTENTIAL DEGRADATION OF ELECTRICAL PENETRATION ASSEMBLY (EPA) SEALS AND DETERMINED: LUBRICANTS USED TO INSTALL THE SEALS (IN ACCORDANCE WITH ORIGINAL MANUFACTURER'S RECOMMENDATIONS) MAY CAUSE SEAL DEGRADATION; AND THE SEAL MANUFACTURER INDICATES THE SEALS MAY DEGRADE IF SUBJECTED TO DESIGN BASIS ACCIDENT (DBA) MOISTURE AND/OR TEMPERATURE CONDITIONS. THE PRIMARY CAUSE OF THIS CONDITION WAS THAT THE ORIGINALLY PROVIDED SEAL MATERIAL WAS INAPPROPRIATE FOR THE SELECTED APPLICATION. CORRECTIVE ACTIONS INCLUDE: REPLACEMENT OF THE EPA SEALS WITH REPLACEMENT SEALS OF A MORE DURABLE MATERIAL, ADDITION OF A SEAL BACKUP O-RING IN THE EPA MODULE ASSEMBLIES, AND ENVIRONMENTAL QUALIFICATION OF THE REVISED EPA SEAL DESIGN, AND PERFORMANCE OF AN INVESTIGATION TO IDENTIFY CONTRIBUTING CAUSES.

[181] TROJAN DOCKET 50-344 LER 91-012
INADEQUATE PROCEDURE ALLOWS TRANSFER OF INSTRUMENT BUS POWER SUPPLY WHICH CAUSES MOMENTARY DE-ENERGIZATION OF CHLORINE DETECTORS WHICH RESULTS IN ISOLATION OF CONTROL ROOM VENTILATION.
EVENT DATE: 051391 REPORT DATE: 061291 NSSS: WE TYPE: PWR

(NSIC 222347) ON MAY 13, 1991 THE TROJAN NUCLEAR PLANT WAS IN MODE 6 (REFUELING) WITH NO FUEL MOVEMENT IN PROGRESS. AT APPROXIMATELY 1346 AN ISOLATION OF THE CONTROL ROOM NORMAL VENTILATION SYSTEM OCCURRED DUE TO AN OUTSIDE AIR TOXIC GAS ISOLATION SIGNAL. THE SIGNAL RESULTED FROM A MOMENTARY DE-ENERGIZATION OF A CHLORINE DETECTOR WHILE TRANSFERRING POWER SUPPLIES TO ITS INSTRUMENT BUS. THE ROOT CAUSE WAS CONCLUDED TO BE INADEQUATE OPERATING PROCEDURES WHICH DID NOT IDENTIFY THAT THE ISOLATION COULD OCCUR. CORRECTIVE ACTIONS INCLUDE PROCEDURE CHANGES. BECAUSE THE DETECTORS AND VENTILATION SYSTEM OPERATED AS DESIGNED, THIS EVENT WAS NOT SAFETY SIGNIFICANT.

[182] TROJAN DOCKET 50-344 LER 91-021
INADEQUATE ANALYSIS OF SHUTDOWN METHODOLOGY RESULTS IN POTENTIAL INABILITY TO DEPRESSURIZE AND COOLDOWN THE PLANT IN ACCORDANCE WITH 10 CFR PART 50 APPENDIX R.
EVENT DATE: 052491 REPORT DATE: 062491 NSSS: WE TYPE: PWR

(NSIC 222378) ON MAY 24, 1991 THE TROJAN NUCLEAR PLANT WAS IN MODE 6, DURING THE 1991 REFUELING AND MAINTENANCE OUTAGE. A VALIDATION OF EMERGENCY FIRE PROCEDURE (EFP)-1, "ALTERNATIVE SHUTDOWN FOR CONTROL ROOM EVACUATION CAUSED BY FIRE" WAS BEING PERFORMED, ON THE TROJAN PLANT SIMULATOR, FOLLOWING PROCEDURE CHANGES WHICH WERE MADE TO ADDRESS NRC IDENTIFIED DEFICIENCIES. DURING THE SIMULATION IT COULD NOT BE VERIFIED THAT THE PLANT COULD BE COOLED DOWN AND DEPRESSURIZED TO RESIDUAL HEAT REMOVAL (RHR) SYSTEM CONDITIONS, USING EXISTING PROCEDURES AND THE PLANT COMPONENTS ANALYZED TO BE AVAILABLE FOLLOWING A POSTULATED WORST CASE FIRE. THIS EVENT WAS THE RESULT OF AN APPARENT FAILURE TO PERFORM AN ADEQUATE ANALYSIS TO DEMONSTRATE THAT THE REACTOR COOLANT SYSTEM COULD BE DEPRESSURIZED UNDER THE CONDITIONS ASSUMED TO EXIST FOLLOWING A POSTULATED WORST CASE FIRE IN THE PLANT. THE PRESSURIZER POWER OPERATED RELIEF VALVE (PORV) CONTROL SYSTEMS WILL BE MODIFIED TO ENABLE CREDITING AT LEAST ONE OF THE PRESSURIZER PORVS FOR POST-FIRE REACTOR COOLANT SYSTEM DEPRESSURIZATION. THE TROJAN NUCLEAR STEAM SUPPLY SYSTEM (NSSS) VENDOR WILL PERFORM AN ANALYSIS DEMONSTRATING THE CAPABILITY TO PERFORM A CONTROLLED NATURAL CIRCULATION COOLDOWN AND DEPRESSURIZATION UNDER A REVISED METHODOLOGY UTILIZING THE PORVS.

[183] TURKEY POINT 4 DOCKET 50-251 LER 91-003
DE-ENERGIZATION OF INTAKE COOLING WATER SYSTEM PUMP CAUSES THE INTERRUPTION OF SPENT FUEL POOL COOLING DUE TO PERSONNEL ERROR.
EVENT DATE: 062691 REPORT DATE: 062891 NSSS: WE TYPE: PWR

(NSIC 222402) THIS VOLUNTARY LER IS BEING SUBMITTED FOLLOWING THE GUIDANCE PROVIDED BY NUREG 1022, SUPPLEMENT 1, ITEM 19.1. ON 6/26/91, AT 0308 EDT, WITH BOTH UNITS DEFUELED, THE UNIT 4, 4D 4KV BUS LOCKED OUT. THE BUS LOCKOUT RESULTED IN THE DE-ENERGIZING OF THE 4C INTAKE COOLING WATER PUMP AND THE 4C COMPONENT COOLING WATER (CCW) PUMP. CCW PUMP 4A WAS IN SERVICE AND REMAINED IN SERVICE DURING THIS EVENT . WITH 4A AND 4B ICW PUMPS OUT OF SERVICE FOR MAINTENANCE, THE DE-ENERGIZING OF THE 4C ICW PUMP MADE THE SPENT FUEL POOL (SFP) COOLING SYSTEM INOPERABLE SINCE THE NORMAL HEAT SINK WAS NOT AVAILABLE. ON 6/26/91, AT 0403, THE 4C CC W PUMP WAS STARTED AND AT 0412, 4C ICW PUMP WAS STARTED RESTORING THE NORMAL HEAT SINK FOR THE SFP COOLING SYSTEM. THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE ADMINISTRATIVE CONTROLS. A SENSITIVE EQUIPMENT LIST USED IN PREPARING A WORK PACKAGE WAS SUPERSEDED WHICH RESULTED IN A BUMPED RELAY 174, DEVICE RB INSIDE CUBICLE 4AD06. RELAY 174, DEVICE RB TRIPPED CAUSING THE LOCKOUT OF THE 4D 4KV BUS. ALL WORK IN THE AREA OF THE RELAY THAT ACTUATED (RELAY 174, DEVICE RB) WAS STOPPED SO THE CAUSE OF THE ACTUATION COULD BE EVALUATED. WORK WILL NOT RESUME UNTIL THE EVENT HAS BEEN REVIEWED WITH APPLICABLE CONTRACTOR PERSONNEL EMPHASIZING THE SENSITIVITY OF THE EQUIPMENT IN THE AREAS BEING WORKED AND ALSO EMPHASIZING ATTENTION TO DETAIL.

[184] VERMONT YANKEE DOCKET 50-271 LER 90-010 REV 02
UPDATE ON FAILURE TO MEET TECHNICAL SPECIFICATIONS FOR DIESEL GENERATOR TESTING.
EVENT DATE: 081690 REPORT DATE: 062691 NSSS: GE TYPE: BWR
VENDOR: FAIRBANKS MORSE

(NSIC 222407) ON AUGUST 16, 1990, AT APPROXIMATELY 1630 HOURS, WITH THE REACTOR OPERATING AT 89.1% POWER, IT WAS IDENTIFIED THAT THE REQUIRED MONTHLY OPERATIONAL READINESS TESTS FOR THE "A AND "B" EMERGENCY DIESEL GENERATORS MAY NOT HAVE BEEN PERFORMED IN ACCORDANCE WITH TECHNICAL SPECIFICATION SECTION 4.10.A.1.A. IT WAS BELIEVED THAT THE EXPECTED MAXIMUM EMERGENCY LOAD USED IN THE SURVEILLANCE PROCEDURE WAS NOT THE TRUE MAXIMUM BASED UPON THE VALUE STATED IN THE FSAR. THE PROCEDURE WAS CHANGED TO INCORPORATE THE VALUE OF 3200 KW AT UNITY POWER FACTOR. AFTER A RE-REVIEW INTO THE EVENT IT WAS DETERMINED THAT THE NEW VALUE OF 3200 KW EXCEEDED THE CONTINUOUS RATING OF THE DIESEL GENERATOR. AN INSPECTION OF THE DIESELS WAS PERFORMED AS A RESULT OF THE OVERLOAD AND THE DIESELS WERE FOUND TO MEET OR EXCEED CRITERIA PROVIDED BY THE MANUFACTURER. THE CONTINUOUS AND OVERLOAD RATINGS WERE CLARIFIED WITH THE AIDE OF THE VENDOR AND THE PROCEDURE WAS THEN AGAIN REVISED TO REFLECT A VALUE OF 2650 - 2750 KW FOR THE TESTING OF THE DIESEL GENERATORS. CONFUSION OF THE INTENT OF THE MONTHLY SURVEILLANCE TEST EXISTED ALONG WITH INCOMPLETE AND ERRONEOUS DIESEL GENERATOR RATINGS INFORMATION. NEW NAME PLATE INFORMATION WILL BE INSTALLED ON THE DIESEL GENERATORS.

[185] VERMONT YANKEE DOCKET 50-271 LER 91-006 REV 01
UPDATE ON LOSS OF 'B' LOOP SHUTDOWN COOLING DUE TO PRESSURE SWITCH ACTIVATION.
EVENT DATE: 031491 REPORT DATE: 070191 NSSS: GE TYPE: BWR

(NSIC 222408) ON 03/14/91 AT 0450 HOURS, WITH REACTOR VESSEL COOLDOWN IN PROGRESS FOLLOWING A REACTOR SCRAM ON 03/13/91 (SUBJECT OF LICENSEE EVENT REPORT 91-05), AND WITH THE "B" LOW RESIDUAL HEAT REMOVAL (RHR) (BO*) SYSTEM FLUSHED AND LINED UP FOR SHUTDOWN COOLING, A GROUP 4 PRIMARY CONTAINMENT ISOLATION SIGNAL (PCIS) (JM)* WAS RECEIVED DURING TWO ATTEMPTED STARTS OF THE "B" RHR PUMP. THE GROUP 4 ISOLATION SIGNAL RESULTED IN A TRIP OF THE "B" PUMP AND CLOSURE OF SHUTDOWN COOLING SUCTION ISOLATION VALVES. THE SECOND FAILED PUMP ATTEMPT WAS INITIATED AT 0455 HOURS. AT 0504 HOURS, AFTER ISOLATIONS WERE RESET A SECOND TIME, THE "D" RHR PUMP WAS SUCCESSFULLY STARTED ON THE "B" RHR LOOP. SHUTDOWN COOLING REMAINED IN OPERATION ON THE "B" RHR LOOP FOR THE DURATION OF THE SHUTDOWN. THE REACTOR RETURNED TO CRITICAL ON 03/18/91 AT 0055 HOURS. THE "ROOT CAUSE" OF THIS EVENT HAS NOT YET BEEN DETERMINED. RESULTS OF ROOT CAUSE INVESTIGATIONS AND TESTING COMPLETED TO DATE HAVE BEEN INCONCLUSIVE IN IDENTIFYING THE EXACT CAUSE OF THE PRESSURE SURGE IN THE "B" RHR LOOP WHICH OCCURRED AFTER THE "B" RHR AS STARTED. MORE COMPREHENSIVE TESTING IS PRESENTLY BEING PLANNED FOR THE UPCOMING 1992 REFUELING OUTAGE. APPROPRIATE CORRECTIVE ACTIONS WILL BE DEVELOPED BASED UPON ANALYSIS OF TEST RESULTS AND WILL BE REPORTED TO THE COMMISSION IN A SUPPLEMENTAL LER.

[186] VERMONT YANKEE DOCKET 50-271 LER 91-009
REACTOR SCRAM DUE TO LOSS OF NORMAL OFF-SITE POWER CAUSED BY INADEQUATE PROCEDURE GUIDELINE.
EVENT DATE: 042391 REPORT DATE: 052391 NSSS: GE TYPE: BWR
VENDOR: EXIDE ELECTRONICS CORP
WESTINGHOUSE ELEC CORP.-NUCLEAR ENERGY SYS

(NSIC 222200) ON 04/23/91 AT 1448 HRS, DURING NORMAL OPERATION WITH REACTOR POWER AT 100%, A REACTOR SCRAM OCCURRED AS A RESULT OF A GENERATOR/TURBINE TRIP ON GENERATOR LOAD REJECT TO THE RECEIPT OF A 345KV BREAKER FAILURE SIGNAL. THE FAILURE SIGNAL WAS THE RESULT OF BREAKER FAILURE INTERLOCK (BFI) SIGNALS THAT OCCURRED SIMULTANEOUSLY IN THE 345KV AND 115KV BREAKER CONTROL CIRCUITRY DURING THE RESTORATION OF A BATTERY BANK TO SWITCHYARD BUS DC 4A. CUMULATIVE EFFECTS OF BOTH BFI SIGNALS RESULTED IN A TOTAL LOSS OF 345KV AND 115KV OFF-SITE POWER. AN UNUSUAL EVENT WAS DECLARED AT 1507 HRS. BOTH EMERGENCY DIESEL GENERATORS PROVIDED POWER FOR ESSENTIAL SAFETY RELATED SYSTEMS DURING THE LNP UNTIL APPROXIMATELY 0430 HRS ON 04/24/91 AT WHICH POINT OFF-SITE 345KV POWER WAS RESTORED AND BACKFED THROUGH THE STATION AUXILIARY TRANSFORMER. DURING THE EVENT, TORUS WATER VOLUME EXCEEDED THE TECHNICAL SPECIFICATION LIMIT OF 70,000 CUBIC FT. THE UNUSUAL EVENT TERMINATED AT 1950 HRS ON 04/24/91. THE REACTOR REACHED COLD SHUTDOWN AT 0357 HRS ON 04/25/91 AND WAS RETURNED TO CRITICAL AT 0300 HRS ON 04/30/91. THE ROOT CAUSE OF THIS EVENT IS FAILURE OF THE REPAIR DEPARTMENT PERSONNEL TO RECOGNIZE CONSEQUENCES OF OPERATING A DC BUS WITHOUT A CONNECTED BATTERY BANK. CORRECTIVE ACTIONS TO PREVENT RECCURRENCE ARE PRESENTLY BEING FINALIZED AND WILL BE PRESENTED IN A SUPPLEMENTAL REPORT.

[187] VERMONT YANKEE DOCKET 50-271 LER 91-013
RPS MG "A" OUTPUT BREAKER TRIP.
EVENT DATE: 052891 REPORT DATE: 062091 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 222304) AT APPROXIMATELY 0839 HOURS ON 05/28/91 WITH THE PLANT AT FULL POWER, A REACTOR PROTECTION SYSTEM (RPS) MOTOR GENERATOR SET (MG) OUTPUT BREAKER (BKR) TRIPPED. THE RESULTING LOSS OF POWER TO HALF OF RPS CAUSED A HALF SCRAM, A GROUP 3 PRIMARY CONTAINMENT DILATION, SHUTDOWN OF REACTOR BUILDING VENTILATION AND START OF THE STANDBY GAS TREATMENT SYSTEM. EXTENSIVE TESTING/INSPECTION OF THIS BREAKER AND THE ASSOCIATED MG PROTECTIVE CIRCUITRY HAS NOT RESULTED IN ANY EVIDENCE WHICH WOULD ALLOW FOR ROOT CAUSE ANALYSIS. THEREFORE, THE CAUSE OF THE TRIPPED OUTPUT BREAKER IS INDETERMINATE. THE EVENTS OF THIS REPORT DID NOT HAVE ADVERSE SAFETY IMPLICATIONS. THE RPS AND PRIMARY CONTAINMENT ISOLATOR STEM (PCIS) OPERATED AS DESIGNED UPON EXPERIENCING A POWER LOSS TO HALF OF RPS.

[188] VOGTLE 1 DOCKET 50-424 LER 90-021 REV 02
UPDATE ON PERSONNEL ERRORS LEAD TO MISSED SPECIAL CONDITION SURVEILLANCE.
EVENT DATE: 120290 REPORT DATE: 062491 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 222391) DURING THE PERIOD BETWEEN 1718 CST ON 12-2-90 AND 0602 CST ON 12-3-90, GEORGIA POWER COMPANY FAILED TO COMPLY WITH A TECHNICAL SPECIFICATION 4.1.3.2 SPECIAL CONDITION SURVEILLANCE WHICH IS APPLICABLE WHEN THE ROD POSITION DEVIATION MONITOR IS INOPERABLE. THIS SURVEILLANCE REQUIRES THAT THE DEMAND POSITION INDICATION SYSTEM AND THE DIGITAL ROD POSITION INDICATION SYSTEM BE COMPARED AT LEAST ONCE PER 4 HOURS. CONTROL ROOM PERSONNEL HAD INADVERTENTLY MADE THE ROD POSITION DEVIATION MONITOR INOPERABLE WHEN ATTEMPTING TO REINSERT CONTROL ROD POSITION INDICATOR VALUES INTO THE PROTEUS (PLANT STATUS) COMPUTER. THE CAUSE OF THIS EVENT WAS THE FAILURE OF THE UNIT SHIFT SUPERVISOR (USS) TO FOLLOW PROCEDURE WHEN REENTERING THE ROD POSITION VALUES INTO THE PROTEUS COMPUTER. ADDITIONALLY, THE SHIFT SUPERINTENDENT FAILED TO ENSURE THAT THE USS COMPLIED WITH THE PROCEDURE. THESE PERSONNEL HAVE UNDERGONE POSITIVE DISCIPLINE REGARDING THE IMPORTANCE OF PROCEDURAL COMPLIANCE.

[189] VOGTLE 2 DOCKET 50-425 LER 91-007
GENERATOR LOSS OF FIELD RESULTS IN AUTOMATIC REACTOR TRIP.
EVENT DATE: 050791 REPORT DATE: 060491 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.
MASONEILAN INTERNATIONAL, INC.

(NSIC 222186) ON 5-7-91 AT 0948 CDT, SEVERAL ANNUNCIATORS AND ALARMS WERE RECEIVED ON THE ELECTRICAL AUXILIARY BOARD IN THE UNIT 2 CONTROL ROOM. IMMEDIATELY AFTER RECEIPT OF THE ALARMS, AN AUTOMATIC REACTOR TRIP OCCURRED DUE TO A TRIP OF THE UNIT 2 TURBINE. EXCEPT FOR THE TRIPPING OF NON-1E BREAKERS WHICH SUPPLY TURBINE BUILDING LIGHTING AND THE MALFUNCTION OF A STEAM-DUMP-TO-CONDENSER VALVE, NO OTHER EQUIPMENT MALFUNCTIONS OR ABNORMALITIES OCCURRED FOLLOWING THE REACTOR TRIP, AND BY 1010 CDT THE PLANT WAS STABILIZED IN HOT STANDBY. A POST-TRIP REVIEW OF COMPUTER DATA, THE PLANT FAULT RECORDER, AND THE PROTECTION RELAY PANEL REVEALED THAT A GENERATOR LOSS OF FIELD OCCURRED JUST PRIOR TO THE EVENT, CAUSING A GENERATOR TRIP WHICH THEN CAUSED THE TURBINE/REACTOR TRIP. THE ROOT CAUSE FOR THE GENERATOR LOSS OF FIELD IS INDETERMINATE. EXTENSIVE TROUBLESHOOTING UNDER THE DIRECTION OF A GENERAL ELECTRIC REPRESENTATIVE WAS PERFORMED ON 5-8-91 WITH THE GENERATOR SHUT DOWN AND ON 5-9-91 WITH THE GENERATOR AT RATED SPEED AND VOLTAGE. NO PROBLEM WITH THE GENERATOR FIELD EXCITATION CIRCUITRY WAS FOUND, AND AT 1941 CDT ON 5-9-91 THE GENERATOR WAS TIED TO THE GRID. CONTINUOUS MONITORING VIA INSTALLED RECORDERS WAS THEN PERFORMED WITH THE GENERATOR ONLINE. AFTER NO ABNORMALITY WAS DETECTED THAT MIGHT HAVE CAUSED THE EVENT, THE RECORDERS WERE REMOVED ON 5-19-91.

[190] WATERFORD 3 DOCKET 50-382 LER 90-004 REV 01
UPDATE ON INADVERTENT EMERGENCY FEEDWATER SYSTEM ACTUATION DUE TO TEST CIRCUIT MALFUNCTION.
EVENT DATE: 040890 REPORT DATE: 062591 NSSS: CE TYPE: PWR
VENDOR: CUTLER-HAMMER

(NSIC 222383) AT 0444 HOURS ON APRIL 8, 1990, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% POWER WHEN A PARTIAL ACTUATION OF THE EMERGENCY FEEDWATER (EFW) SYSTEM OCCURRED DURING A SCHEDULED FUNCTIONAL TEST OF THE PLANT PROTECTION SYSTEM (PPS). THIS EVENT IS REPORTABLE AS AN UNPLANNED ENGINEERED SAFETY FEATURE (ESF) ACTUATION. THE ACTUATION WAS NOT OF SUFFICIENT DURATION TO ALLOW EFW FLOW TO THE STEAM GENERATORS (SG). THE ROOT CAUSE OF THIS EVENT IS TEST CIRCUIT MALFUNCTION. THE MATRIX TEST MODULE RELAY HOLD PUSHBUTTON ASSEMBLY WAS REPLACED AND THE PPS FUNCTIONAL TEST WAS COMPLETED SATISFACTORILY. FURTHER DIAGNOSTIC TESTING AND EVALUATION OF THE RELAY HOLD PUSHBUTTON ISOLATED THE CAUSE OF THE CIRCUIT MALFUNCTION TO BE THE INCORRECT ASSEMBLY OF THE EARLY CLOSING CONTACT BLOCK. BECAUSE THE CIRCUIT MALFUNCTION WAS ASSOCIATED ONLY WITH THE TEST CIRCUITRY AND THE EFW SYSTEM WAS CAPABLE OF RESPONDING IN THE SAFE DIRECTION, THIS EVENT DID NOT THREATEN THE HEALTH AND SAFETY OF THE GENERAL PUBLIC OR PLANT PERSONNEL.

[191] WATERFORD 3 DOCKET 50-382 LER 91-007
INCORRECT ELECTRONIC CURRENT SENSOR RATING ON A 480 V SAFETY RELATED BUS DUE TO IMPROPER ARCHITECT ENGINEER REVIEW.
EVENT DATE: 051091 REPORT DATE: 061091 NSSS: CE TYPE: PWR

(NSIC 222304) ON 5/10/91, UNIT 3 WAS SHUTDOWN IN MODE 5, WHEN A REVIEW OF CALCULATION EC-E89-007, RELAY SETTINGS AND COORDINATION CURVES FOR 6.9 KV, 4.16 KV AND 480 V BUSSES, REVEALED THAT THE ELECTRONIC CURRENT SENSOR (ECS) CURRENT TRANSFORMER (CT) RATING INSTALLED ON MOTOR CONTROL CENTER BUS (MCC) 3AB311-S WAS 300 AMPS VICE THE REQUIRED 600 AMPS. MCC 3AB311-S SUPPLIES POWER TO EQUIPMENT NECESSARY TO RUN THE "AB" ESSENTIAL SERVICES CHILLER. THE POTENTIAL FOR MCC 3A B311-S TO EXCEED 300 AMPS EXISTED IN THE EVENT OF A LOSS OF OFF-SITE POWER WITH THE MAIN TURBINE TRIPPED. THE SUBSEQUENT AUTOMATIC RESTORATION OF ELECTRICAL LOADS COULD HAVE RESULTED IN THE MCC 3AB311-S BREAKER OPENING DUE TO AN OVERCURRENT CONDITION. THIS EVENT IS REPORTABLE AS A CONDITION WHICH COULD HAVE CAUSED ONE TRAIN TO BECOME INOPERABLE FOR SYSTEMS DESIGNED TO REMOVE RESIDUAL HEAT OR SHUTDOWN THE REACTOR AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION. THE ROOT CAUSE OF THIS EVENT WAS A FAILURE OF THE ARCHITECT ENGINEER TO PROPERLY VERIFY CT AMPERAGE RATINGS, IN ACCORDANCE WITH DESIGN CALCULATIONS, WHEN THE CTS WERE ORDERED FOR INITIAL INSTALLATION. CORRECTIVE ACTION INCLUDED INSTALLING THE PROPER RATED CT AND VERIFYING SAFETY RELATED BUSES FOR CORRECT CT RATING. SINCE ANOTHER TRAIN WOULD HAVE BEEN AVAILABLE DURING THIS TIME, THIS EVENT WOULD NOT HAVE THREATENED THE HEALTH AND SAFETY OF THE PUBLIC.

[192] WATERFORD 3 DOCKET 50-382 LER 91-008
REACTOR COOLANT SYSTEM LEAKAGE IN EXCESS OF TECHNICAL SPECIFICATIONS DUE TO CHECK VALVE LEAKAGE.
EVENT DATE: 051791 REPORT DATE: 061791 NSSS: CE TYPE: PWR

(NSIC 222359) AT 1335 HOURS ON MAY 17, 1991, WATERFORD STEAM ELECTRIC STATION UNIT 3 IN HOT SHUTDOWN (MODE 4), AN UNUSUAL EVENT WAS DECLARED DUE TO REACTOR COOLANT SYSTEM LEAKAGE FROM PRESSURIZER SPRAY CHECK VALVE RC-303, IN EXCESS OF TECHNICAL SPECIFICATION (TS) REQUIREMENTS. TS 3.4.5.2(D) LIMITS REACTOR COOLANT SYSTEM (RCS) LEAKAGE TO 10 GPM. THE LEAKAGE FROM RC-303 WAS CALCULATED TO BE APPROXIMATELY 20 GALLONS PER MINUTE (GPM), REQUIRING PLANT SHUTDOWN TO COLD SHUTDOWN (MODE 5) PER TS'S. DURING THIS EVENT, TS 3.0.3 WAS ENTERED DUE TO CLOSING THE SIT OUTLET ISOLATION VALVES. THE ROOT CAUSE OF THIS EVENT IS UNDERDEVELOPED TRAINING ON PRESSURE SEAL VALVE INSTALLATION. CORRECTIVE ACTIONS ARE TO SUBMIT A TRAINING REQUEST TO TRAIN ALL MECHANICAL MAINTENANCE PERSONNEL ON PRESSURE SEAL VALVE INSTALLATION AND TO REVISE THE TECHNICAL MANUAL TO INCLUDE MORE DETAIL INFORMATION ON PRESSURE SEAL GASKETS. AFTER RC-303 BEGAN TO LEAK, RCS PRESSURE AND TEMPERATURE WAS REDUCED, CONTAINMENT WAS EVACUATED, AND RC-303 WAS REPAIRED; THEREFORE, THE HEALTH AND SAFETY OF THE GENERAL PUBLIC WAS NOT JEOPARDIZED.

[193] WATERFORD 3 DOCKET 50-382 LER 91-009
SAFETY INJECTION TANK 1A AND 1B INOPERABLE DUE TO RELIEF VALVE MALFUNCTION.
EVENT DATE: 052191 REPORT DATE: 062091 NSSS: CE TYPE: PWR
VENDOR: CROSBY VALVE & GAGE CO.

(NSIC 222360) ON MAY 21, 1991, THE PLANT WAS SHUTDOWN IN MODE 3 AT A REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE OF 2250 PSIA AND 540 DEGREES FAHRENHEIT, RESPECTIVELY. SAFETY INJECTION TANK (SIT) 1A WAS OUT OF SERVICE DUE TO AN INOPERABLE NITROGEN RELIEF VALVE. AT 1411 HOURS THE RELIEF VALVE FOR SIT 1B LIFTED, LOWERING SIT 1B PRESSURE BELOW 600 PSIG. THE RESULTING CONDITION OF TWO SITS INOPERABLE UNDER THE EXISTING PLANT CONDITIONS IS REPORTABLE UNDER 10CFR50.73(A)(2)(I) AS A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS (TSS) (ENTRY INTO TS 3.0.3). RCS PRESSURE WAS LOWERED TO LESS THAN 1750 PSIA, SIT 1A AND 1B WERE DECLARED OPERABLE, AND TECHNICAL SPECIFICATION 3.0.3 WAS EXITED. THE RELIEF VALVES FOR SIT 1A AND 1B WERE LATER REMOVED, TESTED, AND RETURNED TO SERVICE. THE MOST LIKELY ROOT CAUSE OF THIS EVENT IS EQUIPMENT MALFUNCTION DUE TO ABNORMAL VIBRATION OF THE RELIEF VALVE OR VALVE TAILPIECE WHEN AGITATED BY SCAFFOLDING MOVEMENT. THIS AGITATION RESULTED IN THE SPURIOUS OPENING OF THE SIT 1A AND 1B RELIEF VALVES AT A PRESSURE LOWER THAN THE REQUIRED SETPOINT. DUE TO

THE PROMPT ACTIONS OF OPERATIONS PERSONNEL TO PLACE THE PLANT IN A STABLE CONDITION IN WHICH THE SITS WERE ALL OPERABLE, THERE WAS NO DANGER TO THE HEALTH AND SAFETY OF THE GENERAL PUBLIC OR SITE PERSONNEL.

[194] WATERFORD 3 DOCKET 50-382 LER 91-010
REACTOR TRIP DUE TO PLANT PROTECTION SYSTEM ACTUATION DURING ANNUNCIATOR GROUND ISOLATION.
EVENT DATE: 052691 REPORT DATE: 062591 NSSS: CE TYPE: PWR

(NSIC 222385) AT 0102 ON 5/26/91, AN AUTOMATIC REACTOR TRIP OF WATERFORD STEAM ELECTRIC STATION UNIT 3 OCCURRED WHILE IN START-UP (MODE 2) AT 0% POWER. THE TRIP WAS INITIATED BY THE PLANT PROTECTION SYSTEM IN RESPONSE TO HIGH LOG POWER ON CHANNELS C AND D. THIS EVENT IS REPORTABLE AS AN UNPLANNED ACTUATION OF AN ENGINEERED SAFETY FEATURE UNDER 10 CFR 50.73(A)(2)(IV). THE ROOT CAUSE OF THIS EVENT WAS THE ISOLATION OF GROUNDS ON THE MAIN ANNUNCIATOR SYSTEM DURING LOW POWER OPERATION, WHICH INTRODUCED ELECTRICAL NOISE INTO THE EXCORE LOG POWER CIRCUITRY. JUST PRIOR TO THE REACTOR TRIP, ELECTRICAL MAINTENANCE TECHNICIANS WERE TROUBLESHOOTING THE MAIN ANNUNCIATOR SYSTEM BY ISOLATING GROUNDS TO VARIOUS POWER SUPPLIES. UPON ISOLATING THE GROUNDS, A SPURIOUS SIGNAL WAS INTRODUCED INTO THE LOG POWER CIRCUITRY, AND THE PLANT TRIPPED ON HIGH LOG POWER. IMMEDIATE CORRECTIVE ACTIONS WERE TAKEN TO CAUTION TAG THE ANNUNCIATOR CABINETS TO PROHIBIT ISOLATION OF GROUNDS WHILE AT A POWER LEVEL BELOW THE LOG POWER TRIP SETPOINT OF .257%. LONG TERM ACTION TO PREVENT RECURRENCE INVOLVES REVISING ME-004-455 TO INCLUDE A CAUTION TO PREVENT ISOLATION OF GROUNDS BELOW THE LOG POWER TRIP SETPOINT. THE PLANT PROTECTIVE FEATURES FUNCTIONED AS DESIGNED.

[195] WATERFORD 3 DOCKET 50-382 LER 91-011
REACTOR TRIP DUE TO FAULTY RELAY.
EVENT DATE: 052891 REPORT DATE: 062791 NSSS: CE TYPE: PWR
VENDOR: STRUTHERS DUNN, INC.

(NSIC 222386) AT 0809 HOURS ON MAY 28, 1991, THE WATERFORD STEAM ELECTRIC STATION UNIT 3 MAIN TURBINE WAS MANUALLY TRIPPED DUE TO A LOSS OF GOVERNOR CONTROL. AFTER THE TURBINE TRIP, AN AUTOMATIC REACTOR TRIP OCCURRED WHEN THE ELECTRICAL BUS POWERING TWO REACTOR COOLANT PUMPS FAILED TO TRANSFER FROM THE UNIT AUXILIARY TRANSFORMERS TO THE STARTUP TRANSFORMERS UPON LOSS OF THE MAIN TURBINE. THE ROOT CAUSE OF THIS EVENT WAS A FAULTY RELAY THAT PREVENTED THE ELECTRICAL BUS TRANSFER. A CONTRIBUTING CAUSE TO THIS EVENT WAS AN INADEQUATE PROCEDURE THAT, DURING TROUBLESHOOTING OF THE MAIN TURBINE DIGITAL ELECTRO HYDRAULIC SYSTEM, CAUSED THE LOSS OF GOVERNOR CONTROL. ANOTHER CONTRIBUTING CAUSE TO THIS EVENT WAS A GROUND IN A MAINTENANCE CABINET THAT CAUSED MAIN TURBINE LOAD FLUCTUATIONS. THE FAULTY RELAY WAS REPLACED, THE PROCEDURE REVISED AND THE CAUSE OF THE GROUND REMOVED. PLANT PROTECTIVE FEATURES FUNCTIONED AS DESIGNED; THEREFORE, THIS EVENT DID NOT THREATEN THE HEALTH OR SAFETY OF THE GENERAL PUBLIC OR PLANT PERSONNEL.

[196] WOLF CREEK 1 DOCKET 50-482 LER 91-006
AUXILIARY FEEDWATER ACTUATION AND CONTROL ROOM VENTILATION ISOLATION WHEN POWER SUPPLY IN ENGINEERED SAFETY FEATURES ACTUATION SYSTEM FAILED.
EVENT DATE: 051291 REPORT DATE: 060791 NSSS: WE TYPE: PWR
VENDOR: CONSOLIDATED CONTROLS CORP.

(NSIC 222191) ON MAY 12, 1991 AT 1800 CDT, A "B" TRAIN CONTROL ROOM VENTILATION ISOLATION OCCURRED, THE "B" TRAIN AUXILIARY FEEDWATER PUMP STARTED, THE TURBINE DRIVEN AUXILIARY FEEDWATER PUMP STARTED, AND "B" TRAIN STEAM GENERATOR BLOWDOWN AND SAMPLING ISOLATED, AS A RESULT OF A POWER SUPPLY FAILURE IN ENGINEERED SAFETY FEATURES (ESF) ACTUATION SYSTEM CABINET SA036D. AS A RESULT OF THE LOSS OF THE "A" TRAIN LOGIC AND ACTUATION RELAYS A PLANT SHUTDOWN WAS INITIATED AS REQUIRED BY TECHNICAL SPECIFICATIONS, AND A NOTIFICATION OF UNUSUAL EVENT WAS DECLARED. THE POWER SUPPLY WAS REPLACED BY 2150 CCT AND THE NOTIFICATION OF UNUSUAL EVENT TERMINATED AT 2208 CDT. ALL ESF EQUIPMENT FUNCTIONED PROPERLY. THE CAUSE OF THE ESF ACTUATION WAS A FAULTY PRE-REGULATOR BOARD IN THE 15 VOLT DC POWER SUPPLY IN CABINET SA036D. ROOT CAUSE OF THE FAILURE COULD NOT BE DETERMINED. TESTING AND

MONITORING OF THIS TYPE OF POWER SUPPLY HAS INDICATED THAT A GENERIC PROBLEM DOES NOT EXIST.

[197] WOLF CREEK 1 DOCKET 50-482 LER 91-007
INADEQUATE TESTING OF COMPONENT COOLING WATER TO REACTOR COOLANT PUMP THERMAL BARRIER CHECK VALVES.
EVENT DATE: 052291 REPORT DATE: 062191 NSSS: WE TYPE: PWR

(NSIC 222331) ON MAY 22, 1991, IT WAS DISCOVERED THAT INSERVICE TESTING (IST) PROCEDURE (IST) PROCEDURE EG-206, REVISION 0, "COMPONENT COOLING WATER SYSTEM INSERVICE VALVE TEST", DOES NOT ADEQUATELY TEST THE COMPONENT COOLING WATER TO REACTOR COOLANT PUMP THERMAL BARRIER CHECK VALVES IN THEIR CLOSED POSITION BECAUSE OF AN INADEQUATE TEST BOUNDARY. THIS TEST DEFICIENCY, WHICH HAS BEEN PRESENT IN THE FIVE TIMES THIS TEST HAS BEEN PERFORMED, IS CONTRARY TO TECHNICAL SPECIFICATION 4.0.5. THIS TEST DEFICIENCY RESULTED FROM PERSONNEL NOT PROVIDING SUFFICIENT TECHNICAL CONTENT DURING INITIAL DEVELOPMENT OF THE IST PROCEDURE. THE ROOT CAUSE OF THIS INITIAL TEST DEFICIENCY COULD NOT BE DETERMINED BECAUSE THE PERSONNEL INVOLVED ARE NO LONGER EMPLOYED BY WOLF CREEK NUCLEAR OPERATING CORPORATION. BECAUSE OF THIS DEFICIENCY AND OTHER IDENTIFIED DEFICIENCIES THAT DID NOT RESULT IN NONCOMPLIANCE, A COMPLETE TECHNICAL IST PROCEDURES WILL BE PERFORMED. CERTAIN INSERVICE INSPECTION PROGRAM PROCEDURES WILL ALSO BE INCLUDED AMONG THIS REVIEW.

[198] WPPSS 2 DOCKET 50-397 LER 91-011
ENGINEERED SAFETY FEATURE ACTUATION DUE TO INADVERTENT WIRE CUTTING DURING PLANT MODIFICATION WORK.
EVENT DATE: 042991 REPORT DATE: 052991 NSSS: GE TYPE: BWR

(NSIC 222176) ON 4/29/91, AT 1249 HOURS, WHILE IN OPERATIONAL CONDITION 5 (REFUELING) WITH THE REACTOR HEAD REMOVED, THE REACTOR CAVITY FLOODED UP AND THE FUEL POOL GATES REMOVED, THE OUTBOARD RESIDUAL HEAT REMOVAL (RHR) SHUTDOWN COOLING VALVE (RHR-V-8) AUTOMATICALLY ISOLATED, CAUSING A LOSS OF SHUTDOWN COOLING, AN ACTUATION. THE EVENT OCCURRED WHEN A CONTRACTOR MAINTENANCE ELECTRICIAN, WORKING UNDER AN APPROVED MAINTENANCE WORK REQUEST (MWR), ACCIDENTALLY CUT THROUGH THE INSULATION OF A WIRE IN THE RHR ISOLATION CONTROL LOGIC CIRCUIT WHILE CUTTING TIE WRAPS AROUND A WIRE BUNDLE. CONTACT OF THE CUTTING TOOL WITH THE BARE WIRE RESULTED IN A BLOWN FUSE IN THE RHR CONTROL LOGIC CIRCUIT. THIS LOSS OF POWER CAUSED RHR-V-8 TO CLOSE, TRIPPING THE RHR SHUTDOWN COOLING PUMP (RHR-P-2B). AT THE TIME OF THE EVENT, THE REACTOR VESSEL TEMPERATURE WAS 94.2F AND REACTOR PRESSURE WAS ATMOSPHERIC. TEMPERATURE INCREASE DURING THE TIME SHUTDOWN COOLING WAS LOST (24 MIN.) WAS NEGLIGIBLE. PLANT OPERATORS RESTORED SHUTDOWN COOLING AT 1313 HOURS. THE ROOT CAUSE OF THE EVENT WAS EQUIPMENT/DESIGN DEFICIENCY IN THAT THERE WAS INADEQUATE ACCESSIBILITY TO THE WIRE BUNDLE FOR THE CONTRACTOR MAINTENANCE ELECTRICIAN TO GET TO THE TIE WRAP TO BE CUT AND ORIENT THE SIDE CUTTERS PARALLEL TO THE TIE WRAP. A CONTRIBUTING FACTOR MAY HAVE BEEN INSUFFICIENT ATTENTION APPLIED TO THE TASK BY THE ELECTRICIAN.

[199] WPPSS 2 DOCKET 50-397 LER 91-013
TECHNICAL SPECIFICATION - SURVEILLANCE PROCEDURE VERIFICATION PROGRAM IDENTIFICATION OF NONCONFORMING CONDITIONS.
EVENT DATE: 050791 REPORT DATE: 060691 NSSS: GE TYPE: BWR
VENDOR: KAMAN SCIENCES CORP.

(NSIC 222178) ON 5/7/91 THE FIRST OF SEVERAL ITEMS OF NON-COMPLIANCE WITH THE WNP-2 TECH SPEC (TS) WAS IDENTIFIED AS PART OF A PROGRAM OF SURVEILLANCE PROCEDURE VERIFICATION. THIS SURVEILLANCE PROCEDURE VERIFICATION PROGRAM WAS INITIATED BY THE SUPPLY SYSTEM AS A RESULT OF PROBLEMS FOUND IN RECENT MONTHS. 115 POTENTIAL DEFICIENCIES WERE IDENTIFIED BY CONTRACT ENGINEERS AND THIS EFFORT IS COMPLETE. EVALUATION BY THE PLANT STAFF WAS PERFORMED ON EACH POTENTIAL PROBLEM ITEM TO IDENTIFY THE VALIDITY AND NECESSARY FOLLOW-UP ACTIONS. A TOTAL OF 7 ITEMS WERE IDENTIFIED AS REPORTABLE PROBLEMS BY THIS PROCESS WHEN THE TIME CUTOFF DATE FOR THIS LER OCCURRED. THE 7 ITEMS COVER A VARIETY OF SUBJECTS

INVOLVING ITEMS OF NON-COMPLIANCE. SOME OF THE ITEMS ARE ISSUES WHERE COMPLIANCE IS NOT POSSIBLE AND A TS CHANGE OR OTHER RELIEF WILL BE REQUESTED THIS LER SHOULD BE CONSIDERED A PRELIMINARY DOCUMENT AS THE INVESTIGATION OF THESE ITEMS IS STILL IN PROGRESS. A REVISION TO THIS LER WILL BE SUBMITTED PRIOR TO 7/3/91 TO PROVIDE AN UPDATE OF THESE INITIAL 7 ITEMS AND ADDITIONAL REPORTABLE ITEMS THAT ARE DISCOVERED BY THE REVIEW. IMMEDIATE AND FURTHER CORRECTIVE ACTIONS INCLUDE ADDITIONAL TESTING, PLANT PROCEDURE CHANGES, REQUESTS FOR CHANGES OR OTHER RELIEF, AND POSSIBLE DESIGN CHANGES. GENERAL CORRECTIVE ACTIONS HAVE NOT BEEN IDENTIFIED TO DATE BUT WILL BE INCLUDED IN THE REVISION.

[200] WPPSS 2 DOCKET 50-397 LER 91-014
RESIDUAL HEAT REMOVAL SHUTDOWN COOLING CONTAMINENT ISOLATION ACTUATION DUE TO LESS THAN ADEQUATE DESIGN DRAWING INFORMATION.
EVENT DATE: 051091 REPORT DATE: 061091 NSSS: GE TYPE: BWR

(NSIC 222179) ON 5/10/91 AT 2126 HOURS WHILE THE PLANT WAS IN OPERATIONAL MODE 5 (REFUELING) WITH RESIDUAL HEAT REMOVAL (RHR) LOOP B OPERATING IN THE SHUTDOWN COOLING MODE, AN RHR SHUTDOWN COOLING ISOLATION OCCURRED. THIS EVENT OCCURRED WHEN A REACTOR PROTECTION SYSTEM (RPS) CIRCUIT BREAKER WAS OPENED AS PART OF A CLEARANCE ORDER AND DE-ENERGIZED RELAYS IN THE NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS). THIS LED TO THE CLOSURE OF THE RHR SHUTDOWN COOLING SUCTION LINE ISOLATION VALVES. THIS SHUTDOWN COOLING ISOLATION WAS AN UNPLANNED AUTOMATIC ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF) SYSTEM. AS IMMEDIATE CORRECTIVE ACTIONS, PLANT LICENSED CONTROL ROOM OPERATORS REESTABLISHED SHUTDOWN COOLING BY 2133 HOURS. THE ROOT CAUSE OF THIS EVENT WAS A LESS THAN ADEQUATE DESIGN DRAWING. FURTHER CORRECTIVE ACTION WILL BE TO REVISE THE AFFECTED DRAWINGS AND TO UPDATE THE TAGGING SYSTEM TO INDICATE THIS SYSTEM INTERRELATIONSHIP. THERE WAS MINIMAL S[illegible] SIGNIFICANCE ASSOCIATED WITH THE EVENT. AT THE TIME OF THE EVENT THE REACTOR WATER LEVEL WAS SUFFICIENT TO PROVIDE ADEQUATE CORE COOLING FOR AN EXTENDED PERIOD OF TIME. RHR SHUTDOWN COOLING WAS FULLY RESTORED IN SEVEN MINUTES. THIS EVENT POSED NO THREAT TO THE HEALTH AND SAFETY OF EITHER THE PUBLIC OR PLANT PERSONNEL.

[201] WPPSS 2 DOCKET 50-397 LER 91-015
HIGH PRESSURE CORE SPRAY SYSTEM PUMP SUCTION VALVE SWITCHOVER ACTUATION ON HIGH SUPPRESSION POOL LEVEL DUE TO PERSONNEL ERROR.
EVENT DATE: 060291 REPORT DATE: 062891 NSSS: GE TYPE: BWR

(NSIC 222502) ON 6/2/91 AT 0106 HOURS WHILE THE PLANT WAS SHUTDOWN FOR MAINTENANCE, A HIGH PRESSURE CORE SPRAY (HPCS) SYSTEM PUMP SUCTION SWITCHOVER FROM THE CONDENSATE STORAGE TANKS TO THE SUPPRESSION POOL OCCURRED DUE TO A SUPPRESSION POOL HIGH WATER LEVEL CONDITION. PLANT CONFIGURATION AT THE TIME WAS SUCH THAT HPCS SUCTION WAS LINED UP TO THE CONDENSATE STORAGE TANKS (CSTS) WITH CST SUCTION VALVE HPC S-V-1 OPEN AND SUPPRESSION POOL SUCTION VALVE HPCS-V-15 CLOSED, THE NORMAL SYSTEM LINEUP (REFERENCE FIGURE 1). THE SWITCHOVER, AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION, WAS THE AUTOMATIC CLOSURE OF HPCS-V-1 AND THE OPENING OF HPCS-V-15. DURING THE EVENT PERIOD, PLANT CONTROL ROOM OPERATORS WERE LOWERING REACTOR VESSEL LEVEL THROUGH THE RESIDUAL HEAT REMOVAL (RHR) "A" HEAT EXCHANGER VENTS TO THE SUPPRESSION POOL FOLLOWING EXCESS FLOW CHECK (EFC) VALVE TESTING. DURING THIS PERIOD THE SUPPRESSION POOL HIGH LEVEL ALARM ANNUNCIATOR HAD SEALED IN (THE ALARM ANNUNCIATES AT +0.5 INCHES (0.5 INCHES ABOVE NORMAL POOL LEVEL)). HOWEVER, AFTER ACKNOWLEDGING AND RESETTING THE ALARM, A PLANT CONTROL ROOM OPERATOR FAILED TO TAKE ACTION TO EITHER LOWER SUPPRESSION POOL LEVEL OR MANUALLY SWITCHOVER THE HPCS SUCTION VALVES AS DIRECTED BY PROCEDURE. WHEN SUPPRESSION POOL WATER VOLUME REACHED APPROXIMATELY +3.0 INCHES INDICATED LEVEL, THE AUTOMATIC SWITCHOVER OCCURRED.

[202] ZION 1 DOCKET 50-295 LER 91-007
CONTROL ROOM HVAC ENVELOPE UNFILTERED INLEAKAGE FOUND TO BE HIGHER THAN THE VALUE ASSUMED IN THE DOCKETED ANALYSIS.
EVENT DATE: 050791 REPORT DATE: 061791 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 2 (PWR)

(NSIC 222322) IN FEBRUARY A CONTROL ROOM ENVELOPE INLEAKAGE TEST WAS CONDUCTED TO DETERMINE IF GENERAL DESIGN CRITERIA (GDC)-19 COULD BE MET WITH THE EXISTING CONTROL ROOM HVAC (HEATING, VENTILATING AND AIR CONDITIONING) SYSTEM (VI) (PV). THE TEST RESULTS WERE RECEIVED FROM THE TESTING CONTRACTOR IN EARLY MAY, AND THEY SHOWED THAT THE AIR INLEAKAGE RATES INTO THE CONTROL ROOM ENVELOPE WERE HIGHER THAN THE DOCKETED ANALYSIS. THE CONTROL ROOM POSITIVE PRESSURE AND THE EMERGENCY MAKE-UP AIR FLOW RATES ALSO WERE LOWER THAN THE VALUES STATED IN THE UFSAR (UPDATED FINAL SAFETY ANALYSIS REPORT). ENGINEERING AND NUCLEAR CONSTRUCTION (ENC) PERFORMED AN OPERABILITY ANALYSIS ON THE PV SYSTEM AND DETERMINED THAT THE SYSTEM WAS OPERABLE AND THERE WAS NO SAFETY SIGNIFICANCE. THIS EVENT WAS CAUSED BY INCORRECT ASSUMED VALUES USED IN THE CONTROL ROOM HABITABILITY ANALYSIS. CORRECTIVE ACTIONS INCLUDED REPAIRS TO FAN FLEX JOINTS AND DOOR GASKETS THAT WERE IDENTIFIED AS AREAS OF INLEAKAGE, INVESTIGATION INTO ADDITIONAL METHODS OF FURTHER REDUCING IN-LEAKAGE AND REDOING THE CONTROL ROOM HABITABILITY ANALYSIS USING ASSUMPTIONS THAT MORE CLOSELY MATCH ACTUAL CONDITIONS, WITHIN CURRENT REGULATORY GUIDANCE.

[203] ZION 1 DOCKET 50-295 LER 91-009
MISSED RADIATION MONITOR SURVEILLANCE DUE TO SAMPLE BEING TAKEN ON WRONG MONITOR.
EVENT DATE: 050991 REPORT DATE: 061091 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 2 (PWR)

(NSIC 222374) ON 5/9/91 AT APPROX. 0015 HOURS, THE UNIT 1 LICENSED SHIFT SUPERVISOR (LSS) NOTIFIED THE RADIATION PROTECTION (RP) SUPERVISOR VIA TELEPHONE THAT THE MISCELLANEOUS VENTS NOBLE GAS RADIATION MONITOR, ORT-PR18B, WAS BEING DECLARED OUT OF SERVICE (OOS). THE RP SUPERVISOR ASSEMBLED THE NECESSARY RP OOS SURVEILLANCE PAPERWORK PER THE PHONE CONVERSATION WITHOUT WAITING FOR THE APPROPRIATE OPERATING PAPERWORK. THE RP OOS SURVEILLANCE PAPERWORK WAS INADVERTENTLY INITIATED ON OR T-PR18A, THE MISCELLANEOUS VENTS PARTICULATE RADIATION MONITOR. WHEN ALL OF THE NECESSARY PAPERWORK WAS ASSEMBLED FOR THE OOS PACKAGE NEITHER THE LSS OR THE RP SUPERVISOR REALIZED THE ERROR. THE APPROPRIATE SURVEILLANCES WERE PERFORMED ON ORT-PR18A FOR FIVE SHIFTS BUT NO OOS SURVEILLANCES WERE PERFORMED ON ORT-PR18B. ON 5/11/91 THE SAME RP SUPERVISOR WHO INITIATED THE OOS PACKAGE DISCOVERED THE ERROR. THE CORRECT PAPERWORK FOR ORT-PR18B WAS ASSEMBLED AND THE APPROPRIATE SAMPLES WERE TAKEN. THE EVENT WAS CAUSED BY MISCOMMUNICATION AND LACK OF STRICT PROCEDURE ADHERENCE BY THE RP SUPERVISOR IN INITIATING THE RADIATION MONITOR OOS SURVEILLANCE.

[204] ZION 1 DOCKET 50-295 LER 91-008
INADVERTENT SAFETY INJECTION DUE TO A.C. INSTRUMENT INVERTER FAILURE.
EVENT DATE: 051091 REPORT DATE: 061091 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 222373) AT 0020 ON MAY 10, 1991, WITH THE UNIT IN HOT STANDBY AND RTD CROSS-CALIBRATION TESTING IN PROGRESS, A.C. INSTRUMENT INVERTER 112 FAILED CAUSING THE HI-STEAM FLOW BISTABLES TO TRIP. WITH THE LO-LO BISTABLES PREVIOUSLY TRIPPED PER THE RTD CROSS-CALIBRATION TEST PROCEDURE, THE 2 OF 4 COINCIDENCE NECESSARY FOR SAFETY INJECTION ACTUATION WAS ESTABLISHED. THE OPERATORS ENTERED THE EMERGENCY PROCEDURES AND SAFETY INJECTION WAS TERMINATED AT 0029. TOTAL DURATION OF THE SAFETY INJECTION WAS 9 MINUTES. DURING THE TRANSIENT, BOTH PORVS WERE DISABLED IN THE AUTOMATIC MODE DUE TO THE LOSS OF INSTRUMENT BUS 112. THE OPERATING CREW ALERTLY OPENED BOTH PORVS MANUALLY TO CONTROL RCS PRESSURE. RCS PRESSURE PEAKED AT 2411 PSIG AND PRESSURIZER LEVEL INCREASED 7.3 PERCENT. ALL SAFETY RELATED EQUIPMENT OPERATED AS DESIGNED DURING THE EVENT. AN INVESTIGATIVE TEAM CONSISTING OF HIGHLY EXPERIENCED OFFSITE CORPORATE PERSONNEL AND ZION SYSTEM EXPERTS WAS FORMED TO INVESTIGATE THIS EVENT. DESPITE EXTENSIVE TROUBLESHOOTING AND TESTING, THE EXACT CAUSE OF THE INVERTER FAILURE HAS NOT BEEN PINPOINTED. HOWEVER, SEVERAL WEAKENED COMPONENTS WITHIN THE INVERTER WERE IDENTIFIED AND REPAIRED OR REPLACED. THOROUGH SUBSEQUENT TESTING HAS SHOWN THE INVERTER TO BE FULLY OPERABLE.

COMPONENT INDEX

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

COMPONENT INDEX

SYSTEM INDEX

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

SYSTEM INDEX

SYSTEM INDEX

KEYWORD INDEX

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

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

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VENDOR CODE INDEX

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BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

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11a. TYPE OF REPORT
Monthly Report

b. PERIOD COVERED (Inclusive dates)
August 1991

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

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

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS
Licensee Event Report
Sequence Coding and Search System
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Reactor, BWR
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Components
Operating Experience
Event Compilation

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