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Licensee Event Report (LER) Compilation

For month of January 1992

Oak Ridge National Laboratory

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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 Operations Analysis Center (NOAC) 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:

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[1] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 91-010
OFFSITE POWER SURGE POTENTIALLY UNABLE TO MAINTAIN ADEQUATE VOLTAGE TO SAFETY LOADS DURING SPECIFIC ACCIDENT CONDITIONS DUE TO INCREASED OFFSITE GRID LOADING.
EVENT DATE: 100591 REPORT DATE: 110591 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 2 (PWR)

(NSIC 223432) ON OCTOBER 5, 1991, AN ANO ENGINEERING EVALUATION IDENTIFIED SPECIFIC CONDITIONS WHICH COULD JEOPARDIZE THE CAPABILITY OF AN OFFSITE POWER SOURCE TO MEET THE REQUIREMENTS OF GENERAL DESIGN CRITERIA 17 FOR ANO-1 AND ANO-2. IT WAS DETERMINED THAT DURING PEAK SUMMERTIME LOAD CONDITIONS THE 161 KV OFFSITE POWER SOURCE (STARTUP TRANSFORMER 2) MIGHT NOT BE ABLE TO MAINTAIN ADEQUATE VOLTAGE TO ANO LOADS DURING ACCIDENT CONDITIONS WITH BOTH UNITS OFF LINE AND THE 500KV POWER SOURCE (AUTOTRANSFORMER) UNAVAILABLE. THIS CONDITION DEVELOPED AS A RESULT OF INCREASED 161KV GRID LOADING WITH TIME. THE ROOT CAUSE OF THIS CONDITION IS STILL BEING EVALUATED AND WILL BE INCLUDED IN A SUPPLEMENT TO THIS REPORT WHICH IS EXPECTED TO BE SUBMITTED BY MARCH 30, 1992. CORRECTIVE ACTIONS INCLUDED PLACING THE FEEDER BREAKERS FROM STARTUP TRANSFORMER 2 IN THE "PULL TO LOCK" POSITION, WHICH PREVENT SIT FROM AUTOMATICALLY PICKING UP ANO LOADS DUE TO LOSS OF POWER TO THE 6.9KV OR 4.16KV BUSES. PROCEDURAL GUIDANCE WAS ALSO IMPLEMENTED TO RESTRICT THE LOADING OF STARTUP TRANSFORMER 2 WHEN THE 500KV SOURCE IS UNAVAILABLE TO ENSURE ITS CAP ABILITY TO SUPPLY ADEQUATE VOLTAGE TO SAFETY LOADS. THIS REPORT IS ALSO INTENDED TO MEET THE SPECIAL REPORTING REQUIREMENTS OF ANO-1 TECHNICAL SPECIFICATION 3.7.2.H.

[2] BEAVER VALLEY 1 DOCKET 50-334 LER 91-029
AUXILIARY FEEDWATFR PUMP ACTUATION DUE TO LOW-LOW STEAM GENERATOR LEVEL.
EVENT DATE: 110691 REPORT DATE: 120691 NSSS: WE TYPE: PWR

(NSIC 223521) ON 11/6/91 WHILE IN MODE 3, HEAT UP OF THE REACTOR COOLANT SYSTEM (RCS) WAS TERMINATED AND A PLANT COOLDOWN TO MODE 5 WAS COMMENCED. STEAM GENERATOR LEVELS WERE BEING MAINTAINED WITH CONDENSATE PUMP FLOW THROUGH THE BYPASS FEEDWATER REGULATING VALVES. A MAIN STEAM ISOLATION VALVE BYPASS VALVE WAS OPENED TO FACILITATE THE RCS COOLDOWN. CONCURRENTLY, STEAM GENERATOR BLOWDOWN SYSTEM FLOW WAS INCREASED TO AID IN STEAM GENERATOR CHEMISTRY CONTROL. STEAM GENERATOR LEVELS BEGAN TO DECREASE. OPERATORS TOOK MANUAL CONTROL OF THE BYPASS FEEDWATER REGULATING VALVES BUT WERE UNABLE TO RESTORE STEAM GENERATOR LEVEL. GENERATOR LEVEL CONTINUED TO DECREASE UNTIL THE A STEAM GENERATOR REACHED THE LOW-LOW LEVEL SETPOINT AND STARTED THE STEAM DRIVEN AUXILIARY FEEDWATER PUMP. STEAM GENERATOR LEVELS WERE RESTORED TO NORMAL PRIOR TO SECURING THE AUXILIARY FEEDWATER PUMP. THE CAUSE OF THE EVENT WAS DUE TO A LACK OF SECONDARY FLOW TO THE STEAM GENERATORS. CONDENSATE FLOW WAS INEFFECTIVE DUE TO THE STEAM GENERATOR PRESSURE BEING NEAR THE SHUTOFF HEAD OF THE CONDENSATE PUMPS. AT NO TIME WAS THE MARGIN OF SAFETY TO THE GENERAL PUBLIC REDUCED, AS THE AUXILIARY FEEDWATER PUMP STARTED AS DESIGNED. THE EVENT WILL BE REVIEWED IN A FUTURE OPERATOR LICENSE RETRAINING SESSION.

[3] BRAIDWOOD 1 DOCKET 50-456 LER 91-012
REACTOR TRIP AS A RESULT OF A FEEDWATER PUMP TRIP DUE TO A LOSS OF EH FLUID PRESSURE.
EVENT DATE: 110691 REPORT DATE: 120591 NSSS: WE TYPE: PWR

(NSIC 223580) PRIOR TO THE EVENT, SIGNIFICANT LEAKAGE HAD BEEN IDENTIFIED ON MAIN FEEDWATER (FW) MOTOR-OPERATED ISOLATION VALVE 1FW006A. THE VALVE IS IN THE FLOWPATH TO THE 1A STEAM GENERATOR (S/G). ATTEMPTS TO REDUCE LEAKAGE HAD BEEN UNSUCCESSFUL DUE TO THE HIGH TEMPERATURE AND PRESSURE OF THE WATER SPRAYING FROM THE VALVE BODY. REPAIRING THE VALVE REQUIRED A UNIT SHUTDOWN. THE PLANT WAS SCHEDULED TO BE TAKEN OFF-LINE ON 11/2/91. ON 11/1/91 DUE TO A LOW RESERVE MARGIN ON THE COMMONWEALTH EDISON CO. SYSTEM GRID, THE LOAD DISPATCHER REQUESTED THAT THE OUTAGE BE POSTPONED. THE LEAKAGE WAS NOT AFFECTING UNIT CAPACITY, SINCE ALL S/G LEVELS COULD BE MAINTAINED AT THEIR NORMAL LEVEL (66%). AT 0013 ON 11/6/91 AN ELECTRO-HYDRAULIC (EH) FLUID LEAK TRIPPED THE 1B FW PUMP. CONTROL ROOM PERSONNEL INITIATED A TURBINE RUNBACK. S/G LEVELS IN ALL FOUR S/G'S DECREASED DUE TO THE REDUCTION IN TOTAL FW FLOW. AFTER THE TURBINE RUNBACK WAS COMPLETED, S/G LEVELS IN ALL FOUR S/G'S DECREASED DUE TO THE REDUCTION IN TOTAL FW FLOW.

AFTER THE TURBINE RUNBACK WAS COMPLETED, S/G LEVELS IN THE 1B, 1C AND 1D S/G'S STABILIZED AT 45%. HOWEVER, THE 1A S/G LEVEL CONTINUED TO DECREASE. AT 0017 THE LEVEL IN THE 1A S/G REACHED THE LOW-LOW LEVEL REACTOR TRIP SETPOINT (40.8%) AND A REACTOR TRIP OCCURRED. THE ROOT CAUSE OF THE 1B FW PUMP TRIP WAS A CRACK IN A 5/8 INCH SUPPLY LINE. THE LINE WAS REPAIRED.

[4] BROWNS FERRY 2 DOCKET 50-260 LER 91-018
FAILURE TO OPEN THE GENERATOR EXCITER FIELD BREAKER AFTER MANUAL TURBINE TRIP RESULTED IN A REACTOR SCRAM REQUIRED BY TECHNICAL SPECIFICATION.
EVENT DATE: 101891 REPORT DATE: 111591 NSSS: E TYPE: BWR

(NSIC 223471) ON OCTOBER 18, 1991 AT 0458 HOURS, DURING A CONTROLLED SHUTDOWN, BROWNS FERRY UNIT 2 REACTOR WAS MANUALLY SCRAMMED FROM APPROXIMATELY EIGHT PERCENT THERMAL POWER DUE TO UNEXPECTED EQUIPMENT RESPONSES. THE ROOT CAUSE OF THIS EVENT WAS INAPPROPRIATE PERSONNEL ACTION. A PLANT OPERATOR FAILED TO FOLLOW THE PROCEDURE FOR TURBINE GENERATOR SHUTDOWN WHEN HE DID NOT OPEN THE EXCITER FIELD BREAKER AFTER THE MANUAL TURBINE GENERATOR TRIP. THIS ACTION CAUSED UNEXPECTED EQUIPMENT RESPONSES INCLUDING ACTUATION OF THE GENERATOR REVERSE POWER RELAY AND SUBSEQUENT TRIPPING OF THE REACTOR RECIRCULATION PUMPS. THE REVERSE POWER RELAY WAS TESTED. THE EVALUATION AND TESTING OF THE RECIRCULATION PUMP BOARDS WAS COMPLETED. OPERATIONS PERSONNEL WILL RECEIVE LIVE-TIME TRAINING ON THE EVENT. TVA WILL CHANGE THE PLANT DESIGN TO INCLUDE AUTOMATIC TRIPPING OF THE GENERATOR FIELD BREAKER WHEN THE TURBINE GENERATOR IS TRIPPED. THE BROWNS FERRY SIMULATOR WILL BE UPGRADED TO MATCH PLANT CONDITIONS.

[5] BROWNS FERRY 3 DOCKET 50-296 LER 91-004
INABILITY TO IDENTIFY AND ESTABLISH COMPENSATORY MEASURES FOR LOSS OF HIGH PRESSURE FIRE PROTECTION SYSTEM HOSE STATIONS WHICH LEAD TO A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS.
EVENT DATE: 110491 REPORT DATE: 120391 NSS.. GE TYPE: BWR

(NSIC 223493) ON NOVEMBER 4, 1991 AT 1310 HOURS, THE UNIT 3 CONTROL ROOM WAS NOTIFIED OF A LEAK IN A THREE-INCH HIGH PRESSURE FIRE PROTECTION SYSTEM PIPE. A UTILITY-LICENSED ASSISTANT SHIFT OPERATIONS SUPERVISOR DISPATCHED TO THE SCENE ISOLATED THE LEAK. PARALLEL ACTIONS WERE TAKEN TO INITIATE REQUIRED COMPENSATORY MEASURES; HOWEVER, REQUIRED COMPENSATORY MEASURES WERE NOT ESTABLISHED WITHIN ONE HOUR AS REQUIRED BY TECHNICAL SPECIFICATIONS. THE ROOT CAUSE OF THIS EVENT WAS AN INADEQUATE METHOD FOR IDENTIFYING WHICH PARTS OF THE SYSTEM REQUIRED ONE HOUR COMPENSATORY ACTIONS AND WHICH ALLOWED 24 HOURS. THIS RESULTED IN AN EXCESSIVE AMOUNT OF TIME REQUIRED TO IDENTIFY AND IMPLEMENT THE ONE HOUR COMPENSATORY MEASURES FOR ISOLATED FIRE HOSE STATIONS. CONTRIBUTING TO THE FAILURE TO MEET COMPENSATORY MEASURE REQUIREMENTS IS THE EXISTING DESIGN OF THE OPEN NOZZLE FIXED SPRAY SYSTEMS. THE IMMEDIATE CORRECTIVE ACTIONS INCLUDED ISOLATING THE LEAK, INITIATING ESTABLISHMENT OF COMPENSATORY ACTIONS AND THE INSTALLATION OF A TEMPORARY PATCH. TVA WILL DEVELOP A METHOD TO ENABLE PERSONNEL TO BEGIN COMPENSATORY MEASURES IMMEDIATELY WHEN A PORTION OF THE HIGH PRESSURE FIRE PROTECTION IS ISOLATED. AN APPROPRIATE AMOUNT OF COMPENSATORY ACTION HOSE KITS WILL BE STAGED. TVA HAS STAGED PATCH KITS.

[6] BRUNSWICK 1 DOCKET 50-325 LER 91-026
REACTOR SHUTDOWN REQUIRED WHEN EMERGENCY DIESEL GENERATOR MAINTENANCE WAS DETERMINED TO REQUIRE MORE TIME THAN THE TECHNICAL SPECIFICATION LIMITING CONDITION OF OPERATION WOULD ALLOW WITH A UNIT IN POWER.
EVENT DATE: 101591 REPORT DATE: 111591 NSSS: GE TYPE: BWR
VENDOR: WISCONSIN BRIDGE & IRON

(NSIC 223503) AT 0700 ON OCTOBER 1, 1991, #3 EMERGENCY DIESEL GENERATOR (EDG) WAS DECLARED INOPERABLE TO ALLOW FOR SCHEDULED MAINTENANCE AS PART OF THE UNIT 2 REFUELING OUTAGE. BY 0600 ON 10/15/91, IT WAS APPARENT THAT THE #3 EDG WOULD NOT BE RETURNED TO SERVICE IN TIME TO AVOID THE REACTOR SHUTDOWN REQUIRED BY THE LIMITING CONDITION FOR OPERATION (LCO) ACTION STATEMENT OF THE ELECTRICAL POWER SYSTEMS A. C. SOURCES TECHNICAL SPECIFICATION. THE TIME REQUIRED TO COMPLETE THE ORIGINAL WORK SCOPE THAT INCREASED DUE TO THE ITEMS DETAILED IN A MANAGEMENT

MEETING HELD WITH THE NRC STAFF AT THE BRUNSWICK SITE ON 11/01/91. ON 10/15/91 AT 0600, UNIT 1 WAS AT 100% REACTOR POWER WITH THE EMERGENCY CORE COOLING SYSTEMS (ECCS) OPERABLE. UNIT 2 WAS DEFUELED IN A REFUELING OUTAGE. THE DECISION WAS MADE TO COMMENCE A REACTOR SHUTDOWN BEGINNING AT 0630 ON 10/15/91. UNIT 1 ENTERED HOT SHUTDOWN AT 1803 ON 10/15/91, AND COLD SHUTDOWN AT 1740 ON 10/16/91. UPON COMPLETION OF 3 EDG MAINTENANCE ADD TESTING, THE EDG WAS DECLARED OPERABLE AT 1850 ON 10/20/91. UNIT 1 REACTOR STARTUP WAS COMMENCED AT 0048 ON 10/21/91 AND THE UNIT WAS SYNCHRONIZED TO THE SYSTEM GRID AT 1724 ON 10/21/91.

[7] BRUNSWICK 1 DOCKET 50-325 LER 91-027
TWO INOPERABLE CONTROL ROD ACCUMULATORS RESULT IN ENTRY INTO TECHNICAL SPECIFICATION 3.0.3.
EVENT DATE: 110591 REPORT DATE: 112791 NSSS: GE TYPE: BWR

(NSIC 223502) ON NOVEMBER 5, 1991, UNIT 1 WAS AT 100% REACTOR POWER. AT 0040 ON NOVEMBER 5, 1991, AS CONTROL ROD DRIVE (CRD) HYDRAULIC CONTROL UNIT (HCU) 34-19 WAS BEING RECHARGED WITH NITROGEN, A SECOND HCU ACCUMULATOR 46-27 LOW NITROGEN PRESSURE CAUSED A CRD ACCUMULATOR LOW PRESSURE/HIGH LEVEL ALARM IN THE CONTROL ROOM. WITH TWO OR MORE CONTROL ROD DRIVE SCRAM ACCUMULATORS INOPERABLE, TECHNICAL SPECIFICATION 3.0.3 WAS INVOKED. THIS REQUIRED PLACING THE UNIT IN A SIX HOUR TO HOT SHUTDOWN CONDITION. AFTER VERIFYING THE SECOND ALARM WAS DUE TO LOW NITROGEN PRESSURE, THE AUXILIARY OPERATOR CONTINUED RECHARGING THE FIRST HCU 34-19 WITH NITROGEN. THERE WAS NO INTENT TO COMMENCE A UNIT SHUTDOWN UNLESS PROBLEMS WERE ENCOUNTERED DURING THE RECHARGING EVOLUTION. AT 0055, ON NOVEMBER 5, 1991, THE ACCUMULATOR RHCL 34-19 WAS SUCCESSFULLY RECHARGED AND TECHNICAL SPECIFICATION 3.0.3 WAS EXITED. THE CAUSE OF THIS EVENT IS THE SIMULTANEOUS INOPERABILITY OF TWO CONTROL ROD DRIVE HYDRAULIC CONTROL UNITS DUE TO THE DROP IN THE REACTOR BUILDING AMBIENT TEMPERATURE. A COLD FRONT RESULTED IN AREAS OF THE REACTOR BUILDING REACHING NEW SEASONAL LOWS (AS RECORDED BUILDING TEMPERATURES DROPPED APPROX. 7 DEG F OVER THE PREVIOUS 3 DAYS). PREVIOUS LER WAS 1-91-015.

[8] BRUNSWICK 1 DOCKET 50-325 LER 91-028
COMPONENT FAILURE OF A REACTOR WATER CLEANUP SYSTEM ISOLATION LOGIC RELAY RESULTED IN AN UNPLANNED ENGINEERED SAFETY FEATURE ACTUATION.
EVENT DATE: 110591 REPORT DATE: 120591 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223501) ON NOVEMBER 5, 1991, AT APPROX. 1518 HOURS (EST), THE UNIT 1 REACTOR WAS AT 100% POWER. THE REACTOR WATER CLEANUP (RWCU) SYSTEM WAS IN SERVICE WITH BOTH FILTER/DEMINERALIZES ON-LINE. THE EMERGENCY CORE COOLING SYSTEMS (ECCS), THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM, AND THE EMERGENCY DIESEL GENERATORS (EDGS) WERE OPERABLE AND IN STANDBY READINESS. AT THIS TIME, THE PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) RECEIVED A SPURIOUS VALVE GROUP 3 ISOLATION SIGNAL RESULTING IN THE AUTOMATIC CLOSURE OF THE RWCU SYSTEM INLET INBOARD ISOLATION VALVE (1-G31-F001). THE UNIT CONTROL OPERATOR (CO) BECAME AWARE OF THIS UNPLANNED ESF ACTUATION BY CONTROL ROOM ANNUNCIATION AND INDICATION. IE SPURIOUS ISOLATION SIGNAL WAS GENERATED PER DESIGN WHEN A NORMALLY ENERGIZED RWCU ISOLATION LOGIC RELAY (1-A71-R37) COIL BURNED UP CAUSING FAILURE IN THE DEENERGIZED DEPOSITION, WHICH THEN ACTUATED THAT PORTION OF THE RWCU LEAK DETECTION LOGIC THAT COMMANDS 1-G31-F001 TO CLOSE. INSPECTION OF THE CONTROL ROOM BACKPANEL AREAS REVEALED A FAINT BURNING SMELL; HOWEVER, THERE WAS NO VISIBLE SMOKE OR FIRE FROM THE FAILED GENERAL ELECTRIC (GE) MODEL NO. CR120 RELAY. THE RELAY FAILURE WAS DETERMINED TO BE NORMAL END OF COIL LIFE FAILURE DUE TO AGING. THE COIL FOR THE FAILED RELAY WAS REPLACED WITH A NEW ONE FROM STOCK FOLLOWING SATISFACTORY POSTMAINTENANCE TESTING.

[9] BRUNSWICK 2 DOCKET 50-324 LER 91-005 REV 01
UPDATE ON ESF ACTUATIONS CAUSED BY VOLTAGE REGULATOR TRANSIENT WITH FAILURE OF PRIMARY CONTAINMENT ISOLATION SOLENOID OPERATED VALVES.
EVENT DATE: 063091 REPORT DATE: 112691 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 1 (BWR)
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)

(NSIC 2235\5) A SURVEILLANCE TO CHECK THE OPERATION OF THE MANUAL AND AUTO MAIN GENERATOR VOLTAGE REGULATORS WAS IN PROGRESS. FOLLOWING A STEP OF THE PROCEDURE WHICH REQUIRED THE VOLTAGE REGULATOR MODE SELECTOR SWITCH TO BE PLACED IN THE MANUAL POSITION, THE TRANSFER VOLTMETER DEFLECTED SIGNIFICANTLY. EMERGENCY BUS E4 WAS LOST AND THE REACTOR PROTECTION SYSTEM MOTOR-GENERATOR SET 2B TRIPPED. EMERGENCY DIESEL GENERATOR #4 AUTO-STARTED AND A HALF SCRAM SIGNAL WAS RECEIVED ALONG WITH PRIMARY CONTAINMENT SYSTEM ISOLATION SIGNALS; GROUP 3 (REACTOR WATER CLEANUP), GROUP 6 (CONTAINMENT ATMOSPHERIC CONTROL), HALF GROUP 1 (MAIN STEAM LINE), GROUP 2-DIVISION II (TRANSVERSE INCORE PROBE AND DRYWELL FLOOR/EQUIPMENT DRAINS). MAIN GENERATOR OUTPUT VOLTAGE WAS RAISED USING THE MANUAL VOLTAGE REGULATOR AND THE VOLTAGE REGULATOR WAS TRANSFERRED TO THE AUTOMATIC MODE. ESF ACTUATIONS OCCURRED EXCEPT FOR THE OUTBOARD DRYWELL FLOOR DRAIN ISOLATION VALVE WAS CLOSED AFTER FOUR ATTEMPTS AND THE LINE WAS ISOLATED. THE CAUSE OF THIS EVENT IS STILL UNDER INVESTIGATION. DUE TO THE PLANT BEING AT FULL POWER, THE DECISION WAS MADE TO ALLOW THE VOLTAGE REGULATOR TO REMAIN IN THE AUTOMATIC MODE UNTIL TROUBLESHOOTING TAKES PLACE DURING THE SEPTEMBER, 1991 SCHEDULED SHUTDOWN FOR A REFUELING. SIMILAR LERS ARE 2-88-001, 2-81-01, 1-86-024, AND 2-90-015.

[10] BRUNSWICK 2 DOCKET 50-324 LER 91-017
SAFETY RELIEF VALVES TESTED IN WYLE LABORATORIES EXCEEDED TECHNICAL SPECIFICATION LIMIT DUE TO PILOT DISC-TO-SEAT BONDING.
EVENT DATE: 101791 REPORT DATE: 111691 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 223506) UNIT 2 WAS SHUT DOWN AND IN DAY 36 OF A 77 DAY SCHEDULED REFUELING OUTAGE. 11 SAFETY RELIEF VALVE (SRV) PILOT ASSEMBLIES WERE REMOVED FROM THE UNIT AND SHIPPED TO WYLE LABORATORIES FOR SET PRESSURE TESTING AND RECERTIFICATION. 'AS-RECEIVED' TESTING WAS PERFORMED BY WYLE LABORATORIES USING WYLE TEST PROCEDURE NO. 1025 WITH THE FOLLOWING RESULTS: TWO OF THE SRV PILOTS HAD A SETPOINT DRIFT OF LESS THAN 1%, FIVE OF THE SRV PILOTS HAD A SETPOINT DRIFT OF BETWEEN 1.6% AND 3.6%, ONE SRV PILOT HAD A SETPOINT DRIFT OF 8.6%, THE REMAINING THREE SRVE PILOTS DID NOT INITIALLY LIFT AT STEAM TEST PRESSURES OF 1207, 1303 AND 1260 PSIG RESPECTIVELY. SPECIAL DIAGNOSTIC TESTING WAS PERFORMED ON THE THREE SRV PILOT ASSEMBLIES THAT DID NOT INITIALLY LIFT. THE AVERAGE PERCENT DRIFT FOR ALL SRV'S IS 7.0%, WITH THE HIGHEST DRIFT FOR ANY SRV AT 20.2%. AN ENGINEERING EVALUATION HAS DETERMINED THAT OXYGEN INDUCED PILOT-DISC-TO-SEAT BONDING PREVENTED THE THREE PILOTS FROM INITIALLY ACTUATING. FOUR OF THE PILOT DISCS HAVE BEEN REPLACED. PH13-8MO DISC HAS BEEN REPLACED WITH A STELLITE 6B DISC. OTHER SRV PILOTS SENT TO WYLE HAVE ALSO BEEN RECERTIFIED. CP&L WILL REQUEST A T.S. CHANGE FOR SETPOINT TOLERANCES FROM +/- 1% TO +/- 3%.

[11] BRUNSWICK 2 DOCKET 50-324 LER 91-018
REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TRAIN START DUE TO LOW LEVEL #2 ACTUATION CAUSED BY REACTOR LEVEL INSTRUMENTATON REFERENCE LEG PERTURBATION.
EVENT DATE: 102091 REPORT DATE: 111991 NSSS: GE TYPE: BWR

(NSIC 223504) UNIT 2 WAS IN REFUEL (MODE 5) WITH THE REACTOR VESSEL DEFUELED. AT 1208 ON 10/20/91, WHILE TURNING TO SERVICE AN A LOOP REACTOR PRESSURE INDICATOR FOLLOWING TECHNICAL SPECIFICATION REQUIRED CALIBRATION TESTING, A HYDRAULIC PERTURBATION OF A SHARED REFERENCE LEG OCCURRED. THE PERTURBATION RESULTED IN ISOLATION OF THE UNIT 2 REACTOR BUILDING VENTILATION SYSTEM AUTOMATIC START OF THE 2A AND 2B STANDBY GAS TRAINS (BG). SPECIAL PROCEDURE WAS DEVELOPED TO SUPPORT ROOT CAUSE DETERMINATION. THE RESULTS OF THE TECHNICAL PROCEDURE INDICATE THAT THE PERTURBATION WAS DUE TO A LACK OF ATTENTION TO DETAIL WHILE VENTING APPLIED CALIBRATION PRESSURE FROM A REACTOR PRESSURE INDICATOR. IMMEDIATE CORRECTIVE ACTION WAS IMPLEMENTED TO ENSURE THAT THE B LOOP REACTOR PRESSURE INDICATOR LIBERATION PROCESS WAS MONITORED BY THE RESPONSIBLE MAINTENANCE SUPERVISOR. ADDITIONAL CORRECTIVE ACTIONS INCLUDE COUNSELING OF THE RESPONSIBLE INSTRUMENTATION AND CONTROL (I&C) TECHNICIAN AND PERFORMANCE OF A HUMAN PERFORMANCE EVALUATION. A REVIEW OF THE EVENT WILL BE CONDUCTED WITH APPROPRIATE MAINTENANCE PERSONNEL TO STRENGTHEN THEIR AWARENESS OF THE POTENTIAL FOR PERTURBATION OF REACTOR VESSEL LEVEL INSTRUMENTATION DURING SHUTDOWN CONDITIONS.

[12] BYRON 1 DOCKET 50-454 LER 91-004
INADVERTENT SAFETY INJECTION DURING REFUELING DUE TO PERSONNEL ERROR.
EVENT DATE: 101691 REPORT DATE: 111591 NSSS: WE TYPE: PWR

(NSIC 223409) ON OCTOBER 16, 1991 AT APPROXIMATELY 1304, WITH UNIT 1 IN MODE 6 (REFUELING) A NUCLEAR STATION OPERATOR (NSO) WAS ATTEMPTING TO RESET A STATUS LIGHT FOR "AUTO SI BLOCKED" FOLLOWING A TRAIN A ENGINEERED SAFEGUARDS FEATURES ACTUATION SYSTEM (ESFAS) SURVEILLANCE. CYCLING THE REACTOR TRIP BREAKER CONTROL SWITCH DID NOT RESET THE LIGHT, BECAUSE THE BREAKERS WERE OUT-OF-SERVICE. THE NSO THEN RESET THE BLOCK ON LOW STEAMLINE PRESSURE SI. TRAIN B SI ACTUATED ON LOW STEAMLINE PRESSURE. THE 1B DIESEL GENERATOR AND SEVERAL OTHER PIECES OF EQUIPMENT ACTUATED. MOST OF THE TRAIN B SI EQUIPMENT WAS OUT-OF-SERVICE FOR THE OUTAGE AND DID NOT ACTUATE. THE SI SIGNAL WAS RESET AND EQUIPMENT WAS RESTORED. THE CAUSE OF THE EVENT WAS COGNITIVE PERSONNEL ERROR. CONTRIBUTING TO THIS EVENT WAS A LACK OF GUIDANCE IN THE PROCEDURES ON HOW TO CLEAR THIS LIGHT WHEN THE SURVEILLANCE WAS PERFORMED WITH THE REACTOR TRIP BREAKERS OPEN AS DURING AN OUTAGE. THE PROCEDURE WILL BE REVISED. ADDITIONAL TRAINING WILL BE PROVIDED ON OPERATION OF THE AUTO SI BLOCK. THIS EVENT IS REPORTABLE PER 10CFR50.73 (A)(2)(IV) ANY EVENT OR CONDITION THAT RESULTED IN MANUAL OR AUTOMATIC ACTUATION OF ANY ENGINEERED SAFEGUARDS FEATURE.

[13] BYRON 1 DOCKET 50-454 LER 91-005
MODE CHANGE MADE WITH CONTAINMENT SPRAY SYSTEM INOPERABLE DUE TO PERSONNEL ERROR.
EVENT DATE: 102791 REPORT DATE: 112091 NSSS: WE TYPE: PWR

(NSIC 223579) ON OCTOBER 27, 1991 AT 1655, HOT SHUTDOWN (MODE 4) WAS ENTERED WITH THE CONTAINMENT SPRAY SYSTEM AND EMERGENCY CORE COOLING SYSTEM RECIRCULATION SUMP ISOLATION VALVES INOPERABLE. TECHNICAL SPECIFICATIONS REQUIRE THESE OPERABLE FOR A MODE CHANGE FROM COLD SHUTDOWN (MODES) TO HOT SHUTDOWN (MODE 4). THIS EVENT WAS CAUSED BY FAILURE OF CONTROL ROOM PERSONNEL TO FOLLOW THE PROCEDURE THAT CONTROLLED THE MODE CHANGE. SHIFT PERSONNEL DIRECTED THEIR ATTENTION TO COMPLETING THE ASSOCIATED MODE CHANGE CHECKLIST WHICH IS A SUB-DOCUMENT OF THE MODE CHANGE PROCEDURE. MODE CHANGE CHECKLISTS WILL BE REVISED TO SPECIFICALLY REFERENCE THE GOVERNING MODE CHANGE PROCEDURE TO ENSURE ALL APPROPRIATE STEPS ARE COMPLETED PRIOR TO CHANGING MODES. IN ADDITION, SHIFT BRIEFINGS WILL BE HELD PRIOR TO MODE CHANGES TO DISCUSS RELATED ACTIVITIES AND REQUIREMENTS. THIS EVENT IS REPORTABLE PURSUANT TO 10C-R50.73(A)(2)(II) AN EVENT ON CONDITION THAT RESULTS IN BEING IN AN UNANALYZED CONDITION.

[14] BYRON 2 DOCKET 50-455 LER 90-006 REV 01
UPDATE ON INADVERTENT TRAIN "A" SAFETY INJECTION SIGNAL DUE TO MISCOMMUNICATION AND PROCEDURAL DEFICIENCY.
EVENT DATE: 090390 REPORT DATE: 103091 NSSS: WE TYPE: PWR
VENDOR: GOULDS CO.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 223343) ON 9-03-90 AT APPROX. 0805, WITH UNIT 2 IN MODE 5 2BOS 3.2.1.1.A-1, WAS BEING PERFORMED PER THE REFUELING OUTAGE SCHEDULE. AFTER PERFORMING THE NORMAL SAFETY INJECTION (SI) (JE). IT WAS NOTED THAT THE 2C REACTOR CONTAINMENT FAN COOLER (RCFC) LOW SPEED FAN BREAKER DID NOT CLOSE. ATTEMPTS TO CLOSE THE BREAKER WERE UNSUCCESSFUL. AT 0820, THE 480 VOLT BUS THAT FEEDS THE BREAKER WAS DE-ENERGIZED TO ALLOW REMOVAL OF THE BREAKER. AT 0850, WHILE STRIPPING THE BUS OF ITS ALTERNATING CURRENT LOAD, INSTRUMENT INSERTERS 211 AND 213 WERE DE-ENERGIZED DUE TO A COMMUNICATIONS BREAKDOWN. WHEN THE INSTRUMENT BUSES WERE DE-ENERGIZED, THE PRESSURIZER PRESSURE LOW SI AND STEAMLINE PRESSURE LOW SI BLOCKS WERE LOST ON TRAIN A. THE UNIT REACTOR OPERATOR WAS UNAWARE THAT THE BLOCKS HAD BEEN LOST AND THE SURVEILLANCE DID NOT CONTAIN AN EMERGENCY EXIT SECTION TO PROVIDE RESTORATION GUIDANCE. AT 0902, THE TRAIN A REACTOR TRIP BREAKER WAS CLOSED PER THE SURVEILLANCE AND A SAFETY INJECTION SIGNAL RESULTED DUE TO A LOSS OF THE REACTOR TRIP INTERLOCK (P-4) WHILE CYCLING THE REACTOR TRIP BREAKER CORRECTIVE ACTIONS INCLUDE A PROCEDURE REVISION TO THE MANUAL SI SURVEILLANCE TO INCLUDE AN EMERGENCY EXIT SECTION. THIS EVENT WILL ALSO BE INCLUDED IN OPERATOR REQUIRED LISTENING.

[15] BYRON 2 DOCKET 50-455 LER 91-003
EXCESSIVE RCS LEAKAGE DUE TO A VALVE PACKING LEAK.
EVENT DATE: 102791 REPORT DATE: 112091 NSSS: WE TYPE: PWR
VENDOR: ANDERSON, GREENWOOD & CO.

(NSIC 223562) ON OCTOBER 26, 1991, UNIT 2 EXPERIENCED A REACTOR COOLANT SYSTEM (RCS) LEAK WHICH WAS IN EXCESS OF TECHNICAL SPECIFICATION REQUIREMENTS FOR AN UNIDENTIFIED LEAK. THE LEAK WAS INDICATED BY A HIGH LEAK DETECTION ALARM FOR THE CONTAINMENT FLOOR DRAIN, AND WAS VERIFIED USING THE 2BOS 4.6.2.1.D-1 RCS LEAK RATE TEST. THE LEAK WAS APPROXIMATELY 2 GPM AND WAS BELIEVED TO BE LOCATED IN THE 2C RTD BYPASS MANIFOLD AREA BECAUSE THE RTD BYPASS FLOW LOW ALARM FOR LOOP C HAD ANNUNCIATED SHORTLY BEFORE LEAKAGE INCREASED. LCOAR 2BOS 4.6.2-1A WAS ENTERED AT 0122 ON 10/27/91. AN INITIAL CONTAINMENT ENTRY WAS MADE TO INVESTIGATE THE SUSPECTED AREA OF LEAKAGE OUTSIDE OF THE MISSILE BARRIER. NO MAJOR CONTRIBUTIONS TO THE LEAK WERE FOUND. THE PLANT STARTED A POWER RAMP DOWN TO SUPPORT A PENDING CONTAINMENT ENTRY INSIDE THE MISSILE BARRIER. SHORTLY AFTER ENTRY INTO MODE 3 (HOT STANDBY), A CONTAINMENT ENTRY INSIDE THE MISSILE BARRIER INDICATED THAT THE 2RC029C VALVE HAD A PACKING LEAK. THE VALVE WAS ISOLATED AND SUBSEQUENTLY REPAIRED BY MECHANICAL MAINTENANCE. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(A).

[16] CALLAWAY 1 DOCKET 50-483 LER 91-006
A REACTOR TRIP DUE TO A FAILURE OF A GATING/SEQUENCING CARD IN THE INVERTER FOR A 120 VOLT AC INSTRUMENT BUS.
EVENT DATE: 110591 REPORT DATE: 120591 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELEC CORP.-NUCLEAR ENERGY SYS

(NSIC 223567) ON 11/5/91, AT 1031 CST, A REACTOR TRIP OCCURRED DUE TO THE FAILURE OF A GATING/SEQUENCING CARD IN THE INVERTER (NN12) SUPPLYING A 120 VOLT AC SAFETY-RELATED INSTRUMENT BUS (NN02). THE NN02 BUS WAS DE-ENERGIZED. THIS FAILED A CONTROLLING FEEDWATER CHANNEL RESULTING IN A HIGH WATER LEVEL IN THE 'A' STEAM GENERATOR WHICH CAUSED A TURBINE TRIP SIGNAL. THE REACTOR TRIP OCCURRED ON A UNIT TRIP/TURBINE TRIP SIGNAL. A FEEDWATER ISOLATION AND AN AUXILIARY FEEDWATER ACTUATION WERE GENERATED BY DESIGN. THE PLANT WAS IN MODE 1 - POWER OPERATIONS AT 100 PERCENT REACTOR POWER. THE REACTOR COOLANT SYSTEM TEMPERATURE WAS 588 DEGREES F AND THE PRESSURE WAS 2237 PSIG. THE LICENSED OPERATORS RECOVERED FROM THE TRIP AND THE ENGINEERED SAFETY FEATURE ACTUATIONS VIA PLANT PROCEDURES. THE NN02 BUS WAS ENERGIZED FROM BACKUP POWER VIA THE SOLA TRANSFORMER AT 1109. THE FAILED INVERTER CARD WAS REPLACED AND THE INVERTER LINED UP TO NN02 AT 0242 ON 11/6/91. THE PLANT WAS RETURNED TO MODE 1 - POWER OPERATIONS AT 2018 ON 11/6/91. CORRECTIVE ACTIONS INCLUDE: AN ANALYSIS OF THE CARD FAILURE; A DETERMINATION IF ADDITIONAL PREVENTIVE MAINTENANCE FOR THIS CARD IS NECESSARY; PROCEDURES WILL BE, DEVELOPED WITH NN BUS LOAD INFORMATION FOR OPERATOR USE; AND OPERATOR TRAINING ON THIS TYPE OF EVENT WILL BE DEVELOPED AND PERFORMED.

[17] CALVERT CLIFFS 2 DOCKET 50-318 LER 91-008
STEAM GENERATOR EFFLUENT FLOW RATE NOT ESTIMATED EVERY FOUR HOURS PER TS ACTION STATEMENT DUE TO OPERATOR ERROR AND INSUFFICIENT PROCEDURE CLARITY.
EVENT DATE: 103091 REPORT DATE: 112791 NSSS: CE TYPE: PWR

(NSIC 223512) ON OCTOBER 30, 1991, AT APPROXIMATELY 0810, A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS (TS) WAS IDENTIFIED AT CALVERT CLIFFS UNIT 2. WHILE DRAINING STEAM GENERATOR (SG) NO. 21, FLOW RATE WAS NOT ESTIMATED EVERY 4 HOURS WHEN THE DISCHARGE FLOW RATE MEASURING DEVICE WAS NOT OPERABLE AS REQUIRED BY TS 3.3.3.10. AT THE TIME, UNIT 2 WAS IN COLD SHUTDOWN AT A TEMPERATURE OF 102 DEGREES FAHRENHEIT AND A REACTOR COOLANT PRESSURE OF 100 PSIA. THE CAUSES OF THE EVENT WERE INSUFFICIENT OPERATOR ATTENTION TO DETAIL AND INSUFFICIENT CLARITY OF THE APPLICABLE OPERATING INSTRUCTION. THERE WERE NO SAFETY CONSEQUENCES ASSOCIATED WITH THIS EVENT. CORRECTIVE ACTIONS INCLUDE PERSONNEL ACTION, REEMPHASIZING TO THE OPERATORS THE IMPORTANCE OF ATTENTION TO DETAIL, EVALUATING SG DRAINING RELATED PROCEDURES AND CLARIFYING THEM AS NECESSARY, INSTRUCTING PERSONNEL TO IDENTIFY SIMILAR PROBLEMS IN OTHER PROCEDURES FOR CLARIFICATION, ESTABLISHING A OPERATIONS POLICY REQUIRING A BACKUP FOR SENIOR REACTOR OPERATOR

REVIEWS OF TSS REFERENCED BY PROCEDURES, AND PROVIDING THE DETAILS OF THIS EVENT TO ALL LICENSED OPERATORS.

[18] CATAWBA 1 DOCKET 50-413 LER 91-028
TECHNICAL SPECIFICATION VIOLATION DUE TO EXCEEDING LIQUID WASTE RELEASE LIMITS AS A RESULT OF INAPPROPRIATE ACTIONS.
EVENT DATE: 102491 REPORT DATE: 112091 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 223552) ON 10/24/91, BETWEEN 1000 HOURS AND 1120 HOURS, WITH UNIT 1 IN MODE 1, POWER OPERATION, AT 100% POWER AND UNIT 2 IN MODE 5, COLD SHUTDOWN, A TECHNICAL SPECIFICATION (T/S) VIOLATION OCCURRED DUE TO EXCEEDING THE LIQUID WASTE RELEASE (LWR) LIMITS BECAUSE INADEQUATE DILUTION FLOW WAS AVAILABLE FOR THE LWR DISCHARGE RATE SELECTED FOR THE LWR. THE INCIDENT WAS DISCOVERED WHEN RADIATION PROTECTION (RP) SPECIALIST B REVIEWED THE LWR BECAUSE A CHEMISTRY ASSISTANT TECHNICIAN QUESTIONED THE VALIDITY OF THE DILUTION FLOW REQUIREMENTS SPECIFIED BY THE LWR COMPUTER. RP SPECIALIST B NOTIFIED APPROPRIATE RP MANAGEMENT OF THE INCIDENT. THE EXCESSIVE DISCHARGE FLOW WAS SELECTED FOR THE LWR BECAUSE CHEMISTRY SPECIALIST A MISREAD THE EXPONENT ON THE DILUTION FLOW REQUIRED FOR THE DISCHARGE RATE SELECTED. CHEMISTRY ASSISTANT TECHNICIAN B INDEPENDENTLY VERIFIED THAT THE SELECTED DISCHARGE RATE AND DILUTION FLOW WERE CORRECT. THIS INCIDENT IS ATTRIBUTED TO INAPPROPRIATE ACTIONS BECAUSE THE EXPONENT WAS MISREAD. CORRECTIVE ACTIONS INCLUDED COUNSELING THE CHEMISTRY SPECIALIST A AND ASSISTANT TECHNICIAN B AND APPROPRIATE NOTIFICATIONS OF THE INCIDENT. PLANNED CORRECTIVE ACTIONS INCLUDE PROCEDURE ENHANCEMENTS TO HP/0/B/1004/04, RADIOACTIVE LIQUID WASTE RELEASE, AND OP/0/B/6500/60, DISCHARGE OF AN AMT TO THE ENVIRONMENT, AND CHANGES TO THE LWR COMPUTER PROGRAM TO IMPROVE THE INFORMATION PROVIDED.

[19] CATAWBA 2 DOCKET 50-414 LER 91-012
UNEXPECTED EMERGENCY SAFEGUARD FEATURE ACTUATION DURING SHUTDOWN FOR REFUELING.
EVENT DATE: 101791 REPORT DATE: 111891 NSSS: WE TYPE: PWR

(NSIC 223408) ON OCTOBER 17, 1991, WITH UNIT 2 IN MODE 1, POWER OPERATION, WHILE IN THE PROCESS OF REDUCING REACTOR POWER IN PREPARATION FOR THE END OF CYCLE (EOC) 4 REFUELING OUTAGE, AN UNEXPECTED EMERGENCY SAFEGUARD FEATURE (ESF) ACTUATION OCCURRED AT APPROXIMATELY 1953 HOURS. CONTROL ROOM OPERATORS (CROS) HAD SUCCESSFULLY REDUCED REACTOR POWER LEVEL FROM 100 TO 45 PERCENT. TURBINE IMPULSE PRESSURE (TIP) WAS APPROXIMATELY 260 PSIG. CROS WERE CLOSELY MONITORING TIP IN ANTICIPATION OF AN AUTOMATIC RESET OF THE ANTICIPATED TRANSIENT WITHOUT SCRAM MITIGATION SYSTEM AND ACTUATION CIRCUIT (AMSAC). THE AUTOMATIC RESET SHOULD OCCUR APPROXIMATELY TWO MINUTES AFTER TIP IS LESS THAN 260 PSIG. WHEN THE CROS REALIZED THAT AMSAC DID NOT PROPERLY RESET, ACTION WAS TAKEN, PER PROCEDURE, TO MANUALLY BYPASS AND DEACTIVATE THE AMSAC CIRCUIT. SUBSEQUENTLY, AN UNEXPECTED, PARTIAL ACTUATION OF AMSAC OCCURRED RESULTING IN AN AUTOSTART OF THE AUXILIARY FEEDWATER (CA) SYSTEM PUMP 2B AND ISOLATION OF THE STEAM GENERATOR BLOWDOWN (BB) SYSTEM AND STEAM GENERATOR SAMPLING PORTION OF THE NUCLEAR SAMPLING (NM) SYSTEM. THE CA PUMP 2A WAS ALREADY INSERVICE. THIS INCIDENT IS ATTRIBUTED TO A FUNCTIONAL DESIGN DEFICIENCY OF THE AMSAC CIRCUIT. CORRECTIVE ACTIONS INVOLVED SECURING THE CA PUMP 2B, REALIGNING THE BB AND NM SYSTEMS, AND INVESTIGATION OF THE AMSAC CIRCUIT ANOMALY.

[20] CATAWBA 2 DOCKET 50-414 LER 91-013
TECHNICAL SPECIFICATION VIOLATION DUE TO INAPPROPRIATE ACTION RENDERING ONE TRAIN OF THE BORON DILUTION MITIGATION SYSTEM INOPERABLE.
EVENT DATE: 110291 REPORT DATE: 112791 NSSS: WE TYPE: PWR

(NSIC 223554) ON SUNDAY, NOVEMBER 2, 1991, UNIT 2 WAS IN MODE 6, REFUELING. DURING THE MORNING, SPORADIC AND SPURIOUS ALARMS WERE RECEIVED IN TRAIN A BORON DILUTION MITIGATION SYSTEM (BDMS). DURING AN INVESTIGATION BY OPERATIONS, VALVE 2NV252A (CHEMICAL VOLUME AND CONTROL-NV-PUMP SUCTION FROM REFUELING WATER STORAGE TANK-FWST) WAS FOUND RED TAGGED CLOSED AND DEENERGIZED. THE VALVE AND ITS BREAKER WERE RED TAGGED SINCE OCTOBER 26, 1991. THIS VALVE IS REQUIRED TO OPEN ON A HIGH FLUX SIGNAL FROM BDMS. THEREFORE, TRAIN A BDMS WAS INOPERABLE PER TECHNICAL

SPECIFICATION (T/S). TWO TRAINS OF BDMS ARE REQUIRED FOR MODES 5 AND 6. NO ACTION STATEMENT WAS ENTERED AND, HENCE, A T/S VIOLATION OCCURRED. B TRAIN WAS OPERABLE AND UNAFFECTED. TO RETURN TRAIN A TO OPERABILITY, THE RED TAGS WERE REMOVED FROM THE VALVE AND BREAKER AND THE BREAKER ENERGIZED. THE BDMS TEST PUSHBUTTON WAS ENGAGED AND THE INTERLOCKS WERE VERIFIED: 1) BOTH REACTOR MAKEUP WATER PUMPS STOPPED, 2) 1NV188A (VOLUME CONTROL TANK OUTLET ISOLATION) CLOSED AND 3) 1NV252A OPENED. THIS INCIDENT IS ATTRIBUTED TO IMPROPER OR INADVERTENT ACTION DUE TO LACK OF ATTENTION TO DETAIL DURING PREPARATION OF THE TAGOUT. THE PLANNED CORRECTIVE ACTION IS TO EVALUATE WHAT REFERENCE MATERIALS SHALL REFLECT BDMS INTERLOCKS.

[21] COMANCHE 1 DOCKET 50-445 LER 91-025
COMPONENT COOLING WATER STOP CHECK VALVES INOPERABLE DUE TO LESS THAN ADEQUATE PREVENTIVE MAINTENANCE.
EVENT DATE: 102991 REPORT DATE: 112991 NSSS: WE TYPE: PWR
VENDOR: ROCKWELL MANUFACTURING COMPANY

(NSIC 223576) AT 1215 ON OCTOBER 29, 1991, THE STOP CHECK VALVES (SCV) IN THE COMPONENT COOLING WATER LINES TO THE REACTOR COOLANT PUMP THERMAL BARRIERS WERE BEING TESTED TO SATISFY INSERVICE TESTING REQUIREMENTS. DURING THE TEST, FIVE OF THE EIGHT SCVS FAILED TO STROKE CLOSED. THE FIVE SCVS WERE SUBSEQUENTLY MANUALLY EXERCISED, AFTER WHICH THEY OPERATED AS DESIGNED. ON NOVEMBER 6, 1991, TWO OF THE SCVS WERE INSPECTED. A SMALL ACCUMULATION OF CORROSION PRODUCTS BETWEEN THE PLUG AND STEM OF THE VALVES, AND A SLIGHT SCALING ALONG THE BORE, WERE FOUND. LARGER ACCUMULATIONS OF CORROSION PRODUCTS THAT MAY HAVE BEEN PRESENT WERE FLUSHED OUT WHEN THE VALVES WERE MANUALLY EXERCISED. AT APPROXIMATELY 0215 ON NOVEMBER 23, 1991, ALL FIVE SCVS WERE IN SERVICE AND DETERMINED TO BE CAPABLE OF PERFORMING THEIR INTENDED FUNCTION. THE ROOT CAUSE OF THIS EVENT IS BELIEVED TO BE THE ACCUMULATION OF CORROSION PRODUCTS IN THE SCVS DUE TO LESS THAN ADEQUATE PREVENTIVE MAINTENANCE (PM). CORRECTIVE ACTION INCLUDES THE DEVELOPMENT OF A PM PROCEDURE TO MANUALLY EXERCISE THESE VALVES AND MONITORING OF THE EFFECTIVENESS OF THE PM.

[22] COMANCHE 1 DOCKET 50-445 LER 91-026
PRESSURIZER SAFETY VALVES DISCOVERED TO BE INOPERABLE USING NEW TEST METHODOLOGY.
EVENT DATE: 103091 REPORT DATE: 112991 NSSS: WE TYPE: PWR

(NSIC 223577) ON OCTOBER 18, 1988, ALL THREE PRESSURIZER SAFETY VALVES (PSV) WERE TESTED AND ADJUSTED AT THE CROSBY VALVE AND GAGE COMPANY USING 200 +/- 10 DEGREE FAHRENHEIT WATER AT THE VALVE INLET, APPROXIMATING INSTALLED PLANT CONDITIONS FOR PSVS WITH HOT WATER LOOP SEALS. ON OCTOBER 28, 1991, USING THE STATE-OF-THE-ART TEST METHODOLOGY (SATURATED STEAM AS THE TEST MEDIUM), ALL THREE PSVS WERE FOUND TO HAVE SETPOINTS ONE TO FOUR PERCENT LESS THAN THE SETPOINT TOLERANCE ALLOWED BY TECHNICAL SPECIFICATIONS (TS). ALL THREE PSV SETPOINTS WERE SUBSEQUENTLY RESET TO TS LIMITS. THE ROOT CAUSE OF THIS EVENT WAS A LESS THAN ADEQUATE PSV TEST METHODOLOGY/PROCEDURE. CORRECTIVE ACTION INCLUDED TESTING OF THE PSVS TO STATE-OF-THE-ART METHODOLOGY AND RESETTING THE PSVS TO TS LIMITS.

[23] COMANCHE 1 DOCKET 50-445 LER 91-027
SOURCE RANGE FLUX DOUBLING ACTUATION CAUSED BY CIRCUIT RESPONSE DURING POWER SUPPLY TRANSFER.
EVENT DATE: 110791 REPORT DATE: 120991 NSSS: WE TYPE: PWR

(NSIC 223578) ON NOVEMBER 7, 1991, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 WAS IN MODE 6, REFUELING, WITH ALL FUEL REMOVED FROM THE REACTOR VESSEL. OPERATIONS PERSONNEL TRANSFERRED THE POWER SOURCE FOR A PORTION OF THE NUCLEAR INSTRUMENTATION FROM THE ALTERNATE POWER SUPPLY TO THE NORMAL POWER SUPPLY. COINCIDENT WITH THE SWITCHOVER, A SOURCE RANGE FLUX DOUBLING SIGNAL WAS INITIATED, ACTUATING THE INADVERTENT BORON DILUTION MITIGATION FUNCTION. THE CAUSE OF THE EVENT WAS DETERMINED TO BE THE NUCLEAR INSTRUMENTATION CIRCUIT RESPONSE TO POWER SUPPLY SWITCHOVER. CORRECTIVE ACTION CONSISTS OF PROCEDURE CHANGES TO ACCOUNT FOR CIRCUIT RESPONSE.

[24] CONNECTICUT YANKEE DOCKET 50-213 LER 91-022
LEAKING FLANGE ON RESIDUAL HEAT REMOVAL TO CHARGING SYSTEM CROSSTIE PIPING.
EVENT DATE: 110491 REPORT DATE: 120491 NSSS: WE TYPE: PWR

(NSIC 223461) ON NOVEMBER 4, 1991, AT 0015, WITH THE PLANT IN MODE 6 (REFUELING), A BLANK FLANGE ON A LINE WHICH CONNECTS THE DISCHARGE OF THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM TO THE CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS) WAS FOUND TO BE LEAKING. LEAKAGE WAS ESTIMATED TO BE APPROXIMATELY 1 GALLON PER MINUTE. AT THE TIME OF THE EVENT, THE RHR SYSTEM WAS IN OPERATION AND SEAL WATER TO THE REACTOR COOLANT PUMPS (RCP'S) WAS BEING SUPPLIED BY THE RHR PUMPS VIA THE CROSS-TIE LINE TO THE CVCS. INITIAL CORRECTIVE ACTION INVOLVED STARTING A CHARGING PUMP TO SUPPLY RCP SEAL WATER FROM THE VOLUME CONTROL TANK AND THEN ISOLATING AND DEPRESSURIZING THE RHR TO CVCS CROSS-TIE LINE TO STOP THE LEAKAGE FROM THE FLANGE. A PROMPT REPORT OF THIS EVENT WAS MADE UNDER 10 CFR 50.72(B)(2)(I). THIS EVENT IS REPORTABLE UNDER 10 CFR 50.73(A)(2)(II) AS A CONDITION THAT COULD HAVE RESULTED IN THE PLANT BEING IN AN UNANALYZED CONDITION, SINCE THE RHR TO CVCS CROSS-TIE LINE IS ONE OF THE PATHWAYS FOR CONTAINMENT SUMP RECIRCULATION FLOW FOLLOWING A LOSS OF COOLANT ACCIDENT. LEAKAGE FROM THIS FLANGE DURING SUMP RECIRCULATION OPERATION AFTER AN ACCIDENT COULD HAVE RESULTED IN ADVERSE RADIOLOGICAL CONSEQUENCES. CORRECTIVE ACTION WILL INVOLVE A PLANT MODIFICATION TO REPLACE THE BLANK FLANGE WITH A WELDED CAP.

[25] COOK 1 DOCKET 50-315 LER 91-007 REV 01
UPDATE ON SHUTDOWN RODS MISPOSITIONED DURING ATTEMPT TO MOVE CONTROL RODS DUE TO MALFUNCTION OF MULTIPLEXING RELAY IN THE ROD CONTROL SYSTEM.
EVENT DATE: 081991 REPORT DATE: 111591 NSSS: WE TYPE: PWR
VENDOR: CLARE RELAYS CO.

(NSIC 223520) THIS REVISION PROVIDES INFORMATION ON THE RESULTS OF THE ROOT CAUSE ANALYSIS AND CORRECTS THE FAILED COMPONENT IDENTIFICATION INFORMATION. ON 8/19/91, DURING ROUTINE SURVEILLANCE TESTING OF THE FULL LENGTH RODS, CONTROL BANK A WAS ORDERED INTO THE CORE BY THE ROD CONTROL SYSTEM BUT APPEARED NOT TO MOVE INTO THE CORE BY OBSERVATION OF THE ANALOG ROD POSITION INDICATION. FURTHER POSITIONING OF CONTROL BANK A RESULTED IN AN URGENT ALARM BEING RECEIVED. INITIAL TROUBLESHOOTING OF THE ROD CONTROL SYSTEM FOUND NO REASON FOR THE URGENT ALARM AND IT WAS RESET. THE RODS WERE MOVED PER THE TEST REQUIREMENTS AND THE SURVEILLANCE COMPLETED. AT APPROX. 1200, A ROD MISALIGNMENT IN SHUTDOWN BANK A, GROUP 2 WAS SUSPECTED DUE TO ADDITIONAL SMALL AMOUNT OF DILUTION REQUIRED TO KEEP THERMAL POWER AT THE PRE-SURVEILLANCE LEVEL AND A DECREASE IN THE ANALOG ROD POSITION INDICATION FOR SHUTDOWN BANK A, GROUP 2 IN RELATION TO PRE-SURVEILLANCE READINGS. AT 1305 HOURS THE FLUX MAPPING SYSTEM WAS USED TO DETERMINE THAT THE 4 RODS OF SHUTDOWN BANK A, GROUP 2 WERE INSERTED APPROXIMATELY 6 STEPS INTO THE CORE. FURTHER TROUBLESHOOTING OF THE ROD CONTROL SYSTEM DISCOVERED A FAILURE OF THE 2AC POWER CABINET MULTIPLEXING RELAY, MXR-1. IT WAS REPLACED AND PROPER OPERATION OF THE ROD CONTROL SYSTEM WAS VERIFIED.

[26] COOK 1 DOCKET 50-315 LER 91-010
LIQUID RELEASE TO UNRESTRICTED AREA IN VIOLATION OF TECHNICAL SPECIFICATION DUE TO POOR HUMAN FACTORS IN THE DESIGN AND OPERATION OF THE RELEASE MONITOR.
EVENT DATE: 101691 REPORT DATE: 111591 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)
VENDOR: EBERLINE INSTRUMENT CORP.

(NSIC 223433) ON 10/16/91 APPROX. 1310 HOURS, A SAMPLED BUT UNMONITORED LIQUID RELEASE OCCURRED WHEN THE LIQUID RADWASTE EFFLUENT HEADER SAMPLE GAMMA RADIATION DETECTOR RRS-1000 WENT INTO AN EXTERNAL FAIL STATUS CHANGE ALARM. AT THAT TIME THE ALARM WAS ACKNOWLEDGED BY DEPRESSING THE ACKNOWLEDGE BUTTON. THIS ACTION ALSO BYPASSES THE TRIP FUNCTIONS WHICH ALLOWS OPENING OF THE WASTE RELEASE ISOLATION VALVE TO INITIATE SAMPLE FLOW TO ABOVE THE LOW FLOW TRIP SETPOINT OF 3.1 GPM. TO RE-ARM THE TRIP FUNCTION THE SAMPLE FLOW MUST BE INCREASED TO 5.5 GPM. DURING THE 6 MINUTES OF RELEASE PRIOR TO DETECTING THE MONITORS INOPERABLE STATUS, THE REARMING FLOWRATE HAD NOT BEEN REACHED, AND ONLY 598 GALLONS OF THE 16,134 GALLON TANK HAD BEEN RELEASED. THE EVENT WAS CAUSED BY POOR HUMAN FACTORS IN THE DESIGN AND OPERATION OF THE SYSTEM WHICH MADE IT DIFFICULT FOR THE OPERATORS TO

DETERMINE THEY HAD BYPASSED THE TRIP FUNCTION WHEN THE ALARM ACKNOWLEDGE BUTTON WAS DEPRESSED. ENHANCEMENTS WERE MADE TO THE LIQUID WASTE RELEASE PROCEDURE TO REQUIRE INCREASING THE SAMPLE FLOW RATE TO THE LEVEL REQUIRED FOR ARMING THE TRIP SETPOINT. IN ADDITION, TO AID THE OPERATORS IN IDENTIFYING THE ALARM CONDITION, THE DURATION OF THE AUDIBLE ALARM WILL BE INCREASED FROM SEVEN TO FIFTEEN SECONDS.

[27] COOK 2 DOCKET 50-316 LER 91-008
CONTAINMENT PRESSURE RELIEF PERFORMED WITH THE RADIATION MONITOR TRIP FUNCTION BLOCKING DUE TO PERSONNEL ERROR.
EVENT DATE: 110591 REPORT DATE: 112791 NSSS: WE TYPE: PWR

(NSIC 223513) ON NOVEMBER 5, 1991 AT 1906 HOURS, THE CONTAINMENT PRESSURE RELIEF VALVES WERE OPENED TO PERFORM A CONTAINMENT PRESSURE RELIEF WHILE THE VALVE'S ASSOCIATED RADIATION MONITORING SYSTEM TRIP/BLOCK SWITCHES WERE IN THE BLOCK POSITION. THE PRESSURE RELIEF VALVES ARE ALLOWED TO BE OPENED BY TECHNICAL SPECIFICATION 3.6.3.1, CONTAINMENT ISOLATION VALVES, ON AN INTERMITTENT BASIS UNDER ADMINISTRATIVE CONTROL. THE OPENING OF THESE VALVES WITH THE RADIATION MONITORING SYSTEM SWITCHES IN BLOCK WAS NOT THE PROPER ADMINISTRATIVE CONTROL FOR THE CONDITIONS THAT EXISTED AT THE TIME. DUE TO A MISINTERPRETATION OF A PROCEDURE STEP, A REACTOR OPERATOR (RO) TRAINEE SIGNED OFF THE STEP IN THE CONTAINMENT PRESSURE RELIEF PROCEDURE WHILE THE SWITCHES WERE IN THE BLOCK POSITION; HOWEVER, THE SWITCHES SHOULD HAVE BEEN IN THE NORMAL POSITION. THIS ERROR WAS NOT IDENTIFIED BY THE UNIT SUPERVISOR (US) OR THE RO UNTIL THE CONTAINMENT PRESSURE RELIEF WAS COMPLETED. THE CAUSE OF THIS EVENT WAS IMPROPER TRAINEE CONTROL. NEITHER THE US NOR THE RO WERE IN DIRECT CONTACT WITH THE RO TRAINEE WHEN THE PROCEDURAL ERROR OCCURRED. THE OPERATIONS DEPARTMENT SUPERINTENDENT ISSUED LETTERS TO ALL LICENSED OPERATORS, RO TRAINEES AND SHIFT SUPERVISORS EMPHASIZING INDIVIDUAL RESPONSIBILITY FOR CONTROLLING TRAINEE ACTIVITIES IN THE CONTROL ROOM.

[28] COOPER DOCKET 50-298 LER 91-010
REACTOR WATER CLEANUP ISOLATION DUE TO HIGH SYSTEM TEMPERATURE CAUSED BY BACKFLOW FROM THE REGENERATIVE HEAT EXCHANGERS.
EVENT DATE: 100291 REPORT DATE: 103191 NSSS: GE TYPE: BWR

(NSIC 223357) ON OCTOBER 2, 1991, AT 10:35 A.M., A GROUP 3 ISOLATION (REACTOR WATER CLEANUP (RWCU) SYSTEM) OCCURRED DUE TO HIGH TEMPERATURE CONDITIONS (140 DEGREES FAHRENHEIT) DOWNSTREAM OF THE NON-REGENERATIVE HEAT EXCHANGERS. THE RWCU SYSTEM HAD BEEN REMOVED FROM SERVICE AND RAPIDLY DEPRESSURIZED TO ALLOW MAINTENANCE ON AN RWCU PUMP SUCTION VALVE WHICH HAD DEVELOPED A STEAM LEAK. BOTH RWCU SYSTEM ISOLATION VALVES WERE CLOSED AT THE TIME OF THE EVENT. DURING THIS EVENT, THE REACTOR WAS AT APPROXIMATELY 72 PERCENT POWER, 517 DEGREES FAHRENHEIT AND 973 PSIG. THE HIGH TEMPERATURE CONDITION WAS THE RESULT OF HOT WATER BEING DRAWN BACKWARDS FROM THE REGENERATIVE HEAT EXCHANGERS INTO THE SECTION OF PIPING CONTAINING THE HIGH TEMPERATURE ISOLATION SWITCH. THE BACKFLOW WAS A RESULT OF VOID FORMATION CAUSED BY THE RAPID PRESSURE REDUCTION AND SUBSEQUENT COOLING OF THE SYSTEM. THE EFFECT OF BACKFLOW ON THE TEMPERATURE SWITCH WAS AN UNFORESEEN RESULT OF THE PHYSICAL ARRANGEMENT OF THE SYSTEM. A PROCEDURE CHANGE WILL BE PROCESSED TO IDENTIFY THE NEED FOR A SLOW DEPRESSURIZATION, WHEN POSSIBLE, SINCE A SLOW DEPRESSURIZATION SHOULD NOT RESULT IN AN ISOLATION SIGNAL. A CAUTION WILL BE ADDED TO EXPECT A GROUP 3 ISOLATION SHOULD A RAPID PRESSURE REDUCTION BE NECESSARY. THIS EVENT WILL ALSO BE DISCUSSED IN INDUSTRY EVENTS TRAINING.

[29] COOPER DOCKET 50-298 LER 91-014
UNPLANNED GROUP 6 ISOLATION FROM SHORT CIRCUIT DURING DESIGN CHANGE ACTIVITIES.
EVENT DATE: 103091 REPORT DATE: 112791 NSSS: GE TYPE: BWR

(NSIC 223495) ON OCTOBER 30, 1991, AT 11:36 A.M., WITH THE PLANT IN COLD SHUTDOWN FOR A REFUELING OUTAGE, AN UNPLANNED ACTUATION OF THE GROUP 6 ISOLATION LOGIC OCCURRED, RESULTING IN ISOLATION OF THE SECONDARY CONTAINMENT AND STARTUP OF THE B TRAIN OF STANDBY GAS TREATMENT. THE A TRAIN OF STANDBY GAS TREATMENT WAS OUT OF SERVICE FOR MAINTENANCE. CONTRACT ELECTRICIANS, WORKING IN A CONTROL ROOM PANEL, WERE PREPARING TO INSTALL WIRING FOR A DESIGN CHANGE. UPON MOVING THE WIRING, AN

UNINSULATED METAL CLIP (USED FOR SURVEILLANCE TESTING) ON THE END TERMINAL OF A TERMINAL BOARD CAME IN CONTACT WITH THE TERMINAL BOARD MOUNTING HARDWARE, RESULTING IN A BLOWN FUSE AND THE GROUP 6 ISOLATION. AFTER DETERMINING THE CAUSE, THE FUSE WAS REPLACED, THE ISOLATION RESET, AND THE REACTOR BUILDING VENTILATION SYSTEM AUTOMATICALLY RETURNED TO ITS NORMAL LINEUP. THE CAUSE OF THIS EVENT IS THE IMPROPER SELECTION OF AN UNINSULATED METAL CLIP FOR SURVEILLANCE TESTING. THE USE OF THIS STYLE CLIP DID NOT CONSIDER THE POTENTIAL CONSEQUENCES OF ITS USE. A TAB ON THE TERMINAL BOARD MOUNTING HARDWARE WAS BENT IN TOWARD THE TERMINALS, AND THE SCREW HOLDING THE CLIP MAY NOT HAVE BEEN TIGHT DUE TO INSTALLATION ACTIVITIES OR AN OVERSIZE WIRE LUG. CONGESTED CONDITIONS IN THE PANEL PREVENTED THE CLIP AND BENT TAB ON THE TERMINAL BOARD MOUNTING HARD WARE FROM BEING DISCOVERED.

[30]   COOPER   DOCKET 50-298   LER 91-015
SAFETY/RELIEF AND SAFETY VALVE SETPOINT VARIANCE NOT WITHIN TECHNICAL SPECIFICATION LIMITS.
EVENT DATE: 110591   REPORT DATE: 120591   NSSS: GE   TYPE: BWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV
TARGET ROCK CORP.

(NSIC 223496) WHILE THE PLANT WAS SHUTDOWN FOR THE 1991 REFUELING OUTAGE, TESTING OF SAFETY/RELIEF VALVES (S/RVS) AND A SAFETY VALVE (SV) WAS CONDUCTED AT WYLE LABORATORIES, HUNTSVILLE, ALABAMA. THE AS - FOUND SETPOINT FOR THE S/RVS WAS NOT WITHIN THE TECHNICAL SPECIFICATION LIMIT OF +/- 1 PERCENT (11 PSI). ADDITIONALLY, THE SETPOINT FOR THE SV WAS NOT WITHIN THE TS LIMIT OF +/- 1 PERCENT (13 PSI). THIS IS A VOLUNTARY REPORT IN ACCORDANCE WITH 10CFR50.73 GUIDELINES. NO SPECIFIC DISCREPANCIES WERE IDENTIFIED DURING TESTING OR SUBSEQUENT DISASSEMBLY AND INSPECTION EFFORTS. REVIEW OF THE TEST DATA INDICATES THE SAFETY RELIEF VALVES GENERALLY DISPLAYED PERFORMANCE CHARACTERISTICS TYPICAL OF INDUSTRY-WIDE PROBLEMS WITH CORROSION BONDING OF THE PILOT DISC. TO DATE, THE ROOT CAUSE AND CORRECTIVE ACTION FOR THIS INDUSTRY-WIDE PROBLEM REMAINS UNRESOLVED. ALSO NOTED WERE PERFORMANCE CHARACTERISTICS TYPICAL OF LABYRINTH SEAL INDUCED FRICTION. THE FAILURE OF THE SAFETY VALVE IS CONSIDERED RANDOM SETPOINT DRIFT. THE VALVES WERE INSPECTED, REFURBISHED AND SATISFACTORILY RETESTED AT WYLE LABORATORIES BEFORE THEIR RETURN TO THE SITE. A TS CHANGE REQUEST, INCREASING THE SETPOINT TOLERANCE TO +/- 3 PERCENT, HAD PREVIOUSLY BEEN SUBMITTED TO THE NRC ON JANUARY 26, 1990. THE PROPOSED CHANGE WAS IDENTIFIED AS CORRECTIVE ACTION IN LER 89-015. NO CORRECTIVE ACTION IS PLANNED NOW.

[31]   CRYSTAL RIVER 3   DOCKET 50-302   LER 91-011
DISCOVERY OF CONDITION HAVING THE POTENTIAL TO COMPROMISE LONG TERM COOLING POST-LOCA.
EVENT DATE 110491   REPORT DATE: 120491   NSSS: BW   TYPE: PWR

(NSIC 223519) CRYSTAL RIVER UNIT 3 (CR-3) WAS IN MODE 5, COLD SHUTDOWN, ON NOVEMBER 4, 1991. AT 1630 UTILITY PERSONNEL DETERMINED FROM ANALYSES THAT A CONDITION EXISTED WHICH APPEARED TO HAVE THE POTENTIAL TO AFFECT LONG TERM COOLING IF A LOSS OF COOLANT ACCIDENT (LOCA) OCCURRED DUE TO A BREAK IN ONE OF THE REACTOR COOLANT SYSTEM COLD LEGS. THE BASIS FOR CR-3'S OPERATING LICENSE INCLUDES THE POST-LOCA LONG TERM COOLING ANALYSES PERFORMED FOR BABCOCK & WILCOX PLANTS, WHICH INDICATES THAT ACTIONS TO PREVENT BORON PRECIPITATION IN THE CORE [illegible] NOT BE REQUIRED UNTIL APPROXIMATELY FORTY DAYS FOLLOWING A COLD LEG BREAK [illegible] MORE RECENT ANALYSES INDICATED BORON PRECIPITATION COULD BEGIN MUCH SOONER [illegible]NS TO PREVENT BORON PRECIPITATION WOULD BE REQUIRED WITHIN TWENTY FOUR [illegible]DURES FOR CR-3 ALREADY REQUIRED THESE ACTIONS WITHIN TWENTY FOUR [illegible] NOTIFICATION OF THIS PROBLEM, ADDITIONAL ANALYSIS HAS SHOWN THAT [illegible]WITHIN THE REACTOR WILL PREVENT THE BORON CONCENTRATION FROM [illegible] SOLUBILITY LIMIT. THEREFORE, NO BORON PRECIPITATION WOULD OCCUR [illegible]R HARDWARE REVISIONS ARE REQUIRED.

[illegible]1   DOCKET 50-346   LER 91-006
[illegible] BREAK LOCA BORON CONCENTRATION WAS POTENTIALLY
[illegible]T DATE: 120591   NSSS: BW   TYPE: PWR

(NSIC 223527) ON NOVEMBER 5, 1991, WITH THE PLANT IN MODE 2, TOLEDO EDISON DETERMINED THAT ANALYSIS OF BORON CONCENTRATION IN THE CORE AFTER A LARGE BREAK LOCA OF A COLD LEG WAS POTENTIALLY NON-CONSERVATIVE. A PRELIMINARY SAFETY CONCERN (PSC 2-91) WAS INITIALLY EVALUATED BY BABCOCK AND WILCOX NUCLEAR SERVICES COMPANY (NSSS VENDOR). THE EVALUATION CONCLUDED THAT THE BORON DILUTION FLOW PATH THROUGH THE REACTOR VESSEL VENT VALVES WOULD NOT CARRY TWO PHASE FLOW FOR THE FORTY DAY PERIOD PREVIOUSLY PREDICTED. THE NSSS VENDOR HAD NOT ACCOUNTED FOR THE EFFECT OF THE PLENUM CYLINDER IN THE REACTOR VESSEL WHEN MODELING THE BOILING IN THE PLENUM REGION. SUBSEQUENT TO THE EVALUATION, A PROCEDURE CHANGE TO THE LARGE LOCA SECTION OF DB-OP-02000 (SFAS, SFRCS, RPS TRIP, OR SG TUBE RUPTURE), WAS ISSUED. THIS CHANGE PROVIDES ADMINISTRATIVE REQUIREMENTS TO ESTABLISH A LONG TERM BORON DILUTION FLOW PATH, VIA THE DECAY HEAT DROP LINE OR THE PRESSURIZER AUXILIARY SPRAY, AT THE EARLIEST TIME POSSIBLE FOLLOWING THE TRANSFER TO THE EMERGENCY SUMP, WHICH WILL NORMALLY BE WITHIN 2 HOURS. RECENT ANALYSIS HAS CONFIRMED THE GAP BETWEEN THE PLENUM AND THE HOT LEG NOZZLES PROVIDES AN ADEQUATE FLOW PATH TO PREVENT BORON PRECIPITATION IN THE CORE UNTIL EITHER THE DECAY HEAT DROP LINE OR AUXILIARY SPRAY FLOW CAN SUCCESSFULLY PROVIDE ADEQUATE DILUTION FLOW.

[33] DIABLO CANYON 2 DOCKET 50-323 LER 91-011
FAILURE TO TEST A VALVE FOLLOWING MAINTENANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 102091 REPORT DATE: 112091 NSSS: WE TYPE: PWR

(NSIC 223507) ON OCTOBER 20, 1991, AT 1400 PDT, THE 72-HOUR LIMITING CONDITION FOR OPERATION (LCO) OF TECHNICAL SPECIFICATION (TS) 3.5.2, ACTION A, WAS EXCEEDED WHEN VALVE SI-2-8802B WAS NOT TESTED FOLLOWING CORRECTIVE MAINTENANCE. TS 4.0.5 REQUIRED SURVEILLANCE WAS NOT PERFORMED. ON OCTOBER 20, 1991, AT 1724 PDT, TS 4.0.4 WAS NOT MET WHEN UNIT 2 TRANSITIONED FROM MODE 3 (HOT STANDBY) TO MODE 2 (STARTUP) WITH THE REQUIRED POST-MAINTENANCE TEST (PMT) REQUIREMENTS OF VALVE SI-2-8802B NOT COMPLETED. ON OCTOBER 22, 1991, AT 2100 PDT, THE FAILURE TO CONDUCT A PMT WAS IDENTIFIED BY PLANT ENGINEERING WHILE REVIEWING REQUIRED ACTIONS FOR A SECOND TRANSITION INTO MODE 2. THE REQUIRED SURVEILLANCE TEST WAS REVISED TO PERMIT PERFORMANCE IN MODE 3 AND WAS SUCCESSFULLY PERFORMED ON OCTOBER 22, 1991, AT 2159 PDT. THE ROOT CAUSE OF THESE EVENTS WAS PERSONNEL ERROR DUE TO A FAILURE TO RECOGNIZE THAT MAINTENANCE PERFORMED ON VALVE SI-2-8802B WOULD REQUIRE POST-MAINTENANCE OPERABILITY TESTING. INFORMATION SUPPLIED IN THE FIELD COPY OF THE WORK PACKAGE WAS CONFUSING AND NOT SUFFICIENT TO READILY IDENTIFY PMT REQUIREMENTS. TO PREVENT RECURRENCE, (1) AN OPERATIONS EVENT REPORT WILL BE ISSUED, (2) WORK PLANNING WILL ISSUE ENHANCED WORK POLICIES, AND (3) THE ENGINEERING TEST GROUP WILL BE TRAINED PERFORMING MODE TRANSITION SEARCHES.

[34] DIABLO CANYON 2 DOCKET 50-323 LER 91-010
INOPERABLE REACTOR CAVITY SUMP WIDE RANGE LEVEL CHANNEL DUE TO UNKNOWN CAUSE.
EVENT DATE: 102291 REPORT DATE: 112191 NSSS: WE TYPE: PWR

(NSIC 223508) ON OCTOBER 22, 1991, AT 0958 PDT, WITH UNIT 2 IN MODE 2 (STARTUP) AT 0 PERCENT POWER, ACTION STATEMENT A OF TECHNICAL SPECIFICATION (TS) 3.3.3.6 WAS EXCEEDED DUE TO REACTOR CAVITY SUMP WIDE RANGE LEVEL CHANNEL 942A BEING INOPERABLE FOR GREATER THAN SEVEN DAYS IN MODES 1 (POWER OPERATION), 2, OR 3 (HOT STANDBY). ON OCTOBER 15, 1991, AT 0958 PDT, UNIT 2 TRANSITIONED FROM MODE 4 (HOT SHUTDOWN) TO MODE 3, AND CHANNEL 942A WAS THEN REQUIRED TO BE OPERABLE. CHANNEL 942A WAS SUBSEQUENTLY DETERMINED TO HAVE BEEN OUT OF TOLERANCE AND INOPERABLE SINCE OCTOBER 10, 1991. ON OCTOBER 22, 1991, AT 1500 PDT, AN NRC INSPECTOR IDENTIFIED A POTENTIAL PROBLEM WITH REACTOR CAVITY SUMP WIDE RANGE LEVEL CHANNEL ON A SECONDARY SCREEN OF THE SAFETY PARAMETER DISPLAY SYSTEM. THE OUT OF TOLERANCE CONDITION OF CHANNEL 942A WAS VERIFIED ON THE STRIP CHART RECORDER. THE CHANNEL WAS DECLARED INOPERABLE, AND TROUBLESHOOTING EFFORTS WERE INITIATED. THE TROUBLESHOOTING DID NOT IDENTIFY ANY EQUIPMENT PROBLEMS WITH CHANNEL 942A. THE CHANNEL WAS RETURNED TO SERVICE AT APPROXIMATELY 1700 PDT. THE CHANNEL WAS DECLARED OPERABLE AT 2115 PDT ON OCTOBER 22, 1991 BECAUSE THE CHANNEL PARAMETERS WERE DETERMINED TO BE WITHIN ALLOWABLE TOLERANCES. THE ROOT CAUSE OF THE FAILURE IS UNDER INVESTIGATION. THE ROOT CAUSE AND THE CORRECTION ACTIONS WILL BE SUBMITTED IN A SUPPLEMENTAL LER.

UNINSULATED METAL CLIP (USED FOR SURVEILLANCE TESTING) ON THE END TERMINAL OF A TERMINAL BOARD CAME IN CONTACT WITH THE TERMINAL BOARD MOUNTING HARDWARE, RESULTING IN A BLOWN FUSE AND THE GROUP 6 ISOLATION. AFTER DETERMINING THE CAUSE, THE FUSE WAS REPLACED, THE ISOLATION RESET, AND THE REACTOR BUILDING VENTILATION SYSTEM AUTOMATICALLY RETURNED TO ITS NORMAL LINEUP. THE CAUSE OF THIS EVENT IS THE IMPROPER SELECTION OF AN UNINSULATED METAL CLIP FOR SURVEILLANCE TESTING. THE USE OF THIS STYLE CLIP DID NOT CONSIDER THE POTENTIAL CONSEQUENCES OF ITS USE. A TAB ON THE TERMINAL BOARD MOUNTING HARDWARE WAS BENT IN TOWARD THE TERMINALS, AND THE SCREW HOLDING THE CLIP MAY NOT HAVE BEEN TIGHT DUE TO INSTALLATION ACTIVITIES OR AN OVERSIZE WIRE LUG. CONGESTED CONDITIONS IN THE PANEL PREVENTED THE CLIP AND BENT TAB ON THE TERMINAL BOARD MOUNTING HARD WARE FROM BEING DISCOVERED.

[30] COOPER DOCKET 50-298 LER 91-015
SAFETY/RELIEF AND SAFETY VALVE SETPOINT VARIANCE NOT WITHIN TECHNICAL SPECIFICATION LIMITS.
EVENT DATE: 110591 REPORT DATE: 120591 NSSS: GE TYPE: BWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV
TARGET ROCK CORP.

(NSIC 223496) WHILE THE PLANT WAS SHUTDOWN FOR THE 1991 REFUELING OUTAGE, TESTING OF SAFETY/RELIEF VALVES (S/RVS) AND A SAFETY VALVE (SV) WAS CONDUCTED AT WYLE LABORATORIES, HUNTSVILLE, ALABAMA. THE AS - FOUND SETPOINT FOR THE S/RVS WAS NOT WITHIN THE TECHNICAL SPECIFICATION LIMIT OF +/- 1 PERCENT (11 PSI). ADDITIONALLY, THE SETPOINT FOR THE SV WAS NOT WITHIN THE TS LIMIT OF +/- 1 PERCENT (13 PSI). THIS IS A VOLUNTARY REPORT IN ACCORDANCE WITH 10CFR50.73 GUIDELINES. NO SPECIFIC DISCREPANCIES WERE IDENTIFIED DURING TESTING OR SUBSEQUENT DISASSEMBLY AND INSPECTION EFFORTS. REVIEW OF THE TEST DATA INDICATES THE SAFETY RELIEF VALVES GENERALLY DISPLAYED PERFORMANCE CHARACTERISTICS TYPICAL OF INDUSTRY-WIDE PROBLEMS WITH CORROSION BONDING OF THE PILOT DISC. TO DATE, THE ROOT CAUSE AND CORRECTIVE ACTION FOR THIS INDUSTRY-WIDE PROBLEM REMAINS UNRESOLVED. ALSO NOTED WERE PERFORMANCE CHARACTERISTICS TYPICAL OF LABYRINTH SEAL INDUCED FRICTION. THE FAILURE OF THE SAFETY VALVE IS CONSIDERED RANDOM SETPOINT DRIFT. THE VALVES WERE INSPECTED, REFURBISHED AND SATISFACTORILY RETESTED AT WYLE LABORATORIES BEFORE THEIR RETURN TO THE SITE. A TS CHANGE REQUEST, INCREASING THE SETPOINT TOLERANCE TO +/- 3 PERCENT, HAD PREVIOUSLY BEEN SUBMITTED TO THE NRC ON JANUARY 26, 1990. THE PROPOSED CHANGE WAS IDENTIFIED AS CORRECTIVE ACTION IN LER 89-015. NO CORRECTIVE ACTION IS PLANNED NOW.

[31] CRYSTAL RIVER 3 DOCKET 50-302 LER 91-011
DISCOVERY OF CONDITION HAVING THE POTENTIAL TO COMPROMISE LONG TERM COOLING POST-LOCA.
EVENT DATE: 110491 REPORT DATE: 120491 NSSS: BW TYPE: PWR

(NSIC 223519) CRYSTAL RIVER UNIT 3 (CR-3) WAS IN MODE 5, COLD SHUTDOWN, ON NOVEMBER 4, 1991. AT 1630 UTILITY PERSONNEL DETERMINED FROM ANALYSES THAT A CONDITION EXISTED WHICH APPEARED TO HAVE THE POTENTIAL TO AFFECT LONG TERM COOLING IF A LOSS OF COOLANT ACCIDENT (LOCA) OCCURRED DUE TO A BREAK IN ONE OF THE REACTOR COOLANT SYSTEM COLD LEGS. THE BASIS FOR CR-3'S OPERATING LICENSE INCLUDES THE POST-LOCA LONG TERM COOLING ANALYSES PERFORMED FOR BABCOCK & WILCOX PLANTS, WHICH INDICATES THAT ACTIONS TO PREVENT BORON PRECIPITATION IN THE CORE WOULD NOT BE REQUIRED UNTIL APPROXIMATELY FORTY DAYS FOLLOWING A COLD LEG BREAK LOCA. MORE RECENT ANALYSES INDICATED BORON PRECIPITATION COULD BEGIN MUCH SOONER AND ACTIONS TO PREVENT BORON PRECIPITATION WOULD BE REQUIRED WITHIN TWENTY FOUR HOURS. PROCEDURES FOR CR-3 ALREADY REQUIRED THESE ACTIONS WITHIN TWENTY FOUR HOURS. SINCE NOTIFICATION OF THIS PROBLEM, ADDITIONAL ANALYSIS HAS SHOWN THAT FLOW PHENOMENA WITHIN THE REACTOR WILL PREVENT THE BORON CONCENTRATION FROM INCREASING TO THE SOLUBILITY LIMIT. THEREFORE, NO BORON PRECIPITATION WOULD OCCUR SO NO PROCEDURAL OR HARDWARE REVISIONS ARE REQUIRED.

[32] DAVIS-BESSE 1 DOCKET 50-346 LER 91-006
ANALYSIS OF POST LARGE BREAK LOCA BORON CONCENTRATION WAS POTENTIALLY NON-CONSERVATIVE.
EVENT DATE: 110591 REPORT DATE: 120591 NSSS: BW TYPE: PWR

(NSIC 223527) ON NOVEMBER 5, 1991, WITH THE PLANT IN MODE 2, TOLEDO EDISON DETERMINED THAT ANALYSIS OF BORON CONCENTRATION IN THE CORE AFTER A LARGE BREAK LOCA OF A COLD LEG WAS POTENTIALLY NON-CONSERVATIVE. A PRELIMINARY SAFETY CONCERN (PSC 2-91) WAS INITIALLY EVALUATED BY BABCOCK AND WILCOX NUCLEAR SERVICES COMPANY (NSSS VENDOR). THE EVALUATION CONCLUDED THAT THE BORON DILUTION FLOW PATH THROUGH THE REACTOR VESSEL VENT VALVES WOULD NOT CARRY TWO PHASE FLOW FOR THE FORTY DAY PERIOD PREVIOUSLY PREDICTED. THE NSSS VENDOR HAD NOT ACCOUNTED FOR THE EFFECT OF THE PLENUM CYLINDER IN THE REACTOR VESSEL WHEN MODELING THE BOILING IN THE PLENUM REGION. SUBSEQUENT TO THE EVALUATION, A PROCEDURE CHANGE TO THE LARGE LOCA SECTION OF DB-OP-02000 (SFAS, SFRCS, RPS TRIP, OR SG TUBE RUPTURE), WAS ISSUED. THIS CHANGE PROVIDES ADMINISTRATIVE REQUIREMENTS TO ESTABLISH A LONG TERM BORON DILUTION FLOW PATH, VIA THE DECAY HEAT DROP LINE OR THE PRESSURIZER AUXILIARY SPRAY, AT THE EARLIEST TIME POSSIBLE FOLLOWING THE TRANSFER TO THE EMERGENCY SUMP, WHICH WILL NORMALLY BE WITHIN 2 HOURS. RECENT ANALYSIS HAS CONFIRMED THE GAP BETWEEN THE PLENUM AND THE HOT LEG NOZZLES PROVIDES AN ADEQUATE FLOW PATH TO PREVENT BORON PRECIPITATION IN THE CORE UNTIL EITHER THE DECAY HEAT DROP LINE OR AUXILIARY SPRAY FLOW CAN SUCCESSFULLY PROVIDE ADEQUATE DILUTION FLOW.

[33]   DIABLO CANYON 2   DOCKET 50-323   LER 91-011
FAILURE TO TEST A VALVE FOLLOWING MAINTENANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 102091   REPORT DATE: 112091   NSSS: WE   TYPE: PWR

(NSIC 223507) ON OCTOBER 20, 1991, AT 1400 PDT, THE 72-HOUR LIMITING CONDITION FOR OPERATION (LCO) OF TECHNICAL SPECIFICATION (TS) 3.5.2, ACTION A, WAS EXCEEDED WHEN VALVE SI-2-8802B WAS NOT TESTED FOLLOWING CORRECTIVE MAINTENANCE. TS 4.0.5 REQUIRED SURVEILLANCE WAS NOT PERFORMED. ON OCTOBER 20, 1991, AT 1724 PDT, TS 4.0.4 WAS NOT MET WHEN UNIT 2 TRANSITIONED FROM MODE 3 (HOT STANDBY) TO MODE 2 (STARTUP) WITH THE REQUIRED POST-MAINTENANCE TEST (PMT) REQUIREMENTS OF VALVE SI-2-8802B NOT COMPLETED. ON OCTOBER 22, 1991, AT 2100 PDT, THE FAILURE TO CONDUCT A PMT WAS IDENTIFIED BY PLANT ENGINEERING WHILE REVIEWING REQUIRED ACTIONS FOR A SECOND TRANSITION INTO MODE 2. THE REQUIRED SURVEILLANCE TEST WAS REVISED TO PERMIT PERFORMANCE IN MODE 3 AND WAS SUCCESSFULLY PERFORMED ON OCTOBER 22, 1991, AT 2139 PDT. THE ROOT CAUSE OF THESE EVENTS WAS PERSONNEL ERROR DUE TO A FAILURE TO RECOGNIZE THAT MAINTENANCE PERFORMED ON VALVE SI-2-8802B WOULD REQUIRE POST-MAINTENANCE OPERABILITY TESTING. INFORMATION SUPPLIED IN THE FIELD COPY OF THE WORK PACKAGE WAS CONFUSING AND NOT SUFFICIENT TO READILY IDENTIFY PMT REQUIREMENTS. TO PREVENT RECURRENCE, (1) AN OPERATIONS EVENT REPORT WILL BE ISSUED, (2) WORK PLANNING WILL ISSUE ENHANCED WORK POLICIES, AND (3) THE ENGINEERING TEST GROUP WILL BE TRAINED PERFORMING MODE TRANSITION SEARCHES.

[34]   DIABLO CANYON 2   DOCKET 50-323   LER 91-010
INOPERABLE REACTOR CAVITY SUMP WIDE RANGE LEVEL CHANNEL DUE TO UNKNOWN CAUSE.
EVENT DATE: 102291   REPORT DATE: 112191   NSSS: WE   TYPE: PWR

(NSIC 223508) ON OCTOBER 22, 1991, AT 0958 PDT, WITH UNIT 2 IN MODE 2 (STARTUP) AT 0 PERCENT POWER, ACTION STATEMENT A OF TECHNICAL SPECIFICATION (TS) 3.3.3.6 WAS EXCEEDED DUE TO REACTOR CAVITY SUMP WIDE RANGE LEVEL CHANNEL 942A BEING INOPERABLE FOR GREATER THAN SEVEN DAYS IN MODES 1 (POWER OPERATION), 2, OR 3 (HOT STANDBY). ON OCTOBER 15, 1991, AT 0958 PDT, UNIT 2 TRANSITIONED FROM MODE 4 (HOT SHUTDOWN) TO MODE 3, AND CHANNEL 942A WAS THEN REQUIRED TO BE OPERABLE. CHANNEL 942A WAS SUBSEQUENTLY DETERMINED TO HAVE BEEN OUT OF TOLERANCE AND INOPERABLE SINCE OCTOBER 10, 1991. ON OCTOBER 22, 1991, AT 1500 PDT, AN NRC INSPECTOR IDENTIFIED A POTENTIAL PROBLEM WITH REACTOR CAVITY SUMP WIDE RANGE LEVEL CHANNEL ON A SECONDARY SCREEN OF THE SAFETY PARAMETER DISPLAY SYSTEM. THE OUT OF TOLERANCE CONDITION OF CHANNEL 942A WAS VERIFIED ON THE STRIP CHART RECORDER. THE CHANNEL WAS DECLARED INOPERABLE, AND TROUBLESHOOTING EFFORTS WERE INITIATED. TROUBLESHOOTING DID NOT IDENTIFY ANY EQUIPMENT PROBLEMS WITH CHANNEL 942A. THE CHANNEL WAS RETURNED TO SERVICE AT APPROXIMATELY 1700 PDT. THE CHANNEL WAS DECLARED OPERABLE AT 2115 PDT ON OCTOBER 22, 1991 BECAUSE THE CHANNEL PARAMETERS WERE DETERMINED TO BE WITHIN ALLOWABLE TOLERANCES. THE ROOT CAUSE OF THE FAILURE IS UNDER INVESTIGATION. THE ROOT CAUSE AND THE CORRECTION ACTIONS WILL BE SUBMITTED IN A SUPPLEMENTAL LER.

[35] DRESDEN 2 DOCKET 50-237 LER 91-[illegible]
VENT AND PURGE SYSTEM EXHAUST DUCTWORK SEPARATION DUE TO
CONSTRUCTION/INSTALLATION DEFICIENCY.
EVENT DATE: 091891 REPORT DATE: 112591 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 223464) AT 0515 HOURS ON SEPTEMBER 18, 1991, WITH UNIT 2 AT 95% POWER AND UNIT 3 IN A REFUEL OUTAGE, A JOINT IN THE EXHAUST DUCTWORK OF THE UNIT 3 DRYWELL VENT AND PURGE SYSTEM WAS FOUND SEPARATED. THE TWO PIECES OF EXHAUST DUCTWORK WERE TOUCHING AND SLIGHTLY OUT OF ALIGNMENT. REPAIRS WERE COMPLETED BY 0600 HOURS ON SEPTEMBER 18, 1991. THE EXHAUST DUCTWORK HAD BEEN VERIFIED TO BE INTACT ON SEPTEMBER 10, 1991 AT 1600 HOURS BY A TECHNICAL STAFF ENGINEER; THIS INDICATED THAT THE FAILURE OCCURRED SUBSEQUENT TO THAT TIME. INITIAL REVIEW OF THE EVENT FOCUSED ON SCAFFOLDING ERECTION WORK IN THE AREA AS A POTENTIAL CAUSE OF THE DAMAGE. HOWEVER, FURTHER INVESTIGATION INTO THE EXISTING JOINT CONFIGURATION CONCLUDED THAT IT HAD NOT BEEN FASTENED IN ACCORDANCE WITH ITS DESIGN REQUIREMENTS. INSPECTIONS OF THE OTHER DUCTWORK SECTIONS WERE ALSO COMPLETED. ENGINEERING REVIEW CONCLUDED ON NOVEMBER 12, 1991 THAT THE AS-FOUND CONFIGURATION OF THE FAILED JOINT DID NOT CONFORM TO DESIGN BASIS CRITERIA; HOWEVER, FAILURE OF THIS JOINT UNDER DESIGN BASIS CONDITIONS WOULD NOT HAVE PREVENTED THE STANDBY GAS TREATMENT SYSTEM FROM MAINTAINING A SIGNIFICANT VACUUM IN THE REACTOR BUILDING.

[36] DRESDEN 2 DOCKET 50-237 LER 91-031
CONTROL ROD DRIVE R-10 FAILURE TO LATCH DUE TO COLLET PISTON BINDING.
EVENT DATE: 092191 REPORT DATE: 100891 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223340) AT 0327 HOURS ON SEPTEMBER 21, 1991 WITH UNIT 2 AT 85% POWER, CONTROL ROD DRIVE (CRD) EXERCISING WAS BEING DUCTED ON CRD R-10 PER DRESDEN OPERATING SURVEILLANCE (DOS) 300-1, DAILY/WEEKLY CONTROL ROD DRIVE EXERCISE. WHILE ATTEMPTING TO EXERCISE CRD R-10 FROM POSITION "48" (FULL OUT) TO POSITION "46", THE CRD LED TO LATCH AT POSITIONS "46" AND "44". WITH QUALIFIED NUCLEAR ENGINEER (QNE) PERMISSION, AN ATTEMPT TO LATCH AT POSITION "42" WAS ALSO MADE AND FAILURE TO LATCH AGAIN OCCURRED. THE CRD WAS INSERTED AND CONTAINED AT POSITION "00" BY CONTINUOUSLY APPLYING AN EMERGENCY-IN SIGNAL. AT 0535 HOURS, CRD R-10 BEGAN DRIFT OUT; THE CRD WAS SCRAMMED TO POSITION "OO" (FULL IN) AND HELD IN POSITION BY CHARGING WATER PRESSURE. TESTS WERE CONDUCTED PER SPECIAL PROCEDURE (SP) 91-9-114 TO REMATCH CRD R-10 AND FLUSH THE CRD COLLET PISTON. DURING TESTING, CRD R-10 WAS SUCCESSFULLY RELATCHED. TESTING INDICATED THAT THERE IS NO LEAKAGE IN THE DIRECTIONAL CONTROL VALVES. THE CRD COLLET HOUSING WAS DETERMINED TO BE INTACT AND LATCHING FAILURE WAS DETERMINED TO BE A RESULT OF COLLET PISTON BINDING. UPON COMPLETION OF FLUSHING, LATCHING FAILURE AGAIN OCCURRED AND CRD R-10 WAS SCRAMMED TO POSITION "OO" WHERE IT SUCCESSFULLY LATCHED. THIS EVENT HAD MINIMAL SAFETY SIGNIFICANCE.

[37] DRESDEN 2 DOCKET 50-237 LER 91-032
ACTUATION OF GROUP I ISOLATION VALVES DUE TO PERSONNEL ERROR DURING PLACEMENT OF JUMPERS.
EVENT DATE: 102591 REPORT DATE: 112291 NSSS: GE TYPE: BWR

(NSIC 223463) ON 10/25/91 AT 1852 HOURS, WITH UNIT 2 IN COLD SHUTDOWN, DRESDEN OPERATING SURVEILLANCE (DOS) 0250-3, MAIN STEAM ISOLATION VALVE (MSIV) FAIL-SAFE CLOSURE TEST, WAS BEING PERFORMED. THIS TEST REQUIRES JUMPERS TO BE PLACED ACROSS THE CONTROL RELAYS FOR THE MSIVS TO PREVENT THEIR MOVEMENT DURING THE TEST. WHILE INSTALLING TO JUMPERS, TWO FUSES OPENED, RESULTING IN CLOSURE SIGNALS BEING SENT TO CERTAIN PRIMARY CONTAINMENT GROUP I ISOLATION VALVES. THE PERSONNEL INVOLVED INITIALLY BELIEVED THAT THE CAUSE OF THE OPEN FUSES WAS INADVERTENT GROUNDING OF THE JUMPERS DURING INSTALLATION. AT 2049 HOURS, IN PREPARATION FOR DOS 250-3, THE NSO ATTEMPTED TO RESET ONE SIDE OF THE GROUP I ISOLATION LOGIC AND ONE OF THE FUSES OPENED AGAIN. IT WAS THEN DISCOVERED THAT THE JUMPERS HAD BEEN INCORRECTLY INSTALLED ON THE MSIV CONTROL SWITCHES, RATHER THAN THE CONTROL RELAYS SPECIFIED IN DOS 250-3, WHICH RESULTED IN A PARTIAL GROUP I ISOLATION SIGNAL. THE JUMPERS WERE REMOVED AND THE CIRCUIT FUNCTION WAS VERIFIED. BOTH OF THE ELECTRICIANS INVOLVED HAVE BEEN COUNSELLED BY THE MASTER ELECTRICIAN AND APPROPRIATE

DISCIPLINARY ACTION WAS TAKEN. OPERABILITY OF THE ISOLATION LOGIC WAS NOT REQUIRE DURING COLD SHUTDOWN. A PREVIOUS EVENT INVOLVING AN UNRELATED SURVEILLANCE ERROR RESULTING IN A UNPLANNED START OF THE STANDBY GAS TREATMENT SYSTEM WAS REPORTED BY LER 91-4/050249.

[38] FARLEY 1 DOCKET 50-348 LER 91-011
INADEQUATE TEST OF CCW SAMPLE COOLING SUPPLY CHECK VALVES.
EVENT DATE: 102391 REPORT DATE: 112291 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: FARLEY 2 (PWR)
VENDOR: KEROTEST MANUFACTURING CORP.

(NSIC 223528) ON 10-23-91, DURING A PROCEDURE REVIEW, IT WAS DISCOVERED THAT FNP-2-STP-23.8, "COMPONENT COOLING WATER VALVE INSERVICE TEST" CONTAINED AN INADEQUATE SYSTEM LINEUP FOR DETERMINING IF COMPONENT COOLING WATER (CCW) CHECK VALVE Q2P17V288 WOULD SEAT AGAINST REVERSE FLOW AS REQUIRED. FNP-1-STP-23.8 WAS FOUND TO HAVE THE SAME DEFICIENCY. AS A RESULT OF THIS REVIEW IT WAS DETERMINED THAT CHECK VALVES Q1P17V288 AND Q2P17V288 HAD NOT BEEN ADEQUATELY TESTED PER TECHNICAL SPECIFICATION 4.0.5. BOTH CHECK VALVES WERE DECLARED INOPERABLE AND MANUALLY ISOLATED. EACH UNIT INITIATED A 72 HOUR LIMITING CONDITION FOR OPERATION (LCO) FOR THE ON-SERVICE TRAIN OF CCW AT 1630 ON 10-23-91. THESE CHECK VALVES ARE LOCATED IN THE CCW RETURN LINE FROM THE GROSS FAILED FUEL DETECTOR (GFFD) AND SAMPLE COOLERS FOR THEIR RESPECTIVE UNIT. THIS EVENT WAS CAUSED BY PROCEDURAL INADEQUACY. FNP-1-STP-23.8 AND FNP-2-STP-23.8 WERE REVISED ON 10-23-91 TO PROVIDE A CORRECT SYSTEM ALIGNMENT FOR THE DETERMINATION THAT CHECK VALVE Q1(2)P17V288 PREVENTS CCW FLOW IN THE REVERSE DIRECTION. A TEST OF THE UNIT 1 VALVE USING THE REVISED PROCEDURE WAS PERFORMED SATISFACTORILY, AND THE CORRESPONDING LCO WAS CLEARED AT 2340 ON 10-23-91. THE UNIT 2 VALVE, HOWEVER, DID NOT PASS THE REVISED PROCEDURE CRITERIA. A MAINTENANCE WORK REQUEST (MWR) WAS WRITTEN, AND CORRECTIVE MAINTENANCE WAS PERFORMED.

[39] FITZPATRICK DOCKET 50-333 LER 91-015 REV 01
UPDATE ON HIGH PRESSURE COOLANT INJECTION SYSTEM INOPERABLE DUE TO SLOW RESPONSE TIME.
EVENT DATE: 081991 REPORT DATE: 112591 NSSS: GE TYPE: BWR

(NSIC 223499) SURVEILLANCE OF THE HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI) (BJ), REQUIRED DURING A REACTOR START-UP, WAS PERFORMED ON 8/19/91 IN ACCORDANCE WITH TECHNICAL SPECIFICATION 4.5.C.1. TEST DATA INDICATED THAT THE FULL FLOW INJECTION TIME OF 30 SECONDS, SPECIFIED IN THE FINAL SAFETY ANALYSIS REPORT (FSAR) WAS EXCEEDED BY 0.84 SECONDS. THE TURBINE STOP VALVE OPENING TIME OF 25.3 SECONDS SPECIFIED BY THE INSERVICE TESTING (IST) PROGRAM WAS EXCEEDED BY 2.46 SECONDS. ALSO, THE MAIN PUMP DISCHARGE INBOARD ISOLATION VALVE (23MOV-19) CALCULATED RESPONSE TIME OF 32.7 SECONDS EXCEEDED THE IST REQUIREMENT BY 2.7 SECONDS. HPCI WAS DECLARED INOPERABLE INITIATING A 7-DAY LIMITING CONDITION FOR OPERATION (LCO). DURING THE NEXT 2 DAYS, HPCI WAS SUBJECTED TO 3 TEST RUNS. THE EXTENDED TIME TO FULL FLOW WAS CAUSED BY THE EXCESSIVE TIME PERIOD FOR THE TURBINE STOP VALVE (23HOV-1) TO COME OFF ITS FULLY SHUT SEAT. OIL PRESSURE TO OPEN THE STOP VALVE WAS DELAYED DUE TO AIR IN THE DUPLEX FILTER HOUSING. HPCI MET THE IST, ST, AND FSAR CRITERIA AND WAS DECLARED OPERABLE ON 8/21/91. HPCI WAS INOPERABLE FOR 45.75 HOURS. CORRECTIVE ACTIONS INCLUDE ADJUSTING OIL SYSTEM PRESSURE AND PLACING STANDBY FILTER IN SERVICE. LERS 87-010, 89-019, AND 89-025 ARE RELATED.

[40] FITZPATRICK DOCKET 50-333 LER 91-022
PRIMARY CONTAINMENT VENT AND PURGE VALVES ISOLATED BY SPURIOUS HIGH SIGNAL TRIP FROM CONTAINMENT HIGH RANGE RADIATION MONITOR DUE TO ELECTRICAL NOISE DURING TESTING.
EVENT DATE: 101591 REPORT DATE: 111391 NSSS: GE TYPE: BWR

(NSIC 223497) A SPURIOUS PRIMARY CONTAINMENT (NH) HIGH RADIATION MONITOR A (IL) ISOLATION SIGNAL OCCURRED AT 0201 ON 10/15/91. THE PRIMARY CONTAINMENT VENT AND PURGE VALVES WHICH ARE ISOLATED BY THIS SIGNAL WERE ALREADY IN THE CLOSED POSITION. REDUNDANT INSTRUMENTATION CONFIRMED THAT CONTAINMENT RADIATION LEVELS

WERE NORMAL AND LOGIC CIRCUITRY RESET IMMEDIATELY. THE TRANSIENT INPUT SIGNAL WHICH ACTIVATED THE MONITOR TRIP IS BELIEVED TO HAVE BEEN GENERATED BY THE ACTUATION OF THE OFF-GAS VENT LINE ISOLATION DELAY TIMER WHICH WAS INITIATED DURING THE PERFORMANCE OF SURVEILLANCE TESTING. THE EXACT NATURE OF THE INTERACTION HAS NOT BEEN IDENTIFIED AT THIS TIME. A PROCEDURE IS BEING DEVELOPED TO MORE CLOSELY EXAMINE THE RELATIONSHIP OF ACTUATION TO THE NOISE TRANSIENT. DURING THE 1992 REFUELING OUTAGE INSTRUMENT PANELS CONTAINING THE PRIMARY CONTAINMENT HIGH RANGE RADIATION MONITORS WILL BE INSPECTED AND EVALUATED TO DETERMINE IF IMPROVEMENTS ARE POSSIBLE WHICH WILL REDUCE RANDOM TRANSIENT OR SPURIOUS ELECTRIC NOISE WHICH CAN CAUSE ISOLATION SIGNALS. RELATED LERS: 90-028, 91-001, AND 91-018.

[41] FT. CALHOUN 1 DOCKET 50-285 LER 91-010 REV 01
UPDATE ON AUXILIARY STEAM PIPING IN ROOM 57 OUTSIDE DESIGN BASIS.
EVENT DATE: 051791 REPORT DATE: 112291 NSSS: CE TYPE: PWR

(NSIC 223482) 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 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 STEAM SUPPLY TO THE AS PIPING IN THE AFFECTED ROOM. LONG TERM CORRECTIVE ACTIONS INCLUDED COMPLETION OF AN ENGINEERING ANALYSIS TO EVALUATE THE REMAINDER OF THE AS SYSTEM SIMILAR HELB CONCERNS AND ISOLATION OF AS SUPPLY PIPING TO THE DIESEL GENERATOR ROOMS PRIOR TO ENERGIZING THE AS HEADER TO THE AUXILIARY BUILDING.

[42] FT. CALHOUN 1 DOCKET 50-285 LER 91-026
EXPIRED NRC LICENSED OPERATOR MEDICAL EXAMINATION.
EVENT DATE: 110591 REPORT DATE: 120591 NSSS: CE TYPE: PWR

(NSIC 223484) ON 11/5/91, WHILE FORT CALHOUN STATION (FCS) WAS OPERATING AT 100 PERCENT POWER (MODE 1), IT WAS DISCOVERED THAT A SENIOR REACTOR OPERATOR (SRO) AT FCS DID NOT HAVE A CURRENT NRC LICENSED OPERATOR MEDICAL EXAMINATION AS REQUIRED IN 10 CFR 55.21. BEFORE, IT WAS CONSERVATIVELY ASSUMED THAT FCS DID NOT MEET THE MANNING REQUIREMENTS SPECIFIED IN TECHNICAL SPECIFICATION TABLE 5.2-1 "MINIMUM SHIFT CREW COMPOSITION" DURING TIMES THE SRO STOOD WATCH. THIS REPORT IS BEING SUBMITTED PURSUANT TO CFR 50.73(A)(2)(I)(B). THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL BASED ON THE FACT THAT THE SRO INVOLVED PASSED A SUBSEQUENT NRC PHYSICAL WITH NO RESTRICTIONS OR DISCREPANCIES NOTED BY THE ATTENDING PHYSICIAN. THE ROOT CAUSE OF THIS INCIDENT IS A LACK OF SINGLE POINT CONTROL (PROCEDURE) OF THE NRC LICENSED OPERATOR (LO) MEDICAL EXAMINATION PROGRAM. CONTRIBUTING CAUSES INCLUDE THE FOLLOWING: ADMINISTRATIVE PROCEDURES WITH INADEQUATELY DEFINED RESPONSIBILITIES, ADEQUATE TRACKING PROGRAM FOR NRC LICENSED OPERATORS (LO) PHYSICALS, AND A LACK OF UNDERSTANDING OF NRC LO PHYSICAL REQUIREMENTS BY PERSONNEL INVOLVED. CORRECTIVE ACTIONS CONSIST OF ESTABLISHING SINGLE DEFINITIVE PROCEDURE TO CONTROL THE PROCESS, UPGRADING THE OPTIM (ONLINE PERSONNEL TRAINING INFORMATION MANAGEMENT) COMPUTER PROGRAM, AND ISSUING A MANAGEMENT MEMORANDUM THAT ENSURES THAT LICENSED OPERATORS ARE AWARE OF THE REQUIREMENTS OF 10 CFR 55.21.

[43] FT. CALHOUN 1 DOCKET 50-285 LER 91-024
UNPLANNED ENGINEERED SAFETY FEATURE ACTUATION AFTER PULLING FUSES.
EVENT DATE: 110691 REPORT DATE: 120691 NSSS: CE TYPE: PWR

(NSIC 223483) AT 0831 HOURS, ON NOVEMBER 6, 1991, WHILE FORT CALHOUN STATION WAS OPERATING AT 100 PERCENT POWER (MODE 1), AN ELECTRICIAN WAS REMOVING FUSES FROM INSIDE A CONTROL PANEL IN CONTROL ROOM WHEN HE PULLED A FUSE FROM THE WRONG FUSE BLOCK. THE REMOVAL OF THE CAUSED THE VENTILATION ISOLATION ACTUATION SIGNAL (VIAS) RELAYS B/94-1, B/94-2 AND B/94-3 TO DE-ENERGIZE. LOSS OF POWER TO THE VIAS RELAYS CAUSED TWO CONTAINMENT RADIATION MONITOR SAMPLE VALVES TO CLOSE AND THE CONTROL ROOM AIR CONDITIONING UNIT AND RADIATION MONITOR RM-065 TO START. THE ELECTRICIAN WAS INSTRUCTED TO REINSTALL THE FUSE AND POWER WAS RESTORED TO THE VIAS RELAYS ALLOWING THE OPERATORS TO RESET THE COMPONENTS. THE COMPONENTS WERE RETURNED TO THEIR NORMAL CONFIGURATION WITHIN TWO MINUTES OF THEIR ACTUATION. THIS REPORT IS BEING SUBMITTED PURSUANT TO 10 CFR 50.73(A)(2)(IV). THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL BASED ON THE SHORT DURATION OF THE EVENT AND THE COMPONENTS WHICH WERE AFFECTED. THE CAUSE OF THIS EVENT WAS ATTRIBUTED TO PERSONNEL ERROR. THE ELECTRICIAN FAILED TO CHECK THE DANGER TAG SHEET TO ENSURE THAT HE WAS IN THE CORRECT FUSE BLOCK PRIOR TO PULLING THE FUSE. CORRECTIVE ACTIONS INCLUDE A PROCEDURE CHANGE AND AN EVALUATION OF THE NEED TO ESTABLISH A FORMAL SELF-CHECKING PROGRAM FOR THE ELECTRICAL MAINTENANCE AND INSTRUMENT AND CONTROL GROUPS.

[44] HATCH 1 DOCKET 50-321 LER 91-025
LESS THAN ADEQUATE PROCEDURE RESULTS IN TECHNICAL SPECIFICATIONS NONCOMPLIANCE.
EVENT DATE: 102491 REPORT DATE: 112291 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 2 (BWR)

(NSIC 223511) ON 10/24/91, AT 1300 CDT, UNIT 1 WAS IN A SCHEDULED MAINTENANCE AND REFUELING OUTAGE WITH THE VESSEL DISASSEMBLED AND WITH NO FUEL IN THE VESSEL. UNIT 2 WAS IN THE RUN MODE AT APPROXIMATELY 2436 CMWT (APPROXIMATELY 100 PERCENT OF RATED THERMAL POWER). AT THAT TIME, NONLICENSED PERSONNEL DETERMINED THAT THE AUTOMATIC FUNCTION OF MAIN CONTROL ROOM ENVIRONMENTAL CONTROL SYSTEM (MCRECS) EXHAUST DAMPER 1Z41-F018B WAS NOT BEING TESTED AS REQUIRED BY THE PLANT'S TECHNICAL SPECIFICATIONS. UNIT 1 TECHNICAL SPECIFICATIONS TABLE 4.2-8 AND UNIT 2 TECHNICAL SPECIFICATIONS SECTION 4.7.2.E.3 REQUIRE A LOGIC SYSTEM FUNCTIONAL TEST OF THE PRESSURIZATION MODE FUNCTION OF MCRECS TO BE PERFORMED AT LEAST ONCE PER 18 MONTHS. DAMPER 1Z41-F018B AUTOMATICALLY CLOSES AS PART OF THE AUTOMATIC TRANSFER OF MCRECS TO THE PRESSURIZATION MODE. HOWEVER, ITS AUTOMATIC FUNCTION WAS NOT BEING CHECKED DURING THE ASSOCIATED LOGIC SYSTEM FUNCTIONAL TESTING. LICENSED PERSONNEL WERE NOTIFIED OF THE DEFICIENCY. AT 1500 CDT, A CLEARANCE WAS ISSUED TO ISOLATE THE MOTIVE FORCE (THE ACTUATOR'S AIR SUPPLY) FROM THE DAMPER ACTUATOR TO ENSURE THAT THE DAMPER WAS MAINTAINED IN THE CLOSED POSITION UNTIL IT COULD BE PROPERLY TESTED. THE CAUSE OF THE EVENT WAS A LESS THAN ADEQUATE PROCEDURE. PROCEDURE 42SV-Z41-001-0S, DID NOT INCLUDE VERIFICATION OF THE DAMPER CLOSING FOLLOWING AN AUTOMATIC TRANSFER OF MCRECS TO THE PRESSURIZATION MODE.

[45] HATCH 1 DOCKET 50-321 LER 91-026
PRESSURE PERTURBATION CAUSES FALSE LOW REACTOR WATER LEVEL SIGNAL AND ESF ACTUATIONS.
EVENT DATE: 110691 REPORT DATE: 120491 NSSS: GE TYPE: BWR

(NSIC 223510) ON 11/6/91 AT 0441 CST, UNIT 1 WAS IN THE REFUEL MODE WITH THE CORE COMPLETELY LOADED AND REACTOR PRESSURE VESSEL REASSEMBLY IN PROGRESS. AT THAT TIME, A FULL REACTOR SCRAM SIGNAL OCCURRED FROM A FALSE LOW WATER LEVEL SIGNAL CAUSED BY A PRESSURE PERTURBATION ON A COMMON SENSING LINE. THIS OCCURRED WHEN A NEW INSTRUMENT ISOLATION VALVE WAS OPENED DURING ROUTINE VALVE LINEUPS. ALL CONTROLS RODS WERE FULLY INSERTED; THUS, NO ROD MOTION RESULTED. SEVERAL GROUP 2 PRIMARY CONTAINMENT ISOLATION SYSTEM VALVES ALSO RECEIVED AN ISOLATION SIGNAL ON LOW WATER LEVEL AND VALVES 1E11-F008, SHUTDOWN COOLING SUCTION OUTBOARD ISOLATION VALVE, AND 1E11-F015B, RESIDUAL HEAT REMOVAL INBOARD INJECTION VALVE, CLOSED PER DESIGN. THIS RESULTED IN INTERRUPTION OF THE NORMAL SHUTDOWN COOLING FLOW PATH. THE TRIP SIGNAL CLEARED 12 SECONDS AFTER ITS RECEIPT. LICENSED OPERATIONS PERSONNEL THEN RESET THE SCRAM AND ISOLATION SIGNALS AND, AT 0517 CST, RESTORED SHUTDOWN COOLING FLOW. DURING THE 36 MINUTES SHUTDOWN COOLING WAS OUT OF SERVICE, REACTOR WATER TEMPERATURE INCREASED FROM 105 TO ONLY 109 DEG F. THE CAUSE OF THIS EVENT WAS A LACK OF ADEQUATE CONTROLS GOVERNING THE POSITION OF THE NEW

INSTRUMENT ISOLATION VALVE DURING INSTRUMENT VENTING ACTIVITIES. THE VALVE WAS NOT OPENED PRIOR TO INSTRUMENT VENTING ACTIVITIES AS IT SHOULD HAVE BEEN.

[46] HATCH 1 DOCKET 50-321 LER 91-027
IMPROPER SENSING LINE INSTALLATION RESULTS IN ESF ACTUATION.
EVENT DATE: 110891 REPORT DATE: 120491 NSSS: GE TYPE: BWR

(NSIC 223509) ON 11/08/91, UNIT 1 WAS IN THE REFUEL MODE WITH VESSEL REASSEMBLY IN PROGRESS. A FLOW CONTROLLER FOR A REACTOR WATER CLEANUP (RWCU) SYSTEM FLOW CONTROL VALVE, 1G31-F033, HAD BEEN REPAIRED. AT 0020 CST, THE VALVE WAS BEING OPENED FOR TESTING PURPOSES WHEN THE RWCU LEAK DETECTION SYSTEM (LDS) ANNUNCIATOR ALARM WAS RECEIVED FOLLOWED BY THE AUTOMATIC CLOSURE OF GROUP 5 PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) VALVES 1G31-F001 AND 1G31-F004. THE OPERATING RWCU PUMP TRIPPED AS DESIGNED. A SECOND EVENT OCCURRED ON 11/15/91 WITH UNIT 1 IN THE REFUEL MODE AND THE VESSEL FULLY ASSEMBLED. A RESTRICTING ORIFICE WAS TO BE INSTALLED IN THE RWCU SYSTEM PIPING NEAR VALVE 1G31-F033. THE VALVE WAS REQUIRED TO BE OPENED IN ORDER TO ESTABLISH THE CLEARANCE BOUNDARY. WHEN THE VALVE WAS OPENED AT 1341 CST, AN ISOLATION OF THE GROUP 5 PCIS VALVES OCCURRED ALONG WITH A TRIP OF THE OPERATING RWCU PUMP. NO IMMEDIATE OPERATOR ACTIONS WERE REQUIRED IN EITHER EVENT. THE CAUSE OF THIS EVENT WAS CONCLUDED TO BE AN IMPROPER INSTALLATION OF THE INSTRUMENT SENSING LINES FOR DIFFERENTIAL PRESSURE TRANSMITTER 1G31-N012. THIS ROUTING RESULTS IN THE INSTRUMENT TUBING PARTIALLY DRAINING WHENEVER THE RWCU SYSTEM PIPING AT THE INSTRUMENT LINE TAPS IS DRAINED.

[47] HATCH 2 DOCKET 50-366 LER 91-020
SPURIOUS BREAKER TRIP RESULTS IN ENGINEERED SAFETY FEATURES ACTUATION.
EVENT DATE: 110591 REPORT DATE: 120291 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 1 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223534) ON 11/05/91 AT 0350 CST, UNIT 2 WAS IN THE RUN MODE AT A POWER LEVEL OF 2436 CMWT (100% RATED THERMAL POWER). AT THAT TIME, PROTECTIVE BREAKER 2C71-52-3D ON THE OUTPUT OF THE 'B' MOTOR-GENERATOR SET IN THE REACTOR PROTECTION SYSTEM (RPS) POWER SUPPLY TRIPPED. THIS CAUSED A LOSS OF POWER TO THE 'B' CHANNELS OF THE RPS, PROCESS RADIATION MONITORS, NEUTRON MONITORING SYSTEM (NMS), PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS), AND OFFGAS RADIATION MONITORING SYSTEM. THE "FAIL-SAFE" DESIGN OF THESE SYSTEMS RESULTS IN THEIR ASSUMING THE 'TRIPPED' STATE WHEN POWER IS INTERRUPTED. THEREFORE, THESE TRIPS ALSO RESULTED IN CLOSURE OF VARIOUS PCIS VALVES AND IN THE MAIN CONTROL ROOM ENVIRONMENTAL CONTROL SYSTEM (MCREC) ENTERING THE PRESSURIZATION MODE. LICENSED SHIFT PERSONNEL RESTORED POWER TO RPS BUS 'B' VIA ITS ALTERNATE SUPPLY BY 0355 CST PER PROCEDURE 34AB-OPS-066-2S, "LOSS OF RPS BUS," AND ALL AFFECTED EQUIPMENT WAS SUBSEQUENTLY RESTORED TO ITS NORMAL CONFIGURATION. RPS BUS 'B' WAS RESTORED TO ITS NORMAL POWER SUPPLY BY 1148 CST. THE CAUSE OF THIS EVENT WAS A SPURIOUS TRIP OF RPS PROTECTIVE CIRCUIT BREAKER 2C71-52-3D. INVESTIGATION INDICATED THAT NO SYSTEM CONDITION EXISTED WHICH WOULD HAVE RESULTED IN A VALID TRIP. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED ADJUSTING THE RPS OUTPUT VOLTAGE AS A PRECAUTION. THIS ACTION IS COMPLETE. IN ADDITION, A DESIGN REVIEW OF THE BREAKER APPLICATION WILL BE PERFORMED BY 02/28/92.

[48] HATCH 2 DOCKET 50-366 LER 91-021
PERSONNEL ERROR RESULTS IN TECHNICAL SPECIFICATION NONCOMPLIANCE.
EVENT DATE: 110591 REPORT DATE: 120491 NSSS: GE TYPE: BWR

(NSIC 223535) ON 11/5/91, AT 2230 CST, UNIT 2 WAS IN THE RUN MODE AT 2436 CMWT 100% RATED THERMAL POWER). AT THAT TIME, THE UNIT 2 LICENSED SHIFT SUPERVISOR DETERMINED THAT THE PLANT HAD NOT BEEN IN COMPLIANCE WITH ACTION A. OF UNIT 2 TECHNICAL SPECIFICATIONS SECTION 3.3.6.4. SPECIFICALLY, POST-ACCIDENT MONITORING SYSTEM RECORDER 2T48-R608 HAD BEEN INOPERABLE FOR GREATER THAN 30 DAYS. SINCE THE CONDITION WAS NOT IDENTIFIED, THE REQUIRED TECHNICAL SPECIFICATIONS ACTION WAS NOT TAKEN. RECORDER 2T48-R608 PROVIDES POST-ACCIDENT MONITORING CAPABILITY OF THE DRYWELL AND SUPPRESSION POOL PRESSURES. THE SHIFT SUPERVISOR INITIATED A DEFICIENCY CARD TO ENSURE THAT THE EVENT WAS PROPERLY INVESTIGATED. BY THEN THE

RECORDER HAD ALREADY BEEN RETURNED TO SERVICE; THEREFORE, NO LIMITING CONDITIONS FOR OPERATION WERE ENTERED. THE CAUSE OF THE EVENT WAS COGNITIVE PERSONNEL ERROR ON THE PART OF CONTROL ROOM PERSONNEL. IT WAS CONCLUDED FROM AN INVESTIGATION THAT THE RECORDER WAS MOST LIKELY TURNED OFF ON 9/26/91 IN ORDER TO REPLACE ONE OF THE RECORDER PENS AND WAS INADVERTENTLY NOT TURNED BACK ON. SUBSEQUENTLY, OPERATORS NOTED THAT THE CHART PAPER WAS NOT ADVANCING BUT THEY BELIEVED THE PENS WERE TRACKING PROPERLY THE OPERATORS CONSIDERED THE RECORDER OPERABLE AND INITIATED A DEFICIENCY CARD ON THE CHART PAPER NOT ADVANCING. THE PROBLEM WAS NOT INVESTIGATED AND, THUS, THE RECORDER NOT FOUND SWITCHED OFF UNTIL 11/5/91.

[49] INDIAN POINT 2 DOCKET 50-247 LER 91-008 REV 01
UPDATE ON SAFETY INJECTION DESIGN INCONSISTENCIES.
EVENT DATE: 040391 REPORT DATE: 112791 NSSS: WE TYPE: PWR

(NSIC 223467) DURING A REFUELING OUTAGE, AT CON EDISON'S INITIATIVE THE SIX HIGH PRESSURE SAFETY INJECTION FLOW ORIFICES WERE REMOVED AND INSPECTED. IT WAS DETERMINED THAT THREE PLATES WERE INTERCHANGED AND ONE WAS REVERSED. NEW ORIFICE PLATES WERE INSTALLED AND THE FLOW TRANSMITTERS WERE RECALIBRATED. FULL FLOW TESTS AND FLOW BALANCING TESTS WERE CONDUCTED AND THE RESULTS MET THE ACCEPTANCE CRITERIA.

[50] INDIAN POINT 3 DOCKET 50-286 LER 91-011
SAFEGUARDS INITIATION TRAIN 'B' INOPERABLE DUE TO FAULTED INDICATING LAMP.
EVENT DATE: 101591 REPORT DATE: 111491 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223485) ON OCTOBER 15, 1991, WITH THE REACTOR OPERATING AT 100 PERCENT POWER, AN INDICATING LAMP ON SAFEGUARDS INITIATION TRAIN B ELECTRICALLY FAULTED. AS A RESULT, A TEN AMPERE CONTROL POWER FUSE BLEW, RENDERING THE AUTOMATIC INITIATION FEATURE OF SAFEGUARDS TRAIN B INOPERABLE. THE LIMITING CONDITION FOR OPERATION ACTION STATEMENT OF TECHNICAL SPECIFICATIONS FOR SAFETY INJECTION AND RESIDUAL HEAT REMOVAL SYSTEMS WAS ENTERED AND PREPARATIONS FOR A PLANT SHUTDOWN WERE STARTED. AN IDENTICAL REPLACEMENT FUSE WAS NOT AVAILABLE, SO A TEMPORARY MODIFICATION WAS WRITTEN TO PERMIT USE OF A FIVE AMPERE FUSE OF THE SAME CLASS. THE TEMPORARY MODIFICATION VERIFIED THAT THE SMALLER FUSE WAS ADEQUATE FOR THE LOADS OF A SAFEGUARDS ACTUATION AND WOULD NOT DEGRADE ANY PLANT SYSTEM FUNCTION. A TECHNICIAN INSTALLED THE SMALLER FUSE BEFORE ANY CHANGE WAS MADE TO REACTOR POWER. TECHNICIANS EXCHANGED THE FUSE WITH THE CORRECT FUSE ON OCTOBER 18, 1991. TECHNICIANS REPLACED THE DAMAGED LAMP SOCKET ON OCTOBER 19, 1991 WHILE THE PLANT WAS AT HOT SHUTDOWN DUE TO A PROBLEM IN THE MAIN GENERATOR. THE PLANT STAFF WILL DETERMINE IF A BETTER QUALITY INDICATING LAMP CAN BE PROCURED FOR USE IN THE SAFEGUARDS INITIATION RACKS.

[51] KEWAUNEE DOCKET 50-305 LER 91-009
ERROR IN SAFETY INJECTION ACCUMULATOR LEVEL INDICATION CAUSED BY NOT COMPENSATING FOR EFFECTS OF NITROGEN DENSITY DURING CALIBRATION.
EVENT DATE: 090991 REPORT DATE: 111591 NSSS: WE TYPE: PWR
VENDOR: I.T.T. BARTON, INC.

(NSIC 223518) ON SEPTEMBER 9, 1991, WITH THE PLANT AT 99% POWER, AN ERROR IN THE CALIBRATION OF THE SAFETY INJECTION ACCUMULATOR LEVEL TRANSMITTERS WAS IDENTIFIED. THE ERROR WAS CAUSED BY THE FAILURE TO CONSIDER THE EFFECTS OF THE NITROGEN DENSITY AT 750 PSIG ON THE ITT BARTON MODEL NO. 384352 DIFFERENTIAL PRESSURE TRANSMITTERS USED FOR LEVEL MEASUREMENT. THE LEVEL TRANSMITTERS ARE USED TO MEASURE LIQUID LEVEL IN THE ACCUMULATORS OVER A TWENTY-EIGHT (28) INCH SPAN AND PROVIDE AN INDICATION OF 0-100% LEVEL. WITHOUT COMPENSATION FOR THE EFFECTS OF NITROGEN DENSITY AT 750 PSIG, THE LEVEL ERROR, WHICH IS LINEAR OVER THE MEASURED SPAN, IS 5.52% HIGH AT 0% ACTUAL LEVEL AND 0% ERROR AT 100% ACTUAL LEVEL. THE ROOT CAUSE FOR THE EVENT COULD NOT BE CONCLUSIVELY DETERMINED. THE METHODOLOGY FOR DETERMINING THE CALIBRATION REFERENCE VALUES WAS DEVELOPED PRIOR TO INITIAL PLANT STARTUP AND HAD REMAINED UNCHANGED UNTIL THE IDENTIFICATION OF THIS EVENT. AS AN INTERIM MEASURE, TO ENSURE THAT THE TECHNICAL SPECIFICATION FOR MINIMUM ACCUMULATOR WATER VOLUME IS SATISFIED THE LOW LEVEL ALARM SETPOINTS HAVE

BEEN SET AT 6.5% ABOVE THE LEVEL REQUIRED BY THE SPECIFICATION. DURING THE 1992 REFUELING OUTAGE THE TRANSMITTERS WILL BE RECALIBRATED TO COMPENSATE FOR THE LEVEL ERROR DUE TO THE NITROGEN DENSITY.

[52] KEWAUNEE DOCKET 50-305 LER 91-011
REACTOR PROTECTION SYSTEM OVERPOWER AND OVERTEMPERATURE DELTA-T TRIP SETPOINT FOUND LESS CONSERVATIVE THAN TECHNICAL SPECIFICATION REQUIREMENTS.
EVENT DATE: 102391 REPORT DATE: 112291 NSSS: WE TYPE: PWR

(NSIC 223517) AT 1000 ON OCTOBER 23, 1991, WITH THE PLANT AT 100% POWER, IT WAS DETERMINED THAT DURING THE PERIOD OF MAY 10-17, 1991, THREE OF THE FOUR CHANNELS FOR REACTOR COOLANT SYSTEM (RCS) OVERTEMPERATURE AND OVERPOWER (OT/OP) DELTA-T PROTECTION HAD REACTOR TRIP SETPOINTS AT VALUES LESS CONSERVATIVE THAN THAT REQUIRED BY THE TECHNICAL SPECIFICATIONS (TS). DURING THE 1991 REFUELING OUTAGE A NUMBER OF PREVIOUSLY PLUGGED STEAM GENERATOR (S/G) TUBES WERE RESTORED TO SERVICE BY SLEEVING THE DEFECTIVE TUBES. EDDY CURRENT TESTING IDENTIFIED OTHER TUBES WHICH REQUIRED PLUGGING. HOWEVER, THE NET RESULT WAS AN INCREASE IN AREA FOR RCS FLOW. THIS WAS CONFIRMED ON MAY 17, 1991, BY RCS FLOW TESTING WHICH INDICATED A 1.4% INCREASE FROM THE PREVIOUS FUEL CYCLE FLOW. THE INCREASED FLOW CAUSED A CORRESPONDING DECREASE IN THE RCS DELTA-T AT RATED POWER. THE TECHNICAL SPECIFICATIONS REQUIRE THAT THE OP DELTA-T AND OT DELTA-T TRIP SETPOINTS BE </= 1.10 AND </= 1.11, RESPECTIVELY, OF INDICATED DELTA-T AT RATED POWER. ON OCTOBER 23, 1991, FOLLOWING THE REVIEW OF THE NEWLY DETERMINED RATED POWER DELTA-T VALUES, IT WAS DISCOVERED THAT DURING THE PERIOD OF MAY 10 (PLANT STARTUP) TO MAY 17, 1991, THREE OF THE FOUR CHANNELS HAD OP/OT DELTA-T TRIP SETPOINTS WHICH WERE AN AVERAGE OF 0.56% LESS CONSERVATIVE THAN THE TECHNICAL SPECIFICATIONS LIMITS.

[53] LA SALLE 1 DOCKET 50-373 LER 91-014
LOST STATION VENT STACK PARTICULATE COMPOSITE SAMPLE BY OFFSITE VENDOR.
EVENT DATE: 101591 REPORT DATE: 111491 NSSS: GE TYPE: BWR

(NSIC 223539) ON OCTOBER 15, 1991, THE CHEMISTRY SERVICES DEPARTMENT AT LASALLE COUNTY STATION RECEIVED A TELEPHONE CALL FROM A REPRESENTATIVE OF TMA/NORCAL, (A VENDOR THAT PROVIDES RADIOANALYTICAL ANALYSES SERVICE TO THE STATION) INFORMING THE STATION THAT THEY COULD NOT FIND THE MONTHLY STATION VENT STACK PARTICULATE COMPOSITE SAMPLE FOR JULY, 1991. DUE TO THE LOSS OF THIS SAMPLE, THE SURVEILLANCE REQUIREMENTS IN TECHNICAL SPECIFICATION TABLE 4.11.2-1, FOR A MONTHLY GROSS ALPHA ANALYSIS AND A QUARTERLY STRONTIUM 89/90 ANALYSIS ON COMPOSITE PARTICULATE SAMPLES COULD NOT BE PERFORMED. THE EXACT CAUSE OF THE LOST SAMPLE IS NOT KNOWN. THE CHEMISTRY SERVICES DEPARTMENT BELIEVES THAT THE SAMPLE WAS IN THE SHIPPING CONTAINER WHEN IT LEFT THE SITE. THE VENDOR CANNOT DETERMINE IF THE SAMPLE WAS LOST BEFORE RECEIPT, OR WHETHER IT WAS LOST AT THEIR FACILITY. GAMMA ISOTOPIC AND TRITIUM ANALYSES OF THE SAMPLES OBTAINED FROM THE STATION VENT STACK WERE REVIEWED FOR THE MONTHS OF JUNE, JULY AND AUGUST 1991 AND SHOWED A NORMAL RELEASE TREND. THEREFORE, IT IS BELIEVED THAT THE LOST DATA WOULD ALSO BE NORMAL. A REVISION TO CHEMISTRY PROCEDURE, LCP-410-9, WILL BE MADE REQUIRING THE STATION QUALITY CONTROL DEPARTMENT TO REVIEW THE SHIPMENT OF MONTHLY TECHNICAL SPECIFICATION COMPOSITES PRIOR TO LEAVING THE STATION TO ENSURE ALL THE SAMPLES ARE IN THE SHIPMENT WHEN IT LEAVES THE SITE.

[54] LA SALLE 1 DOCKET 50-373 LER 91-017
REACTOR CORE ISOLATION COOLING OVERSPEED TRIP DUE TO GOVERNOR VALVE STICKING OPEN.
EVENT DATE: 102391 REPORT DATE: 112191 NSSS: GE TYPE: BWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV

(NSIC 223541) ON OCTOBER 23, 1991, AT 1031 HOURS, WHILE UNIT 1 WAS IN OPERATIONAL CONDITION 1 (RUN), AT 100% POWER, THE REACTOR CORE ISOLATION COOLING (RCIC) (BN) TURBINE TRIPPED ON MECHANICAL OVERSPEED. THIS EVENT OCCURRED WHILE THE OPERATING DEPARTMENT WAS PERFORMING THE QUARTERLY SURVEILLANCE THAT DEMONSTRATES THE COLD QUICK START CAPABILITY OF THE RCIC SYSTEM, LASALLE OPERATING SURVEILLANCE, LOS-RI-Q4, "REACTOR CORE ISOLATION COOLING SYSTEM COLD QUICK START." THE CAUSE FOR THE MECHANICAL OVERSPEED DURING THE SURVEILLANCE WAS FAILURE OF THE GOVERNOR VALVE TO CLOSE DURING THE START OF THE SYSTEM. IN AN ATTEMPT TO IDENTIFY THE ROOT

CAUSE OF THE PROBLEM, THE VALVE WAS DISASSEMBLED, AND INSPECTED. FOLLOWING INSPECTION OF THE MECHANICAL VALVE, THE GOVERNOR VENDOR, WOODWARD GOVERNOR COMPANY, INSPECTED THE ELECTRONIC CONTROLS AND THE ELECTRONIC GOVERNOR ACTUATOR FOR THE SYSTEM. AT THE TIME OF THIS INCIDENT THE HIGH PRESSURE CORE SPRAY (HPCS) SYSTEM, AND THE OTHER EMERGENCY CORE COOLING SYSTEMS WERE FULLY OPERABLE. THE RCIC SYSTEM WAS DECLARED INOPERABLE AND WORK REQUESTS WERE WRITTEN TO INVESTIGATE AND REPAIR THE PROBLEM. THIS EVENT IS REPORTED TO THE NUCLEAR REGULATORY COMMISSION AS A LICENSEE EVENT REPORT IN ACCORDANCE WITH 10CFR50.73(A)(2)(V) DUE TO RCIC BEING DECLARED INOPERABLE (LOSS OF A SAFETY SYSTEM FUNCTION).

[55]   LA SALLE 1   DOCKET 50-373   LER 91-016
DIVISION 3 125 VDC BATTERY INOPERABLE DUE TO A LOOSE CONNECTION.
EVENT DATE: 102491   REPORT DATE: 112291   NSSS: GE   TYPE: BWR

(NSIC 223540) ON OCTOBER 24, 1991, WITH UNIT 1 IN OPERATIONAL CONDITION 1 (RUN) AT 98% POWER, THE DIVISION 3 125 VOLT DC BATTERY WAS DECLARED INOPERABLE AT 1045 HOURS WHEN A LOOSE CABLE CONNECTION ON CELL #1 WAS REPORTED. SINCE THE DIVISION 3 BATTERY IS THE EMERGENCY DC POWER SUPPLY FOR THE HIGH PRESSURE CORE SPRAY (HPCS, HP)(BG) SYSTEM, THE HPCS SYSTEM WAS DECLARED INOPERABLE. THE REACTOR CORE ISOLATION COOLING (RCIC, RI) (BN) SYSTEM WAS ALREADY INOPERABLE DUE TO A FAILED OPERABILITY SURVEILLANCE. THE BATTERY CABLE CONNECTION WAS QUICKLY RETORQUED TO THE PROPER VALUE AND RESISTANCE MEASUREMENTS VERIFIED ACCEPTABLE. THE DIVISION 3 BATTERY WAS DECLARED OPERABLE AT 1143 HOURS ON OCTOBER 24, 1991. THE CAUSE OF THIS EVENT IS UNKNOWN. THE CONSEQUENCES OF THIS EVENT WERE MINIMAL, SINCE DIVISION 1 AND DIVISION 2 EMERGENCY CORE COOLING SYSTEMS WERE FULLY OPERABLE DURING THIS EVENT. THE UNIT 1 AND UNIT 2 DIVISION 3 BATTERIES WERE INSPECTED TO VERIFY THE BATTERY CONNECTIONS WERE TIGHT. TECHNICAL SPECIFICATIONS REQUIRE THE UNIT TO BE SHUTDOWN WITH BOTH HPCS AND RCIC SYSTEMS INOPERABLE. THIS EVENT IS REPORTABLE TO THE NUCLEAR REGULATORY COMMISSION AS A LICENSEE EVENT REPORT IN ACCORDANCE WITH 10CR50.73(A)(2)(V)(A) DUE TO A DECREASE IN SAFE SHUT DOWN CAPABILITY.

[56]   LA SALLE 2   DOCKET 50-374   LER 91-014
REACTOR SCRAM DUE TO A FALSE HIGH VIBRATION TRIP SIGNAL.
EVENT DATE: 102991   REPORT DATE: 112791   NSSS: GE   TYPE: BWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 223542) AT 0725 HOURS, ON OCTOBER 29, 1991, WITH UNIT 2 IN OPERATIONAL CONDITION ONE (RUN) AT 100% POWER (1122 MWE), THE REACTOR SCRAMMED ON TURBINE STOP VALVE CLOSURE DUE TO THE MAIN TURBINE TRIPPING ON HIGH #6 BEARING VIBRATION. AT THE TIME OF THIS EVENT, NO RELATED TESTING OR MAINTENANCE WAS BEING PERFORMED. ALL OTHER EQUIPMENT RESPONDED AS DESIGNED. ALL CONTROL RODS INSERTED, THE MAIN TURBINE BYPASS VALVES OPENED AS REQUIRED, THE MOTOR DRIVEN REACTOR FEEDWATER PUMP STARTED AND MAINTAINED REACTOR WATER LEVEL, AND MAIN STEAM SAFETY RELIEF VALVES K, E, D, C, S AND V CYCLED TO CONTROL PRESSURE AND THEN RESEATED. THE ROOT CAUSE OF THE SCRAM WAS A SPURIOUS TRIP FROM A TURBINE SUPERVISORY SYSTEM VIBRATION AMPLIFIER CIRCUIT CARD. THE CARD HAS BEEN REPLACED AND ITS TRIP TEMPORARILY DEFEATED. ADMINISTRATIVE CONTROLS ARE IN PLACE TO MANUALLY TRIP THE TURBINE SHOULD ACTUAL HIGH VIBRATION BE EXPERIENCED ON THIS BEARING. THE TURBINE SUPERVISORY SYSTEM WAS SUPPLIED BY GENERAL ELECTRIC. THIS EVENT IS REPORTABLE PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV) DUE TO AN AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM.

[57]   LIMERICK 1   DOCKET 50-352   LER 91-024
EQUIPMENT FAILURE AND PERSONNEL ERRORS CAUSE 'A' LOOP OF THE EMERGENCY SERVICE WATER SYSTEM TO BE INOPERABLE.
EVENT DATE: 102591   REPORT DATE: 112791   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 223529) ON OCTOBER 25, 1991, AN EMERGENCY SERVICE WATER (ESW) SYSTEM SURVEILLANCE TEST (ST) PROCEDURE WAS PERFORMED FOR ITS NORMALLY SCHEDULED QUARTERLY PERFORMANCE BY A NON-LICENSED OPERATOR. THE NON-LICENSED OPERATOR OBSERVED PRESSURIZED FLOW THROUGH A CHECK VALVE AND NOTED THIS IN THE ST

PROCEDURE. MAIN CONTROL ROOM OPERATIONS PERSONNEL REVIEWED THE TEST AND DETERMINED IT WAS UNSATISFACTORY DUE TO THE NOTED ST PROCEDURE STEP, BUT FAILED TO DECLARE EQUIPMENT INOPERABLE. ON OCTOBER 29, 1991, AN EVALUATION OF THE ST PROCEDURE BY THE PLANT STAFF DETERMINED THAT THE ESW SYSTEM CHECK VALVE FAILED TO PERFORM AS DESIGNED AND WAS THEREFORE INOPERABLE. THE PLANT STAFF ALSO DETERMINED THAT A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS EXISTED. CORRECTIVE ACTIONS WERE IMMEDIATELY TAKEN BY OPERATIONS PERSONNEL TO ADDRESS THIS CONDITION. THE ACTUAL CONSEQUENCES OF THIS EVENT WERE MINIMAL IN THAT NO ACCIDENT OR OPERATING TRANSIENT OCCURRED REQUIRING OPERATION OF THE AFFECTED SAFETY SYSTEMS. THE CAUSES OF THIS EVENT WERE AN EQUIPMENT FAILURE AND A FAILURE TO FOLLOW PROCEDURES.

[58] LIMERICK 2 DOCKET 50-353 LER 91-012 REV 02
UPDATE ON THE REMOVAL OF TWO FLOOR DRAIN PLUGS ASSOCIATED WITH REACTOR ENCLOSURE (RE) SECONDARY CONTAINMENT (SC), RENDERING RE SC INOPERABLE DUE TO PERSONNEL ERROR.
EVENT DATE: 062491 REPORT DATE: 112791 NSSS: GE TYPE: BWR

(NSIC 223530) ON JUNE 27, 1991, THE RADWASTE SYSTEM ENGINEER (SE) RECOGNIZED THAT TWO NORMALLY LOCKED IN PLACE FLOOR DRAIN PLUGS ASSOCIATED WITH MAINTAINING UNIT 2 REACTOR ENCLOSURE (RE) SECONDARY CONTAINMENT (SC) INTEGRITY HAD BEEN REMOVED BY MAINTENANCE PERSONNEL ON JUNE 24, 1991, AT APPROXIMATELY 1300 HOURS. THE SE IMMEDIATELY INSTRUCTED MAINTENANCE PERSONNEL TO RE-INSTALL AND LOCK IN PLACE THE TWO UNIT 2 FLOOR DRAIN PLUGS. THE TWO UNIT 2 FLOOR DRAIN PLUGS WERE REINSTALLED AND LOCKED IN PLACE AT 1000 HOURS, ON JUNE 27, 1991, THEREBY RESTORING UNIT 2 RE SC INTEGRITY. THIS EVENT RESULTED IN A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS. THE ACTUAL CONSEQUENCES OF THE EVENT WERE MINIMAL IN THAT NO RADIOACTIVE RELEASE OCCURRED DUE TO THE UNIT 2 RE BEING MAINTAINED AT THE REQUIRED NEGATIVE DIFFERENTIAL PRESSURE FOR THE DURATION OF THIS EVENT. THE CAUSE OF THIS EVENT IS PERSONNEL ERROR INVOLVING THE FOLLOWING CAUSAL FACTORS: LESS THAN ADEQUATE JOB PLANNING AS A RESULT OF INADEQUATE FLOOR DRAIN SYSTEM DESIGN DRAWINGS, INADEQUATE COMMUNICATION, A SENSE OF URGENCY FOR JOB COMPLETION, FAILURE TO COMPLY WITH ADMINISTRATIVE (A) PROCEDURE A-8, DUE TO A LACK OF TRAINING, LESS THAN ADEQUATE ATTENTION TO DETAIL, AND A LACK OF A QUESTIONING ATTITUDE BY MAINTENANCE PERSONNEL ON ADMINISTRATIVE PROCEDURE A-8.

[59] MAINE YANKEE DOCKET 50-309 LER 91-008
APPENDIX R DIESEL FUEL OIL TANK COATING FAILURE.
EVENT DATE: 080891 REPORT DATE: 111591 NSSS: CE TYPE: PWR

(NSIC 223516) THIS VOLUNTARY LER IS SUBMITTED TO ALERT LICENSEES TO POTENTIAL PROBLEMS WITH FUEL OIL TANKS. ON AUGUST 8, 1991, MAINE YANKEE DISCOVERED FOREIGN MATERIAL IN THE FUEL OIL TANK SUPPLYING DG-2, THE APPENDIX R ALTERNATE SHUTDOWN DIESEL. MAINE YANKEE PURSUED INSPECTION PLANS AND EVALUATED FOR POTENTIAL FAILURE MODES. SUBSEQUENTLY, DG-2 SUCCESSFULLY PERFORMED A FOUR HOUR OPERABILITY RUN. THE FUEL OIL TANK WAS THEN DRAINED AND INSPECTED. IT WAS DETERMINED THAT THE FOREIGN MATERIAL WAS A LOOSE SECTION OF INTERNAL TANK EPOXY COATING. THE TANK LINING WAS THEN STRIPPED TO BARE CARBON STEEL. A FUEL OIL SAMPLE WAS SENT OUT FOR ANALYSIS AND THE LINING WENT OUT FOR MATERIALS TESTING. THE TANK WAS REFILLED WITH NEW FUEL OIL AND PLACED BACK IN SERVICE. AFTER ANOTHER OPERABILITY RUN, FUEL OIL FILTERS WERE CHANGED AND THE DIESEL WAS DECLARED OPERABLE.

[60] MAINE YANKEE DOCKET 50-309 LER 91-011
EMERGENCY DIESEL GENERATOR BREAKER LOCKOUT.
EVENT DATE: 101191 REPORT DATE: 111591 NSSS: CE TYPE: PWR

(NSIC 223515) DURING A REVIEW OF THE EMERGENCY DIESEL GENERATOR (EDG) OUTPUT BREAKER CIRCUITRY, MAINE YANKEE POSTULATED THAT A LOCKOUT OF THE EDG BREAKER MAY OCCUR WHEN THE BREAKER IS REQUIRED TO BE CLOSED. IF THE EDG IS PHASED ONTO THE BUS AND A PLANT TRIP OCCURS CONCURRENT WITH A LOSS OF OFFSITE POWER, THE COINCIDENT OPEN AND CLOSE SIGNALS FOR THE BREAKER MAY LOCK OUT THE BREAKER DUE TO ITS ANTI-PUMPING FEATURE. AT THE TIME OF DISCOVERY, ONE EDG HAD BEEN RUNNING PHASED TO ITS EMERGENCY ELECTRICAL BUS FOR APPROXIMATELY 24 HOURS WHILE

MAINTENANCE WAS BEING PERFORMED ON THE REDUNDANT EDG. OPERABILITY OF BOTH EDGS HAD BEEN QUESTIONED DUE TO CONCERNS WITH THE AIR START SYSTEMS. SINCE THE POSTULATED LOCKOUT COULD OCCUR, THE RUNNING EDG WAS ALSO CONSERVATIVELY CONSIDERED INOPERABLE AND THEREFORE WAS IMMEDIATELY UNLOADED BY DISCONNECTING FROM ITS ELECTRICAL BUS. THE EDG WAS LEFT RUNNING AT HALF SYNCHRONOUS SPEED IN ACCORDANCE WITH MANUFACTURER'S RECOMMENDATIONS TO RESOLVE ANY QUESTION OF DIESEL OPERABILITY. ON OCTOBER 16, 1991, MAINE YANKEE VERIFIED THAT THE OPERATING TIMES OF THE RELAYS COULD PERMIT BREAKER LOCKOUT IF THE EDG IS PHASED TO ITS BUS AND A PLANT TRIP OCCURS WITH A LOSS OF OFFSITE POWER. THE PLANT THEREFORE HAD OPERATED IN A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS SINCE BOTH EDGS WERE INOPERABLE FOR GREATER THAN ONE HOUR.

[61] MCGUIRE 1 DOCKET 50-369 LER 91-015
AN INADVERTENT TRAIN A ENGINEERED SAFETY FEATURES ACTUATION OCCURRED BECAUSE OF AN INAPPROPRIATE ACTION.
EVENT DATE: 101391 REPORT DATE: 111291 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 223437) ON OCTOBER 13, 1991, AT 0411, OPERATIONS (OPS) PERSONNEL WERE DE-ENERGIZING UNIT 1 TRAIN A 4160 VOLT BUS 1ETA FOR MAINTENANCE ACTIVITY. UNIT 1 WAS IN NO MODE AND UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER. UPON OPENING BREAKER 1EPC BK 1ETA2, STANDBY FEEDER BREAKER FOR 1ETA SWITCHGEAR, A BLACKOUT SIGNAL WAS SENT TO DIESEL LOAD SEQUENCER 1A. CONSEQUENTLY, AN ENGINEERED SAFETY FEATURES ACTUATION (ESFA) OCCURRED ON TRAIN A CAUSING ASSOCIATED VALVES AND EQUIPMENT TO BE MOVED OR ACTUATED AS REQUIRED FOR A BLACKOUT CONFIGURATION. THIS EVENT IS ASSIGNED A CAUSE OF INAPPROPRIATE ACTION BECAUSE CONTROL POWER WAS NOT ISOLATED TO DIESEL GENERATOR LOAD SEQUENCER 1A PRIOR TO DE-ENERGIZING UNIT 1 TRAIN A 4160 BUS 1ETA. OPS MANAGEMENT WILL REVIEW THIS EVENT WITH APPROPRIATE OPS PERSONNEL.

[62] MCGUIRE 2 DOCKET 50-370 LER 91-010
MANUAL REACTOR TRIP RESULTING FROM POSSIBLE DESIGN, MANUFACTURING, CONSTRUCTION/INSTALLATION DEFICIENCY.
EVENT DATE: 092591 REPORT DATE: 102591 NSSS: WE TYPE: PWR

(NSIC 223311) A UNIT 2 REACTOR TRIP WAS MANUALLY INITIATED AT 1450:59 ON SEPTEMBER 25,1991. PRIOR TO THE EVENT, UNIT 2 WAS OPERATING IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER. THE TRIP WAS MANUALLY INITIATED BY OPERATIONS CONTROL ROOM PERSONNEL UPON RECEIPT OF ALARM PANEL 2AD-4, ANNUNCIATOR C-4 (STEAM GENERATOR D FLOW MISMATCH LOW CF FLOW). THE ANNUNCIATION RESULTED FROM MAIN FEEDWATER VALVE 2CF-26, STEAM GENERATOR D CONTAINMENT ISOLATION VALVE, MOVEMENT TO ITS FAIL-SAFE (CLOSED) POSITION UPON LOSS OF POWER TO THE CONTROLLING SOLENOID. THE LOSS OF POWER WAS CAUSED BY A VOLTAGE SURGE THAT BLEW FUSE BA 11-12 IN TERMINAL BOARD (TB) 1209. THE VOLTAGE SURGE WAS CAUSED BY GROUNDING BOTH LEGS OF VITAL BATTERY EVCD AS A RESULT OF THE REPLACEMENT OF THE CONTROL BOARD INDICATOR LAMP FOR VALVE 2VX-33B, CONTAINMENT SAMPLE SUPPLY INSIDE ISOLATION. THIS EVENT HAS BEEN ASSIGNED A CAUSE OF POSSIBLE DESIGN, MANUFACTURING, CONSTRUCTION/INSTALLATION DEFICIENCY BECAUSE OF DAMAGED INSULATION RESULTING IN A POSITIVE LEG GROUND ON THE SOLENOID ASSOCIATED WITH VALVE 2CF-26. CORRECTIVE ACTIONS INCLUDE INSPECTION OF SURPLUS SOLENOIDS TO ENSURE SERVICEABLE CONDITION OF THE INSULATION.

[63] MCGUIRE 2 DOCKET 50-370 LER 91-011
AN ESF ACTUATION WAS EXPERIENCED WHEN AN ENGINEERED SAFEGUARDS FEATURES ACTUATION WHEN OPERATIONS PERSONNEL MANUALLY STARTED THE AUXILIARY FEEDWATER PUMPS BECAUSE OF A DEFECTIVE PROCEDURE.
EVENT DATE: 100491 REPORT DATE: 110491 NSSS: WE TYPE: PWR

(NSIC 223374) ON OCTOBER 4, 1991, AT 1202, OPERATIONS (OPS) CONTROL ROOM (CR) PERSONNEL MANUALLY STARTED THE UNIT 2 MOTOR DRIVEN AUXILIARY FEEDWATER (CA) PUMPS. THIS WAS CONSIDERED AN ENGINEERED SAFEGUARDS FEATURES (ESF) ACTUATION. UNIT 2 WAS IN MODE 3 (HOT STANDBY) AT THE TIME OF THIS EVENT. CR INSTRUMENTATION INDICATED DECREASING AUXILIARY STEAM HEADER PRESSURE. SUBSEQUENTLY, A DECREASE IN

CONDENSER VACUUM, AND A DECREASE IN THE STEAM SEAL PRESSURE ON THE MAIN TURBINE AND MAIN FEEDWATER (CF) PUMPS OCCURRED. OPS CR PERSONNEL TRIPPED CF PUMP B AND GRADUALLY REMOVED CF PUMP A FROM THE FEEDWATER HEADER ONCE THE AUXILIARY STEAM HEADER PRESSURE RETURNED TO NORMAL, OPS CR PERSONNEL PLACED CF PUMP A BACK INTO THE FEEDWATER HEADER AND SECURED THE CA PUMPS THIRTY MINUTES LATER. THIS EVENT HAS BEEN ASSIGNED A CAUSE OF DEFECTIVE PROCEDURE BECAUSE NO PROCEDURAL GUIDANCE IS PROVIDED TO ALLOW OPS PERSONNEL TO USE THE CA SYSTEM AS AN OPTION DURING NORMAL UNIT SHUTDOWN WHEN THE CF SYSTEM IS UNAVAILABLE. AS A CORRECTIVE ACTION, OPS MANAGEMENT WILL REVISE THE UNIT 1 AND 2 UNIT SHUTDOWN PROCEDURE TO REFLECT THIS OPTION.

[64] MCGUIRE 2 DOCKET 50-370 LER 91-012
AN AUTOMATIC REACTOR TRIP WAS INITIATED DUE TO A CONTROL ROD FAILURE CAUSED BY AN EQUIPMENT FAILURE.
EVENT DATE: 110891 REPORT DATE: 120991 NSSS: WE TYPE: PWR

(NSIC 223538) ON 11/8/91, AT APPROX. 1142, OPERATIONS (OPS) CONTROL ROOM PERSONNEL WERE PERFORMING THE MONTHLY PROCEDURE PT/2/A/4600/01, ROD CLUSTER CONTROL ASSEMBLY MOVEMENT TEST. SHUTDOWN BANKS A AND B MOVED PROPERLY. HOWEVER, SHUTDOWN BANK C DROPPED INTO THE CORE AS THE REACTOR OPERATOR AT THE CONTROLS ATTEMPTED TO STEP THIS BANK IN. THE SOLID STATE PROTECTION SYSTEM SUBSEQUENTLY GENERATOED A NEGATIVE FLUX RATE TRIP SIGNAL WHICH AUTOMATICALLY TRIPPED THE REACTOR. OPS CONTROL ROOM PERSONNEL ALSO INITIATED A MANUAL REACTOR TRIP AS REQUIRED BY THE REACTOR TRIP PROCEDURE. DIESEL GENERATOR 2B WAS RUNNING AT THE TIME AND THE EMERGENCY BREAKER FOR DIESEL GENERATOR 2B OPENED DUE TO AN OVERLOAD WHEN A LOW-LOW LEVEL IN STEAM GENERATOR A CAUSED THE AUXILIARY FEEDWATER AND NUCLEAR SERVICE WATER PUMPS TO START. UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT 100% POWER PRIOR TO THE TRIP. THE UNIT PERFORMED AS EXPECTED DURING THE TRANSIENT AND THERE WERE NO NUCLEAR SAFETY SIGNIFICANT PROBLEMS. AN INVESTIGATION WAS PERFORMED AND IT WAS DETERMINED THAT AN ELECTRONIC EQUIPMENT FAILURE WITH THE SHUTDOWN BANK C INPUT LOGIC CIRCUIT HAD CAUSED THE STATIONARY ROD GRIPPER COILS TO STAY DE-ENERGIZED TOO LONG. AN INDEPENDENT SAFETY REVIEW WAS PERFORMED AND IT WAS DETERMINED IT WAS SAFE AND PRUDENT TO RESTART THE REACTOR. UNIT 2 WAS RETURNED TO MODE 1 ON 11/9/91, AT 0510.

[65] MILLSTONE 1 DOCKET 50-245 LER 91-006 REV 01
UPDATE ON UNJACKETED CABLE RESULTING IN LOSS OF ENVIRONMENTAL QUALIFICATION.
EVENT DATE: 030191 REPORT DATE: 111391 NSSS: GE TYPE: BWR

(NSIC 223465) ON MARCH 19, 1991, AT 1400 HOURS, WITH THE PLANT AT 89% POWER (530 DEGREES FAHRENHEIT AND 1030 PSIG), THE AS BUILT CONFIGURATION OF THE PIGTAIL TO FIELD TERMINATION FOR A CABLE IN THE DRYWELL WAS DETERMINED NOT TO BE FULLY ENVIRONMENTAL QUALIFIED. THIS CONDITION IS REPORTABLE UNDER 10CFR50.73. ON FEBRUARY 25, 1991, AN EVALUATION WAS INITIATED WHEN A QUESTION AROSE AS TO WHETHER A CABLE JACKET WAS RECONSTRUCTED ON KERITE CABLE AT THE PINTAIL TO FIELD TERMINATION FOLLOWING THE INSTALLATION OF RAYCHEM HEAT SHRINK TUBING SLEEVE. KERITE CABLE MUST BE FULLY JACKETED TO TAKE CREDIT FOR BETA RADIATION SHIELDING TO THE CABLE INSULATION TO ENSURE ENVIRONMENTAL QUALIFICATION. SPECIFICALLY THAT CABLE INSULATION EXPOSED TO AN ACCIDENT DRYWELL ENVIRONMENT. OPERABILITY OF THE CABLE END DEVICE COULD BE AFFECTED AND THUS THE ABILITY OF THE DEVICE TO IMITIATE THE CONSEQUENCES OF A LOSS OF COOLANT ACCIDENT. AN OPERABILITY EVALUATION WAS PERFORMED ON THE UNJACKETED KERITE CABLE FOUND ON 19 END DEVICES LOCATED IN THE DRYWELL. OF THE 19 END DEVICES EVALUATED ALL WERE FOUND TO REMAIN OPERABLE TO PERFORM THEIR DESIGN BASIS ACCIDENT FUNCTION WITH THE EXCEPTION OF THE FIELD CABLE TO THE AIR SOLENOID ON VALVE 1-CU-2A. THE SOLENOID SUPPLIES AIR TO CONTAINMENT ISOLATION VALVE 1-CU-2A. 1-CU-2A IS A 1/2" BYPASS VALVE AROUND 1-CU THE RWCU SYSTEM IN-BOARD CONTAINMENT ISOLATION VALVE.

[66] MILLSTONE 1 DOCKET 50-245 LER 91-027
345KV SYSTEM INSTABILITY.
EVENT DATE: 110191 REPORT DATE: 120291 NSSS: GE TYPE: BWR

(NSIC 223466) ON 11/1/91, AT 1340 HOURS, WITH THE PLANT SHUTDOWN (142 F AND 0

PSIG), AN ENGINEERING EVALUATION CONCLUDED A SINGLE ELECTRICAL FAULT COULD RESULT IN A LOSS OF ALL OFFSITE POWER. THIS CONDITION IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(II). DURING A RECENT DESIGN REVIEW IN AUGUST 1991 OF THE MILLSTONE UNIT ONE 345KV TRANSMISSION SYSTEM FOR STABILITY, IT WAS RECOGNIZED A TRANSFER TRIP TIME OF 8.5 CYCLES FOR THE SWITCHYARD BREAKERS TO OPEN UNDER A THREE PHASE BOLTED FAULT INSIDE THE MAIN STEP-UP TRANSFORMER COULD RESULT IN LOSS OF OFFSITE POWER SOURCES TO MILLSTONE ONE AND LOSS OF THE MILESTONE SWITCHYARD. THE LOSS OF THE MILLSTONE SWITCHYARD COULD THEN CAUSE INSTABILITY IN THE 315KV SYSTEM IN THE NORTHEAST. A PLANT DESIGN CHANGE HAS BEEN IMPLEMENTED TO CORRECT THE DEFICIENCY. THIS EVENT WAS THE SUBJECT OF A REPORTABILITY EVALUATION IN ACCORDANCE WITH NORTHEAST UTILITIES PROCEDURE (NEO 2.25). THE EVALUATION WAS INITIATED AUGUST 21, 1991, HOWEVER, THE ACTUAL REPORT WAS NOT MADE UNTIL NOVEMBER 1, 1991 DUE TO A FAILURE TO RECOGNIZE THE EXISTENCE OF SUFFICIENT TECHNICAL DATA TO MAKE A REPORTABILY DETERMINATION. IN ADDITION THE REPORTABILITY TIMELINESS LIMITS WERE NOT MET PER PROCEDURE NEO 2.25. THESE WERE AN OVERSIGHT. RECENT IMPROVEMENTS HAVE BEEN INITIATED TO CORRECT THE TIMELINESS ISSUE.

[67] MILLSTONE 3 DOCKET 50-423 LER 91-028
IMPROPER POWER RANGE TRIP RATE DECAY TIME TRIP SETPOINT DUE TO IMPROPER WORK PRACTICES.
EVENT DATE: 121590 REPORT DATE: 111891 NSSS: WE TYPE: PWR

(NSIC 223559) ON 10/18/91, AT 1100 HOURS WHILE IN COLD SHUTDOWN, AT 93 DEGREES AND ATMOSPHERIC PRESSURE, INSTRUMENT AND CONTROL TECHNICIANS DISCOVERED THAT THE POWER RANGE ANALOG CHANNEL OPERATION TEST WAS PERFORMED INCORRECTLY. THE TEST HAS BEEN PERFORMED INCORRECTLY SINCE 12/15/90. THE NEUTRON FLUX POWER RANGE HIGH RATE TRIP WAS NOT CALIBRATED WITHIN THE TECHNICAL SPECIFICATION TRIP SETPOINT. IT WAS CALIBRATED WITHIN THE TECHNICAL SPECIFICATION ALLOWABLE VALUE. THERE WERE NO IMMEDIATE OPERATOR ACTIONS REQUIRED. THE CALIBRATED VALUES WERE ALWAYS WITHIN THE VALUES CREDITED IN THE SAFETY ANALYSIS. THEREFORE, SAFETY FUNCTIONS WERE MAINTAINED AT ALL TIMES. THE ROOT CAUSE OF THIS EVENT IS IMPROPER WORK PRACTICES. PERSONNEL DID NOT CORRECTLY FOLLOW THE PROCEDURE. THE POWER RANGE CHANNELS WERE RECALIBRATED AND PERSONNEL WERE INFORMED OF THE PROBLEM AND INSTRUCTED ON THE PROPER CALIBRATION TECHNIQUE. THE PROCEDURES WILL BE MODIFIED TO INCLUDE A MORE DETAILED DESCRIPTION OF THE RATE TRIP DECAY TIME CALIBRATION. THE COMPANY'S POSITION ON PROCEDURAL COMPLIANCE HAS BEEN EMPHASIZED TO ALL PLANT PERSONNEL AND DISCOVERY OF THIS EVENT IS THE RESULT OF DILIGENCE TOWARDS THE COMPLIANCE POSITION. ADDITIONALLY, WHILE INVESTIGATING THIS EVENT, IT WAS DISCOVERED THAT THE POWER RANGE RESPONSE TIME TEST DID NOT ENCOMPASS THE ENTIRE POWER RANGE CIRCUIT.

[68] MILLSTONE 3 DOCKET 50-423 LER 91-027
SOURCE RANGE REACTOR TRIP DUE TO INTERFERENCE CAUSED BY MALFUNCTIONING WELDING MACHINE.
EVENT DATE: 101491 REPORT DATE: 111391 NSSS: WE TYPE: PWR

(NSIC 223558) ON 10/14/91, AT 1104 HOURS WHILE IN MODE 5 (COLD SHUTDOWN), AT 93F AND ATMOSPHERIC PRESSURE, A REACTOR TRIP SIGNAL WAS GENERATED DUE TO HIGH SOURCE RANGE CHANNEL 32 DETECTOR COUNTS. THE REACTOR TRIP BREAKERS WERE OPEN PREVIOUS TO THIS EVENT. THE PROBABLE ROOT CAUSE OF THIS EVENT IS ELECTRO-MAGNETIC INTERFERENCE ON THE SOURCE RANGE SIGNAL CAUSED BY A MALFUNCTIONING WELDING MACHINE. A WELDING MACHINE FAN CAUSED A SHORT ON ONE LEG OF THE THREE PHASE 480 VOLT NON-VITAL AC POWER SUPPLY. THE POWER CABLE AND SOURCE RANGE CABLE ROUTING HAS A MINIMUM 15 FEET SEPARATION. PROCEDURES HAVE BEEN MODIFIED TO BYPASS THE SOURCE RANGE TRIP WHEN THE REACTOR TRIP BREAKERS ARE OPEN AND THE SOURCE RANGE TRIP IS NOT REQUIRED, EXCEPT FOR PLANNED TESTING AS DICTATED BY PLANT PROCEDURES. EXISTING PROCEDURES PREVENT WELDING IN CONTAINMENT WITH THE REACTOR TRIP BREAKERS CLOSED DURING A REACTOR STARTUP.

[69] NINE MILE POINT 1 DOCKET 50-220 LER 91-006 REV 01
UPDATE ON CONTAINMENT HYDROGEN/OXYGEN MONITORING INOPERABLE DUE TO NON-SAFETY RELATED GAS.
EVENT DATE: 050891 REPORT DATE: 111491 NSSS: GE TYPE: BWR

(NSIC 223462) DURING THE PERFORMANCE OF N1-ISP-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.

[70]   NINE MILE POINT 2   DOCKET 50-410   LER 91-016 REV 02
UPDATE ON LOSS OF CONFIGURATION CONTROL CAUSED BY MISPOSITIONED VALVES DUE TO INADEQUATE WORK PRACTICES.
EVENT DATE: 070291   REPORT DATE: 112191   NSSS: GE   TYPE: BWR

(NSIC 223550) ON 7/2/91, AT 0545, DURING THE PERFORMANCE OF PROCEDURE N2-OSP-SWP-M001, "SERVICE WATER VALVE POSITION VERIFICATION," NMP2 OPERATIONS PERSONNEL DISCOVERED THAT THE COOLING WATER INLET AND OUTLET VALVES FOR REACTOR BUILDING UNIT COOLER 2HVR*UC404D WERE NOT IN THEIR NORMAL OPERATING POSITIONS. THESE CONDITIONS RESULTED IN THE UNIT COOLER BEING INOPERABLE. AT THE TIME THE CONDITIONS WERE DISCOVERED, THE REACTOR MODE SWITCH WAS IN THE "RUN" POSITION (MODE 1) WITH THE REACTOR OPERATING AT 100% RATED THERMAL POWER. THE ROOT CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE PERSONNEL ERROR DUE TO INADEQUATE WORK PRACTICES. A CONTRIBUTING CAUSE HAS BEEN DETERMINED TO BE PROCEDURAL INADEQUACY. THESE ERRORS RESULTED IN A LOSS OF UNIT COOLER 2HVR*UC404D CONFIGURATION CONTROL. THE IMMEDIATE CORRECTIVE ACTIONS WERE FOR PLANT OPERATORS TO CORRECTLY POSITION THE COOLING WATER VALVES FOR UNIT COOLER 2HVR*UC404D, AND TO DECLARE THE UNIT COOLER OPERABLE. ADDITIONAL CORRECTIVE ACTIONS INCLUDE: REVIEWING THE INCIDENT WITH NMP2 OPERATIONS PERSONNEL; ISSUING A LESSONS LEARNED TRANSMITTAL; COUNSELING AND DISCIPLINING PERSONNEL INVOLVED WITH THE MARKUP; HOLDING DEPARTMENTAL MEETINGS WITH NMP2 PERSONNEL TO EMPHASIZE EXPECTED WORK PRACTICES; ISSUING AN INTERIM INSTRUCTION FOR PLANT OPERATORS TO ALLEVIATE THE PROCEDURAL INADEQUACY; REVISING THE APPROPRIATE PROCEDURE.

[71]   NORTH ANNA 2   DOCKET 50-339   LER 91-011
PLANT SHUTDOWN REQUIRED BY TECHNICAL SPECIFICATIONS DUE TO RESIDUAL HEAT REMOVAL VALVE PACKING FAILURE.
EVENT DATE: 110391   REPORT DATE: 112791   NSSS: WE   TYPE: PWR
VENDOR: CHESTERTON PACKING & SEALS
        COPES-VULCAN, INC.

(NSIC 223523) ON NOVEMBER 3, 1991, AT 0803 HOURS, WITH UNIT 2 OPERATING AT 100 PERCENT POWER, OPERATIONS DEPARTMENT PERSONNEL NOTICED INCREASED MAKEUP FLOW TO THE REACTOR COOLANT SYSTEM (RCS) AND AN INCREASED PUMPING FREQUENCY OF THE PRIMARY DRAINS TRANSFER TANK (PDTT). AT 0841 HOURS IDENTIFIED LEAKAGE WAS DETERMINED TO BE GREATER THAN THE 10 GPM LIMIT OF TECHNICAL SPECIFICATION (TS) 3.4.6.2, AND THE ACTION STATEMENT WAS ENTERED. A CONTAINMENT ENTRY TEAM IDENTIFIED THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM INLET MOTOR OPERATED VALVE AS HAVING AN ELEVATED LEAK-OFF TEMPERATURE. AT 1456 HOURS THE UNIT WAS PLACED IN HOT STANDBY (MODE 3) IN ACCORDANCE WITH THE ACTION STATEMENT. THIS EVENT IS REPORTABLE PURSUANT TO 10 CFR 50.73 (A)(2)(I)(A). A ONE HOUR NOTIFICATION WAS MADE PURSUANT TO 10 CFR 50.72 (B)(1)(I)(A). THE CAUSE OF THE EVENT WAS FAILURE OF THE RHR SYSTEM INLET ISOLATION VALVE PACKING. AFTER REPAIRING THE VALVE PACKING, THE RCS LEAKAGE WAS DETERMINED TO BE WITH IN TECHNICAL SPECIFICATION LIMITS AND THE ACTION STATEMENT WAS CLEARED. NO SIGNIFICANT SAFETY CONSEQUENCES RESULTED FROM THIS EVENT BECAUSE THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL TO THE ENVIRONMENT. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED AT ANY TIME DURING THIS EVENT.

[72] OCONEE 1 DOCKET 50-269 LER 91-012
DESIGN DEFICIENCY IN ESTABLISHING RELIEF VALVE SETPOINT RESULTS IN TECHNICAL INOPERABILITY OF ALTERNATE REACTOR COOLANT MAKEUP SYSTEM.
EVENT DATE: 101491 REPORT DATE: 112691 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 223476) ON OCTOBER 14, 1991, DUKE POWER DESIGN ENGINEERING (DE) INITIATED AN INVESTIGATION INTO A POSSIBLE INADEQUATELY LOW ACTUATION SETPOINT ON A RELIEF VALVE WHICH WAS INSTALLED ON THE REACTOR COOLANT MAKEUP SYSTEM (RCMU). RCMU IS A PORTION OF THE STANDBY SHUTDOWN FACILITY (SSF) USED TO ESTABLISH HOT SHUTDOWN CONDITIONS FOLLOWING A FIRE, FLOOD, OR SABOTAGE EVENT. IT INJECTS BORATED WATER INTO THE SEALS OF THE REACTOR COOLANT PUMPS (RCPS). THE RCMU PUMP IS REQUIRED BY A PROPOSED TECHNICAL SPECIFICATION. ON OCTOBER 28, 1991, WITH ALL THREE UNITS AT 100 PERCENT FULL POWER, AN OPERABILITY EVALUATION PERFORMED BY DE DECLARED THE RCMU SYSTEM ON ALL THREE UNITS INOPERABLE BASED ON THE INABILITY TO PROVIDE ADEQUATE RCP SEAL FLOW TO PREVENT SEAL FAILURE AND A LOSS OF REACTOR COOLANT UNDER CERTAIN ACCIDENT CONDITIONS. THE RCMU SYSTEM WAS TECHNICALLY INOPERABLE SINCE ITS ORIGINAL DESIGN IN 1981. CONDITIONAL OPERABILITY FOR ALL UNITS WAS ATTAINED ON NOVEMBER 4, 1991, BY VERIFYING NEW TEST ACCEPTANCE CRITERIA FOR SSF EQUIPMENT AND BY IMPLEMENTING SSF EMERGENCY OPERATING PROCEDURE CHANGES. THE ROOT CAUSE OF THE EVENT WAS A FUNCTIONAL DESIGN DEFICIENCY. PROPERLY SET RELIEF VALVES WILL BE INSTALLED ON EACH UNIT.

[73] OCONEE 1 DOCKET 50-269 LER 91-013
POST LOCA DECAY HEAT REMOVAL SYSTEM DECLARED TECHNICALLY INOPERABLE DUE TO DESIGN DEFICIENCY.
EVENT DATE: 110491 REPORT DATE: 112691 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 223477) ON NOVEMBER 4, 1991, AT 1645 HOURS, BABCOCK AND WILCOX (B&W) NOTIFIED UTILITIES THAT BORON PRECIPITATION INSIDE THE CORE WOULD OCCUR SOONER THAN PREVIOUSLY ANALYZED AFTER CERTAIN LOCA SCENARIOS. AS A RESULT, THE POST LOCA BORON DILUTION SYSTEM (BDS) MUST BE IN SERVICE SIGNIFICANTLY EARLIER THAN PREVIOUSLY REQUIRED TO AVOID BORON CRYSTALLIZATION ON FUEL ASSEMBLIES AND THE RESULTING DEGRADED HEAT TRANSFER. OPERATORS WERE INSTRUCTED TO PLACE BDS IN SERVICE WITHIN 90 MINUTES OF THE LOCA IN ACCORDANCE WITH PRELIMINARY GUIDANCE FROM B&W. ON NOVEMBER 5, 1991, DUKE POWER CALCULATED THAT BDS MUST BE IN SERVICE WITHIN 9 (RATHER THAN 24) HOURS AFTER THE LOCA. CORRECTIVE ACTION WAS TO REVISE PROCEDURES TO INITIATE SYSTEM OPERATION WITHIN THE NEW TIME LIMIT. THE LOW PRESSURE INJECTION SYSTEM WAS DECLARED TECHNICALLY INOPERABLE FROM INSTALLATION OF THE BDS (IN 1976) UNTIL NOVEMBER 5, 1991, BECAUSE THE BDS MAY NOT HAVE BEEN PLACED IN SERVICE WHEN NEEDED. ALL THREE OCONEE UNITS WERE AT 100% FULL POWER WHEN NOTIFIED. THE ROOT CAUSE IS DESIGN DEFICIENCY.

[74] PALO VERDE 1 DOCKET 50-528 LER 91-010
REACTOR TRIPS CAUSED BY GRID PERTURBATION.
EVENT DATE: 102791 REPORT DATE: 112691 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 2 (PWR)
PALO VERDE 3 (PWR)

(NSIC 223572) ON 10/27/91, AT APPROX. 0722 MST, PALO VERDE UNITS 1 AND 3 WERE OPERATING AT APPROX. 100% POWER WHEN A GRID PERTURBATION CAUSED THE MAIN TURBINE CONTROL SYSTEM TO FAST CLOSE AND IMMEDIATELY REOPEN THE TURBINE CONTROL VALVES (TVCS). THE MOMENTARY REDUCTION IN STEAM FLOW CAUSED THE STEAM BYPASS CONTROL VALVES IN UNITS 1 AND 3 TO QUICK OPEN. A REACTOR POWER CUTBACK OCCURRED IN UNIT 3, BUT NOT IN UNIT 1. REACTOR TRIPS IN UNITS 1 AND 3 OCCURRED WHEN REACTOR POWER EXCEEDED THE CORE PROTECTION CALCULATOR VARIABLE OVERPOWER TRIP SETPOINTS. IMMEDIATELY FOLLOWING THE TRIPS, SAFETY INJECTION ACTUATION SYSTEM (SIAS) AND CONTAINMENT ISOLATION ACTUATION SYSTEM (CIAS) ENGINEERED SAFETY FEATURE ACTUATION SYSTEM ACTUATIONS OCCURRED ON LOW PRESSURIZER PRESSURE. ALL SAFETY SYSTEM COMPONENTS ACTUATED AS DESIGNED IN EACH UNIT. BY APPROX. 0805 MST ON 10/27/91, THE PLANTS WERE STABILIZED IN MODE 3 (HOT STANDBY). THE CAUSE OF THE EVENT WAS

DETERMINED TO BE THE EXPECTED PLANT RESPONSE TO A UNIQUE COMBINATION OF CIRCUMSTANCES. THE EVENT WAS PRECIPITATED BY A GRID FAULT RESULTING FROM A LIGHTNING STRIKE ON A SUBSTATION FEEDER LINE. THE FAULT WHICH OCCURRED WAS DIFFERENT FROM PREVIOUS GRID DISTURBANCE EVENTS (I.E., FAULT WITHOUT GROUND). THE GENERATOR OUTPUT CURRENT AT PALO VERDE DECREASED TRIGGERING A MOMENTARY POWER/LOAD UNBALANCE TURBINE PROTECTION ACTUATION.

[75] PALO VERDE 1 DOCKET 50-528 LER 91-011
LOSS OF ESSENTIAL AIR HANDLING UNIT DUE TO POSTULATED FIRE.
EVENT DATE: 102991 REPORT DATE: 112791 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 2 (PWR)
PALO VERDE 3 (PWR)

(NSIC 223573) ON OCTOBER 29, 1991, PALO VERDE UNITS 1 AND 3 WERE IN MODE 3 AND UNIT 2 WAS IN MODE 6 AT APPROXIMATELY 96 DEGREES FAHRENHEIT AND DEPRESSURIZED TO ATMOSPHERIC PRESSURE WHEN APS ENGINEERING PERSONNEL DETERMINED THAT A DESIGN BASIS APPENDIX R FIRE IN THE CONTROL ROOM COULD RESULT IN THE LOSS OF ONE TRAIN "B" ESSENTIAL AIR HANDLING UNIT (AHU). THE TRAIN "B" ESSENTIAL AHU PROVIDES COOLING TO TRAIN "B" ENGINEERED SAFETY FEATURES (ESF) EQUIPMENT, TRAIN "B" DC EQUIPMENT, AND TRAIN "B " DC BATTERY ROOMS. THE TRAIN "B" EQUIPMENT IS NECESSARY FOR THE SAFE SHUTDOWN OF THE PLANT IF THERE WAS A FIRE IN THE CONTROL ROOM. UPON DISCOVERY OF THIS POTENTIAL EVENT, APPROPRIATE COMPENSATORY MEASURES WERE ESTABLISHED IN ACCORDANCE WITH THE PVNGS FIRE PROTECTION PROGRAM. THE CAUSE OF THIS POSTULATED EVENT WAS A FAILURE OF THE ORIGINAL APPENDIX R EVALUATION TO RECOGNIZE THE CONTROL CIRCUIT FOR THE ESSENTIAL AHU BEING IN THE CONTROL ROOM. A PREVIOUS SIMILAR EVENT WAS REPORTED PURSUANT TO TECHNICAL SPECIFICATION 6.9.3 IN LER 528/91-008-01.

[76] PALO VERDE 2 DOCKET 50-529 LER 91-005 REV 01
UPDATE ON MSSV AND PSV SETPOINTS OUT OF TOLERANCE.
EVENT DATE: 100991 REPORT DATE: 120491 NSSS: CE TYPE: PWR
VENDOR: DRESSER INDUSTRIES, INC.

(NSIC 223560) ON 10/9/91, WHILE UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 97.5 PERCENT POWER, AN ENGINEERING EVALUATION OF ASME SURVEILLANCE TESTING RESULTS DETERMINED THAT ELEVEN (11) OF THE TWENTY (2) 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 OCTOBER 8 THROUGH OCTOBER 9, 1991, WHILE UNIT 2 WAS IN MODE 1, TO VERIFY THE RELIEF SETTINGS OF THE MSSVS. ON NOVEMBER 4 AND 5, 1991, WHILE UNIT 2 WAS IN MODE 6 WITH THE CORE OFFLOADED ASME SURVEILLANCE TESTING WAS PERFORMED ON THE PRESSURIZER SAFETY VALVES. THREE OF FOUR VALVES WERE OUT OF TOLERANCE ON THE INITIAL LIFT. THE CAUSE OF THE EVENT IS SETPOINT DRIFT. AS IMMEDIATE CORRECTIVE ACTION THE MSSVS AND PSVS HAVE BEEN ADJUSTED AND TESTED SATISFACTORILY. PREVIOUS SIMILAR EVENTS WERE REPORTED IN MSSV LERS 528/88-014-1, 528/89-010-00, 529/89-002-00, 529/89-007-00 AND 530/91-001-00.

[77] PALO VERDE 2 DOCKET 50-529 LER 91-006
TECHNICAL SPECIFICATION VIOLATIONS DURING CORE ALTERATIONS.
EVENT DATE: 102791 REPORT DATE: 112691 NSSS: CE TYPE: PWR

(NSIC 223574) ON 10/27/91, PALO VERDE UNIT 2 WAS IN MODE 6 (REFUELING) WITH THE REACTOR VESSEL HEAD REMOVED DURING A PLANNED REFUELING OUTAGE. AT APPROX. 1220 MST, THE CONTROL ROOM SHIFT SUPERVISOR WAS NOTIFIED THAT THE CONTROL ELEMENT ASSEMBLIES HAD BEEN WITHDRAWN APPROXIMATELY ONE (1) FOOT OUT OF THE CORE DURING DEFUELING OPERATIONS WITHOUT DIRECT COMMUNICATION ESTABLISHED WITH THE CONTROL ROOM DURING THIS CORE ALTERATION AND WITHOUT THE REFUELING SENIOR REACTOR OPERATOR PRESENT TO OBSERVE AND DIRECTLY SUPERVISE THE CORE ALTERATION. THIS IS CONTRARY TO THE REQUIREMENTS OF TECHNICAL SPECIFICATION (TS) LIMITING CONDITION FOR OPERATION (LCO) 3.9.5 AND TS ADMINISTRATIVE CONTROL 6.2.2B. CORE ALTERATIONS WERE IMMEDIATELY SUSPENDED PER THE ACTION REQUIREMENT OF LCO 3.9.5. FOLLOWING RESUMPTION OF CORE ALTERATIONS AFTER IMPLEMENTATION OF IMMEDIATE CORRECTIVE

ACTIONS, CORE ALTERATIONS WERE NOT SUSPENDED AT 1716 MST CONTRARY TO THE ACTION REQUIREMENTS OF TS 3.8.3.2 WHEN ONE (1) OF THE TWO (2) REQUIRED 120 VOLT VITAL AC BUSSES TRANSFERRED FROM ITS NORMAL POWER SOURCE TO ITS ALTERNATE POWER SOURCE. THE CAUSE OF THESE EVENTS WAS DUE TO A COMBINATION OF COGNITIVE PERSONNEL ERRORS. THE CONTROL ROOM SHIFT SUPERVISOR SUSPENDED CORE ALTERATIONS AND PLANT MANAGEMENT SUSPENDED ALL WORK AFFECTING OR INVOLVING CORE ALTERATIONS. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED PURSUANT TO 10CFR50.73.

[78] PALO VERDE 3 DOCKET 50-530 LER 91-011
AUXILIARY FEEDWATER FLOW TRANSMITTERS EQUALIZING VALVE IMPROPERLY ALIGNED.
EVENT DATE: 110491 REPORT DATE: 113091 NSSS: CE TYPE: PWR

(NSIC 223575) ON 11/4/91, AT APPROX. 2342 MST, PALO VERDE UNIT 3 WAS IN MODE 1 (POWER OPERATION) OPERATING AT APPROX. 100% POWER WHEN AN AUXILIARY OPERATOR (AO) ASSISTING IN THE PERFORMANCE OF A SURVEILLANCE TEST (ST) DISCOVERED THAT THE CHANNEL "B" AUXILIARY FEEDWATER FLOW TRANSMITTER AFBFT-41B EQUALIZING VALVE WAS IMPROPERLY ALIGNED. THE AO NOTIFIED THE CONTROL ROOM REACTOR OPERATOR DESIGNATED AS THE ST TEST LEADER OF THE IMPROPER VALVE ALIGNMENT. WHILE WAITING FOR CONTROL ROOM RESPONSE, THE AO DISCOVERED THAT THE OTHER CHANNEL "B" AUXILIARY FEEDWATER FLOW TRANSMITTER AFB-FT-41A EQUALIZING VALVE WAS ALSO IMPROPERLY ALIGNED. SINCE BOTH CHANNEL "B" FLOW TRANSMITTERS WERE FOUND TO BE IMPROPERLY ALIGNED, ONE LESS THAN THE REQUIRED NUMBER OF CHANNELS (I.E., TWO) SPECIFIED IN THE TECH SPEC (TS) LIMITING CONDITION FOR OPERATION (LCO) 3.3.3.6 WAS OPERABLE. SINCE IT HAS NOT BEEN DETERMINED WHEN THE FLOW TRANSMITTER EQUALIZING VALVES WERE MANIPULATED, IT IS ASSUMED THAT THE OPERABILITY REQUIREMENTS AND THE ASSOCIATED SEVEN (7) DAY ACTION WERE NOT MET FOR TS LCO 3.3.3.6 SINCE THE LAST TIME THAT THE FLOW TRANSMITTERS WERE DETERMINED TO BE OPERATING PROPERLY ON OCTOBER 27, 1991. BASED UPON THE RESULTS OF THE INVESTIGATION, THE CAUSE OF THE AUXILIARY FEEDWATER FLOW TRANSMITTER EQUALIZING VALVES BEING IMPROPERLY ALIGNED HAS NOT BEEN DETERMINED. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED PURSUANT TO 10CFR50.73.

[79] PEACH BOTTOM 2 DOCKET 50-277 LER 91-035
ENGINEERED SAFETY FEATURE ACTUATIONS WHEN THE INCORRECT TEST PLUG WAS INSTALLED DURING A SURVEILLANCE TEST.
EVENT DATE: 102591 REPORT DATE: 112191 NSSS: GE TYPE: BWR

(NSIC 223478) ON 10/25/91 AT 1230 HOURS, A TRIP OF AN EMERGENCY BUS BREAKER CAUSED AN AUTOMATIC 4KV FAST TRANSFER WHICH RESULTED IN A CONTROL ROOM EMERGENCY VENTILATION SYSTEM ACTUATION. THE CAUSE OF THE EVENT WAS THAT THE I&C TECHNICIAN INCORRECTLY INSTALLED TEST EQUIPMENT DUE TO INADEQUATE LABELING. THE TEST PLUGS AND CONNECTIONS HAVE BEEN RELABELED. THE EVENT HAS BEEN DISCUSSED WITH THE INVOLVED INDIVIDUALS AND THE PERTINENT INFORMATION HAS BEEN PROVIDED TO THE APPROPRIATE I&C PERSONNEL. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. THERE WERE NO PREVIOUS SIMILAR LERS IDENTIFIED.

[80] PEACH BOTTOM 2 DOCKET 50-277 LER 91-037
'2B' REACTOR PROTECTION SYSTEM MOTOR GENERATOR SET TRIP ON UNDERVOLTAGE DUE TO DEGRADED DIODES.
EVENT DATE: 110491 REPORT DATE: 112991 NSSS: GE TYPE: BWR

(NSIC 223480) ON 11/4/91 AT 1535 HOURS, A UNIT 2 "B" CHANNEL REACTOR PROTECTION SYSTEM (RPS) HALF SCRAM OCCURRED WHEN THE '2B' RPS MOTOR GENERATOR (M/G) SET OUTPUT BREAKER TRIPPED. THE TRIPPED BREAKER CAUSED A PRIMARY CONTAINMENT ISOLATION SYSTEM HALF GROUP III ISOLATION WHICH INCLUDED A STANDBY GAS TREATMENT SYSTEM INITIATION. THE CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE AN UNDERVOLTAGE CONDITION ON THE OUTPUT OF THE '2B' RPS M/G SET DUE TO DEGRADED DIODES. FOLLOWING THE EVENT, THE '2B' RPS WAS ALIGNED TO ITS ALTERNATE FEED WHICH ALLOWED THE HALF SCRAM AND ISOLATIONS TO BE RESET. SUBSEQUENTLY, THE AFFECTED SYSTEMS WERE RESTORED TO NORMAL. SEVERAL DIODES WERE REPLACED, A TEMPORARY PLANT ALTERATION WAS INSTALLED TO MONITOR THE 2B M/G SET. THE RPS WAS RESTORED TO ITS NORMAL POWER SUPPLY. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THESE EVENTS. NO PREVIOUS SIMILAR LERS HAVE BEEN IDENTIFIED.

[81] PERRY 1 DOCKET 50-440 LER 91-011 REV 01
UPDATE ON REACTOR WATER CLEANUP CONTAINMENT ISOLATION OCCURRED DURING PLANT SHUTDOWN DUE TO HIGH DIFFERENTIAL FLOW.
EVENT DATE: 041691 REPORT DATE: 120691 NSSS: GE TYPE: BWR

(NSIC 223556) ON APRIL 16, 1991 AT 0706 HOURS DURING A MANUAL PLANT SHUTDOWN, A REACTOR WATER CLEANUP (RWCU) SYSTEM CONTAINMENT ISOLATION OCCURRED DUE TO HIGH DIFFERENTIAL FLOW. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO VERIFY THAT NO ACTUAL SYSTEM LEAKAGE HAD OCCURRED. THE RWCU SYSTEM WAS SECURED AND SUBSEQUENTLY RETURNED TO SERVICE . THIS EVENT WAS DUE TO A DESIGN DEFICIENCY ASSOCIATED WITH THE REDUCED FEEDWATER TEMPERATURE MODE OF OPERATION. THE RWCU SYSTEM WAS NOT ORIGINALLY INTENDED TO BE OPERATED IN A REDUCED FEEDWATER TEMPERATURE MODE, WHICH WAS ADDED AFTER INITIAL CONSTRUCTION IN ORDER TO MINIMIZE THE POTENTIAL FOR FEEDWATER SYSTEM THERMAL STRATIFICATION AND STRESS. HOWEVER, OPERATION IN THE REDUCED FEEDWATER TEMPERATURE MODE OF OPERATION INTRODUCED UNFORESEEN SYSTEM VOIDING WHICH CAUSED RWCU ISOLATION ON HIGH DIFFERENTIAL FLOW. AFTER EVALUATING A NUMBER OF PROPOSED RECOMMENDATIONS FOR REDUCING SUCH EVENTS, THE OPTION OF PURSUING A TECHNICAL SPECIFICATION CHANGE TO INCREASE THE DIFFERENTIAL FLOW TIMER SETPOINT WAS CHOSEN. ADDITIONALLY, A VENDOR ANALYSIS DETERMINED THAT NO LONG-TERM DELETERIOUS EFFECTS HAD RESULTED OR WOULD RESULT FROM OCCASIONAL RWCU SYSTEM TRANSIENTS INVOLVING FLASHING OR VOIDING.

[82] PERRY 1 DOCKET 50-440 LER 91-023
MISADJUSTED LEAK DETECTION INSTRUMENTATION RESULTS IN TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 110591 REPORT DATE: 120591 NSSS: GE TYPE: BWR
VENDOR: TRANSMATION, INC.

(NSIC 223557) ON 11/5/91, AT 1911, THE MISADJUSTMENT OF TWO MAIN STEAM LINE ISOLATION ACTUATION INSTRUMENTATION TURBINE BUILDING MAIN STEAM LINE TEMPERATURE - HIGH CHANNELS RESULTED IN THE INOPERABILITY OF BOTH CHANNELS AND IN VIOLATION OF TECH SPECS. THESE CHANNELS HAD BEEN CONSIDERED OPERABLE FROM 11/5/91 AT 1811 UNTIL 1917 AT WHICH TIME THEY WERE DECLARED INOPERABLE AGAIN FOR TESTING. HOWEVER, DURING THIS PERIOD, WERE IN FACT INOPERABLE AND THE ACTION TO PLACE AT LEAST ONE TRIP SYSTEM IN THE TRIPPED CONDITION WITHIN ONE HOUR, AS REQUIRED BY TECH SPEC LCO 3.3.2 ACTION C, WAS NOT TAKEN. THE CAUSE OF THIS EVENT IS EQUIPMENT PROBLEM (OTHER). A TRANSLATION MODEL 1010 REFERENCE CELL, USED AS PART OF THE TEST EQUIPMENT FOR SURVEILLANCE TESTING, HAD INTERMITTENT MALFUNCTION WHICH CAUSED THE REFERENCE CELL TO PROVIDE AN INACCURATE OUTPUT DURING THE TEST. THIS CAUSED THE INSTRUMENTATION CHANNELS TO APPEAR TO REQUIRE ADJUSTMENT AND RESULTED IN THE SUBSEQUENT INOPERABILITY OF BOTH CHANNELS. A REVIEW OF THE PREVIOUS USES OF THE REFERENCE CELL IN QUESTION HAS BEEN CONDUCTED AND THE OPERABILITY OF OTHER PLANT EQUIPMENT HAS NOT BEEN AFFECTED. ADDITIONALLY, ALL APPLICABLE REFERENCE CELLS HAVE BEEN CHECKED TO ENSURE THAT BATTERY CONNECTIONS ARE CLEAN AND TIGHT. TO PREVENT RECURRENCE, QUALIFICATION TRAINING ON THE REFERENCE CELL IS BEING ENHANCED TO INCLUDE DIRECTION ON HOW TO CHECK THE REFERENCE CELL.

[83] PILGRIM 1 DOCKET 50-293 LER 91-024
LOSS OF PREFERRED AND SECONDARY OFFSITE POWER DUE TO SEVERE COASTAL STORM WHILE SHUTDOWN.
EVENT DATE: 103091 REPORT DATE: 112991 NSSS: GE TYPE: BWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 223488) ON OCTOBER 30, 1991 AT 1942 HOURS A LOSS OF PREFERRED OFFSITE 345 KV POWER OCCURRED WHILE SHUT DOWN DURING A SEVERE COASTAL STORM. THE LOSS OF PREFERRED OFFSITE POWER RESULTED IN DESIGNED RESPONSES INCLUDING AUTOMATIC ACTUATIONS OF THE REACTOR PROTECTION SYSTEM, PRIMARY AND SECONDARY CONTAINMENT ISOLATION CONTROL SYSTEMS, AND EMERGENCY DIESEL GENERATORS. THE CAUSE OF THE LOSS OF PREFERRED OFFSITE POWER WAS THE FLASHOVER OF A 345 KV SWITCHYARD [illegible]ULATOR, AND SEPARATE OPERATION OF A STUCK BREAKER CIRCUIT. THE FLASHOVER [illegible]ED THREE SWITCHYARD AIR CIRCUIT BREAKERS (ACBS) TO OPEN AS DESIGNED. A FOURTH [illegible] OPENED ABOUT 1.4 SECONDS LATER (STUCK BREAKER CIRCUIT) EVEN THOUGH THE [illegible]ED ACB OPENED AS DESIGNED. THE MOST PROBABLE CAUSE OF THE STUCK BREAKER

CIRCUIT OPERATION WAS 345 KV ELECTRICAL NOISE COUPLED INTO THE STUCK BREAKER CIRCUIT. CORRECTIVE ACTIONS PLANNED INCLUDE THE INSTALLATION OF A HIGH SPEED RECORDER TO MONITOR SWITCHYARD CIRCUITRY. A LOSS OF THE SECONDARY SOURCE OF OFFSITE POWER OCCURRED AT 1953 HOURS AND AN UNUSUAL EVENT WAS DECLARED AT 2003 HOURS. THE CAUSE OF THE LOSS OF SECONDARY OFFSITE POWER (23 KV) WAS ALSO STORM RELATED, WHEN A TREE FELL ONTO A 23 KV LINE. PREFERRED OFFSITE POWER WAS RESTORED AT 2142 HOURS AND THE UNUSUAL EVENT WAS TERMINATED AT 2230 HOURS. THE LOSS OF PREFERRED OFFSITE POWER OCCURRED ABOUT 2.5 HOURS AFTER THE REACTOR WAS SHUTDOWN.

[84] PILGRIM 1 DOCKET 50-293 LER 91-025
REACTOR CORE ISOLATION COOLING SYSTEM BECAME INOPERABLE DUE TO OVERSPEED TRIP AND INVERTER TRIP.
EVENT DATE: 103091 REPORT DATE: 112991 NSSS: GE TYPE: BWR
VENDOR: TOPAZ ELECTRONICS

(NSIC 223489) ON 10/30/91 AT 1942 HOURS, THE REACTOR CORE ISOLAT' N COOLING (RCIC) SYSTEM TURBINE TRIPPED AS OPERATORS MANUALLY STARTED THE SYSTEM FOR REACTOR VESSEL (RV) WATER LEVEL CONTROL. THE TRIP WAS RESET AND THE RCIC SYSTEM WAS MANUALLY STARTED. HOWEVER, THE RCIC INVERTER HAD TRIPPED WHEN THE "A" RESIDUAL HEAT REMOVAL (RHR) PUMP WAS STARTED AT 1946 HOURS RESULTING IN THE RCIC SYSTEM NOT REACHING RATED FLOW. THE OPERATORS MANUALLY SHUT DOWN THE RCIC SYSTEM, RESET THE INVERTER AND RESTARTED THE SYSTEM SUCCESSFULLY. THE RCIC TURBINE TRIPPED DUE TO MECHANICAL OVERSPEED. THE LICENSED OPERATOR DID NOT OPEN THE INJECTION VALVE WITHIN FOUR (4) SECONDS AFTER OPENING THE TURBINE STEAM INLET VALVE. DURING SYSTEM RESTORATION THE RCIC INVERTER (TOPAZ ELECTRONICS , MODEL 125-GW-125(60]) TRIPPED DUE TO A DC VOLTAGE TRANSIENT CAUSED WHEN THE "A" RHR PUMP WAS STARTED. THE BATTERY CHARGER WAS NOT ORIGINALLY DESIGNED TO MAINTAIN DC OUTPUT DURING AC VOLTAGE TRANSIENTS CAUSED BY STARTING LARGE AC MOTORS. CORRECTIVE ACTIONS FOR THE OVERSPEED TRIP INCLUDE CHANGING THE PROCEDURE TO REQUIRE VALVES TO BE OPENED SIMULTANEOUSLY. A MODIFICATION WAS COMPLETED THAT INSTALLED NEW RCIC AND HIGH PRESSURE COOLANT INJECTION INVERTERS HAVING A HIGHER TRIP SETPOINT AND AN AUTOMATIC RESET FUNCTION. EXTENSIVE TESTING WAS PERFORMED TO ENSURE THE INVERTERS WILL NOT TRIP DUE TO DC VOLTAGE TRANSIENTS.

[85] PILGRIM 1 DOCKET 50-293 LER 91-026
AUTOMATIC ACTUATIONS OF PRIMARY CONTAINMENT SYSTEM GROUP 6 ISOLATION VALVE.
EVENT DATE: 103091 REPORT DATE: 112791 NSSS: GE TYPE: BWR

(NSIC 223490) ON OCTOBER 30 AND 31, 1991 TWO AUTOMATIC ACTUATIONS OCCURRED IN THE REACTOR WATER CLEANUP (RWCU) SYSTEM PORTION OF THE PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM (PCIS). THE ACTUATIONS RESULTED IN THE AUTOMATIC CLOSURE OF ISOLATION VALVES IN THE RWCU PORTION OF THE PRIMARY CONTAINMENT SYSTEM (PCS). THE FIRST ACTUATION OCCURRED ON OCTOBER 30, 1991 AT 2335 WHEN THE RWCU SYSTEM WAS BEING PUT INTO SERVICE. THE SECOND ACTUATION OCCURRED ON OCTOBER 31, 1991 AT 0248 HOURS WHEN THE POSITION OF A RWCU VALVE WAS BEING ADJUSTED. THIS CAUSED INTERRUPTIONS IN THE OPERATION OF THE RWCU SYSTEM. THE CAUSE OF THE ACTUATIONS IS BELIEVED TO HAVE BEEN MOMENTARY FLUCTUATIONS IN RWCU SYSTEM FLOW. THE NUCLEAR ENGINEERING DEPARTMENT IS EVALUATING THE HIGH FLOW DESIGN SETPOINTS AND THE SYSTEM DESIGN TO ENHANCE DETECTION CAPABILITY WITHOUT UNNECESSARY ISOLATIONS. THE ACTUATIONS OCCURRED WHILE SHUTDOWN WITH THE REACTOR MODE SELECTOR SWITCH IN THE SHUTDOWN POSITION. DURING THE FIRST ISOLATION, REACTOR PRESSURE WAS ABOUT 250 PSIG AND REACTOR WATER TEMPERATURE WAS ABOUT 400 DEGREES FAHRENHEIT. DURING THE SECOND ISOLATION, REACTOR PRESSURE WAS ABOUT 165 PSIG AND REACTOR WATER TEMPERATURE WAS ABOUT 365 DEGREES FAHRENHEIT. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(IV) AND THE ACTUATIONS POSED NO THREAT TO THE HEALTH AND SAFETY OF THE PUBLIC.

[86] POINT BEACH 1 DOCKET 50-266 LER 91-012 REV 01
UPDATE ON NUCLEAR INSTRUMENTATION TURBINE RUNBACK.
EVENT DATE: 092491 REPORT DATE: 112791 NSSS: WE TYPE: PWR
VENDOR: ELGAR, CORP.

(NSIC 223474) AT 1027 ON 9/24/91 UNIT 1 EXPERIENCED A TURBINE RUNBACK FROM 100%

TO APPROXIMATELY 80% POWER DURING MAINTENANCE ON INVERTER DYOD. THE RUNBACK WAS CAUSED BY A VOLTAGE LOSS ON INSTRUMENT BUSES 1Y04 AND 1Y104 WHEN THE SUPPLY BREAKER FROM DC BUS D04 TO INVERTER DYOD WAS CLOSED. PRIOR TO THIS EVENT, SWING INVERTER DYOD (ELGAR MODEL 253-1-103) HAD BEEN TAKEN OUT OF SERVICE FOR MAINTENANCE. THE TURBINE RUNBACK OCCURRED WHEN THE D04 TO DYOD SUPPLY BREAKER WAS CLOSED WHILE BRINGING THE INVERTER BACK INTO SERVICE. THE INPUT CIRCUIT BREAKER FILTER CAPACITORS TO INVERTER DYOD HAD NOT BEEN CHARGED PRIOR TO ATTEMPTED CLOSURE. THE UNCHARGED CAPACITORS CAUSED A VOLTAGE SPIKE TO OCCUR ON THE DC SUPPLY BUS TO INVERTER 1DY04 (BUS D04). THIS VOLTAGE SPIKE SCRAMBLED THE INVERTER'S LOGIC, BLOWING A FUSE AND SHUTTING DOWN INVERTER 1DY04. ALSO, THE UNCHARGED CAPACITOR CREATED AN EXCESSIVE CURRENT DRAW ON INVERTER DYOD, CAUSING THE DC SUPPLY BREAKER TO TRIP ON OVERCURRENT. POWER TO YELLOW INSTRUMENT BUSES 1Y04 AND 1Y104, WHICH WAS BEING SUPPLIED BY INVERTER 1DY04, WAS SUBSEQUENTLY LOST. THIS LOSS OF POWER PRODUCED A NEGATIVE VOLTAGE SPIKE ON NUCLEAR INSTRUMENT CHANNEL N44, ACTIVATING THE DROPPED ROD TURBINE RUNBACK CIRCUITRY. ALL INSTRUMENTATION AND CONTROL SYSTEMS OPERATED AS DESIGNED. UNIT 1 WAS RETURNED TO FULL POWER AT 1257.

[87] POINT BEACH 1 DOCKET 50-266 LER 91-014
NUCLEAR INSTRUMENTATION TURBINE RUNBACK CAUSED BY A VOLTAGE SPIKE ON POWER RANGE CHANNEL N44.
EVENT DATE: 110391 REPORT DATE: 112791 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 223475) AT 1140 ON NOVEMBER 3, 1991, WHILE UNIT 1 WAS OPERATING AT 100% POWER, A 20% RUN BACK OF THE TURBINE GENERATOR OCCURRED. THIS RUNBACK WAS CAUSED BY AN OUTPUT TRANSIENT ON POWER RANGE NUCLEAR INSTRUMENTATION CHANNEL N44. THE REACTOR PROTECTION SYSTEM SENSED A DROPPED CONTROL ROD AND INITIATED THE TURBINE RUNBACK. FOLLOWING THE RUNBACK, THE LOAD INCREASE WAS COMMENCED AT 1145. FULL POWER WAS ATTAINED AT 1228.

[88] POINT BEACH 2 DOCKET 50-301 LER 91-004
CONTAINMENT ISOLATION VALVE LEAKAGE IN EXCESS OF TECHNICAL SPECIFICATION LIMITS.
EVENT DATE: 100891 REPORT DATE: 111591 NSSS: WE TYPE: PWR
VENDOR: GRINNELL INDUSTRIAL PIPING, INC.
MASSACHUSETTS ENGINEERING CO., INC.

(NSIC 223650) ON 10/1/91, DURING THE UNIT 2 REFUELING SHUTDOWN, OPERATIONS REFUELING TEST (ORT) 67, "COMPONENT COOLING WATER TO AND FROM THE EXCESS LETDOWN HEAT EXCHANGER, UNIT 2," WAS IN PROGRESS. THIS TEST, AS WELL AS THE OTHER TESTS DISCUSSED IN THIS REPORT, WAS BEING PERFORMED TO MEET THE REQUIREMENTS OF 10 CFR 50, APPENDIX J FOR TYPE C CONTAINMENT ISOLATION VALVES. DURING THE INITIAL PERFORMANCE OF THIS SEAT LEAKAGE TEST, CHECK VALVE CC-767 FAILED TO SHUT. BECAUSE THE VALVE FAILED TO SHUT, THE REQUIRED TEST PRESSURE COULD NOT BE ACHIEVED; THEREFORE, THE AS-FOUND LEAK RATE COULD NOT BE QUANTIFIED. A SECOND ATTEMPT WAS MADE TO PRESSURIZE THIS LINE. THIS TIME THE LINE PRESSURIZED TO THE REQUIRED PRESSURE. THE LEAK RATE MEASURED DURING THIS SECOND ATTEMPT WAS WITHIN THE LIMITS OF TECHNICAL SPECIFICATIONS SECTION 15.4.4.III.B.

[89] QUAD CITIES 1 DOCKET 50-254 LER 91-022
"B" TRAIN CR HVAC EMERGENCY FILTRATION UNIT UNABLE TO ATTAIN PROPER DIFFERENTIAL TEMPERATURE (DT) ACROSS HEATER DUE TO EXCESSIVELY CONSERVATIVE DT REQUIREMENTS.
EVENT DATE: 110491 REPORT DATE: 120491 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 223470) ON NOVEMBER 4, 1991 AT 1940 HOURS UNIT ONE WAS IN THE RUN MODE AT 100 PERCENT OF RATED CORE THERMAL POWER. UNIT TWO WAS ALSO IN THE RUN MODE AT 79 PERCENT OF RATED CORE THERMAL POWER. AT THIS TIME, THE CONTROL ROOM (VI) (CR) "B" TRAIN AIR FILTRATION UNIT (AFU) WAS DECLARED INOPERABLE. DURING THE SURVEILLANCE TESTING, THE HEATER (EHTR) FAILED TO ATTAIN A 15 DEGREE FAHRENHEIT DIFFERENTIAL TEMPERATURE (DT). AN EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE CALL WAS COMPLETED AT 2104 HOURS PER 10CFR50.72(B)(2)(III)(D). TECHNICAL SPECIFICATION AMENDMENTS 133 AND 128 WERE APPROVED AND IMPLEMENTED ON NOVEMBER 18, 1991 FOLLOWING A 30 DAY

COMMENT PERIOD. THE CAUSE OF THE FAILURE TO REACH A DIFFERENTIAL TEMPERATURE IS UNKNOWN. A SUPPLEMENTAL REPORT WILL BE SUBMITTED AT A LATER DATE. CORRECTIVE ACTIONS INCLUDED THE NEW TECHNICAL SPECIFICATION HEATER DT REQUIREMENTS, REVISING THE MONTHLY SURVEILLANCE TO INCORPORATE THE NEW REQUIREMENTS, AND PERFORMING THE MONTHLY OPERABILITY SURVEILLANCE. THIS REPORT IS BEING SUBMITTED TO COMPLY WITH 10CFR50.73(A)(2)(V).

[90] QUAD CITIES 2 DOCKET 50-265 LER 91-011
HIGH PRESSURE COOLANT INJECTION DECLARED INOPERABLE DUE TO FAILURE OF GLAND SEAL HOTWELL PUMP CAUSED BY HIGH SEAL SWITCH FAILURE.
EVENT DATE: 101591 REPORT DATE: 110891 NSSS: GE TYPE: BWR
VENDOR: MERCOID CORP.

(NSIC 223444) ON OCTOBER 15, 1991, AT 1304 HOURS, UNIT TWO WAS IN THE RUN MODE AT 88 PERCENT OF RATED CORE THERMAL POWER. AT THIS TIME, THE HIGH PRESSURE COOLANT INJECTION (HPCI) GLAND SEAL HOTWELL PUMP WAS OBSERVED TO BE CYCLING ON AND OFF. THE HPCI SYSTEM WAS DECLARED INOPERABLE AFTER FURTHER INVESTIGATION AND A HPCI SYSTEM OUTAGE REPORT WAS INITIATED. OCTOBER 15, 1991 AT 1442 HOURS, AN EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE NOTIFICATION WAS COMPLETED AS REQUIRED BY 10CFR50.72(B)(2)(III)(D). AT 1125 HOURS ON OCTOBER 16, 1991, ELECTRICAL MAINTENANCE (EM) HAD DISCOVERED THE HIGH LEVEL SWITCH FOR THE HPCI GLAND SEAL HOTWELL PUMP HAD SLIPPED OUT OF THE C-CLIP. THE SWITCH WAS REPAIRED AND THE SYSTEM WAS SUCCESSFULLY TESTED. CORRECTIVE ACTIONS INCLUDE THE CONTINUATION OF THE INSPECTION AND PREVENTIVE ACTIONS DOCUMENTED IN LER 2-90-009. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(V)(D).

[91] RIVERBEND 1 DOCKET 50-458 LER 91-020
HYDROGEN IGNITERS DECLARED INOPERABLE DUE TO DISCREPANCIES BETWEEN THE SURVEILLANCE PROCEDURE AND THE TECHNICAL SPECIFICATIONS.
EVENT DATE: 102491 REPORT DATE: 112491 NSSS: GE TYPE: BWR

(NSIC 223563) AT 0800 HOURS ON OCTOBER [illegible] 1991, WITH THE REACTOR IN OPERATIONAL CONDITION 1 (POWER OPERATION), WHILE PERFORMING A REVIEW OF TECHNICAL SPECIFICATION (TS) SECTION 3/4.6.6.3 "PRIMARY CONTAINMENT/DRYWELL HYDROGEN IGNITION SYSTEM", A DISCREPANCY WAS FOUND BETWEEN THE TS AND THE APPLICABLE SURVEILLANCE TEST PROCEDURE (STP). THE STP HAS BEEN NONCONSERVATIVE WITH RESPECT TO THE TS. SIXTY-TWO HYDROGEN IGNITERS WERE DECLARED INOPERABLE AND THE REACTOR WAS SHUTDOWN PURSUANT TO TS SECTION 3.0.3. THEREFORE, THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(I)(A) (PLANT SHUTDOWN REQUIRED BY THE TS) AND 10CFR50.73(A)(2)(I)(B) (OPERATION PROHIBITED BY THE TS). CORRECTIVE ACTIONS INCLUDE REVISION OF THE STP TO RESTORE CONSISTENCY WITH THE TS, ADDITIONAL TRAINING, AND A REVIEW OF A SAMPLE OF STP REVISIONS AND TEMPORARY CHANGE NOTICE FOR 10CFR50.59 APPLICABILITY, AND A VERIFICATION OF A SAMPLE OF STPS AGAINST THE TS. THE REACTOR WAS SHUTDOWN IN ACCORDANCE WITH TS 3.0.3. SUBSEQUENTLY, HYDROGEN IGNITER SYSTEM OPERABILITY WAS VERIFIED PURSUANT TO TS 4.6.6.3.

[92] RIVERBEND 1 DOCKET 50-458 LER 91-019
SURVEILLANCE ON THE RADWASTE BUILDING VENTILATION EXHAUST DUCT NOBLE GAS ACTIVITY MONITOR NOT MET WITHIN THE TECHNICAL SPECIFICATION REQUIRED TIME FRAME.
EVENT DATE: 102691 REPORT DATE: 112491 NSSS: GE TYPE: BWR

(NSIC 223561) AT 2050 ON 10/26/91, WITH THE REACTOR IN OPERATIONAL CONDITION 2 (STARTUP), OPERATIONS PERSONNEL DISCOVERED THAT SURVEILLANCE TEST PROCEDURE (STP)-511-4 516, A QUARTERLY STP WHICH CONCERNS THE RADWASTE BUILDING VENTILATION EXHAUST DUCT NOBLE GAS ACTIVITY MONITOR, HAD NOT BEEN COMPLETED WITHIN THE REQUIRED TECH SPEC TIME FRAME. OPERATIONS PERSONNEL REVIEWED THE TECH SPEC REQUIREMENTS UNDER SECTION 3.3.7.11.B-4A, DECLARED THE MONITOR INOPERABLE AND ENTERED THE ACTION STATEMENT REQUIRING CHEMISTRY SAMPLING UNDER A LIMITING CONDITION FOR OPERATION (LCO). THE STP WAS SUCCESSFULLY PERFORMED BY 2300 HOURS ON 10/28/91. TO DOCUMENT THE NON-COMPLIANCE WITH THE SURVEILLANCE REQUIREMENT, THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(I)(B) AS OPERATION PROHIBITED BY THE TECH SPECS. BASED ON A TASK ANALYSIS OF THIS EVENT AND PERSONNEL INTERVIEWS, THE ROOT CAUSE FOR MISSING THE SCHEDULED PERFORMANCE DUE

DATE WAS PERSONNEL ERROR WHICH INVOLVED TWO PRIMARY CAUSAL FACTORS, SELF-CHECKING AND VERBAL COMMUNICATIONS BETWEEN THE I&C FOREMAN AND CONTROL ROOM PERSONNEL. BETWEEN 10/12/91 AND 10/26/91, THE MONITOR WAS ADMINISTRATIVELY INOPERABLE; HOWEVER, SATISFACTORY PERFORMANCE OF THE STP ON 10/28/91 VERIFIED THAT THE MONITOR WAS FUNCTIONAL DURING THIS TIMEFRAME AND WAS CAPABLE OF PERFORMING ITS INTENDED SAFETY FUNCTION.

[93] ROBINSON 2 DOCKET 50-261 LER 91-009 REV 01
UPDATE ON OVERTEMPERATURE DELTA TEMPERATURE CHANNELS INOPERABLE DUE TO SUMMATOR MODULE LAG CONSTANTS.
EVENT DATE: 081691 REPORT DATE: 112791 NSSS: WE TYPE: PWR

(NSIC 223472) IN AUGUST OF 1991, AN ON-GOING ANALYSIS AND REVIEW WAS IN PROGRESS REGARDING DELAY TIMES AND LAG CONSTANTS ASSOCIATED WITH OVERTEMPERATURE DELTA TEMPERATURE (OTDT) INSTRUMENTATION AND CIRCUITRY. THE OVERALL RESPONSE TIME OF REACTOR COOLANT SYSTEM HOT AND COLD LEG TEMPERATURE INSTRUMENTATION AND CIRCUITRY WAS REVISED BY A 1988 MODIFICATION. HOWEVER, CERTAIN SUMMATOR MODULES WITHIN THIS CIRCUITRY CONTAINED CAPACITORS (LAG CIRCUITS) THAT WERE NOT CONSISTENT WITH THE VALUES ASSUMED WITHIN THE ACCIDENT ANALYSES. THE THREE OTDT CHANNELS WERE DECLARED INOPERABLE AT 1200 HOURS ON AUGUST 16, 1991 AND TECHNICAL SPECIFICATION 3.0 WAS ENTERED. SINCE REPAIRS COULD NOT BE COMPLETED WITHIN EIGHT HOURS AS REQUIRED BY THE TECHNICAL SPECIFICATIONS, A UNIT SHUTDOWN WAS INITIATED AT 1345 HOURS, WITH THE REACTOR BEING MADE SUBCRITICAL AT 1845 HOURS. THE CAPACITORS WERE REMOVED AND THE OTDT CHANNELS WERE DECLARED OPERABLE AT 0630 HOURS ON AUGUST 18, WITH THE UNIT BEING PLACED ON LINE AT 1432 HOURS. AN INDEPENDENT TEAM OF ONSITE AND OFFSITE PERSONNEL COMPLETED AN INVESTIGATION AND ROOT CAUSE ANALYSIS OF THIS OCCURRENCE. ONE OF THE ROOT CAUSES IDENTIFIED BY THE TEAM WAS THE FAILURE OF THE VENDOR WHO DEVELOPED AND DESIGNED THE 1988 MODIFICATION TO IDENTIFY THE REMOVAL OF THE LAG CIRCUITS AS A DESIGN BASIS REQUIREMENT.

[94] SALEM 1 DOCKET 50-272 LER 91-003 REV 01
UPDATE ON TWO STEAM FLOW CHANNELS FOR 1 STEAMLINE INOPERABLE DUE TO PERSONNEL ERROR.
EVENT DATE: 020991 REPORT DATE: 102491 NSSS: WE TYPE: PWR

(NSIC 223333) ON 2/9/91, DURING REACTOR SHUTDOWN (IN SUPPORT OF THE UPCOMING NINTH REFUELING OUTAGE), A NO. 14 STEAM GENERATOR (S/G) STEAMLINE FLOW CHANNEL I TRANSMITTER SENSING LINE WAS ISOLATED DURING INVESTIGATION OF A 14 S/G STEAMLINE FLOW CHANNEL II ERRONEOUS READING. SUBSEQUENTLY, ACTION STATEMENT 3.0.3 WAS ENTERED SINCE THE ACTION STATEMENTS FOR TECH SPECS 3.3.2.1 AND 3.3.3.1 DO NOT ADDRESS REQUIRED ACTIONS WITH MORE THAN ONE INOPERABLE STEAMLINE FLOW CHANNEL FOR ANY ONE S/G. THE ROOT CAUSE OF THIS EVENT IS PERSONNEL ERROR AS ATTRIBUTED TO INAPPROPRIATE SUPERVISORY DIRECTION. THE SUPERVISOR INVOLVED ACTED UPON AN INVALID ASSUMPTION. WHEN THE TRANSMITTER SENSING LINE WOULD NOT STOP VENTING (AFTER THE ROOT VALVE WAS CLOSED) THE SUPERVISOR INCORRECTLY ASSUMED EITHER THE SCHEMATIC WAS NOT READ CORRECTLY, THE VALVES WERE MISLABELED OR THE SCHEMATIC WAS INCORRECT. THE SUPERVISOR DID NOT CONSIDER THAT THE CORRECT TRANSMITTER SENSING LINE ROOT VALVE WAS CLOSED BUT WAS LEAKING BY (DUE TO IT NOT BEING CLOSED TIGHT ENOUGH). CONTRIBUTING TO THIS EVENT WAS THAT THE SCHEMATIC DRAWING, WHICH DETAILS COMPONENT ALIGNMENT, (FOR THE TRANSMITTER SENSING LINES) WAS NOT TAKE TO THE JOB SITE. UPON NOTIFICATION OF THE ISOLATION OF VALVE AND SUCCESSFULLY CLOSED THE CORRECT ROOT VALVE (I.E., THE FIRST ONE CLOSED) AND VENTED THE SUBJECT SENSING LINE.

[95] SALEM 2 DOCKET 50-311 LER 91-016
MISSED TECHNICAL SPECIFICATION, TWO FIRE HOSES NOT INSPECTED IN THE REQUIRED FREQUENCY.
EVENT DATE: 102591 REPORT DATE: 112291 NSSS: WE TYPE: PWR

(NSIC 223431) ON 10/25/91, A STAFF ENGINEER DISCOVERED THAT PROCEDURE M10-SST-023-2 "DETAILED INSPECTION OF TECHNICAL SPECIFICATION RELATED HOSE STATIONS", USED TO SATISFY SURVEILLANCE REQUIREMENTS FOR TECHNICAL SPECIFICATION 4.7.10.4.B EVERY 18 MONTHS, WAS INCORRECT. TWO (2) HOSE STATIONS, 2FP229 AND

2FP230, HAD BEEN INADVERTENTLY DELETED FROM ATTACHMENT 1 OF THE PROCEDURE WHEN IT WAS REVISED IN JUNE 1989. THE SURVEILLANCE WAS REVISED. THE LAST SUCCESSFULLY COMPLETED SURVEILLANCE, PRIOR TO THE INAPPROPRIATE PROCEDURE REVISION, WAS IN MAY 1989. THE OPERATING SHIFT WAS INFORMED AND ACTION STATEMENT 3.7.4.10 WAS ENTERED. THE ROOT CAUSE OF THE INAPPROPRIATE PROCEDURAL REVISION IS PERSONNEL ERROR (INATTENTION TO DETAIL). THE PROCEDURE WRITER DID NOT VERIFY THE ACCURACY OF THE REVISION WHICH DELETED FIRE HOSE STATIONS 2FP229 AND 2FP230 FROM ATTACHEMENT 1. THE PROCEDURE WRITER AND PROCEDURE REVIEWER, WHO PROCESSED THE PROCEDURE REVISION, HAVE BEEN COUNSELED. THE PROCEDURE ERROR WAS CORRECTED AND HOSE STATIONS 2FP229 AND 2FP230 WERE INSPECTED AS PER THE CORRECTED PROCEDURE. THE FIRE PROTECTION DEPARTMENT WILL REVIEW THEIR TECHNICAL SPECIFICATION SURVEILLANCE PROCEDURES REVISED AFTER JULY 1989. THE FIRE PROTECTION DEPARTMENT REVIEWED THIS EVENT WITH THEIR PROCEDURE WRITERS AND PROCEDURE REVIEWERS. PROCEDURE, NC.NA-AP.ZZ-0032(Q) WILL BE REVIEWED AND REVISED.

[96] SALEM 2 DOCKET 50-311 LER 91-017
REACTOR/TURBINE TRIP ON LOW AUTO STOP OIL PRESSURE FOLLOWED BY TURBINE/GENERAL FAILURE.
EVENT DATE: 110991 REPORT DATE: 120991 NSSS: WE TYPE: PWR

(NSIC 223514) ON 11/9/91, A REACTOR TRIP EVENT OCCURRED DURING PERFORMANCE OF THE MONTHLY AUTO STOP OIL SYSTEM PROTECTION DEVICE TESTING. AFTER THE GENERATOR BREAKER OPENED (PER DESIGN), A TURBINE OVERSPEED CONDITION DEVELOPED RESULTING IN FAILURE OF THE TURBINE/GENERATOR. THE REACTOR TRIP INITIATED WHILE THE AUTO STOP OIL (AST) SYSTEM TEST/NORMAL LEVER WAS STILL IN THE "TEST" POSITION. FOLLOWING THE INITIAL DROP IN AST SYSTEM HEADER PRESSURE, PRESSURE RETURNED. ONCE THE AST BISTABLE RESET SETPOINT WAS REACHED, THE TURBINE RELATCHED ALLOWING THE TURBINE VALVE TO REOPEN. WITHIN 15 SECONDS OF THE GENERATOR OUTPUT BREAKERS' OPENING, THE MAIN STOP VALVES FLUCTUATED AT THE 90% OPEN LIMIT. WITHIN THE FOLLOWING 6 SECONDS THE 4 MAIN STOP VALVES HAD GONE FULL OPEN. THE GOVERNOR VALVES THEN OPENED DUE TO FAILURE OF THE 63-3 AST TURBINE PROTECTION PRESSURE SWITCH TO LOCK IN THE TURBINE TRIP SIGNAL IN CONJUNCTION WITH THE FAILURE OF 3 OVERSPEED PROTECTION EHC SOLENOID VALVES (20/ET, 20-1/OPC AND 20-2/OPC) TO OPEN. THE OPERATORS AT THE FRONT STANDARD HEARD SEVERAL HIGHLY UNUSUAL SOUNDS FROM THE TURBINE AND MANUALLY TRIPPED THE TURBINE CAUSING CLOSURE OF THE TURBINE VALVES. THE TRIP CAUSE WAS BLOCKAGE OF THE AST PRESSURE REDUCTION ORIFICE. INVESTIGATION TO IDENTIFY THE SOURCE OF THE FOREIGN MATERIAL IS CONTINUING. SEVERAL CAUSAL FACTORS OF THE TURBINE FAILURE HAVE BEEN IDENTIFIED.

[97] SAN ONOFRE 1 DOCKET 50-206 LER 91-013 REV 01
UPDATE ON MIS-ASSEMBLY OF THE 4160 VOLT ROOM HALON SYSTEM ACTUATION LINES.
EVENT DATE: 070191 REPORT DATE: 120691 NSSS: WE TYPE: PWR

(NSIC 223458) AT 0524 ON 7/1/91, AN INADVERTENT ACTUATION OF THE HALON FIRE SUPPRESSION SYSTEM IN THE UNIT 1 4160 VOLT (4 KV) SWITCHGEAR ROOM OCCURRED. MOISTURE INTRUSION IN TO THE HALON CONTROL PANEL ACTUATION CIRCUITRY CAUSED THE INADVERTENT ACTUATION. CONTRARY TO DESIGN, THE DISCHARGE WAS LIMITED SOLELY TO THE MAIN BANK MASTER BOTTLE. THE MASTER BANK SLAVE BOTTLES FAILED TO PROPERLY ACTUATE AS DESIGNED DUE TO AN INCORRECTLY CONNECTED ACTUATION LINE BETWEEN THE MASTER BOTTLE AND THE SLAVE BOTTLES. THE RESERVE BANK (REDUNDANT) HALON SYSTEM WAS SIMILARLY INCORRECTLY CONFIGURED AND WAS THEREFORE ALSO INCAPABLE OF COMPLETE DISCHARGE. AS A RESULT, BOTH BANKS OF THE 4 KV ROOM HALON SYSTEM WERE INOPERABLE. OUR INVESTIGATION HAS CONCLUDED THAT THE ACTUATION LINES WERE INCORRECTLY CONNECTED DURING MAINTENANCE ACTIVITIES THAT OCCURRED EITHER IN JUNE 1988 OR IN APRIL/MAY 1989, WHEN THE MAIN AND RESERVE BANK MASTER BOTTLES WERE DISCONNECTED AND TRANSPORTED TO AN OFF-SITE VENDOR FOR SERVICING, AND SUBSEQUENTLY REINSTALLED BY SCE. THIS EVENT PRESENTS A CONDITION PROHIBITED BY TECHNICAL SPECIFICATION 3.14.4, "HALON SYSTEMS," PART B, SINCE THE OPERABILITY REQUIREMENTS OF THE HALON SYSTEM HAD NOT BEEN SATISFIED DURING THE PERIOD OF TIME THAT THE SYSTEM WAS NOT PROPERLY CONFIGURED. SCE'S INVESTIGATION HAS IDENTIFIED DEFICIENCIES IN MAINTENANCE PROCEDURES, POST-MAINTENANCE VERIFICATION AND OTHERS.

[98] SAN ONOFRE 1 DOCKET 50-206 LER 91-017
AUTOMATIC REACTOR TRIP UPON TRANSFER OF VITAL BUS #1 WITH PRE-EXISTING FAILURE OF HIGH STARTUP RATE BLOCK RELAY.
EVENT DATE: 101791 REPORT DATE: 111891 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 223459) AT 1314 ON 10/17/91, DURING THE PERFORMANCE OF CORRECTIVE MAINTENANCE ON AN ALARM MODULE POWERED FROM VITAL BUS NO. 1 (VB1), AND WITH A PRE-EXISTING FAILURE OF THE HIGH STARTUP RATE (SUR) BLOCK RELAY (AP4D), UNIT 1 AUTOMATICALLY TRIPPED FROM 91% POWER ON A SPURIOUS HIGH SUR SIGNAL. THE SPURIOUS SUR SIGNAL WAS GENERATED WHEN A MOMENTARY POWER INTERRUPTION TO VB1 OCCURRED DURING AN AUTOMATIC TRANSFER OF VB1 TO ITS ALTERNATE POWER SOURCE, WHICH WAS INITIATED DUE TO A MOMENTARY GROUND FAULT ON THE BUS. AS A TAPED ALARM MODULE POWER LEAD WAS BEING ROUTED THROUGH A GROOMING HOLE IN THE MODULE CHASSIS, THE LEAD ARCED APPARENTLY THROUGH THE TAPE TO THE MODULE CHASSIS, CAUSING THE GROUND. THE REACTOR PROTECTION SYSTEM (RPS) AND AUTOMATIC CONTROL SYSTEMS FUNCTIONED AS DESIGNED, AND THE PLANT WAS STABILIZED IN MODE 3 AT 1342. AN AUXILIARY FEEDWATER (AFW) ACTUATION, OCCURRED DUE TO THE EXPECTED DECREASE IN STEAM GENERATOR LEVELS FOLLOWING THE TRIP. ALL AFW COMPONENTS RESPONDED IN ACCORDANCE WITH DESIGN, AND THE AFW ACTUATION WAS RESET AT 1401. LABORATORY ANALYSIS OF THE TAPED LEAD REVEALED THAT THE INSULATING CAPABILITY OF THE TAPE HAD APPARENTLY BEEN DEGRADED DURING HANDLING. THIS IS POSTULATED TO HAVE OCCURRED WHEN THE TAPED LEAD CONTACTED A SHARP EDGE OF THE GROOMING HOLE ON THE ALARM MODULE CHASSIS WHEN THE LEAD WAS INSERTED THROUGH THE HOLE.

[99] SAN ONOFRE 1 DOCKET 50-206 LER 91-020
DELINQUENT REACTOR COOLANT PUMP FLYWHEEL INSPECTIONS.
EVENT DATE: 102491 REPORT DATE: 112591 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 223533) ON 10/24/91, WITH UNIT 3 IN MODE 1 AT 100% POWER, THE REACTOR COOLANT PUMP (RCP) MOTOR FLYWHEEL INSPECTION INTERVAL, AS SPECIFIED BY TECHNICAL SPECIFICATION (TS) SURVEILLANCE 4.4.9, "REACTOR COOLANT SYSTEM - STRUCTURAL INTEGRITY," AND REGULATORY GUIDE (RG) 1.14, WAS IDENTIFIED TO HAVE BEEN EXCEEDED. IT WAS SUBSEQUENTLY DISCOVERED ON 10/25/91, WITH UNIT 1 IN MODE 1 AT 90% POWER, THAT THE FLYWHEEL INSPECTION INTERVAL, AS SPECIFIED BY TS 4.7, "INSERVICE INSPECTION REQUIREMENTS," AND RG 1.14, FOR UNIT 1 RCPS "B" AND "C" HAD ALSO BEEN EXCEEDED. SINCE THE SURVEILLANCE INTERVALS HAD BEEN EXCEEDED, THE FOUR UNIT 3 RCPS AND UNIT 1 RCPS "B" AND "C" WERE CONSIDERED INOPERABLE. AS UNITS 1 AND 3 TSS DO NOT ADDRESS OPERABILITY REQUIREMENTS FOR RCP MOTOR FLYWHEELS, TS 3.0.3 WAS ALSO APPLICABLE. PLANT SHUTDOWN WAS DELAYED, AS ALLOWED BY TS 4.0.3, WHILE SCE SOUGHT A TEMPORARY WAIVER OF COMPLIANCE FOR UNITS 1 AND 3 TO DEFER SHUTDOWN OF THE UNITS TO INSPECT THE RCP MOTOR FLYWHEELS UNTIL 12/1/91. VERBAL APPROVAL WAS OBTAINED ON 10/25/91, AND ON 10/30/91, THE WAIVER REQUEST WAS FORMALLY GRANTED. THE CAUSE OF THE DELINQUENT UNIT 1 FLYWHEEL INSPECTION HAS BEEN ATTRIBUTED TO A WEAKNESS IN THE TRACKING OF THE FLYWHEEL SURVEILLANCE. THE CAUSE OF THE UNIT 3 INSPECTION INTERVAL BEING EXCEEDED WAS A SCHEDULING OVERSIGHT BY SCE PERSONNEL DUE TO WEAKNESSES IN ADMINISTRATIVE CONTROLS.

[100] SAN ONOFRE 1 DOCKET 50-206 LER 91-019
EMERGENCY LIGHT INOPERABLE CONTRARY TO TECHNICAL SPECIFICATIONS DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 103091 REPORT DATE: 120291 NSSS: WE TYPE: PWR

(NSIC 223460) ON NOVEMBER 1, WITH UNIT 1 IN MODE 1, DURING THE REVIEW OF ROUTINE PREVENTATIVE MAINTENANCE INSPECTION DATA FROM 8-HOUR EMERGENCY LIGHTS, IT WAS DETERMINED THAT EMERGENCY LIGHT L2-25-1-R HAD BEEN INOPERABLE SINCE OCTOBER 23, 1991, WHEN A BATTERY LEAD BECAME DISCONNECTED FROM A SEVERELY CORRODED BATTERY TERMINAL AS IT WAS BEING CLEANED. ON NOVEMBER 1, 1991, THE EMERGENCY LIGHT WAS REPAIRED. THE LIGHT WAS SATISFACTORILY TESTED AND DEMONSTRATED TO BE OPERABLE ON 11/3/91. CONTRARY TO TECHNICAL SPECIFICATION (TS) 3.14.9, "8-HOUR EMERGENCY LIGHTING UNITS," THE REPAIR OF THE LIGHT WAS NOT COMPLETED WITHIN THE 7 DAYS PERMITTED, WHICH ENDED ON OCTOBER 30, 1991. THE INCIDENT RESULTED FROM AN INADEQUATE INSPECTION PROCEDURE THAT FAILED TO PROVIDE APPROPRIATE GUIDANCE

RELATIVE TO OBSERVED 10 CFR 50, APPENDIX R, EMERGENCY LIGHT DEFICIENCIES REQUIRING PROMPT ACTION AND NOTIFICATION. AS A RESULT, INITIATION OF CORRECTIVE ACTION FOR THE IDENTIFIED DEFICIENCY WAS INCORRECTLY DEFERRED FROM THE TIME OF DISCOVERY TO OCTOBER 30, 1991, WHEN THE OTHER EMERGENCY LIGHT INSPECTIONS WERE COMPLETED. AS CORRECTIVE ACTION, A MAINTENANCE SURVEILLANCE TEST PROCEDURE WILL BE DEVELOPED EXCLUSIVELY FOR UNIT TS RELATED 10 CFR 50, APPENDIX R, LIGHTING UNITS. THIS EVENT IS OF NO SAFETY SIGNIFICANCE, SINCE EVEN WITHOUT THE USE OF THE INOPERABLE EMERGENCY LIGHT, ADEQUATE LIGHTING WAS AVAILABLE FOR OPERATORS.

[101] SAN ONOFRE 3 DOCKET 50-362 LER 91-005
DELINQUENT CORE PROTECTION CALCULATOR TECHNICAL SPECIFICATION SURVEILLANCE.
EVENT DATE: 110491 REPORT DATE: 120491 NSSS: CE TYPE: PWR

(NSIC 223531) AT APPROXIMATELY 1940 ON 11/4/91, WITH UNIT 3 IN MODE 1 AT 100% POWER, IT WAS DISCOVERED THAT THE ONCE PER 12-HOUR CORE PROTECTION CALCULATOR (CPC) AUTO-RESTART TECHNICAL SPECIFICATION (TS) SURVEILLANCE HAD EXCEEDED THE SURVEILLANCE INTERVAL BY 1 HOUR AND 5 MINUTES. UNITS 2/3 OPERATIONS DEPARTMENT RECENTLY CHANGED THEIR SHIFT SCHEDULE FROM 8-HOUR SHIFTS TO 12-HOUR SHIFTS. AS A CONSEQUENCE, THE TIMES AT WHICH VARIOUS SURVEILLANCES ARE PERFORMED WERE CHANGED TO ACCOMMODATE THE NEW SHIFT SCHEDULE. IN THIS CASE, THE TIME THAT THE ONCE PER 12-HOUR CPC AUTO-RESTART SURVEILLANCE WAS TO BE PERFORMED WAS CHANGED ON 10/31/91, FROM 0000 AND 1200 HOURS TO 0800 AND 2000 HOURS. WHEN THE SURVEILLANCE INSTRUCTION CHANGE DELINEATING THE REVISED SCHEDULE BECAME EFFECTIVE, THE COMPUTER GENERATED TS SURVEILLANCE SCHEDULE USED THE OPERATORS TO CONTROL TS SURVEILLANCES WAS NOT PROMPTLY UPDATED TO REFLECT THE SCHEDULE CHANGE. ADDITIONALLY, THE OPERATORS WERE NOT REQUIRED TO READ THE SURVEILLANCE OPERATING INSTRUCTION CHANGE NOTICE PRIOR TO ASSUMING SHIFT RESPONSIBILITIES. CONSEQUENTLY, ON 11/4/91, A NEW ONCOMING DAY-SHIFT CREW (WHO HAD BEEN OFF THE PREVIOUS 3 DAYS) PERFORMED THE SURVEILLANCE IN ACCORDANCE WITH THE OUT OF DATE COMPUTER GENERATED SCHEDULE, RESULTING IN THE SURVEILLANCE INTERVAL BEING EXCEEDED. CORRECTIVE ACTIONS INCLUDE ENHANCING ADMINISTRATIVE CONTROLS AND REVIEWING THIS EVENT.

[102] SAN ONOFRE 3 DOCKET 50-362 LER 91-007
FUEL HANDLING ISOLATION SYSTEM TRAIN "A" SPURIOUS ACTUATION DURING RADIATION MONITOR RETURN TO SERVICE.
EVENT DATE: 110791 REPORT DATE: 120691 NSSS: CE TYPE: PWR
VENDOR: MICRO SWITCH

(NSIC 223532) AT 2255 ON 11/7/91, WITH UNIT 3 AT 100% REACTOR POWER, FHIS TRAIN "A" SPURIOUSLY ACTUATED AS RADIATION MONITOR 3RI-7822 WAS BEING RETURNED TO SERVICE FOLLOWING A FILTER CHANGE-OUT FOR THE PARTICULATE/IODINE CHANNEL. THERE WAS NO INDICATION OF INCREASED RADIATION LEVELS IN THE FUEL HANDLING BUILDING (FHB). AT 2314, AFTER THE AIRBORNE ACTIVITY LEVELS IN THE FHB WERE CONFIRMED TO BE NORMAL, FHIS TRAIN "A" WAS RESET AND THE FHB VENTILATION SYSTEM WAS RETURNED TO NORMAL. THE CAUSE OF THE FHIS TRAIN "A" ACTUATION WAS AN INTERMITTENT FAILURE INTERNAL TO THE NORMAL/BYPASS SWITCH ASSOCIATED WITH THE FHIS CIRCUITRY. THE PLANNED CORRECTIVE ACTIONS INCLUDE: 1) REPLACEMENT OF THE FHIS NORMAL/BYPASS HANDSWITCH; AND 2) PERFORMANCE OF A FAILURE ANALYSIS ON THE REPLACED HANDSWITCH TO DETERMINE THE ROOT CAUSE OF THE FAILURE. A SUPPLEMENT TO THIS LER WILL BE SUBMITTED TO UPDATE THE CAUSE AND CORRECTIVE ACTION IF THE ROOT CAUSE IS DETERMINED TO BE OTHER THAN AN ISOLATED FAILURE. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE FHB AIRBORNE ACTIVITY LEVELS REMAINED NORMAL AND ALL FHIS TRAIN "A" COMPONENTS OPERATED IN ACCORDANCE WITH DESIGN.

[103] SEQUOYAH 1 DOCKET 50-327 LER 91-024
INOPERABLE FIRE DETECTOR CIRCUIT SUPERVISION DUE TO INADEQUATE UNDERSTANDING AND REVIEW.
EVENT DATE: 102291 REPORT DATE: 112191 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 223500) ON OCTOBER 2, 1991, WITH UNIT 1 IN MODE 5 AND UNIT 2 IN MODE 1, IT WAS DISCOVERED THAT AN ALARM CONDITION ON A FIRE DETECTION PANEL "MASKS" TROUBLE

ALARMS ON THE PANEL. BECAUSE SEVERAL ALARMS HAVE BEEN IN CONTINUOUSLY, THE DETECTOR SUPERVISORY CIRCUITS REQUIRED BY SURVEILLANCE REQUIREMENT 4.3.3.8.2 HAVE BEEN FUNCTIONALLY INOPERABLE, WITHOUT COMPLYING WITH THE CORRESPONDING ACTION PROVISIONS. THIS CONDITION RESULTED FROM A LACK OF UNDERSTANDING OF THE SPECIFIC FIRE DETECTION SYSTEM FEATURE AND A LACK OF SENSITIVITY TO FULLY EVALUATING CONTINUED OPERATION WITH ALARM CONDITIONS. THE CONTINUOUS ALARM CONDITIONS HAVE BEEN DISABLED, AND OPERATORS HAVE BEEN TRAINED ON THE MASKING FEATURE OF ALARM CONDITIONS. ADDITIONAL CORRECTIVE ACTIONS ARE BEING TAKEN TO INSTITUTIONALIZE THE AWARENESS OF THE MASKING FEATURE.

[104] SHEARON HARRIS 1 DOCKET 50-400 LER 91-019
UNPLANNED ESF ACTUATION DUE TO SPIKE ON RADIATION MONITOR.
EVENT DATE: 102291 REPORT DATE: 112191 NSSS: WE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 223549) ON 10/22/91, AT 1505 HOURS, A CONTAINMENT VENTILATION ISOLATION SIGNAL WAS GENERATED. THIS CONSTITUTED AN UNPLANNED ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF) COMPONENT. THE EVENT OCCURRED WHILE REMOVING ONE OF THE FOUR CONTAINMENT VENTILATION RADIATION MONITORS FROM SERVICE DURING SURVEILLANCE TESTING. THE CAUSE OF THE SIGNAL WAS AN INADVERTENT SPIKE ON ONE OF THE THREE RADIATION MONITORS THAT REMAINED OPERABLE (RM-3561A). THE CONTROL ROOM STAFF IMMEDIATELY VERIFIED THAT ALL ASSOCIATED CONTAINMENT VENTILATION COMPONENTS HAD FUNCTIONED AS REQUIRED AND THAT NO ABNORMAL RADIATION LEVELS ACTUALLY EXISTED. THE SYSTEM WAS THEN REALIGNED TO ITS NORMAL OPERATING LINEUP. CORRECTIVE ACTIONS TO PREVENT RECURRENCE WILL INCLUDE NEEDED REPAIRS AND/OR ADJUSTMENTS TO THE POWER SUPPLY CIRCUITRY FOR RADIATION MONITOR RM-3561A. THERE WERE NO SIGNIFICANT SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT AS THE CONTAINMENT VENTILATION SYSTEM WAS IN THE REQUIRED EMERGENCY MODE HAD AN ACTUAL EVENT OCCURRED. THIS IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73 (A)(2)(IV) AS AN UNPLANNED ACTUATION OF AN ESF COMPONENT.

[105] SOUTH TEXAS 1 DOCKET 50-498 LER 91-002 REV 01
UPDATE ON UNPLANNED SAFETY INJECTION ACTUATION DUE TO LESS THAN ADEQUATE WORK INSTRUCTIONS FOR A MAINTENANCE ACTIVITY.
EVENT DATE: 012691 REPORT DATE: 111891 NSSS: WE TYPE: PWR

(NSIC 223568) ON JANUARY 26, 1991, UNIT 1 WAS IN ITS THIRD REFUELING OUTAGE WITH NO FUEL IN THE REACTOR VESSEL AND THE REACTOR COOLANT SYSTEM VENTED TO ATMOSPHERE. AT 0850 HOURS, DURING THE FIRST PERFORMANCE OF A PREVENTIVE MAINTENANCE (PM) WORK ACTIVITY, AN AUTOMATIC ACTUATION OF THE SAFETY INJECTION (SI) SYSTEM OCCURRED IN ONE OF THREE TRAINS (TRAIN C) AS A RESULT OF LESS THAN ADEQUATE PM WORK INSTRUCTIONS. ALL ASSOCIATED ENGINEERED SAFETY FEATURES (ESF) EQUIPMENT OPERATED AS EXPECTED. THE CAUSE OF THE LESS THAN ADEQUATE WORK INSTRUCTIONS WAS PERSONNEL ERROR IN THAT TWO SUPERVISORS FAILED TO REQUIRE FURTHER REVIEW OF WORK INSTRUCTIONS WHICH THEY BELIEVED HAD POTENTIAL FOR CAUSING AN UNPLANNED ESF ACTUATION. CORRECTIVE ACTIONS INCLUDE INACTIVATING THE SUBJECT PM AND THE ASSOCIATED PMS FOR THE OTHER ACTUATION TRAINS IN BOTH UNITS. THESE PMS WILL BE CORRECTED PRIOR TO FUTURE USE. FURTHER CORRECTIVE ACTIONS WERE TAKEN TO ISSUE A TRAINING BULLETIN TO APPROPRIATE OPERATIONS AND MAINTENANCE SUPERVISORS DESCRIBING THE EVENT, AND TO COUNSEL THE TWO SUPERVISORS ON THE NECESSITY OF PERFORMING THOROUGH REVIEWS OF PROCEDURES AND WORK INSTRUCTIONS THAT HAVE THE POTENTIAL TO CAUSE UNPLANNED ESF ACTUATIONS.

[106] SOUTH TEXAS 1 DOCKET 50-498 LER 91-010 REV 01
UPDATE ON MANUAL ENGINEERED SAFETY FEATURES ACTUATION DUE TO A TOXIC GAS ALARM.
EVENT DATE: 040491 REPORT DATE: 111891 NSSS: WE TYPE: PWR

(NSIC 223569) ON APRIL 4, 1991, UNIT 1 WAS IN MODE 1 AT 13 PERCENT POWER. AT 0843 HOURS, THE MAIN CONTROL ROOM RECEIVED A TOXIC GAS HIGH CONCENTRATION ALARM. THE CONTROL ROOM VENTILATION SYSTEM WAS MANUALLY PLACED INTO THE RECIRCULATION MODE AS A CONSERVATIVE RESPONSE. NO TOXIC GAS WAS DETERMINED TO BE PRESENT AFTER AN IMMEDIATE INVESTIGATION. THE ALARM OCCURRED AS A RESULT OF A FAILURE IN THE EMERGENCY RESPONSE FACILITIES DATA ACQUISITION AND DISPLAY SYSTEM COMPUTER. THE

CAUSE OF THE ALARM WAS A FAILED FIBER OPTICS DATA ACQUISITION CONTROLLER SUBSYSTEM PRINTED CIRCUIT BOARD. THE FAILED PRINTED CIRCUIT BOARD HAS BEEN REPLACED AS A RESULT OF THE EVENT.

[107] SOUTH TEXAS 1 DOCKET 50-498 LER 91-021
REACTOR TRIP DUE TO PERSONNEL ERROR INADVERTENTLY TRIPPING THE FEEDER BREAKER TO AUXILIARY BUS "1J".
EVENT DATE: 101091 REPORT DATE: 110891 NSSS: WE TYPE: PWR

(NSIC 223570) ON OCTOBER 10, 1991, AT 2056, UNIT 1 WAS IN MODE 1 AT 100% POWER WHEN POWER FROM THE 1J BUS WAS LOST. DURING THE PERFORMANCE OF WORK ACTIVITIES, A MAINTENANCE ELECTRICIAN MISAPPLIED MULTIMETER TEST LEADS IN AN ENERGIZED CIRCUIT WITH THE MULTIMETER SET TO READ "RESISTANCE". THE MISAPPLIED TEST LEADS ENERGIZED RELAY 2A WHICH ACTUATED THE 86X LOCKOUT RELAY CAUSING BREAKER P150 TO TRIP AND DE-ENERGIZE THE 1J BUS. UPON LOSS OF POWER ON 1J BUS, REACTOR COOLANT PUMP (RCP) 1D TRIPPED AND CAUSED A REACTOR TRIP DUE TO LOW COOLANT FLOW. THE BUS WAS RE-ENERGIZED AT 2(59 FROM THE UNIT AUXILIARY TRANSFORMER WITH NO FURTHER INCIDENTS. THE PRIMARY CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THE MAINTENANCE ELECTRICIAN'S ATTENTION TO DETAIL DURING WORK PERFORMANCE, ELEMENTARY DRAWING READING AND TROUBLESHOOTING TECHNIQUES WERE LESS THAN ADEQUATE. A TRAINING SESSION IS BEING HELD FOR APPROPRIATE PLANT PERSONNEL STRESSING THE APPLICATION OF SELF VERIFICATION DURING WORK PERFORMANCE. A TESTING PROGRAM WILL BE IMPLEMENTED TO ENSURE THAT APPLICABLE PERSONNEL ARE QUALIFIED TO USE ELEMENTARY DRAWINGS TO AID IN PERFORMANCE OF MAINTENANCE ACTIVITIES.

[108] SOUTH TEXAS 1 DOCKET 50-498 LER 91-023
RESIDUAL HEAT REMOVAL MOTOR LEAD CRACKING ON EPOXY INTERFACE.
EVENT DATE: 102091 REPORT DATE: 111991 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SOUTH TEXAS 2 (PWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223571) ON OCTOBER 20, 1991 UNIT 1 WAS IN MODE 4 AND UNIT 2 WAS IN NO-MODE DURING A SCHEDULED REFUELING OUTAGE, WHEN THE DETERMINATION WAS MADE THAT CRACKS FOUND ON THE RESIDUAL HEAT REMOVAL (RHR) MOTOR "T" LEADS EPOXY INTERFACE WERE REPORTABLE. ON OCTOBER 11, 1991, WHILE PERFORMING WORK ON UNIT 2 RHR MOTOR "A", LEADS FOR THE MOTOR WERE DISCOVERED TO BE DAMAGED. EXAMINATION OF THE REMAINING UNIT 2 RHR PUMP MOTORS AND UNIT 1 TRAIN A AND TRAIN C, REVEALED SIMILAR MOTOR LEAD INSULATION CRACKING ON ALL OF THESE RHR PUMPS. THE APPARENT CAUSE OF THIS CRACKING IS THAT DIFFERENCES IN THE FLEXIBILITY OF THE MOTOR LEAD BETWEEN THE ORIGINAL INSULATION AND INSULATION USING RAYCHEM SLEEVES, CONCENTRATED THE BENDING STRESS IN THE CABLE IN THE AREA ADJACENT TO THE EPOXY CAUSING THE CRACKING. THE UNIT 1 RHR PUMP MOTOR "T" LEAD INSULATION CRACKS HAVE BEEN REPAIRED, AND UNIT 2 RHR MOTOR "T" LEADS WILL BE REPAIRED DURING THE CURRENT REFUELING OUTAGE. THE REPAIRS ARE DESIGNED TO PREVENT RECURRENCE OF THE CRACKING.

[109] ST. LUCIE 1 DOCKET 50-335 LER 91-007
DIESEL GENERATORS ADMINISTRATIVELY DECLARED OUT OF SERVICE BECAUSE OF PARTICULATE CONTAMINATION IN THE DIESEL FUEL OIL DUE TO PROCEDURE DEFICIENCIES.
EVENT DATE: 110491 REPORT DATE: 120291 NSSS: CE TYPE: PWR

(NSIC 223522) ON NOVEMBER 4, 1991 ST. LUCIE NUCLEAR UNIT 1 WAS IN MODE 6 (REFUELING) AND UNIT 2 WAS IN MODE 1 AT 100% POWER WHEN SAMPLE RESULTS TELEPHONED FROM AN OFF-SITE CONTRACT LABORATORY INDICATED OUT OF SPECIFICATION PARTICULATE CONTAMINATION IN THREE OF THE FOUR DIESEL FUEL OIL STORAGE TANKS (FOSTS) ON SITE. BOTH THE UNIT 1 FOSTS AND THE 2B FOST CONTAINED GREATER THAN 10.0 MILLIGRAMS PER LITER (MG/L) PARTICULATES. AS A RESULT OF THIS, THE 1A, 1B, AND 2B EMERGENCY DIESEL GENERATORS WERE ADMINISTRATIVELY DECLARED OUT OF SERVICE. A MOBILE FILTRATION SYSTEM WAS BROUGHT IN FROM A VENDOR TO RECIRCULATE THE FUEL OIL IN THE TANKS AND FILTER OUT THE PARTICULATE CONTAMINATION CONTAINED IN THE FUEL OIL. THE FOSTS WERE CLEANED AND RETURNED TO SERVICE WITHIN THE TIME LIMITS OF THE APPLICABLE TECHNICAL SPECIFICATION ACTION STATEMENT ON BOTH UNITS. THIS EVENT WAS CAUSED BY THE USE OF A CONTAMINATED FUEL OIL TANKER TRUCK TRANSFER PUMP AND HOSE

DURING THE RECEIPT OF DIESEL FUEL OIL ON OCTOBER 10, 1991. THE ROOT CAUSE OF THIS EVENT IS PROCEDURAL DEFICIENCIES IN PLANT CHEMISTRY PROCEDURES.

[110] SUMMER 1 DOCKET 50-395 LER 91-010
INADVERTENT START OF THE ENGINEERED SAFETY FEATURE.
EVENT DATE: 110691 REPORT DATE: 120591 NSSS: WE TYPE: PWR
VENDOR: AUTOMATION INDUSTRIES INC.
FAIRBANKS MORSE

(NSIC 223545) ON NOVEMBER 6, 1991, DURING MAINTENANCE OF THE "A" TRAIN ENGINEERED SAFETY FEATURES LOAD SEQUENCER (ESFLS) (XPN-6020), AN EXTENDER BOARD RESULTED IN THE LOSS OF POWER TO THE CIRCUIT BOARD BEING TESTED WHICH CAUSED THE SEQUENCER TO MALFUNCTION AND THE FOLLOWING TO OCCUR: THE NORMAL INCOMING 7.2 KV CIRCUIT BREAKER TRIPPED OPEN; THE EMERGENCY DIESEL GENERATOR (EDG) STARTED; THE EDG OUTPUT BREAKER CLOSED; AND ALL NORMALLY SEQUENCED LOADS (EXCEPT THOSE WITH BREAKERS RACKED OUT) LOADED ON THE DIESEL GENERATOR BUS IN ONE STEP (APPROXIMATELY 5000 KW IN-RUSH AND 2000 KW RUNNING). OPERATIONS PERSONNEL IMMEDIATELY TOOK ACTION TO RESTORE THE NORMAL OFFSITE POWER TO THE 7.2 KV BUS AND TO SECURE THE EDG. A NONCONFORMANCE NOTICE (NCN) WAS INITIATED TO EVALUATE THE EDG FOR ANY ADVERSE EFFECTS DUE TO THE ABNORMAL LOADING. THE NCN DISPOSITION IDENTIFIED SPECIFIC INSPECTIONS AND TESTS THAT NEEDED TO BE PERFORMED. THE DIESEL GENERATOR WAS INSPECTED, VARIOUS DATA AND MEASUREMENTS TAKEN, AND THE UNIT WAS RUN AT VARIOUS LOADS WITH ADDITIONAL DATA TAKEN. IT WAS DETERMINED THAT THE EDG WAS NOT ADVERSELY AFFECTED AS THE RESULT OF FAST, ABNORMAL LOADING. THIS OCCURRENCE WAS DUE TO TEST EQUIPMENT FAILURE.

[111] SURRY 2 DOCKET 50-281 LER 91-010
LOSS OF CONTAINMENT INTEGRITY CAUSED BY FAILURE OF MAIN STEAM TRIP VALVE BYPASS VALVE.
EVENT DATE: 103191 REPORT DATE: 120291 NSSS: WE TYPE: PWR
VENDOR: CRANE COMPANY

(NSIC 223481) ON OCTOBER 31, 1991, AT 0017 HOURS, WITH UNIT 2 CRITICAL AT 10E-8 AMPS INTERMEDIATE RANGE INDICATION, IT WAS DETERMINED THAT DAMAGE TO THE "C" MAIN STEAM TRIP VALVE (MSTV) BYPASS VALVE, 2-MS-155, RENDERED THE VALVE INCAPABLE OF IMMEDIATE CLOSURE. THE ON-SHIFT OPERATIONS STAFF DETERMINED THAT THE INABILITY TO CLOSE THIS MANUAL VALVE IMMEDIATELY WAS CONTRARY TO THE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.8.A.1 CONCERNING CONTAINMENT INTEGRITY. AN ACTION STATEMENT WAS ENTERED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.0.1. THE "C" MAIN STEAM NON-RETURN VALVE AND ASSOCIATED DRAIN VALVES WERE CLOSED AT 0605 HOURS TO REESTABLISH CONTAINMENT INTEGRITY AND TERMINATE THE ACTION STATEMENT. THE "C" MSTV BYPASS VALVE WAS CLOSED USING A HYDRAULIC JACKING DEVICE AND MECHANICALLY BLOCKED CLOSED BY 0900 HOURS. NO ADVERSE CONSEQUENCES TO PUBLIC HEALTH AND SAFETY WERE CREATED BY THE EVENT SINCE THE NON-RETURN VALVE WAS AVAILABLE TO ISOLATE THE MAIN STEAM LINE. THE EVENT WAS CAUSED BY FAILURE OF THE YOKE BUSHING THREADS IN 2-MS-155, WHICH IS BELIEVED TO BE THE RESULT OF GALLING ON THE VALVE STEM THREADS. A FAILURE EVALUATION WILL BE PERFORMED ON 2-MS-155 AND IT WILL BE REPAIRED OR REPLACED. THIS REPORT IS REQUIRED BY 10 CFR 50.73(A)(2)(I)(B).

[112] SUSQUEHANNA 1 DOCKET 50-387 LER 88-016 REV 01
UPDATE ON MAIN STEAM LINE LEAK DETECTION DIFFERENTIAL TEMPERATURE SYSTEM DESIGN DEFICIENCIES.
EVENT DATE: 072788 REPORT DATE: 120391 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)

(NSIC 223543) ON JULY 26, 1988 A DEFICIENCY WAS DETECTED ON UNIT 1 AND 2 OF THE SUSQUEHANNA STEAM ELECTRIC STATION WITH THE MAIN STEAM LINE LEAK DETECTION SYSTEM. THE DIFFERENTIAL TEMPERATURE (DT) SUBSYSTEM OF THE SUBJECT LEAK DETECTION SYSTEM WAS DETERMINED TO BE INOPERABLE ON BOTH UNIT 1 AND 2 BECAUSE INSTALLATION OF THE TRIP CHANNEL ELEMENTS WERE SUCH THAT AN ACTUAL DT CONDITION WOULD NOT HAVE BEEN DETECTED. THE APPROPRIATE CORRECTIONS WERE MADE BY REWIRING THE CIRCUITS ON JULY 27, 1988. FURTHER EVALUATION DETERMINED THAT THE LOCATION OF THE TEMPERATURE ELEMENTS WHICH INPUT TO THE MAIN STEAM LINE LEAK DETECTION SYSTEM DT SUBSYSTEM ON

UNIT 2 RENDERED THE ISOLATION SETPOINTS NONCONSERVATIVE. THE UNIT 2 TEMPERATURE ELEMENTS WERE RELOCATED TO THEIR PROPER POSITION. A STEAM LEAK DETECTION TASK TEAM WAS FORMED AND EVALUATED ALL STEAM LEAK DETECTION SYSTEMS AT THE STATION. ROOT CAUSES FOR MAIN STEAM LEAK DETECTION PROBLEMS WERE THE LACK OF A LEAK DETECTION SYSTEM FOCAL POINT, NO EXPECTED VALUES FOR LEAK DETECTION INDICATORS DURING NORMAL OPERATION WERE PROVIDED AND THEREFORE "ZERO" WAS ACCEPTED, AND STARTUP TESTING COULD ONLY SIMULATE PROCESS VARIABLES. ENGINEERING RESPONSIBILITIES WERE ASSIGNED TO LEAK DETECTION. NORMAL VALUES FOR LEAK DETECTION SYSTEMS WERE ESTABLISHED. MODIFICATIONS WERE MADE TO SEVERAL SYSTEMS.

[113] THREE MILE ISLAND 1 DOCKET 50-289 LER 91-005
INADVERTENT ESAS ACTUATION CHANNEL "A" DUE TO PERSONNEL ERROR.
EVENT DATE: 110191 REPORT DATE: 112791 NSSS: BW TYPE: PWR

(NSIC 223486) ON NOVEMBER 1, 1991, AT 11:56 P.M. TMI UNIT 1 WAS IN A REFUELING SHUTDOWN CONDITION. THE "C" INVERTER WAS BEING RETURNED TO SERVICE AFTER PREVENTIVE MAINTENANCE WAS COMPLETED. VITAL BUS "C" (VBC) WAS BEING POWERED FROM TRANSFORMER REGULATED BUS "B" (TRB) WHILE THE "C" INVERTER WAS OUT OF SERVICE. ESAS TRAINS "A" AND "B " WERE BOTH BYPASSED. UTILIZING ENCLOSURE V OF PROCEDURE 1107-2 REQUIRES THAT VITAL BUS "C" BE DE-ENERGIZED. THIS CAUSES CHANNEL 3 OF "A" & "B" ESAS TO DE-ENERGIZE, PUTTING EACH TRAIN IN A ONE OUT OF TWO ACTUATION LOGIC. UPON RE-ENERGIZING VBC FROM THE "C" INVERTER, ESAS CHANNEL 3 "A" & "B" MUST BE BYPASSED TO CLEAR ALL ESAS CHANNELS PUTTING THE SYSTEM BACK IN A FULLY BYPASSED CONDITION. THE SHIFT SUPERVISOR DIRECTING THE BYPASS OF CHANNEL 3 "A" AND "B" ESAS OBSERVED THE BYPASS OF TRAIN "B" AND THEN WITNESSED THE ENABLING OF CHANNEL 1 OF THE "A" ESAS TRAIN WHICH ACTUATED THE SYSTEM. CHANNEL 3 AND CHANNEL 1 WERE QUICKLY BYPASSED AND THE AFFECTED COMPONENTS WERE RETURNED TO THEIR DESIRED POSITIONS. THE ROOT CAUSE OF THE EVENT WAS PERSONNEL ERROR. THE CRO MISTAKENLY PUSHED THE WRONG BUTTON. THE EVENT WAS REVIEWED WITH THE INDIVIDUAL INVOLVED BY ADDRESSING THE NEED TO BE MORE AWARE OF HIS ACTIONS. THE EVENT WAS REPORTED PER 10 CFR 50.72 II(B)(2)(II).

[114] THREE MILE ISLAND 1 DOCKET 50-289 LER 91-006
DISCOVERY OF INCREASED POTENTIAL FOR POST LOCA BORON PRECIPITATION DUE TO AN ERROR IN THE NSSS VENDOR'S LICENSING BASIS ANALYSIS.
EVENT DATE: 110491 REPORT DATE: 120491 NSSS: BW TYPE: PWR

(NSIC 223487) TMI-1 WAS SHUTDOWN FOR REFUELING. AT APPROXIMATELY 5:45 PM ON NOVEMBER 4, 1991 GPU NUCLEAR WAS INFORMED BY THE NSSS VENDOR THAT AN UPDATED PRELIMINARY CALCULATION DOES NOT SUPPORT THE PREVIOUS CONCLUSIONS ATTRIBUTED TO BAW 10103A REV. 3 REGARDING BORON CONCENTRATION DURING THE LONG TERM COOLING PERIOD FOLLOWING A LARGE BREAK LOSS OF COOLANT ACCIDENT (LOCA). BASED ON THE REVISED ANALYSIS, THE VENDOR CONCLUDED THAT THE AUXILIARY PRESSURIZER SPRAY MODE OF RECIRCULATION FOR LONG TERM COOLING MAY NOT PROVIDE SUFFICIENT FLOW AND THAT THE TIME ALLOWED TO INITIATE A LONG TERM COOLING MODE WOULD BE SIGNIFICANTLY LESS THAN THAT PREVIOUSLY RELIED UPON FOR OPERATOR ACTION. THIS DEFICIENCY WAS REPORTED AT APPROXIMATELY 6:35 PM ON NOVEMBER 4, 1991 IN ACCORDANCE WITH 10 CFR 50.72(B) (2) (I). PRIOR TO RESTART OF THE UNIT, GPU NUCLEAR PERFORMED CALCULATIONS AND PREPARED A REPORT WHICH JUSTIFIES OPERATION AT FULL POWER. PROCEDURES WERE REVISED TO REFLECT THE CONSERVATIVE TIME PROPOSED BY BWNS AS THAT REQUIRED FOR ACTUATION OF BORON RECIRCULATION AND TO IDENTIFY THE DROP LINE METHOD AS THE PREFERRED MODE OF LONG TERM RECIRCULATION. GPU NUCLEAR'S REPORT CONCLUDES THAT THERE ARE TWO ACTIVE METHODS (DHR SYSTEM RECIRCULATION AND AUXILIARY PRESSURIZER SPRAY) AND A PASSIVE METHOD (CLEARANCE GAP FLOW) AVAILABLE TO MITIGATE BORON PRECIPITATION.

[115] TROJAN DOCKET 50-344 LER 91-018 REV 01
UPDATE ON TECHNICAL SPECIFICATION SURVEILLANCE MISSED DUE TO SENSORS NOT BEING INCLUDED IN THE CALIBRATION OF THE REACTOR VESSEL LEVEL INSTRUMENTATION SYSTEM.
EVENT DATE: 070291 REPORT DATE: 120391 NSSS: WE TYPE: PWR

(NSIC 223525) ON JULY 2, 1991 THE PLANT WAS IN MODE 5 (COLD SHUTDOWN), FOLLOWING CORE RELOAD DURING THE 1991 REFUELING AND MAINTENANCE OUTAGE. DURING AN

INVESTIGATION OF AIR INLEAKAGE INTO REACTOR VESSEL LEVEL INDICATION SYSTEM (RVLIS) SENSING LINES AS A POTENTIAL CAUSE FOR INABILITY TO CALIBRATE TRANSMITTERS, IT WAS DETERMINED THAT THE RVLIS CHANNEL CALIBRATION HAD NOT INCLUDED THE SENSOR BELLOWS LOCATED CLOSEST TO THE REACTOR VESSEL. THE 1991 RVLIS CHANNEL CALIBRATION WHICH INCLUDED THE SENSOR BELLOWS LED TO THE IDENTIFICATION OF AIR INLEAKAGE INTO THE RVLIS SENSING LINES. AIR INLEAKAGE IN THESE SENSING LINES WOULD CAUSE THE RVLIS WATER LEVEL INDICATIONS TO BE INACCURATE UNDER REDUCED-PRESSURE (ACCIDENT) CONDITIONS. CORRECTIVE ACTIONS INCLUDE: ELIMINATION OF POTENTIAL SOURCES OF AIR INLEAKAGE; PURGE, EVACUATION, FILL, AND COMPLETE CHANNEL CALIBRATION OF THE RVLIS; PROCEDURAL CLARIFICATION OF THE REQUIREMENTS FOR PERFORMING RVLIS CHANNEL CALIBRATIONS; A REVIEW FOR SIMILAR OCCURRENCES IN TECHNICAL SPECIFICATION CHANNEL CALIBRATIONS; AND IMPROVED ADMINISTRATIVE CONTROL OF OPERATIONAL ASSESSMENT REVIEWS FOR THOROUGH TECHNICAL EVALUATION AND TARGETED 90-DAY EVALUATION TIME.

[116] TROJAN DOCKET 50-344 LER 91-029 REV 01
UPDATE ON POTENTIAL FOR CONTROL ROOM DOSE TO EXCEED DESIGN LIMITS FOLLOWING A FUEL HANDLING ACCIDENT DUE TO INCORRECT ANALYSIS.
EVENT DATE: 081591 REPORT DATE: 112691 NSSS: WE TYPE: PWR

(NSIC 223526) ON AUGUST 14, 1991, THE TROJAN NUCLEAR PLANT WAS IN MODE 5 (COLD SHUTDOWN) DURING THE 1991 REFUELING OUTAGE. DURING A REVIEW OF A FINAL SAFETY ANALYSIS REPORT (FSAR) AMENDMENT REGARDING CONTROL ROOM HABITABILITY SYSTEMS, THE PLANT REVIEW BOARD RAISED A QUESTION REGARDING THE DESIGN OF THE CONTROL ROOM AREA RADIATION MONITOR (ARM-11). IT WAS QUESTIONED WHETHER ARM-11 MUST BE SAFETY RELATED IN ORDER TO FULFILL ITS DESIGN FUNCTION TO ISOLATE THE CONTROL ROOM UPON DETECTION OF RADIATION ASSOCIATED WITH A FUEL HANDLING ACCIDENT. ARM-11 IS NEITHER SAFETY RELATED, NOR REDUNDANT, AND IT IS NOT PROPERLY CONFIGURED TO PROVIDE AUTOMATIC CONTROL ROOM ISOLATION AS ASSUMED IN FSAR ANALYSES. THEREFORE, MAINTENANCE OF OPERATOR DOSES WITHIN THE LIMITS OF GENERAL DESIGN CRITERION 19 FOLLOWING A FUEL HANDLING ACCIDENT COULD NOT HAVE BEEN ASSURED. THIS CONDITION APPEARS TO BE THE RESULT OF AN ORIGINAL PLANT DESIGN ERROR. THE ORIGINAL FSAR EVALUATION OF A FUEL HANDLING ACCIDENT WAS PERFORMED UNDER THE ASSUMPTION THAT THE CONTROL ROOM WAS ISOLATED. HOWEVER, THERE WAS NO EXPLANATION OF HOW THE ISOLATION WAS TO BE INITIATED, NOR DOES THE PLACEMENT OF THE ARM-11 DETECTOR SUPPORT THIS ASSUMPTION. A SAFETY-RELATED RADIATION MONITORING SYSTEM WILL BE INSTALLED FOR THE CONTROL ROOM.

[117] TURKEY POINT 3 DOCKET 50-250 LER 91-009
AXIAL FLUX DIFFERENCE WAS OUTSIDE THE LIMITS OF TECH SPEC FOR MORE THAN 15 MINUTES.
EVENT DATE: 101891 REPORT DATE: 111591 NSSS: WE TYPE: PWR

(NSIC 223468) ON OCTOBER 18, 1991, AT 0010 EDT, WITH UNIT 3 AT 100% POWER AND UNIT 4 IN MODE 5 (COLD SHUTDOWN), THE 15 MINUTES ALLOWED BY TECH SPEC (TS) 3.2.1 (A) (AXIAL FLUX DIFFERENCE) FOR AXIAL FLUX DIFFERENCE OUTSIDE THE TARGET BAND WAS EXCEEDED. ON OCTOBER 18, 1991 AT 0135 EDT, THE CONDITION WAS DETERMINED TO BE INCONSISTENT WITH THE TS. REACTOR POWER WAS THEN REDUCED TO BELOW 90% AND THE AXIAL FLUX VALUE WAS RETURNED TO WITHIN THE TARGET BAND. AN EVALUATION OF THE EVENT SHOWED THAT FOR THE WORST CASE, INDICATED AXIAL FLUX WAS OUTSIDE THE TARGET BAND FOR ABOUT 2 HOURS. THE TURKEY POINT NSSS VENDOR PERFORMED A SAFETY EVALUATION ON THIS EVENT. THE SAFETY EVALUATION CONCLUDED THAT THE NUCLEAR SAFETY PARAMETERS FQ AND DNBR STAYED WITHIN THEIR LIMITS. THIS EVENT WAS CAUSED BY A PROCEDURE THAT DID NOT MEET THE TS REQUIREMENTS AND AN ERROR THAT WAS DISCOVERED IN THE PLANT TS. A TS AMENDMENT REQUEST HAS BEEN SUBMITTED TO CORRECT THE TS ERROR. A TEMPORARY WAIVER OF COMPLIANCE WAS ISSUED ON OCTOBER 30, 1991, ALLOWING TURKEY POINT TO PERFORM THE SURVEILLANCE. OPERATIONS PROCEDURE OP-12304.8, "INDUCING XENON OSCILLATIONS TO PRODUCE VARIOUS INCORE AXIAL OFFSETS," IS BEING REVISED TO REQUIRE PERFORMANCE OF THE PROCEDURE BELOW 90% POWER.

[118] TURKEY POINT 4 DOCKET 50-251 LER 91-006 REV 01
UPDATE ON AUTOSTART OF AUXILIARY FEEDWATER PUMPS FOLLOWING LOW SUCTION PRESSURE TRIP OF MAIN FEEDWATER PUMP DUE TO MECHANICAL FAILURE OF THE REGULATOR TO THE CONDENSATE POLISHING VESSEL INLET VALVE.
EVENT DATE: 102991 REPORT DATE: 112291 NSSS: WE TYPE: PWR
VENDOR: ECODYNE GRAVER(H2O) DIVISION

(NSIC 223469) ON 10/29/91, AT 2316 EST, WITH UNIT 4 AT APPROXIMATELY 22 PERCENT POWER, AND UNIT 3 AT 100 PERCENT POWER, ALL THREE AUXILIARY FEEDWATER (AFW) PUMPS AUTO STARTED FOLLOWING A TRIP OF THE ONLY OPERATING MAIN FEEDWATER (MFW) PUMP FOR UNIT 4. AUTOSTART OF THE AFW PUMPS IS CLASSIFIED AS AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION. A BACKWASH OF THE 4 "D" CONDENSATE POLISHER VESSEL HAD BEEN INITIATED JUST PRIOR TO THE AUTOSTART OF THE AFW PUMPS. A PLANT OPERATOR IMMEDIATELY STARTED THE 4B MFW PUMP. AT 2323, THE THREE AFW PUMPS WERE SECURED AND RETURNED TO STANDBY MODE. THE NRC WAS NOTIFIED OF THIS EVENT IN ACCORDANCE WITH 10 CFR 50.72 AT 0020 EST, OCTOBER 30, 1991. THE IMMEDIATE CAUSE OF THE AUTOSTART OF THE AFW PUMPS WAS THE TRIP OF THE 4A MFW PUMP UPON LOSS OF SUCTION PRESSURE. THE LOSS OF SUCTION PRESSURE TO THE MFW PUMP WAS CAUSED BY THE DIVERSION OF CONDENSATE FLOW TO THE "D" POLISHER VESSEL, THROUGH THE OPEN INLET VALVE, AND THROUGH THE BACKWASH RECEIVER. THE "D" POLISHER VESSEL INLET VALVE, CV-4-6351D, FAILED TO CLOSE BECAUSE OF THE FAILURE OF THE VALVE LIMIT SWITCH. THE LIMIT SWITCH WAS FOUND TO BE OUT OF ADJUSTMENT. THE LIMIT SWITCH WAS ADJUSTED, TESTED SATISFACTORILY AND RETURNED TO SERVICE.

[119] VOGTLE 1 DOCKET 50-424 LER 91-008
TRANSFORMER FAILURE LEADS TO ESF ACTUATION AND TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 102391 REPORT DATE: 112191 NSSS: WE TYPE: PWR
VENDOR: SOLA BASIC INDUSTRIES

(NSIC 223553) ON 10-23-91 AT 1110 EDT, A TRANSFORMER WAS FOUND TO BE SMOKING HEAVILY. THE INVOLVED TRANSFORMER POWERS RADIATION MONITOR 1RE-2565 AND OTHER RADIATION MONITORS. A CONTAINMENT VENTILATION ISOLATION (CVI) SIGNAL IS INITIATED FROM 1RE-2565. SUPERVISION IN THE CONTROL ROOM ELECTED TO DEENENERGIZE THE TRANSFORMER, IN ORDER TO STOP THE SMOKE PROPAGATION, WITH THE KNOWLEDGE THAT A CVI ACTUATION WOULD OCCUR. AT 1125 EDT, THE TRANSFORMER WAS DEENERGIZED AND A CVI ACTUATION OCCURRED AS EXPECTED. ON 10-24-91, PERSONNEL WERE PREPARING TO TEST ESF SYSTEMS. AS A PART OF THIS TESTING, THE ESF ACTUATION SYSTEM (ESFAS) SEQUENCER SHEDS THE LOADS FROM SOME BUSES. ONE OF THESE BUSES POWERS THE SAMPLE AIR PUMP FOR PLANT VENT RADIATION MONITOR 1RE-12444. THE TECHNICAL SPECIFICATIONS (TS) REQUIRE AT LEAST ONE PLANT VENT MONITOR TO BE IN SERVICE, BUT THE REDUNDANT MONITOR HAD BEEN REMOVED FROM SERVICE BY THE TRANSFORMER FAILURE. AS A RESULT, PLANT VENT MONITORING CEASED WHEN THE ESF TESTING WAS INITIATED. PRIMARY CAUSE OF THE CVI AND CONTRIBUTING CAUSE FOR THE TS VIOLATION WAS THE FAILURE OF THE TRANSFORMER. A TASK FORCE IS INVESTIGATING THE FAILURE.

[120] VOGTLE 1 DOCKET 50-424 LER 91-010
PERSONNEL ERROR LEADS TO INADEQUATE SURVEILLANCE.
EVENT DATE: 110691 REPORT DATE: 120591 NSSS: WE TYPE: PWR

(NSIC 223555) ON OCTOBER 15, 1991, TWO VALVES (11208U44482 AND 11208U4169) IN THE BORATION FLOW PATH FROM THE BORIC ACID STORAGE TANK (BAST) TO THE REACTOR COOLANT SYSTEM (RCS) WERE TAGGED SHUT TO ISOLATE ANOTHER VALVE SO THAT WORK COULD BE PERFORMED. ON NOVEMBER 5, 1991, PERSONNEL VERIFIED THAT TWO BORATION FLOWPATHS TO THE RCS EXISTED PER PROCEDURE. HOWEVER, VALVE POSITION FOR MANUAL VALVES SUCH AS 11208U44 82 AND 11208U4169, WHICH ARE NORMALLY LOCKED OPEN, IS NOT VERIFIED BY THIS PROCEDURE. TECHNICAL SPECIFICATION (TS) 3.1.2.1 REQUIRES ONE FLOWPATH TO BE AVAILABLE DURING SHUTDOWN. THE FLOWPATH CONTAINING THE AFOREMENTIONED MANUAL VALVES ORIGINATED FROM THE BAST AND CONTINUED THROUGH THE POSITIVE DISPLACEMENT PUMP (PDP) TO THE RCS. THEREFORE, THIS FLOWPATH WAS NOT AVAILABLE. ANOTHER FLOWPATH ORIGINATED FROM THE REFUELING WATER STORAGE TANK (RWST) AND CONTINUED THROUGH THE TRAIN B CENTRIFUGAL CHARGING PUMP (CCP) TO THE RCS. ON NOVEMBER 6, 1991, AT 0253 EST, AS PART OF PLANNED MAINTENANCE THE TRAIN B CCP WAS TAKEN OUT OF SERVICE, REMOVING THE ONLY REMAINING AND DEMONSTRATED OPERABLE BORON INJECTION FLOWPATH FROM THE RWST TO THE RCS. THE TRAIN A CCP HAD PREVIOUSLY BEEN TAKEN OUT

OF SERVICE FOR CORRECTIVE MAINTENANCE. THE CAUSE OF THIS EVENT WAS A COGNITIVE PERSONNEL ERROR ON THE PART OF THE UNIT SHIFT SUPERVISOR (USS) AND REACTOR OPERATOR (RO) ON DUTY WHEN THE ADEQUACY OF FLOWPATHS WAS VERIFIED. THE USS AND RO HAVE BEEN COUNSELED.

[121] WOLF CREEK 1 DOCKET 50-482 LER 91-021
EMERGENCY DIESEL GENERATOR "A" RESTORED WITH INCORRECT POST-WELDING HYDROSTATIC TEST PRESSURE ON JACKET WATER AND LUBE OIL HEAT EXCHANGERS.
EVENT DATE: 102191 REPORT DATE: 111891 NSSS: WE TYPE: PWR

(NSIC 223565) ON OCTOBER 29, 1991, WITH THE UNIT SHUTDOWN AND ALL FUEL REMOVED FROM THE CORE, IT WAS IDENTIFIED THAT THE POST-MAINTENANCE HYDROSTATIC TEST FOR THE EMERGENCY DIESEL GENERATOR (EDG) "A" JACKET WATER HEAT EXCHANGER AND LUBE OIL HEAT EXCHANGER WATER BOXES WAS PERFORMED AT AN INCORRECT TEST PRESSURE. EDG "A" HAD BEEN CONSIDERED OPERABLE SINCE COMPLETION OF POST-MAINTENANCE TESTING ON OCTOBER 21, 1991. SUBSEQUENT TO RETURNING EDG "A" TO SERVICE, EDG "B" WAS REMOVED FROM SERVICE ON OCTOBER 21, 1991; THEREFORE, AN OPERABLE EDG WAS NOT AVAILABLE UNTIL EDG "A" WAS AGAIN RETURNED TO SERVICE AN OCTOBER 30, 1991, FOLLOWING COMPLETION OF THE HYDROSTATIC TEST AT THE CORRECT PRESSURE. THIS CONDITION IS NOT IN COMPLIANCE WITH ADMINISTRATIVE CONTROLS AND IT WAS THEREFORE DECIDED TO VOLUNTARILY REPORT THIS CONDITION. USE OF THE INCORRECT TEST PRESSURE RESULTED FROM A COGNITIVE PERSONNEL ERROR BY NON-LICENSED PERSONNEL WHO MISINTERPRETED AN EQUIPMENT DRAWING. AN ENHANCEMENT WILL BE ADDED TO PROCEDURE ADM 08-217, "HYDROSTATIC AND PNEUMATIC TESTING," WHICH WILL INCLUDE GUIDANCE TO BE USED WHEN OBTAINING A SYSTEM'S DESIGN PRESSURE.

[122] WOLF CREEK 1 DOCKET 50-482 LER 91-020
CONTAINMENT ISOLATION VALVES FAILED LOCAL LEAK RATE TEST CAUSING TOTAL PATH LEAKAGE TO BE ABOVE 0.6 LA.
EVENT DATE: 102291 REPORT DATE: 111991 NSSS: WE TYPE: PWR
VENDOR: FISHER CONTROLS CO.

(NSIC 223564) ON OCTOBER 22, 1991, DURING REFUELING OUTAGE V, WITH ALL FUEL REMOVED FROM THE CORE, THE CONTROL ROOM WAS INFORMED THAT THE TOTAL PATH CONTAINMENT LOCAL LEAKAGE RATES FOR TYPE B AND C TESTS WAS ABOVE THE TECHNICAL SPECIFICATION LIMIT OF 0. 6 LA. THIS WAS DETERMINED FOLLOWING THE PERFORMANCE OF A LOCAL LEAK RATE TEST ON CONTAINMENT ISOLATION VALVES EF HV032 AND EF HV034. THESE VALVES, ASSOCIATED WITH PENETRATION 28, ISOLATE CONTAINMENT AIR COOLERS "B" AND "D" FROM ESSENTIAL SERVICE WATER TRAIN "B". ALSO, THESE VALVES ARE NORMALLY OPEN VALVES AND RECEIVE AN OPEN SIGNAL ON A SAFETY INJECTION SIGNAL. THE EXCESSIVE LEAKAGE THROUGH VALVES EF HV032 AND EF HV034 RESULTED FROM EROSION/CORROSION (E/C) OF THE VALVE DISCS. THE VALVE DISCS WERE REPLACED. AN EVALUATION OF E/C DAMAGE TO THE NEWLY INSTALLED DISCS WILL BE CONDUCTED DURING THE NEXT REFUELING OUTAGE.

[123] WOLF CREEK 1 DOCKET 50-482 LER 91-024
DEFICIENCIES DISCOVERED IN MOTOR-OPERATED VALVE TESTING PROGRAM WHICH CAUSED POTENTIAL INOPERABILITY OF SAFETY RELATED VALVES.
EVENT DATE: 110291 REPORT DATE: 120291 NSSS: WE TYPE: PWR

(NSIC 223566) IN OCTOBER AND NOVEMBER OF 1991, WITH THE UNIT IN MODE 6, REFUELING, DEFICIENCIES WERE DISCOVERED IN WOLF CREEK NUCLEAR OPERATING CORPORATION'S PROGRAM FOR IMPLEMENTING COMMITMENTS TO THE PROVISIONS OF GENERIC LETTER 89-10, "SAFETY RELATED MOTOR-OPERATED VALVE TESTING AND SURVEILLANCE". IN RESPONSE TO THESE DEFICIENCIES A TASK FORCE EFFORT WAS INITIATED TO PROVIDE A DETAILED ENGINEERING EVALUATION AND DETERMINE THE OPERABILITY OF VALVES EN HV15, EN HV16, EM HV8807A, EM HV8807 B, BB HV8351A, B, C, D, BB HV8000B, BG HV8111, EJ HV8716A, EJ HV8716B AND OTHER MOTOR-OPERATED VALVES IDENTIFIED BY THE TASK FORCE. OPERABILITY OF ALL VALVES IDENTIFIED TO HAVE DEFICIENCIES WILL BE ASSURED PRIOR TO STARTUP. IT IS NOT POSSIBLE TO DETERMINE THE ROOT CAUSE AND CORRECTIVE ACTIONS FOR THIS EVENT UNTIL FURTHER INFORMATION IS OBTAINED FROM ENGINEERING EVALUATIONS. THEREFORE, A SUPPLEMENT TO THIS REPORT WILL BE SUBMITTED UPON COMPLETION OF THE ENGINEERING EVALUATION BY JANUARY 31, 1992.

[124] WPPSS 2 DOCKET 50-397 LER 91-027
INADEQUATE PROCEDURES REGARDING JET PUMP OPERABILITY.
EVENT DATE: 093091 REPORT DATE: 110691 NSSS: GE TYPE: BWR

(NSIC 223377) ON OCTOBER 15, 1991 IT WAS CONCLUDED, BASED ON THE RESULTS OF A REPORTABILITY EVALUATION, THAT THE PREVIOUSLY ACCEPTED UNDERSTANDING OF TECHNICAL SPECIFICATION REQUIREMENTS FOR JET PUMP TESTING IN MODE 2 WERE NOT IN LITERAL COMPLIANCE WITH THE PLANT TECHNICAL SPECIFICATIONS. ON SEPTEMBER 26, 1991 AT 1636 HOURS WNP-2 ENTERED OPERATIONAL CONDITION 2. AT 2338 HOURS ON SEPTEMBER 29, 1991 WNP-2 ENTERED OPERATIONAL CONDITION 1. ON SEPTEMBER 30, 1991 A QUESTION ON JET PUMP OPERABILITY AND SURVEILLANCE APPLICABILITY IN OPERATIONAL CONDITIONS 1 AND 2 BELOW 25% OF RATED THERMAL POWER WAS RAISED. IT WAS CONCLUDED AFTER EXTENDED EVALUATION THAT THE TECHNICAL SPECIFICATION REQUIRES PERFORMANCE OF THE TESTING WITHIN 24 HOURS AFTER ENTRY INTO MODE 2. THE IMMEDIATE CORRECTIVE ACTIONS FOR THIS EVENT WAS THAT PLANT STARTUP PROCEDURES WERE DEVIATED TO REQUIRE PERFORMANCE OF THE SURVEILLANCE PROCEDURE WITHIN 24 HOURS AFTER ENTRY INTO MODE 2 BUT PRIOR TO EXCEEDING 25% OF RATED THERMAL POWER. THE ROOT CAUSE FOR THIS EVENT WAS PROCEDURES LESS THAN ADEQUATE/INSTRUCTIONS AMBIGUOUS. THE TECHNICAL SPECIFICATION FOR JET PUMPS DOES NOT CLEARLY DEFINE WHEN OPERABILITY MUST BE DEMONSTRATED. IN ADDITION, PLANT PROCEDURES FAILED TO PROVIDE SUFFICIENT INFORMATION TO CLEARLY DEFINE WHEN THE SPECIFICATION IS APPLICABLE.

[125] WPPSS 2 DOCKET 50-397 LER 91-029
INADEQUATE PRIMARY CONTAINMENT HYDROGEN RECOMBINER RECYCLE FLOW CONTROL.
EVENT DATE: 103191 REPORT DATE: 112791 NSSS: GE TYPE: BWR
VENDOR: BAILEY INSTRUMENT CO., INC.

(NSIC 223546) ON OCTOBER 31, 1991, A REPORTABILITY EVALUATION WAS COMPLETED THAT CONCLUDED THAT A PROBLEM ASSOCIATED WITH FLOW CONTROL OF THE PRIMARY CONTAINMENT HYDROGEN RECOMBINERS WAS REPORTABLE. A CONTRACT ENGINEER PERFORMING A SETPOINT CALCULATION REVIEW HAD DISCOVERED THAT INCORRECT CONTAINMENT ATMOSPHERIC CONTROL (CAC) RECYCLE FLOW CONTROL CONTROLLERS (CAC-FC-67A/B) WERE INSTALLED FOR BOTH DIVISIONS IN THE CONTROL ROOM. THE PLANT DESIGN AND OPERATING PROCEDURES REQUIRED THESE INSTRUMENTS TO BE USED IN THE AUTO MODE OF OPERATIONS TO CONTROL RECOMBINER RECYCLE FLOW. IF THESE INCORRECT CONTROLLERS HAD BEEN USED IN THE AUTO MODE, THEY WOULD NOT HAVE CONTROLLED RECYCLE FLOW WHICH COULD HAVE RESULTED IN A REDUCED RECOMBINATION RATE OR POSSIBLE SYSTEM SHUTDOWN DUE TO EXCESSIVE RECOMBINATION. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO CHANGE PLANT PROCEDURES REQUIRING OPERATION OF THESE INSTRUMENTS IN THE MANUAL MODE. THIS ALLOWS PLANT OPERATORS TO CONTROL RECYCLE FLOW FROM THE CONTROL ROOM BY MANUALLY POSITIONING THE RECYCLE FLOW CONTROL VALVE (CAC-FCV-6A/B).

[126] WPPSS 2 DOCKET 50-397 LER 91-030
TECHNICAL SPECIFICATION REQUIRED PLANT SHUTDOWN DUE TO REACTOR PRESSURE BOUNDARY LEAKAGE THROUGH DEFECTIVE WELD ON RESIDUAL HEAT REMOVAL SYSTEM DRAIN LINE PIPING.
EVENT DATE: 110491 REPORT DATE: 112791 NSSS: GE TYPE: BWR

(NSIC 223547) ON NOVEMBER 4, 1991 AT 0423 HOURS, AN INSPECTION TEAM THAT WAS PERFORMING A ROUTINE 1000 PSIG DRYWELL INSPECTION DURING STARTUP FROM AN OUTAGE, INFORMED THE SHIFT MANAGER THAT A SMALL LEAK (APPROXIMATELY 20 DROPS PER MINUTE) HAD BEEN IDENTIFIED IN A WELDED CONNECTION BETWEEN A RESIDUAL HEAT REMOVAL (RHR) SYSTEM DRAIN VALVE AND THE RHR, LOOP A, SHUTDOWN COOLING RETURN LINE. THE SHIFT MANAGER DETERMINED THAT THIS CONDITION REPRESENTED PRESSURE BOUNDARY LEAKAGE. ALTHOUGH THE LEAKAGE COULD HAVE BEEN ISOLATED BY MANUALLY CLOSING AN RHR ISOLATION VALVE, THE DECISION WAS MADE BY PLANT MANAGEMENT TO DECLARE AN UNUSUAL EVENT AND COMMENCE A REACTOR SHUTDOWN BECAUSE OF THE LEAKAGE AND THE EXTENDED LENGTH OF TIME REQUIRED TO CLOSE THE VALVE AND ISOLATE THE LEAKAGE FROM THE REACTOR PRESSURE COOLANT BOUNDARY. AT 0454 HOURS AN UNUSUAL EVENT WAS DECLARED AND PLANT CONTROL ROOM OPERATORS COMMENCED SHUTDOWN OF THE PLANT. AS IMMEDIATE CORRECTIVE ACTIONS, THE PLANT WAS MANUALLY SCRAMMED AT 0525 HOURS AND, AT 0608 HOURS, THE MANUAL ISOLATION VALVE WAS CLOSED TO ISOLATE THE LEAKAGE. AT 0618 HOURS, THE UNUSUAL EVENT WAS TERMINATED.

[127] WPPSS 2 DOCKET 50-397 LER 91-031
INTERMEDIATE RANGE MONITORS CONTROL ROD BLOCK CHANNEL CALIBRATIONS NOT PERFORMED IN THE REQUIRED FREQUENCY.
EVENT DATE: 110491 REPORT DATE: 112791 NSSS: GE TYPE: BWR

(NSIC 223548) ON NOVEMBER 4, 1991 IT WAS DETERMINED THAT THE PROCEDURES PERFORMED TO SATISFY THE INTERMEDIATE RANGE MONITORS (IRMS) CONTROL ROD BLOCK FUNCTION QUARTERLY (EVERY 92 DAYS) CHANNEL CALIBRATION REQUIREMENTS DID NOT MEET THE TECHNICAL SPECIFICATION DEFINITION OF A CHANNEL CALIBRATION. THIS CONDITION IS A DEVIATION FROM THE TECHNICAL SPECIFICATIONS AND IS REPORTABLE PER 10CFR50.73(A)(2)(I)(B). AS AN IMMEDIATE CORRECTIVE ACTION, THE 18-MONTH IRM REACTOR PROTECTION SYSTEM (RPS) CHANNEL CALIBRATION PROCEDURES WERE PERFORMED. THESE PROCEDURES ALSO SATISFY THE QUARTERLY IRM CONTROL ROD BLOCK CHANNEL CALIBRATION REQUIREMENTS. THE ROOT CAUSE OF THIS EVENT WAS PROCEDURES THAT WERE LESS THAN ADEQUATE. THE IRM CONTROL ROD BLOCK QUARTERLY SURVEILLANCE PROCEDURES DID NOT PERFORM A COMPLETE CHANNEL CALIBRATION. PROCEDURES USED TO SATISFY THE IRM CONTROL ROD BLOCK FUNCTION CHANNEL CALIBRATION REQUIREMENTS WILL BE REVISED TO INCLUDE THOSE STEPS NECESSARY TO CONSTITUTE A CHANNEL CALIBRATION. THIS EVENT WAS NOT SAFETY SIGNIFICANT SINCE AN IRM CONTROL ROD BLOCK FUNCTION CHANNEL CALIBRATION WAS PERFORMED EVERY 18 MONTHS AS PART OF THE IRM RPS CHANNEL CALIBRATION. THE IRM CONTROL ROD BLOCK FUNCTION IS NOT REQUIRED FOR ACCIDENT PREVENTION AND MITIGATION SINCE THE LICENSING BASIS ACCIDENTS ARE MITIGATED BY THE RPS SYSTEM TRIPS (AVERAGE POWER RANGE MONITORS (APRMS) AND IRMS).

[128] YANKEE ROWE DOCKET 50-029 LER 91-005
ALL EMERGENCY DIESEL GENERATORS DECLARED INOPERABLE.
EVENT DATE: 110591 REPORT DATE: 120491 NSSS: WE TYPE: PWR
VENDOR: FURNAS ELECTRIC CO.

(NSIC 223457) THE PLANT WAS IN MODE 5, AT 0% REACTOR POWER, WITH A MAIN COOLANT SYSTEM TEMPERATURE OF 103 DEGREES FAHRENHEIT. EDG NO. 2 WAS OUT OF SERVICE. DURING WEEKLY SURVEILLANCE TESTING OF EDG NOS. 1 AND 3, EXCESSIVE ARCING WAS OBSERVED ACROSS CONTACTS IN A CONTROL RELAY. THE EDGS WERE REMOVED FROM SERVICE FOR EVALUATION, AND DECLARED INOPERABLE. AT 1115 HOURS, AN UNUSUAL EVENT WAS DECLARED DUE TO THE LOSS OF ALL THREE EDGS. AT 1300 HOURS, EDG NO.2 WAS RETURNED TO SERVICE AND THE UNUSUAL EVENT WAS TERMINATED. THE ROOT CAUSE OF THE CONTROL RELAY ARCING WAS THE INSTALLATION OF 240/480 VAC STARTING CONTACTOR COILS IN A SYSTEM DESIGNED FOR 125 VDC. SHORT TERM CORRECTIVE ACTION INCLUDES REPLACEMENT OF THE AC COILS WITH DC COILS AND REPLACEMENT OF THE STARTER CONTACTOR WITH ONE RATED FOR DC APPLICATIONS. LONG TERM CORRECTIVE ACTION IS TO UPGRADE THE EDG CIRCUITRY DURING THE PRESENT REFUELING OUTAGE. ALTHOUGH EDGS NOS. 1 AND 3 WERE DECLARED INOPERABLE, THEY REMAINED CAPABLE OF BEING STARTED AND LOADED IF NEEDED. THERE WAS NO ADVERSE IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT. NO SIMILAR EVENTS OF THIS NATURE HAVE BEEN REPORTED IN ANY PREVIOUS LERS AT THE YANKEE NUCLEAR POWER STATION.

[129] ZION 1 DOCKET 50-295 LER 91-016
REACTOR TRIP AND SAFETY INJECTION DUE TO A.C. INSTRUMENT INVERTER FAILURE.
EVENT DATE: 110791 REPORT DATE: 120991 NSSS: WE TYPE: PWR
VENDOR: MOORE PRODUCTS COMPANY
WESTINGHOUSE ELECTRIC CORP.

(NSIC 223491) AT 0434 ON 11/7/91, WITH THE UNIT AT 100% POWER AND STEADY STATE CONDITIONS, A.C. INSTRUMENT INVERTER 114 FAILED, CAUSING THE 'LOW STEAM GENERATOR LEVEL' AND 'LOW STEAM GENERATOR PRESSURE' BISTABLES TO TRIP. SIX SECONDS LATER A STEAM FLOW/FEED FLOW MISMATCH SIGNAL WAS GENERATED DUE TO 1C FEED PUMP FLOW REDUCTION CAUSING A REACTOR TRIP. THE REACTOR TRIP SIGNAL CAUSED THE HIGH STEAM FLOW PROGRAM TO RESET AS DESIGNED, RESULTING IN A HIGH STEAM FLOW SIGNAL. THE HIGH STEAM FLOW SIGNAL COMBINED WITH THE 'LOW STEAM GENERATOR PRESSURE' BISTABLES PREVIOUSLY TRIPPED TO ESTABLISH THE COINCIDENCE NECESSARY TO ACTUATE SAFETY INJECTION AND MAIN STEAMLINE ISOLATION. THE CAUSE OF THE INVERTER FAILURE WAS A COMPONENT FAILURE. A SILICON CONTROLLED RECTIFIER (SCR) IN THE MASTER SECTION OF INVERTER 114 MISGATED CAUSING A REDUCTION IN THE INVERTER OUTPUT VOLTAGE. THIS VOLTAGE REDUCTION WAS SUFFICIENT ENOUGH TO TRIP THE CHANNELS OF REACTOR

PROTECTION AND SAFEGUARDS INSTRUMENTATION FED FROM THIS INVERTER. DURING THIS EVENT ALL SAFETY RELATED EQUIPMENT OPERATED AS DESIGNED AND THERE WAS MINIMAL SAFETY SIGNIFICANCE TO THIS EVENT. CORRECTIVE ACTIONS INCLUDED REPLACING ALL UNIT 1 INVERTER SCRS PRIOR TO STARTUP, INCLUDING THE SCRS IN THE PREVENTATIVE MAINTENANCE PROGRAM AND ENSURING ALL UNIT 2 SCRS WILL BE REPLACED DURING THE NEXT UNIT 2 REFUELING OUTAGE.

[130] ZION 1 DOCKET 50-295 LER 91-017
LOSS OF SERVICE BUS 142 DUE TO A PERSONNEL ERROR.
EVENT DATE: 110891 REPORT DATE: 120991 NSSS: WE TYPE: PWR

(NSIC 223492) AT 1445 ON FRIDAY NOVEMBER 8, 1991, WITH UNIT 1 IN HOT SHUTDOWN, 4KV SERVICE BUS 142 LOST POWER. SERVICE BUS 142 IS THE NORMAL FEED TO ENGINEERED SAFETY FEATURES (ESF) BUS 147. THE LOSS OF POWER TO SERVICE BUS 142 RESULTED IN LOSS OF POWER TO ESF BUS 147. THE LOSS OF ESF BUS 147 RESULTED IN AN AUTO START SIGNAL TO THE '0' EMERGENCY DIESEL GENERATOR (EDG). THE '0' EDG AUTO STARTED DUE TO THE LOW VOLTAGE SENSED ON ESF BUS 147. THE EMERGENCY BREAKER TO BUS 147 CLOSED, ALLOWING 0 EDG TO PICK UP BUS 147. ALL ESF EQUIPMENT OPERATED AS DESIGNED. THIS EVENT WAS CAUSED BY A PERSONNEL ERROR ON THE PART OF THE OPERATIONAL ANALYSIS DEPARTMENT (OAD) ENGINEER. THE ENGINEER DID NOT RESET THE LBB TRIP RELAY PRIOR TO RESTORING THE TEST SWITCHES FOR THE FIVE UNIT 1 SERVICE BUSES. CONTRIBUTING CAUSES INCLUDED OAD ENGINEERS PERFORMING THE 345 KV SWITCHYARD PROTECTIVE RELAY SYSTEMS TEST USING SCHEMATIC DIAGRAMS, JOB SPECIFIC VENDOR MANUALS, AND JOB SPECIFIC TECHNICAL TRAINING, A LACK OF FORMAL WRITTEN PROCEDURES FOR OAD PERSONNEL TO USE WHEN TESTING THE 345 KV SWITCHYARD PROTECTIVE RELAY SYSTEMS, A LACK OF CLEARLY EXPRESSED GUIDANCE FOR CHECKING FOR THE PRESENCE OF TRIP VOLTAGES, AND INADEQUATE COMMUNICATION BETWEEN OAD ENGINEER AND SHIFT PERSONNEL. THE SAFETY SIGNIFICANCE OF THIS EVENT IS BOUNDED BY THE FSAR LOSS OF A ESF BUS DURING HOT SHUTDOWN ANALYSIS.

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

KEYWORD INDEX

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

KEYWORD INDEX

KEYWORD INDEX

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VENDOR CODE INDEX

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(See instructions on the reverse)

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Oak Ridge, TN 37831

9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

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

10. SUPPLEMENTARY NOTES

11. 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 Operations Analysis Center (NOAC) 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 [illegible]ordance with federal regulations. Procedures for LER reporting for revisions to those events occurring prior to 198[illegible] [illegible]e described in NRC Regulatory Guide 1.16 and NUREG-0161, *Instructions for Preparation of Data Entry Sheets for Licensee Event Reports*. For those events occuring 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.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

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Reactor, BWR
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Components
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