NUREG/CR-2000
ORNL/NSIC-200
Vol. 9, No. 8

Licensee Event Report (LER) Compilation

For month of August 1990

Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

Available from

Superintendent of Documents
U.S. Government Printing Office
Post Office Box 37082
Washington, D.C. 20013-7082

A year's subscription consists of 12 issues for
this publication.

Single copies of this publication
are available from National Technical
Information Service, Springfield, VA 22161

NUREG/CR-2000
ORNL/NSIC-200
Vol. 9, No. 8

Licensee Event Report (LER) Compilation

For month of August 1990

Manuscript Completed: September 1990
Date Published: September 1990

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

**Prepared for
Office for Analysis and Evaluation of Operational Data
U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FIN A9135**

Abstract

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

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

Gary T. Mays
Nuclear Operations Analysis Center
Oak Ridge National Laboratory
P. O. Box 2009, Oak Ridge, TN 37831-8065
Telephone: 615/574-0391, FTS Number 624-0391

Questions regarding LER searches should be directed to

W. P. Poore (same address as above)
Telephone: 615/574-0325, FTS Number 624-0325

[1] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 90-004
DEGRADED FIRE BARRIER PENETRATION AS THE RESULT OF PERSONNEL OVERSIGHT AND PROCEDURAL INADEQUACY.
EVENT DATE: 053190 REPORT DATE: 070290 NSSS: BW TYPE: PWR

(NSIC 218761) ON 5/31/90 AT 1330, WHILE CONDUCTING A FIRE BARRIER PENETRATION SEAL INSPECTION AS PART OF A COMPREHENSIVE INSPECTION PROGRAM INITIATED AS PART OF A GENERIC LETTER 86-10 EVALUATION, A DEGRADED FIRE BARRIER WAS DISCOVERED BY PERSONNEL WITHIN THE FIRE PROTECTION GROUP AT ARKANSAS NUCLEAR ONE. DEFICIENT SEAL CONSISTED OF A 2 INCH METAL SLEEVE THROUGH A FLOOR SLAB AND A 1 1/2 INCH CONDUIT CONTAINED WITHIN THE SLEEVE. A REVIEW OF PAST DOCUMENTATION REVEALED THIS CONDITION HAS EXISTED PRIOR TO A GENERAL FIRE BARRIER INSPECTION WALK DOWN CONDUCTED IN 1983. SINCE THIS CONDITION WAS NOT IDENTIFIED DURING THIS WALK DOWN OR SUBSEQUENT TECH SPEC SURVEILLANCES, ROOT CAUSE OF THIS CONDITION HAS BEEN DETERMINED TO BE PERSONNEL ERROR AND OVERSIGHT REGARDING INCORRECT PROCEDURE IDENTIFICATION OF PENETRATION NUMBER 97-0038. UPON DISCOVERY OF THIS CONDITION, CORRESPONDING FIRE DETECTION SYSTEM WAS VERIFIED OPERABLE, A FIRE WATCH WAS POSTED IN ACCORDANCE WITH TECH SPEC REQUIREMENTS, FIRE BARRIER WAS SEALED, AND APPLICABLE FIRE PRINT AND PENETRATION LOG UPDATED. FIRE BARRIER INSPECTION PROCEDURE WILL BE REVISED AND A TRAINING PROGRAM WILL BE IMPLEMENTED FOR FIRE BARRIER INSPECTORS. DEGRADED FIRE BARRIER PENETRATION SEAL IS NOT A SIGNIFICANT SAFETY CONCERN CONSIDERING THE FIRE PREVENTATIVE MEASURES CURRENTLY AVAILABLE BUT IS REPORTABLE PURSUANT TO 10CFR50.73 (A)(2)(I)(B).

[2] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 90-013
DEGRADED FIRE BARRIER PENETRATION CAUSED BY PERSONNEL ERROR ASSOCIATED WITH THE PLANT MODIFICATION PROCESS.
EVENT DATE: 053090 REPORT DATE: 062990 NSSS: CE TYPE: PWR

(NSIC 218767) ON 5/30/90, WHILE PERFORMING A ROUTINE TOUR OF THE ARKANSAS NUCLEAR ONE, UNIT TWO (ANO-2) AUXILIARY BUILDING, AN ANO-2 WASTE CONTROL OPERATOR IDENTIFIED A DEGRADED FIRE BARRIER (PENETRATION, FB-2054-6). A ROVING FIRE WATCH WAS IMMEDIATELY ESTABLISHED AND THE FIRE DETECTION INSTRUMENTATION IN THE AREA WAS VERIFIED OPERABLE. THE PENETRATION IS IN A FOAM BLOCKOUT WITH A TWO INCH CONDUIT PENETRATING A FOUR INCH CORE BORE. THE FIRE BARRIER SEPARATES TWO FIRE AREAS, WITH VERY LOW FIRE LOADING IN EACH AREA. WHILE PERFORMING A PLANT MODIFICATION, THE PENETRATION WAS BREACHED. IN THE PLANT MODIFICATION PROCESS, AT THAT TIME, ADEQUATE PROCEDURAL CONTROLS EXISTED TO ENSURE BREACHED BARRIERS WERE PROPERLY SEALED. AS A RESULT OF A PERSONNEL ERROR, THE PROCESS WAS NOT UTILIZED AND THE PENETRATION WAS NOT PROPERLY SEALED WHEN IT WAS BREACHED. THE DEGRADED PENETRATION FIRE BARRIER HAS BEEN PROPERLY SEALED. THE FIRE AREAS LOCATED ON EITHER SIDE OF THE FIRE BARRIER ARE EQUIPPED WITH FIXED FIRE DETECTION INSTRUMENTATION WHICH ANNUNCIATE IN THE CONTROL ROOM. ADDITIONALLY, FIRE SUPPRESSION EQUIPMENT IS READILY AVAILABLE AND FIRE BRIGADE PERSONNEL, TRAINED IN FIRE FIGHTING, ARE AVAILABLE AT ALL TIMES. THEREFORE, THERE ARE NO SAFETY CONCERNS RELATED TO THIS DEGRADED PENETRATION FIRE BARRIER. THIS CONDITION IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B).

[3] ARNOLD DOCKET 50-331 LER 90-005
AVERAGE POWER RANGE MONITOR FLOW-BIASED TRIP SETPOINT SHIFTED TO THE NON-CONSERVATIVE DIRECTION DUE TO INADEQUATE FLOW UNITS CALIBRATION.
EVENT DATE: 052590 REPORT DATE: 062290 NSSS: GE TYPE: BWR

(NSIC 218763) ON 5/25/90, REACTOR WAS OPERATING AT 83.1% POWER (MWTH). A SYSTEMS ENGINEER, FOLLOWING AN OPERATIONS DEPARTMENT REQUEST TO INVESTIGATE A DIFFERENCE BETWEEN THE RECIRCULATION DRIVEN FLOW INDICATIONS AND THE AVERAGE POWER RANGE MONITORING (APRM) FLOW UNITS, DETERMINED THAT A NON-CONSERVATIVE DEVIATION EXISTED ON THE FLOW BIASED TRIP SETPOINT. DEVIATION BETWEEN THE ACTUAL FLOW BIASED TRIP SETPOINT AND THE REQUIRED SETPOINT WAS 4.1%. THE INTERMEDIATE CAUSE WAS THE FLOW UNITS BEING CALIBRATED TO THE RECIRCULATION DRIVE FLOW REQUIRED TO ACHIEVE 100% CORE FLOW AT FULL POWER DURING THE INITIAL PLANT OPERATION, 26,550 GPM. RECIRCULATION DRIVE FLOW REQUIRED TO ACHIEVE 100% CORE FLOW AT FULL POWER HAS INCREASED SINCE THE INITIAL STARTUP, BUT THE CALIBRATION PROCEDURE WAS NOT UPDATED ACCORDINGLY. THE ROOT CAUSE IS FAILURE TO RECOGNIZE THE POTENTIAL FOR

CHANGES IN RECIRCULATION DRIVING FLOW AND HAVE AN ESTABLISHED PROGRAM TO MONITOR AND FEED BACK THE CHANGES INTO THE CALIBRATION PROCEDURE. AS AN IMMEDIATE CORRECTIVE ACTION, THE GAIN ADJUSTMENT FACTOR (GAF) FOR THE APRM SETTING WAS ADJUSTED UPWARD TO COMPENSATE FOR THE DEVIATION IN THE FLOW UNITS. LONG TERM CORRECTIVE ACTIONS ARE TO REVISE THE FLOW BIAS UNIT CALIBRATION PROCEDURE AND TO MONITOR THE RATIO OF DRIVEN FLOW OVER DRIVING FLOW AND ANY CHANGES WILL BE EVALUATED BY DESIGN ENGINEERING.

[4] BEAVER VALLEY 1 DOCKET 50-334 LER 89-017 REV 02
UPDATE ON FEEDWATER ISOLATION DUE TO SWELL PHENOMENA.
EVENT DATE: 121889 REPORT DATE: 061590 NSSS: WE TYPE: PWR

(NSIC 218494) ON 12/18/89, WITH THE UNIT IN HOT STANDBY (OPERATING MODE 3), OPERATIONS PERSONNEL OPENED THE "C" ATMOSPHERIC STEAM RELIEF VALVE TO LOWER THE "C" STEAM GENERATOR PRESSURE TO FACILITATE OPENING OF THE "C" MAIN STEAM ISOLATION VALVE (MSIV). THIS IS REQUIRED DUE TO AN INTERLOCK WHICH PREVENTS THE VALVE FROM OPENING ON A HIGH DIFFERENTIAL PRESSURE. STEAM GENERATOR LEVEL IN THE "C" STEAM GENERATOR (SGC) WAS INDICATING 55 PERCENT (%) ON THE NARROW RANGE SCALE. UPON OPENING OF THE "C" MSIV, AT 1323 HOURS, STEAM GENERATOR LEVEL IN SGC SWELLED TO 75% CAUSING A FEEDWATER ISOLATION (FWI) SIGNAL. THIS CAUSED THE CLOSURE OF THE FEEDWATER CONTAINMENT ISOLATION VALVES. OPERATIONS PERSONNEL RESET THE FWI SIGNAL. AN INVESTIGATION DETERMINED THAT THE CAUSE ON THIS EVENT WAS SWELL PHENOMENA, DUE TO THE PLANT CONDITIONS AND THE REACTOR COOLANT SYSTEM TEMPERATURE DECREASE EXPERIENCED AFTER THE "C" MSIV WAS OPENED. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. FEEDWATER ISOLATIONS CAUSED BY HI-HI STEAM GENERATOR LEVELS ARE ANALYZED IN BEAVER VALLEY UNIT 1 UFSAR SECTION 14.1.9, "EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS."

[5] BEAVER VALLEY 2 DOCKET 50-412 LER 90-006
OPERATION IN A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS.
EVENT DATE: 060590 REPORT DATE: 070590 NSSS: WE TYPE: PWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)
FISHER CONTROLS CO.

(NSIC 218732) ON 6/5/90 AT 1315 HOURS, WITH THE UNIT AT 86% REACTOR POWER, SURVEILLANCE TESTING OF THE LETDOWN ORIFICE ISOLATION VALVES, 2CHS*AOV200A,B,C WAS IN PROGRESS. THIS TESTING IDENTIFIED THAT ONE SET OF SERIES SOLENOID VALVES, FOR 2CHS*AOV200A,B,C FAILED TO MEET THE TECHNICAL SPECIFICATION (TS) 3.6.3.1 CONTAINMENT ISOLATION PHASE "A" (CIA) CLOSURE LIMIT OF LESS THAN 10 SECONDS. THE VALVES WERE DECLARED INOPERABLE. NORMAL REACTOR COOLANT SYSTEM CHARGING AND LETDOWN WERE ISOLATED AND EXCESS LETDOWN WAS PLACED IN SERVICE TO MAINTAIN REACTOR COOLANT SYSTEM INVENTORY. THE OUTSIDE CONTAINMENT ISOLATION VALVE WAS CLOSED ISOLATING THIS PENETRATION. THE CAUSE FOR THIS EVENT WAS A DEFICIENCY IN THE SURVEILLANCE PROGRAM FOR 2CHS*AOV200A,B,C. A TEMPORARY WAIVER OF COMPLIANCE FOR TS 3.6.3.1 WAS OBTAINED FROM THE NUCLEAR REGULATORY COMMISSION ALLOWING CONTINUED OPERATION WITH NORMAL CHARGING AND LETDOWN IN SERVICE. THERE WERE NO SAFETY IMPLICATIONS AS A RESULT OF THIS EVENT. THE AS FOUND CLOSURE TIMES OF 2CHS*AOV200A,B,C ARE BOUNDED BY THE CLOSURE TIME LIMIT ESTABLISHED FOR THE OUTSIDE CONTAINMENT ISOLATION VALVE, ENSURING THAT THE PENETRATION IS ISOLATED WITH THE TIME REQUIREMENTS STATED IN THE UPDATED FINAL SAFETY ANALYSIS.

[6] BRAIDWOOD 1 DOCKET 50-456 LER 90-007
INADVERTENT START OF THE 1A AUXILIARY FEEDWATER PUMP DUE TO TESTING METHODOLOGY DEFICIENCY.
EVENT DATE: 060190 REPORT DATE: 062890 NSSS: WE TYPE: PWR

(NSIC 218775) ON 6/1/90 A REACTOR OPERATOR (RO) WAS PERFORMING A SLAVE RELAY SURVEILLANCE THAT TESTED THE 1A AUXILIARY FEEDWATER PUMP (AF) STEAM GENERATOR LOW LOW LEVEL AUTO START ACTUATION RELAY, K633. THIS RELAY WAS PART OF THE SOLID STATE PROTECTION SYSTEM. AS PART OF THE RESTORATION PORTION OF THE PROCEDURE THE RESISTANCE ACROSS THE RELAY CONTACTS WAS MEASURED TO VERIFY AN OPEN CIRCUIT PRIOR TO ENABLING THE SYSTEM. AT 0927 THE NSO MEASURED THE RESISTANCE ACROSS TERMINALS 1 AND 2 OF RELAY K633 IN ACCORDANCE WITH THE PROCEDURE. WHILE MAINTAINING ONE OF

THE OHMMETER PROBES ON TERMINAL 2 THE NSO REMOVED THE PROBE FROM TERMINAL 1. DURING THE REMOVAL OF THE PROBE THE NSO INADVERTENTLY TOUCHED THE PROBE TO THE EXPOSED TERMINAL POINT 13 ON RELAY K633. THIS CREATED A CURRENT PATH FROM A PARALLEL CIRCUIT THROUGH THE OHMMETER TO THE STARTING RELAY FOR THE 1A AF PUMP. THIS RESULTED IN THE AUTO START OF THE PUMP. THE 1A AF PUMP CONTROL SWITCH WAS IMMEDIATELY PLACED IN THE "PULLOUT" POSITION AND THE PUMP WAS SECURED. THE CAUSE OF THIS EVENT WAS A METHODOLOGY DEFICIENCY WHICH IS CONSIDERED A PROGRAMMATIC DEFICIENCY. OHMMETER PROBES HAVE BEEN MODIFIED TO REDUCE THE EXPOSED METAL SURFACES TO A SMALL PORTION OF THE TIP. ALTERNATE LOCATIONS FOR OBTAINING THE READINGS WILL BE USED. TRAINING WILL BE PROVIDED. PREVIOUS CORRECTIVE ACTIONS ARE NOT APPLICABLE.

[7] BRAIDWOOD 1 DOCKET 50-456 LER 90-008
REACTOR TRIP DUE TO LIGHTNING INDUCED VOLTAGE TRANSIENT IN THE ROD CONTROL SYSTEM.
EVENT DATE: 060890 REPORT DATE: 070290 NSSS: WE TYPE: PWR

(NSIC 218808) ON 6/8/90 THERE WAS HEAVY THUNDERSTORM ACTIVITY IN THE BRAIDWOOD STATION AREA. AT 0618 A REACTOR TRIP OCCURRED ON UNIT 1 DUE TO A HIGH FLUX RATE TRIP SIGNAL FROM THE POWER RANGE NUCLEAR INSTRUMENTATION. THE REACTOR OPERATORS VERIFIED ALL AUTOMATIC ACTIONS. ALL SYSTEMS FUNCTIONED AS DESIGNED. STABLE PLANT CONDITIONS WERE IMMEDIATELY ESTABLISHED. AN EXAMINATION OF THE ROD DRIVE (RD) POWER CABINETS INDICATED THE POWER SUPPLY OVER VOLTAGE PROTECTORS (PSOVP) HAD TRIPPED IN THREE OF THE RD CABINETS. THERE WERE NO BLOWN FUSES IDENTIFIED IN ANY OF THE CABINETS. THE PSOVP'S WERE RESET AND ALL VOLTAGES WERE CHECKED. THE ROOT CAUSE OF THIS EVENT WAS A VOLTAGE TRANSIENT. IT IS BELIEVED THAT LIGHTNING STRUCK THE UNIT 1 CONTAINMENT AND CAUSED A VOLTAGE SURGE IN THE STATION GROUND SYSTEM. THIS CAUSED THE ACTIVATION OF PSOVPS IN THE 3 RD POWER CABINETS. THIS SHUT OFF THE CURRENT TO THE STATIONARY GRIPPER COILS OF THE RODS POWERED BY THE CABINETS, AND CAUSED THEM TO DROP, RESULTING IN A NEGATIVE FLUX RATE. THE NEGATIVE FLUX RATE WAS OF SUFFICIENT MAGNITUDE TO ACTIVATE THE REACTOR TRIP SIGNAL FROM THE POWER RANGE NUCLEAR INSTRUMENTATION. NO DAMAGE OCCURRED TO THE RD SYSTEM. A REVIEW OF THE STATION LIGHTNING PROTECTION SYSTEM IS BEING CONDUCTED. THE STATION HAD MADE MODIFICATIONS TO THE RD POWER SUPPLY SYSTEM AND THE STATION GROUNDING SYSTEM AS CORRECTIVE MEASURES FROM PREVIOUS EVENTS.

[8] BRAIDWOOD 1 DOCKET 50-456 LER 90-009
FAILURE TO PERFORM CONTAINMENT AIR LOCK LEAK TEST WITHIN THE ALLOWABLE TIME DUE TO PROGRAMMATIC DEFICIENCY.
EVENT DATE: 061590 REPORT DATE: 070690 NSSS: WE TYPE: PWR

(NSIC 218809) A SYSTEM ENGINEER (STE) HAD BEEN MONITORING THE HIGH HIGH RADIATION AREA (HHRA) KEY CONTROL LOG, ON A REGULAR BASIS TO DETERMINE WHEN A LEAKAGE TEST OF THE CONTAINMENT PERSONNEL AIR LOCK (CPA) WAS REQUIRED. THIS WAS THE EXISTING PROGRAM. DURING THE WEEK OF JUNE 11 THE STE REVIEWED THE HHRA KEY LOG ON A REGULAR BASIS AND DID NOT OBSERVE ANY KEY ISSUANCE FOR CONTAINMENT ENTRY. ON 6/15/90 THE STE MADE HIS HHRA KEY LOG REVIEW AND DID NOT OBSERVE ANY NEW ENTRIES. LATER THAT MORNING THE STE WAS INFORMED THAT A CONTAINMENT ENTRY HAD BEEN MADE. AT 1030 THE STE REVIEWED THE HHRA KEY LOG. HE OBSERVED THAT THE LAST ENTRY ON THE LOG SHEET WAS DATED 6/8/90. PUZZLED BY THIS DISCREPANCY, THE STE LEAFED THROUGH THE BLANK PAGES OF LOG SHEETS. WHEN THE STE TURNED TO THE LAST BLANK LOG SHEET HE DISCOVERED THAT A SECOND LOG SHEET HAD BEEN INITIATED. THERE WERE TWO ENTRIES ON THE SECOND LOG SHEET, ONE FOR THAT MORNING, AND ONE FOR A CONTAINMENT ENTRY MADE ON 6/8/90. THE STE IMMEDIATELY PERFORMED THE CPA LEAK TEST AND THE RESULTS WERE FOUND ACCEPTABLE. THIS RESULTED IN EXCEEDING THE ALLOWED TIME BY 3 DAYS. THE CAUSE OF THIS EVENT WAS A PROGRAMMATIC DEFICIENCY. THE PROGRAM WILL BE REVISED. DIVIDERS HAVE BEEN PLACED IN THE HHRA KEY LOG AS AN INTERIM ACTION. PREVIOUS CORRECTIVE ACTIONS ARE NOT APPLICABLE.

[9] BRAIDWOOD 2 DOCKET 50-457 LER 90-008
SPURIOUS ACTUATION OF THE UNIT IN POWER PERMISSIVE CIRCUIT DURING COLD SHUTDOWN ACTIVITIES DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 051790 REPORT DATE: 061490 NSSS: WE TYPE: PWR

(NSIC 218556) THE REACTOR TRIP BREAKER AND GRIPPER COIL RESPONSE TIME MEASUREMENT SURVEILLANCE WAS IN PROGRESS. THE ROD DRIVE SYSTEM WAS NOT CAPABLE OF ROD WITHDRAWAL AS THE DISCONNECT SWITCHES FOR THE ROD LIFT CIRCUITS WERE OPEN FOR ALL ROD DRIVE POWER CABINETS AS SPECIFIED IN THE "LIMITATIONS AND ACTIONS". THE PROCEDURE REQUIRED SWITCH S501 OF THE SOLID STATE PROTECTION SYSTEM (SSPS), BE RETURNED TO THE OFF POSITION. AT 0334 ON 5/17/90 AS SWITCH S501 WAS BEING ROTATED FROM POSITION 7 TO OFF IN A CLOCKWISE DIRECTION, THE UNIT AT POWER PERMISSIVE CIRCUIT (P-10) ACTUATED ON TRAIN "A" OF THE SSPS. REACTOR TRIPS THAT ARE NORMALLY AUTOMATICALLY BLOCKED WHEN POWER IS LESS THAN 10% WERE ENABLED FOR LESS THAN ONE SECOND. AS A RESULT, A REACTOR TRIP SIGNAL WAS GENERATED. NO COMPONENTS REPOSITIONED SINCE THE REACTOR TRIP BREAKERS WERE ALREADY OPEN. ANNUNCIATOR RESET WAS ACCOMPLISHED BY THE TIME S501 WAS IN THE OFF POSITION. IN AN EFFORT TO DUPLICATE THE MALFUNCTION, SWITCH S501 WAS SLOWLY ROTATED THROUGH EACH POSITION FROM 7 TO OFF IN A CLOCKWISE DIRECTION. AT 0420 WHEN THE SWITCH WAS PLACED IN POSITION 21, THE SYMPTONS WERE REPEATED. CAUSE OF THIS EVENT WAS A PROCEDURAL DEFICIENCY IN THAT THE DIRECTION OF SWITCH ROTATION WAS NOT SPECIFIED. THE PROCEDURE WILL BE REVISED. THERE HAVE BEEN NO PREVIOUS SIMILAR OCCURRENCES.

[10] BRAIDWOOD 2 DOCKET 50-457 LER 90-006
CONTAINMENT VENTILATION ISOLATION DUE TO A FAILED DETECTOR ON CONTAINMENT AREA RADIATION MONITOR.
EVENT DATE: 052190 REPORT DATE: 061890 NSSS: WE TYPE: PWR

(NSIC 218776) AT 1117 ON 5/21/90 A CONTAINMENT - FUEL HANDLING INCIDENT AREA RADIATION MONITOR FAILED ITS AUTOMATIC CHECKSOURCE TEST DUE TO A LOW COUNTRATE. THE CHECKSOURCE TEST IS GENERATED BY THE MICROPROCESSOR CIRCUITRY OF THE MONITOR EVERY 24 HOURS. UPON FAILURE, THE MONITOR REVERTED TO THE INTERLOCK POSITION. THIS GENERATED A CONTAINMENT VENTILATION ISOLATION SIGNAL FOR TRAIN "A" WHICH RESULTED IN THE CLOSURE OF MINI-PURGE EXHAUST VALVES AND TRIPPED THE RUNNING MINI-PURGE EXHAUST FAN. THE OPERATOR ACKNOWLEDGED THE CHECKSOURCE FAILURE ALARM AND VERIFIED ALL AUTOMATIC ACTIONS. THE CAUSE OF THIS EVENT WAS COMPONENT FAILURE. THE DETECTOR CHECKSOURCE CIRCUIT FAILED CAUSING THE RADIATION MONITOR TO REVERT TO THE INTERLOCK CONDITION WHICH RESULTED IN THE TRAIN "A" CONTAINMENT VENTILATION ISOLATION. THE DETECTOR WAS REPLACED, RECALIBRATED AND RETURNED TO SERVICE. THE CONTAINMENT FUEL HANDLING INCIDENT RADIATION MONITOR DETECTORS ARE REPLACED ON AN 18 MONTH FREQUENCY IN ACCORDANCE WITH THE ENVIRONMENTAL QUALIFICATION PROGRAM. THE FAILED DETECTOR IN THIS EVENT HAD BEEN IN SERVICE SINCE 4/26/90. A REVIEW OF COMPONENT FAILURE HISTORIES DID NOT IDENTIFY ANY ADVERSE TREND CONCERN FOR THIS TYPE OF DETECTOR. PREVIOUS CORRECTIVE ACTIONS WERE NOT APPLICABLE TO THIS EVENT.

[11] BRAIDWOOD 2 DOCKET 50-457 LER 90-007
CONTAINMENT VENTILATION ISOLATION DUE TO A FAILED DETECTOR ON CONTAINMENT AREA RADIATION MONITOR.
EVENT DATE: 052390 REPORT DATE: 062090 NSSS: WE TYPE: PWR
VENDOR: ITT-BARTON

(NSIC 218777) ON 5/23/90 A PLANT HEATUP WAS IN PROGRESS. AT 0325 THE REACTOR OPERATOR (NSO) OBSERVED THAT PRESSURIZER PRESSURE CHANNEL 458 HAD FAILED LOW. THE CHANNEL WAS DECLARED INOPERABLE. AT 0340 THE NSO OBSERVED THAT ANOTHER PRESSURIZER PRESSURE CHANNEL, 455, HAD DEVIATED IN EXCESS OF 6% ABOVE THE REMAINING TWO CHANNELS. THE 455 CHANNEL WAS DECLARED INOPERABLE. THE TECH SPEC DID NOT PROVIDE FOR BOTH CHANNELS TO BE INOPERABLE WITH PRESSURIZER PRESSURE ABOVE THE PRESSURIZER PRESSURE PERMISSIVE (P-11) SETPOINT OF 1930 PSIG. LIMITING CONDITION FOR OPERATION (LCO) 3.0.3. WAS ENTERED. A PRESSURE REDUCTION WAS INITIATED. AT 0401 PRESSURE WAS REDUCED BELOW P-11 AND LCO 3.0.3 WAS EXITED. THE CAUSE OF THIS EVENT WAS COMPONENT FAILURE. THE 458 CHANNEL HAD A DEFECTIVE WIRE ON THE INTERNAL PORTION OF THE TRANSMITTER'S ELECTRICAL PENETRATION PLUG REFERRED TO AS A "PIGTAIL". THE 455 CHANNEL HAD FAILED AT A SOLDERED CONNECTION INSIDE THE TRANSMITTER. THE PIGTAIL AND THE TRANSMITTER WERE REPLACED ON 458. THE REPLACEMENT MODEL CONTAINED A WELDED CONNECTION INSTEAD OF THE SOLDERED CONNECTION. THE 455 WAS REPLACED WITH A SIMILAR TRANSMITTER. THIS WAS DUE TO THE UNAVAILABILITY OF ADDITIONAL WELDED TYPE TRANSMITTERS. THE WELDED TYPE

TRANSMITTERS ARE BEING EVALUATED FOR INSTALLATION IN THE REMAINING PRESSURIZER PRESSURE CHANNELS. THERE HAVE BEEN NO PREVOUS SIMILAR OCCURRENCES.

[12] BRAIDWOOD 2 DOCKET 50-457 LER 90-009
MAIN STEAMLINE ISOLATION VALVE INOPERABLE DUE TO A FAILED FOUR WAY HYDRAULIC VALVE.
EVENT DATE: 060490 REPORT DATE: 070290 NSSS: WE TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 218810) AT 0830 ON 6/4/90, WITH UNIT 2 IN MODE 4, DURING A ROUTINE INSPECTION OF THE 2A MAIN STEAMLINE ISOLATION VALVE (MSIV) IT WAS IDENTIFIED THAT THE PRESSURES FOR THE ACTIVE AND STANDBY ACCUMULATORS WERE 3400 PSIG AND 4600 PSIG RESPECTIVELY. THIS WAS BELOW THE 4800 PSIG REQUIRED FOR OPERABILITY. THE MSIVS ARE NOT REQUIRED TO BE OPERABLE IN MODE 4. AT APPROXIMATELY 0910 IT WAS IDENTIFIED THAT THE CAUSE OF THE LOW ACCUMULATOR PRESSURE WAS THE "N" FOUR WAY HYDRAULIC VALVE. THE "N" VALVE HAD AN INTERNAL LEAK WHICH ALLOWED HYDRAULIC FLUID TO BE PORTED TO THE RESERVOIR. AN EVALUATION WAS CONDUCTED TO DETERMINE HOW THE VALVE WOULD HAVE PERFORMED IN ITS AS FOUND CONDITION. IT WAS CONCLUDED THAT THE VALVE WOULD HAVE BEEN INCAPABLE OF CLOSURE FROM THE ACTIVE ACCUMULATOR. THERE WAS ADEQUATE HYDRAULIC FLUID AND PRESSURE IN THE STANDBY ACCUMULATOR TO ENSURE 95% OF STROKE TRAVEL IN THE CLOSED DIRECTION BUT IT COULD NOT BE ASSURED THAT 100% CLOSURE WOULD BE ACHIEVED. ADDITIONALLY, WITH THE HYDRAULIC LEAK ON THE "N" VALVE THE AIR DRIVEN HYDRAULIC PUMP DISCHARGE WOULD HAVE BEEN DIRECTED BACK TO THE RESERVOIR INSTEAD OF ASSISTING WITH THE FINAL 5% OF VALVE CLOSURE. THE CAUSE OF THE EVENT WAS COMPONENT FAILURE. THE FOUR WAY VALVE AND THE HYDRAULIC OIL FILTER FOR THE 2A MSIV WERE REPLACED. SINCE REPLACEMENT, THE 2A MSIV HAS PERFORMED SATISFACTORILY. PREVIOUS CORRECTIVE ACTIONS ARE NOT APPLICABLE.

[13] BRAIDWOOD 2 DOCKET 50-457 LER 90-010
REACTOR TRIP DURING PLANT START UP FROM LOW STEAM GENERATOR WATER LEVEL DUE TO A MALFUNCTIONING BYPASS FEEDWATER REGULATING VALVE.
EVENT DATE: 060990 REPORT DATE: 070390 NSSS: WE TYPE: PWR
VENDOR: FISHER CONTROLS CO.

(NSIC 218811) A UNIT START UP WAS IN PROGRESS. FEEDWATER (FW) FLOW WAS BEING CONTROLLED IN AUTOMATIC BY THE BYPASS FW REGULATING VALVES (BFRV). STEAMLINE HEADER PRESSURE WAS BEING CONTROLLED IN AUTOMATIC BY THE STEAM DUMPS. AT 0030 ON 6/9/90 A REACTOR OPERATOR (RO), WHO WAS MONITORING THE FW PANEL, OBSERVED THAT INDICATED LEVEL ON THE 2B STEAM GENERATOR (SG) HAD DECREASED TO 35%. THIS WAS BELOW THE SET POINT OF 50%. THE RO PLACED THE CONTROLLER IN MANUAL AND INCREASED THE OUTPUT TO RAISE SG LEVEL. 2B SG LEVEL CONTINUED TO DECREASE FROM THE 'SHRINK' EFFECT OF THE COLD FW. THE SUPERVISOR (SRO) DIRECTED THE RO WHO WAS MONITORING REACTOR CONTROL PANEL, TO WITHDRAW CONTROL RODS TO INCREASE TEMPERATURE AND "SWELL" THE LEVEL. SG LEVEL INCREASED FROM AN INITIAL VALUE OF 20% TO 24%. AT 0039 THE EFFECTS OF THE INCREASED HEAT INPUT CAUSED STEAMLINE PRESSURE TO INCREASE WHICH CAUSED THE STEAM DUMP VALVES TO CYCLE. THIS CREATED A LEVEL PERTURBATION WHICH CAUSED THE LEVEL IN THE 2B SG TO DECREASE BELOW THE REACTOR TRIP SET POINT OF 17% AND A REACTOR TRIP OCCURRED. THE CAUSE OF THE EVENT WAS A MALFUNCTIONING BFRV WHICH WOULD STICK DURING OPERATION IN THE LOWER THIRD OF VALVE TRAVEL. THE VALVE PACKING WAS LOOSENED AND VALVE TRAVEL WAS SMOOTH AND ACCEPTABLE. OPERATOR TRAINING WILL BE PROVIDED. PREVIOUS CORRECTIVE ACTIONS ARE NOT APPLICABLE.

[14] BROWNS FERRY 2 DOCKET 50-260 LER 90-004
UNPLANNED ENGINEERED SAFETY FEATURE ACTUATION CAUSED BY A DESIGN OVERSIGHT.
EVENT DATE: 060190 REPORT DATE: 070290 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 1 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 218696) ON 6/1/90 AT 1326 HRS, DURING AN AIR TEST ON THE HIGH PRESSURE FIRE PROTECTION (HPFP) SYSTEM, AN UNPLANNED ENGINEERED SAFETY FEATURE ACTUATION OCCURRED. ACTUATION WAS A RESULT OF WATER FROM THE HPFP SYSTEM IN CABLE SPREADING ROOM "A" DRIPPING ONTO THE EMERGENCY CORE COOLING SYSTEM ANALOG TRIP

UNITS (ATU) IN THE UNIT 2 AUXILIARY INSTRUMENT ROOM IN THE ELEVATION BELOW. THE WATER DRIPPING ON THE ATU PANELS CAUSED CIRCUIT CARDS IN THE ATUS TO GENERATE SPURIOUS SIGNALS WHICH RESULTED IN UNNECESSARY START OF EMERGENCY EQUIPMENT COOLING WATER PUMP A1 AND EMERGENCY DIESEL GENERATORS "A", "C", AND "D". THE ROOT CAUSE OF THE EVENT IS DESIGN OVERSIGHT. THE DESIGN OF THE SEISMIC GAP IS SUCH THAT WATER FROM THE CABLE SPREADING ROOM IN THE ELEVATION ABOVE THE UNIT 2 AUXILIARY INSTRUMENT ROOM LEAKED DIRECTLY ONTO THE ATU CABINETS. THIS EFFECT HAS THE POTENTIAL TO CAUSE THE ATU CARDS TO SHORT AND INITIATE ERRONEOUS SIGNALS TO VARIOUS EMERGENCY CORE COOLING SYSTEMS (ECCS) EQUIPMENT. CORRECTIVE ACTION FOR THE EVENT INCLUDED TERMINATING THE AIR TEST ON THE HPFP SYSTEM, WIPING UP THE WATER SPILL, REMOVING THE AFFECTED ATU CABINETS FROM SERVICE FOR EVALUATION AND COVERING THE CABINETS WITH PLASTIC SHEETING. FURTHER CORRECTIVE ACTION TO PREVENT REOCCURRENCE WILL INCLUDE: PERFORM A TECHNICAL ASSESSMENT OF THE SAFETY ASPECTS OF WATER INTRUSION INTO THE ATU PANELS, AND ASSESS THE DESIGN OF THE SEISMIC GAP SLIP JOINT.

[15] BROWNS FERRY 3 DOCKET 50-296 LER 89-005 REV 01
UPDATE ON UNPLANNED ENGINEERED SAFETY FEATURE ACTUATIONS DUE TO LOSS OF POWER TO RADIATION MONITORS.
EVENT DATE: 102589 REPORT DATE: 070290 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 1 (BWR)
BROWNS FERRY 2 (BWR)

(NSIC 218683) AT APPROXIMATELY 0202 HOURS ON OCTOBER 25, 1989, TWO RADIATION MONITORS FOR THE UNIT 3 REACTOR AND REFUELING ZONE EXPERIENCED A LOSS OF POWER AND TRIPPED UPSCALE. AS DESIGNED, AN UPSCALE TRIP OF A SINGLE RADIATION MONITOR RESULTS IN THE AUTOMATIC ACTUATION OF THE LOGIC FOR SEVERAL ENGINEERED SAFETY FEATURES (ESFS). THE ESFS AFFECTED BY THIS EVENT WERE CONTROL ROOM EMERGENCY VENTILATION, STANDBY GAS TREATMENT, REFUEL ZONE VENTILATION ISOLATION, UNIT 3 REACTOR ZONE VENTILATION ISOLATION, AND UNIT 3 GROUP 6 PRIMARY CONTAINMENT ISOLATION. THE CAUSE OF THE EVENT WAS ATTRIBUTED TO THE FAILURE OF FUSE F1 ON THE +24 VOLT REGULATOR BOARD IN THE RADIATION MONITOR POWER SUPPLY. THIS CAUSED A LOSS OF POWER TO THE RADIATION MONITORS. THE POWER SUPPLY WAS REPLACED AND THE INITIATION LOGIC FOR THE ESF SYSTEMS WAS SUBSEQUENTLY RESET. AT THE END OF UNIT 2 CYCLE 6 OUTAGE THE REACTOR AND REFUEL VENTILATION MONITORS WILL BE REPLACED WITH MONITORS POWERED WITH REDUNDANT POWER SUPPLIES HAVING A REVISED LOGIC (ONE OUT OF TWO TAKEN TWICE). DURING THIS EVENT, UNITS 1 AND 3 WERE DEFUELED AND UNIT 2 WAS IN COLD SHUTDOWN WITH FUEL IN THE REACTOR VESSEL AND THE HEAD REMOVED. NO FUEL HANDLING OR OPERATIONS OVER SPENT FUEL WERE PERFORMED DURING THE EVENT.

[16] BRUNSWICK 1 DOCKET 50-325 LER 89-024 REV 01
UPDATE ON FAILURE TO TEST SEVENTEEN PRIMARY CONTAINMENT ISOLATION VALVES PER TECH SPEC 4.6.1.1.A DUE TO FAILURE TO RECOGNIZE TESTING APPLICABILITY.
EVENT DATE: 112789 REPORT DATE: 063090 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)

(NSIC 218782) ON NOVEMBER 27, 1989, IT WAS DETERMINED AS REPORTABLE THAT SEVENTEEN PRIMARY CONTAINMENT ISOLATION (PCIS) VALVES WERE NOT BEING TESTED IN ACCORDANCE WITH TECH SPEC (T/S) SURVEILLANCE REQUIREMENT 4.6.1.1.A. SIXTEEN OF THE SEVENTEEN VALVES WERE INSTALLED BY PLANT MODIFICATIONS (PMS) AND THE SEVENTEENTH WAS REMOVED AS A TEMPORARY REPAIR AND THEN REINSTALLED. THE ROOT CAUSE OF THE FAILURE TO IDENTIFY APPROPRIATE TESTING REQUIREMENTS FOR THE VALVES INSTALLED BY PMS WAS DETERMINED TO BE A FAILURE TO IDENTIFY APPLICABLE DESIGN CRITERIA AND REGULATORY COMMITMENTS IN THE DESIGN BASIS DOCUMENTATION. THE ROOT CAUSE FOR THE FAILURE TO INCORPORATE THE REINSTALLED VALVE BACK INTO THE APPROPRIATE TEST CANNOT BE DETERMINED. A REVIEW OF EACH PRIMARY CONTAINMENT PENETRATION WAS PERFORMED TO ASSESS EACH PCIS VALVE BASED ON CURRENT APPROVED METHODOLOGY AND APPROPRIATE DOCUMENT REVISIONS WERE INITIATED. THE SEVENTEEN PCIS VALVES WERE INCORPORATED INTO THE APPROPRIATE TEST. THIS EVENT HAD MINIMAL SAFETY SIGNIFICANCE.

[17] BRUNSWICK 1 DOCKET 50-325 LER 90-007
FAILURE OF THE CONTROL BUILDING EMERGENCY AIR FILTRATION SYSTEM TO MEET HABITABILITY DESIGN BASIS FOR TOXIC GASES (CHLORINE).
EVENT DATE: 051190 REPORT DATE: 060890 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)

(NSIC 218572) ON 5/11/90 AT 1610, FIELD RESPONSE TESTING OF THE CONTROL BUILDING EMERGENCY AIR FILTRATION (CBEAF) SYSTEM INLET DAMPER 2L-D-CB IDENTIFIED THAT THE DAMPER WOULD FAIL TO CLOSE ON A LOSS OF POWER TO THE DAMPER'S SOLENOID VALVE. THE DAMPER WOULD FAIL IN THE AS-IS POSITION. THIS CONDITION MEANS THAT THE HABITABILITY DESIGN BASIS OF THE CONTROL ROOM ENVELOPE MAY NOT BE MET UNDER A DESIGN BASIS EVENT FOR TOXIC GAS (CHLORINE). IN ADDITION, ON 5/26/90 AT 1426, WHILE INVESTIGATING POSSIBLE CONTROL ROOM ENVELOPE LEAKAGE SOURCES DUE TO FAILURE OF THE CONTROL ROOM ENVELOPE TO MAINTAIN POSITIVE PRESSURE WHILE PERFORMING A PERIODIC SURVEILLANCE, DAMPER 2L-D-CB WAS FOUND TO BE APPROXIMATELY 30 DEGREES OPEN. INVESTIGATION INTO THE 5/26/90 EVENT IS CONTINUING TO DETERMINE ROOT CAUSE OF THE EVENT, AS WELL AS A FINAL EVENT ASSESSMENT. DAMPER 2L-D-CB HAS BEEN MODIFIED TO BE FAIL-SAFE ON LOSS OF POWER. INVESTIGATION IS CONTINUING TO DETERMINE IF ADDITIONAL CORRECTIVE ACTIONS ON THIS EVENT ARE NECESSARY. BOTH THE 5/11/90 AND THE 5/26/90 EVENTS WILL BE UPDATED IN A SUPPLEMENT TO THIS LER BY 8/10/90.

[18] BRUNSWICK 2 DOCKET 50-324 LER 88-001 REV 06
UPDATE ON MANUAL REACTOR SCRAM DUE TO DECREASING MAIN CONDENSER VACUUM AND FAILURE OF PRIMARY CONTAINMENT GROUP 2 VALVES TO CLOSE ON ISOLATION SIGNAL.
EVENT DATE: 010288 REPORT DATE: 061590 NSSS: GE TYPE: BWR
VENDOR: ASCO VALVES
GENERAL ELECTRIC CO.
GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 218493) WHILE PERFORMING A ROUTINE REACTOR SHUTDOWN IN PREPARATION FOR THE UNIT 2 1988 REFUELING/MAINTENANCE OUTAGE, A MANUAL REACTOR PROTECTION SYSTEM TRIP (SCRAM) WAS INITIATED AT 0017 HOURS ON 1/2/88, DUE TO A DECREASING CONDENSER VACUUM. REACTOR POWER WAS APPROX. 55% AND VACUUM HAD DECREASED TO APPROX. -22 INCHES MERCURY. DURING THE EXPECTED VESSEL LEVEL SHRINK FOLLOWING THE SCRAM, VESSEL LEVEL DECREASED TO APPROX. 153 INCHES, INITIATING PRIMARY CONTAINMENT ISOLATION VALVE GROUPS 2, 6, AND 8 AT LOW LEVEL 1 (> 162.5"). OPERATOR VERIFICATION OF THESE VALVE CLOSURES DETERMINED THAT THE GROUP 2 VALVES 2-G16-F003, -F004, -F019, AND -F020 FAILED TO CLOSE. THESE ARE THE INBOARD AND OUTBOARD ISOLATION VALVES FOR THE DRYWELL FLOOR DRAIN SUMP (F003, F004) AND THE DRYWELL EQUIPMENT DRAIN SUMP (F010, F020). THE REMAINING SAFETY SYSTEMS OPERATED AS DESIGNED DURING THIS EVENT. INVESTIGATION OF THE DECREASING VACUUM CONDITION DETERMINED IT RESULTED FROM NUMEROUS LEAKS ON THE MAIN TURBINE AND MAIN STEAM REHEAT INTERCONNECTING PIPING TO THE MAIN TURBINE, WHICH WERE REPAIRED DURING THE UNIT OUTAGE. TO DATE, THE CAUSE OF THE GROUP 2 PCIVS FAILURE TO CLOSE HAS NOT BEEN DETERMINED. BY 8/01/90 A SUPPLEMENT WILL BE ISSUED TO UPDATE THE ROOT CAUSE DETERMINATION OF THE FAILURE OF THE VALVES TO CLOSE.

[19] BRUNSWICK 2 DOCKET 50-324 LER 90-005
UNPLANNED CLOSURE OF HPCI ISOLATION VALVE DUE TO PERSONNEL ERROR DURING SURVEILLANCE TEST.
EVENT DATE: 051490 REPORT DATE: 061390 NSSS: GE TYPE: BWR

(NSIC 218523) AT 0840 ON MAY 14, 1990, AN UNPLANNED CLOSURE OF THE HPCI INBOARD STEAM ISOLATION VALVE, 2-E41-F002, OCCURRED DURING A MAINTENANCE SURVEILLANCE TEST (MSTS). THE EVENT WAS CAUSED BY INATTENTION TO DETAIL ON THE PART OF AN I&C TECHNICIAN WHO PLACED THE RCIC "TEST" SWITCH TO TEST INSTEAD OF THE HPCI "TEST" AS STATED IN THE MST. FACTORS THAT CONTRIBUTED TO THIS EVENT ARE SWITCH LAYOUT, COMMON ANNUNCIATORS FOR HPCI AND RCIC STEAM LEAK DETECTION TEST STATUS AND THE RECENT REVISION OF THE PROCEDURE TO UTILIZE THE TEST SWITCHES. THE UNIT 1 AND UNIT 2 MSTS WILL BE REVISED TO INCLUDE AN INDEPENDENT VERIFICATION OF THE SWITCH PLACEMENT. IN ADDITION, THE SAME OR SIMILAR ACTION WILL BE EVALUATED FOR USE ON THE NSSS/RPS PANELS. DURING THIS EVENT THE REACTOR REMAINED AT 100% POWER AND THE ADS, RHR/LPCI, CS AND RCIC SYSTEMS WERE OPERABLE IN STANDBY READINESS. THIS

EVENT HAD MINIMAL SAFETY SIGNIFICANCE AS THE PLANT IS ANALYZED FOR AND DESIGNED TO RESPOND TO A HPCI FAILURE.

[20] BRUNSWICK 2 DOCKET 50-324 LER 90-006
HYDRAULIC PERTURBATION OF REACTOR VESSEL LEVEL INSTRUMENTATION.
EVENT DATE: 060490 REPORT DATE: 070390 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 1 (BWR)

(NSIC 218791) UNIT 2 WAS IN COLD SHUTDOWN (MODE 4). AT 1603 ON 6-4-90, WHILE VENTING A REACTOR PRESSURE INSTRUMENT FOLLOWING MAINTENANCE, A HYDRAULIC PERTURBATION OF A SHARED REFERENCE LEG RESULTED IN ISOLATION OF THE REACTOR WATER CLEANUP SYSTEM, AN AUTOMATIC INITIATION OF THE STANDBY GAS TREATMENT SYSTEM, AND AN ISOLATION OF THE REACTOR BUILDING VENTILATION SYSTEM. THIS EVENT POSES NO SAFETY SIGNIFICANCE SINCE THE AFFECTED SYSTEMS FUNCTIONED AS REQUIRED AND THE UNIT WAS IN COLD SHUTDOWN. MAINTENANCE HAS GENERATED AN INTERIM POLICY WHICH PROVIDES FOR A DATA GATHERING PHASE TO DETERMINE LONG TERM CORRECTIVE ACTIONS OTHER PERTURBATIONS HAVE BEEN REPORTED IN LERS 1-90-006, 2-89-017, 1-87-017, AND 2-86-020.

[21] CALLAWAY 1 DOCKET 50-483 LER 90-006
TECHNICAL SPECIFICATION ACTION STATEMENT NOT ENTERED FOR INOPERABLE REMOTE MANUAL CONTAINMENT ISOLATION VALVES DUE TO HUMAN PERFORMANCE.
EVENT DATE: 060390 REPORT DATE: 062890 NSSS: WE TYPE: PWR

(NSIC 218745) ON 6-3-90, THE ESSENTIAL SERVICE WATER (ESW) TO CONTAINMENT COOLER ISOLATION VALVES WERE TAGGED WITH WORKER PROTECTION ASSURANCE IN THEIR SAFETY INJECTION SIGNAL (SIS) POSITION, OPEN. THESE VALVES ARE REMOTE MANUAL VALVES AS LISTED IN TECHNICAL SPECIFICATION (T/S) 3.6.3 TABLE 3.6-1. THE PLANT'S TECHNICAL SPECIFICATION INTERPRETATION (TSI) WRITTEN FOR T/S 3.6.3 STATES: "THOSE VALVES WHICH ARE LISTED AS REMOTE MANUAL IN TABLE 3.6-1 DO NOT RECEIVE A CONTAINMENT ISOLATION SIGNAL. THEREFORE, SIMPLY REMOVING POWER FROM THESE VALVES DOES NOT IN AND OF ITSELF MAKE THE VALVES INOPERABLE FROM A CONTAINMENT ISOLATION STANDPOINT." UTILITY LICENSED SCHEDULING AND OPERATIONS PERSONNEL APPLIED THIS TSI TO MEAN THAT THE T/S 3.6.3 ACTION STATEMENT DID NOT NEED TO BE ENTERED FOR THE WORK PERFORMED ON THE VALVES. SUBSEQUENT EVALUATION VERIFIED THAT THE VALVES SHOULD HAVE BEEN CONSIDERED MANUAL CONTAINMENT ISOLATION VALVES, REQUIRING THE VALVES TO BE CLOSED PER T/S 3.6.1.1. T/S 3.6.3 AND T/S 3.6.1.1 WERE VIOLATED AS THEIR ACTION STATEMENTS WERE NEVER ENTERED. THE PLANT WAS IN MODE 1 - POWER OPERATIONS AT 100 PERCENT REACTOR POWER AT THE TIME OF THE EVENT. THE ROOT CAUSE OF THIS EVENT WAS THE TSI FOR T/S 3.6.3 REMOTE MANUAL VALVES WAS INCORRECT. THE TSI WILL BE REVISED TO CLARIFY THE EFFECT OF REMOVING POWER TO REMOTE MANUAL VALVES.

[22] CALLAWAY 1 DOCKET 50-483 LER 90-007
TWO REACTOR TRIPS DUE TO FAILED INPUT BUFFER CARD AND FAULTY SLAVE CYCLER COUNTER CARD AND A MISSED SURVEILLANCE DUE TO COGNITIVE PERSONNEL ERROR.
EVENT DATE: 061190 REPORT DATE: 071090 NSSS: WE TYPE: PWR
VENDOR: CONSOLIDATED CONTROLS CORP.
WESTINGHOUSE ELEC CORP.-NUCLEAR ENERGY SYS

(NSIC 218823) ON 6/11/90, AT 1016 CDT, A REACTOR PROTECTION SYSTEM (RPS) REACTOR TRIP, A FEEDWATER ISOLATION (FWIS), AND AN AUXILIARY FEEDWATER ACTUATION OCCURRED FOLLOWING SIMULTANEOUS CLOSURE OF ALL FOUR MAIN STEAM ISOLATION VALVES (MSIVS). THE PLANT WAS IN MODE 1 - POWER OPERATIONS AT 100 PERCENT REACTOR POWER. THE MSIVS CLOSED DUE TO A FAILED INPUT BUFFER CARD IN THE MSIV MANUAL FAST CLOSE CIRCUITRY IN THE MAIN STEAM AND FEEDWATER ISOLATION SYSTEM CONTROL CABINET. THE FAILED INPUT BUFFER CARD WAS REPLACED. ON 6/12/90, AT 0501 CDT, WHILE SUBCRITICAL DURING REACTOR STARTUP, FOUR BANK 'B' ROD CONTROL CLUSTER ASSEMBLIES DROPPED. OPTIONS WERE EVALUATED AND AT 0545, THE REACTOR TRIP BREAKERS WERE OPENED, MANUALLY TRIPPING THE REACTOR. A FWIS WAS RECEIVED AS ANTICIPATED. THE PLANT WAS IN MODE 2 - STARTUP (SUBCRITICAL). A FAULTY ROD CONTROL SLAVE CYCLER COUNTER CARD FOR THE 1BD CONTROL ROD POWER CABINET WAS IDENTIFIED AND REPLACED. DURING STARTUP OF THE PLANT ON 6/12/90, OPERATIONS SURVEILLANCE PROCEDURE, "MAIN

TURBINE TRIP TESTS" (OSP-AC-00004), WAS NOT PERFORMED. THIS EVENT WAS DISCOVERED ON 6/18/90. THE PLANT WAS IN MODE 2 - STARTUP AT 15 PERCENT REACTOR POWER. THIS EVENT WAS CAUSED BY THE SHIFT TECHNICAL ADVISOR MISTAKENLY READING THE SURVEILLANCE SCHEDULE BOOK. OSP-AC-00004 WAS PERFORMED SATISFACTORILY ON 6/18/90.

[23] CALVERT CLIFFS 1 DOCKET 50-317 LER 87-002 REV 01
UPDATE ON MAIN STEAM PIPING FLAW CAUSED BY GRINDING ON THE EDGE OF THE PIPE.
EVENT DATE: 120386 REPORT DATE: 062090 NSSS: CE TYPE: PWR

(NSIC 218677) DURING REFUELING MODE 6, WHILE CONDUCTING A NON-MANDATORY INSPECTION, A SECTION OF THIN WALL WAS FOUND ON NO. 12 MAIN STEAM LINE. READINGS AS LOW AS 0.86 INCHES WERE FOUND. ANSI B31.1 ALLOWABLE MINIMUM WALL IS 0.95 INCHES. THE PROBABLE CAUSE WAS GRINDING ON THE EDGE OF THE PIPE TO ACHIEVE PROPER FIT-UP FOR WELDING DURING INITIAL CONSTRUCTION. THERE IS NO EVIDENCE OF EROSION, CORROSION, NOR OTHER ACTIVE MECHANISM OF DEGRADATION. THERE IS NO SIGNIFICANT SAFETY HAZARD. INTERIM CODE RELIEF APPROVAL WAS RECEIVED FROM THE NUCLEAR REGULATORY COMMISSION (NRC) ON DECEMBER 19, 1986. FINAL RESULTS OF THE FINITE ELEMENT ANALYSIS SUBSEQUENTLY SHOWED THAT RELIEF WAS NOT REQUIRED. THE FLAW AREA IS NOW CALLED OUT AS AN AREA REQUIRING SPECIAL MONITORING IN BALTIMORE GAS AND ELECTRIC COMPANY'S (BG&ES) EROSION/CORROSION PROGRAM. SINCE RECENT ANALYSIS HAS SHOWN THAT THIS CONDITION MEETS THE INTENT OF THE ASME CODE AND ANSI B31.1, THE PLANT WAS NOT OUTSIDE ITS DESIGN BASIS AND THIS ITEM HAS SUBSEQUENTLY BEEN DETERMINED NOT TO BE REPORTABLE.

[24] CALVERT CLIFFS 1 DOCKET 50-317 LER 89-019 REV 01
UPDATE ON UNIMPLEMENTED REQUIREMENT TO LOCK THE HPSI DISCHARGE HEADER ISOLATION VALVES SHUT IN OPERATION OUTSIDE THE LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM DESIGN BASIS.
EVENT DATE: 112889 REPORT DATE: 061390 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 2 (PWR)

(NSIC 218497) THE LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM IS USED TO PROVIDE OVERPRESSURE PROTECTION FOR THE REACTOR VESSEL AT LOW TEMPERATURES. IF THE LEVEL OF PROTECTION IS INADEQUATE, A RAPIDLY PROPAGATING FRACTURE OF THE REACTOR VESSEL COULD RESULT. IT HAS BEEN DETERMINED THAT FOR A PERIOD OF APPROXIMATELY 10 YEARS (1979-1989) NOT ALL OF THE LTOP CONTROLS ORIGINALLY ESTABLISHED WERE IN PLACE. THE SEVERITY OF THIS LAPSE WAS DESCRIBED TO THE NUCLEAR REGULATORY COMMISSION (NRC) STAFF DURING AN ENFORCEMENT CONFERENCE ON JANUARY 18, 1990. APPROPRIATE CORRECTIVE ACTIONS WERE DETERMINED AND HAVE BEEN IMPLEMENTED FOR UNIT 1. A SEVERITY LEVEL III VIOLATION WAS IMPOSED BY THE NRC (MARCH 6, 1990). AN LTOP SYSTEM FOR UNIT 1 HAS BEEN ESTABLISHED BASED ON CURRENT VESSEL EMBRITTLEMENT CONDITIONS. THE SYSTEM DESCRIPTION AND ASSOCIATED TECHNICAL SPECIFICATION CHANGES ARE UNDER REVIEW BY THE NRC STAFF. METHODS HAVE BEEN ESTABLISHED TO MAINTAIN THE LTOP SYSTEM COMMITMENTS. AN LTOP SYSTEM FOR UNIT 2 IS CURRENTLY UNDER REVIEW. THE PRESSURIZER MANWAY WILL NOT BE INSTALLED ON UNIT 2 WITH FUEL IN THE VESSEL, UNTIL APPLICABLE LTOP ISSUES ARE RESOLVED.

[25] CALVERT CLIFFS 1 DOCKET 50-317 LER 90-018
AXIAL SHAPE INDEX NOT CONTINUOUSLY MONITORED AS REQUIRED BY TECHNICAL SPECIFICATIONS DUE TO INCORRECT LABELING OF POWER RANGE DETECTOR CONNECTORS.
EVENT DATE: 052990 REPORT DATE: 062890 NSSS: CE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 218707) MAY 29, 1990, IT WAS DETERMINED THAT THE UPPER AND LOWER CABLE CONNECTIONS FOR THE UNIT-1 Y CHANNEL EXCORE POWER RANGE DETECTORS WERE REVERSED, MAKING THE POWER RATIO CALCULATOR (PRC) INOPERABLE. THE PRC IS REQUIRED FOR CONTINUOUS AXIAL SHAPE INDEX MONITORING IN MODE 1 WHEN THE PLANT COMPUTER IS NOT AVAILABLE. THERE HAVE BEEN INSTANCES WHEN THE PLANT WAS IN MODE 1 AND THE PLANT COMPUTER WAS NOT AVAILABLE. THIS CONDITION IS NOT IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS. THE ROOT CAUSE OF THE EVENT WAS THE IMPROPER LABELING OF THE DETECTOR CONNECTIONS. A CONTRIBUTING FACTOR WAS THAT THE POST-INSTALLATION TEST OF THE SYSTEM DID NOT DETECT THE PROBLEM. OTHER CAUSES OF THE EVENT ARE BEING INVESTIGATED. CORRECTIVE ACTIONS INCLUDE CORRECTION OF THE DETECTOR CONNECTION

LABELING AND CONNECTING SYSTEM LEADS TO THE CORRECT DETECTOR CONNECTIONS. POST-INSTALLATION TESTING OF THE EXCORE DETECTORS WILL BE REVIEWED AND REVISED. DATA WILL BE COLLECTED FOR UNIT-2 X AND Y CHANNEL EXCORE DETECTORS DURING THE NEXT UNIT-2 STARTUP TO CONFIRM THAT THE PROBLEM DOES NOT APPLY TO UNIT-2. OTHER CORRECTIVE ACTIONS WILL BE SPECIFIED AFTER THE INVESTIGATION IS COMPLETED.

[26] CATAWBA 1 DOCKET 50-413 LER 90-024
COMPLETION OF A TECHNICAL SPECIFICATION REQUIRED SHUTDOWN DUE TO AN INOPERABLE NUCLEAR SERVICE WATER PUMP.
EVENT DATE: 060390 REPORT DATE: 070590 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 218802) ON JUNE 3, 1990, AT 0430 HOURS, WITH UNITS 1 AND 2 IN MODE 1, POWER OPERATION, NUCLEAR SERVICE WATER (RN) PUMP 1B WAS INSERVICE AND FOUND TO BE EXPERIENCING VIBRATION HIGHER THAN NORMAL. AT 0448 HOURS, RN PUMP 2B WAS PLACED INSERVICE. AT 0518 HOURS, RN PUMP 1B WAS SECURED. ON JUNE 5 AT 0300 HOURS, RN PUMP 1B WAS REMOVED FROM SERVICE AND THE MOTOR WAS RUN UNCOUPLED FROM THE PUMP COINCIDENT WITH SCHEDULED PREVENTIVE MAINTENANCE. VIBRATION LEVELS IN THE MOTOR STILL INDICATED HIGHER THAN NORMAL. THE MOTOR WAS REMOVED AND DISASSEMBLED. ABNORMAL WEAR WAS OBSERVED TO HAVE OCCURRED IN THE LOWER BEARING. ATTEMPTS WERE MADE TO REBUILD A SPARE MOTOR AND INSTALL IT ON THE 1B PUMP BEFORE UNIT SHUTDOWN WAS REQUIRED. THESE ATTEMPTS WERE UNSUCCESSFUL AND AT 0922 HOURS, ON JUNE 9, UNIT 1 ENTERED MODE 5, COLD SHUTDOWN, MEETING ALL REQUIREMENTS OF THE TECHNICAL SPECIFICATIONS. THIS INCIDENT IS ATTRIBUTED TO MANUFACTURING DEFICIENCY IN THAT THE MOTORS (PUMP 1B AND SPARE) WERE SUPPLIED WITH A STATOR APPROXIMATELY 0.1 INCHES SHORTER THAN THE ORIGINAL MANUFACTURE SPECIFICATION. THE AFFECTED MOTORS ARE BEING REPAIRED BY THE MANUFACTURER. OTHER RN MOTORS HAVE BEEN EVALUATED TO DETERMINE IF THEY MAY BE AFFECTED BY SIMILAR MANUFACTURING DEFICIENCIES; ACCEPTABLE CONDITIONS WERE FOUND. THIS REPORT IS BEING SUBMITTED PURSUANT TO 10CFR 50.73, SECTION (A)(2)(I)(A).

[27] CLINTON 1 DOCKET 50-461 LER 90-002 REV 01
UPDATE ON INADEQUATE PRE-OPERATIONAL TEST, VALVE POSITIONING, CONSTRUCTION CLEANLINESS, AND CORROSION/SILTING RESULT IN LESS THAN DESIGN FLOW THROUGH HEAT EXCHANGERS.
EVENT DATE: 012490 REPORT DATE: 070290 NSSS: GE TYPE: BWR
VENDOR: AMERICAN AIR FILTER CO., INC.
GENERAL ELECTRIC CO.
SENTRY EQUIPMENT COMPANY
YUBA HEAT TRANSFER

(NSIC 218790) ON MARCH 6, 1990, THE SHIFT SUPERVISOR (SS) DETERMINED AS-FOUND FLOW RATES THROUGH SHUTDOWN SERVICE WATER (SX) PUMP ROOM COOLING COILS 1VH07SA AND 1VH07SB LESS THAN DESIGN WERE REPORTABLE CONDITIONS. TEST ENGINEERS IDENTIFIED THE FLOW RATE PROBLEM OF 1VH07SA ON 1/24/90, DURING HEAT EXCHANGER PERFORMANCE TESTING. AFTER THE SS WAS NOTIFIED OF THIS PROBLEM, HE DIRECTED THAT DESIGN FLOW BE RESTORED THROUGH THE COOLING COILS. INVESTIGATION HAS IDENTIFIED FLOW PROBLEMS IN 27 COMPONENTS USING DIVISION I, II & III OF SX FOR COOLING WATER AND IN 2 COMPONENTS USING CHILLED WATER FOR COOLING. THE CAUSES OF THIS EVENT WERE: INACCURATE PRESSURE DROP DATA AND INADEQUATE TESTING CRITERIA/METHODOLOGY IN PREOPERATIONAL TESTING; VALVES USED FOR THROTTLING PURPOSES WERE NOT IN POSITION AS SPECIFIED BY THE SX SYSTEM PROCEDURE; INADEQUATE CONSTRUCTION CLEANLINESS; AND CORROSION/SILTING. CORRECTIVE ACTIONS INCLUDE: ACHIEVING ACCEPTABLE FLOW RATES FOR COMPONENTS THAT USE THE SX SYSTEM AND CHILLED WATER FOR COOLING WATER; REVIEWING OTHER SAFETY-RELATED PREOPERATIONAL TESTS TO ENSURE THAT ACCEPTANCE CRITERIA IS CORRECT; ESTABLISHING A STANDING ORDER FOR POSITIONING THROTTLE VALVES; INVESTIGATING MARKING THROTTLE VALVES TO ENSURE REPEATABILITY OF POSITIONING; AND CLEANING PIPING/COMPONENTS.

[28] CLINTON 1 DOCKET 50-461 LER 90-011
MISPOSITIONING OF THROTTLE VALVES DUE TO INADEQUATE PROCEDURES, COMMUNICATIONS, AND POST-MAINTENANCE TESTING RESULTS IN INOPERABILITY OF THE DIVISION I AND II DIESEL GENERATORS.

EVENT DATE: 051490 REPORT DATE: 061490 NSSS: GE TYPE: BWR
VENDOR: MORRISON-KNUDSON COMPANY, INC.
POSI-SEAL

(NSIC 218544) ON MAY 11, 1990, WITH THE PLANT IN MODE 4 (COLD SHUTDOWN), FOLLOWING COMPLETION OF REPAIRS TO THE DIVISION I AND II DIESEL GENERATORS (DG1A AND 1B), OPERATORS RESTORED SHUTDOWN SERVICE WATER (SX) FLOW TO THE DGS BY THROTTLING THE SX INLET (BUTTERFLY) VALVES TO THE DG HEAT EXCHANGERS (HXS) IN ACCORDANCE WITH CAUTION TAGS ON THE VALVES. THE POSITIONS SPECIFIED ON THE TAGS WERE THOSE ESTABLISHED BY PLANT STAFF-TECHNICAL PERSONNEL DURING SX FLOW BALANCES. ON MAY 14, 1990, THE PLANT ENTERED MODE 2 (STARTUP). ON MAY 15, 1990, DURING ROUTINE TESTING, DG1A TRIPPED ON HIGH COOLANT TEMPERATURE. INVESTIGATION REVEALED THAT THE SX VALVES WERE NOT OPEN ENOUGH TO PROVIDE ADEQUATE COOLING FLOW TO THE HXS. THE VALVES WERE NOT SUFFICIENTLY OPEN BECAUSE THE REFERENCE USED BY THE OPERATORS IN REPOSITIONING THE VALVES WAS NOT THE SAME AS USED BY TECHNICAL DURING THE FLOW BALANCES. SINCE THE VALVES WERE NOT SUFFICIENTLY OPEN, DG1A AND 1B WERE DECLARED INOPERABLE. THE CAUSE OF THE EVENT IS ATTRIBUTED TO INADEQUATE PROCEDURES, COMMUNICATIONS, AND POST-MAINTENANCE TESTING. A CONTRIBUTING FACTOR WAS THE POOR THROTTLING CHARACTERISTIC OF THE BUTTERFLY VALVES. CORRECTIVE ACTIONS INCLUDE REVISING PROCEDURES AND ISSUING AN OPERATIONS STANDING ORDER TO ESTABLISH GUIDELINES AND CONSISTENCY IN THROTTLE VALVE POSITIONING.

[29] CLINTON 1 DOCKET 50-461 LER 90-012
NORMAL END-OF-LIFE FAILURE OF FEEDWATER (FW) FLOW CHANNEL POWER CONVERTER RESULTS IN SENSED LOW FW FLOW, RECIRCULATION PUMP DOWN-SHIFT AND MANUAL SCRAM.
EVENT DATE: 051790 REPORT DATE: 061890 NSSS: GE TYPE: BWR
VENDOR: AGASTAT RELAY CO.

(NSIC 218744) ON MAY 17, 1990, WITH THE PLANT IN POWER OPERATION AT 43 PERCENT REACTOR POWER, AND BOTH REACTOR RECIRCULATION (RR) PUMPS IN FAST SPEED, REACTOR FEEDWATER SYSTEM (FW) FLOW CHANNEL "B" FAILED AND CAUSED INSTRUMENTATION TO INCORRECTLY SENSE TOTAL FW FLOW AS LOW. AS A RESULT OF THIS, THE LOGIC INITIATED AN AUTOMATIC TRANSFER OF THE "A" AND "B" RR PUMPS TO SLOW SPEED. IN RESPONSE TO THE RR PUMP TRANSFERS, THE CONTROL ROOM OPERATOR INITIATED A MANUAL REACTOR SCRAM IN ACCORDANCE WITH THE OFF-NORMAL PROCEDURE. ADDITIONALLY, GROUPS 2, 3 AND 20 CONTAINMENT ISOLATION VALVES ACTUATED AS A REACTOR VESSEL WATER LOW-LEVEL TRIP OCCURRED. TROUBLESHOOTING DETERMINED THAT A POWER CONVERTER FOR THE FW CONTROL SYSTEM FAILED DUE TO NORMAL END OF LIFE AND CAUSED THE "B" FW FLOW CHANNEL TO FAIL. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO THE NORMAL END-OF-LIFE FAILURE OF THE POWER CONVERTER. CORRECTIVE ACTIONS INCLUDED: REPLACING THE SPECIFIC POWER CONVERTER AND REPLACING ANOTHER POWER CONVERTER OF THE SAME MODEL; INITIATING A PREVENTIVE MAINTENANCE (PM) TASK TO PERIODICALLY REPLACE THE POWER CONVERTER AND ANOTHER CONVERTER OF THE SAME MODEL; AND DETERMINING IF OTHER FW CONTROL SYSTEM POWER SUPPLIES/CONVERTERS MAY NEED PM TASKS FOR PERIODIC REPLACEMENT.

[30] COMANCHE 1 DOCKET 50-445 LER 90-011
CONTAINMENT ISOLATION ROOT VALVE LEFT IN THE INCORRECT POSITION AS A RESULT OF A PROCEDURAL DEFICIENCY.
EVENT DATE: 050490 REPORT DATE: 060490 NSSS: WE TYPE: PWR

(NSIC 218560) ON 5/4/90, DURING THE MONTHLY PERFORMANCE OF THE CONTAINMENT ISOLATION VALVE POSITION VERIFICATION, IT WAS IDENTIFIED THAT A PRESSURE INSTRUMENT ROOT ISOLATION VALVE, 1SI-8961, WAS OPEN WHEN THE PROCEDURE INDICATED IT SHOULD BE LOCKED CLOSED. UPON INVESTIGATION, IT WAS DETERMINED THAT A DESIGN CHANGE NOTICE (DCN) HAD BEEN APPROVED AND ISSUED TO ADDRESS AN NRC CONCERN REGARDING THE POSITION OF VALVE 1SI-8961; HOWEVER, THE FIELD WORK TO CHANGE THE POSITION OF THE VALVE (FROM OPEN TO LOCKED CLOSED) HAD NEVER BEEN PERFORMED. UPON DISCOVERY OF THIS CONDITION, THE VALVE WAS IMMEDIATELY SHUT. THE VALVE HAD BEEN OPEN, IN VIOLATION OF TECH SPEC 3.6.3, SINCE 3/12/90 WHEN UNIT 1 HAD INITIALLY ENTERED MODE 4. THE CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE A LACK OF CLEAR INSTRUCTIONS CONCERNING DOCUMENT CHANGE ONLY DCNS AND THE DEFINITION OF "PHYSICAL WORK". A DOCUMENT CHANGE ONLY DCN SHOULD NOT RESULT IN PHYSICAL CHANGES TO THE PLANT (I.E., VALVE POSITION CHANGE). BASED ON THE INTENT OF THE DOCUMENT CHANGE ONLY DCN PROCESS, NO CLOSURE MECHANISM WAS ESTABLISHED TO

ENSURE THAT PHYSICAL CHANGES TO THE PLANT (RESULTING FROM DOCUMENT CHANGE ONLY DCNS) WERE IMPLEMENTED. SEVERAL REVIEWS WERE CONDUCTED WHICH VERIFIED THAT THIS PROBLEM IS VERY LIMITED IN SCOPE, AND THE PROBLEM WILL BE ADDRESSED BY TRAINING AND A REVISION TO THE DESIGN MODIFICATION PROCESS PROCEDURE.

[31] COMANCHE 1 DOCKET 50-445 LER 90-015
MISSED CHEMISTRY SAMPLE SPECIAL CONDITION SURVEILLANCE DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 052090 REPORT DATE: 061990 NSSS: WE TYPE: PWR

(NSIC 218557) ON 5/20/90 AT 0338, TESTING WAS INITIATED WHICH INVOLVED MEASURING THE CHANGE IN NUCLEAR POWER AS THE VALVES IN EACH STEAM DUMP BANK ARE STROKED FROM A FULL CLOSED POSITION TO A FULL OPEN POSITION. THE TEST PROCEDURE STATES THAT OPENING EACH STEAM DUMP BANK SHOULD RESULT IN A NUCLEAR POWER INCREASE OF APPROXIMATELY 10%. THE TEST WAS COMPLETED AT 0915 ON 5/21. TECH SPEC (TS) 4.4.7, "REACTOR COOLANT SYSTEM SPECIFIC ACTIVITY", TABLE 4.4-1 REQUIRES THAT A SAMPLE OF REACTOR COOLANT FOR ISOTOPIC ANALYSIS IS TO BE TAKEN BETWEEN TWO AND SIX HOURS FOLLOWING A CHANGE EXCEEDING 15% OF RATED THERMAL POWER (RTP) WITHIN A ONE HOUR PERIOD. CHANGES IN EXCESS OF 15% OF RTP WITHIN ONE HOUR PERIOD OCCURRED DUE TO THE TESTING ON 8 OCCASIONS. TS 4.4.7, TABLE 4.4-1 SURVEILLANCE REQUIREMENTS WERE NOT MET WITHIN THE REQUIRED TIME FRAME ON 4 OF THE OCCASIONS. THE MISSED SURVEILLANCES WERE DUE TO A PROCEDURAL DEFICIENCY. CORRECTIVE ACTIONS INCLUDE REVISING THE TEST PROCEDURE TO PROVIDE THE APPROPRIATE CAUTIONS REGARDING THE REQUIRED SAMPLE.

[32] COMANCHE 1 DOCKET 50-445 LER 90-016
THREE OF FOUR STEAM GENERATOR ATMOSPHERIC RELIEF VALVES INOPERABLE DUE TO INSUFFICIENT STROKE LENGTH SETTINGS.
EVENT DATE: 052390 REPORT DATE: 062290 NSSS: WE TYPE: PWR
VENDOR: FISHER CONTROLS CO.

(NSIC 218773) ON 5/21/90, COMMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 WAS CONDUCTING CAPACITY TESTING TO VERIFY THAT THE STEAM GENERATOR ATMOSPHERIC RELIEF VALVES (ARVS) STROKED FULLY, USING FEEDWATER FLOW INCREASE AS A QUALITATIVE INDICATOR OF VALVE STROKE. FOLLOWING THE TEST, ENGINEERING DETERMINED THAT VALVES 1-PV-2325, 1-PV-2326 AND 1-PV-2327 DID NOT PROVIDE THE MINIMUM REQUIRED STEAM FLOW CAPACITY TO SUPPORT THE DESIGN BASIS ACCIDENT ANALYSIS. AS A RESULT, THE THREE ARVS WERE DECLARED INOPERABLE RESULTING IN UNIT 1 ENTRY INTO TECH SPEC LIMITING CONDITION FOR OPERATION (LCO) 3.0.3. THE VALVES WERE SUBSEQUENTLY CALIBRATED AND DECLARED OPERABLE ALLOWING EXIT FROM LCO 3.0.3. THE EVENT RESULTED FROM TWO CAUSES: 1) THE PNEUMATIC CONTROLS FOR THREE OF THE FOUR ARVS HAD DRIFTED OUT OF CALIBRATION, AND 2) THE SPECIFIED STROKE LENGTH FOR TWO OF THESE THREE ARVS WAS REDUCED DUE TO AN INADEQUATE REVIEW AND APPROVAL PROCESS FOR INSTRUMENTATION & CONTROL (I&C) DATA CALIBRATION SHEETS. CORRECTIVE ACTIONS INCLUDE SETTING THE VALVES TO THE APPROPRIATE CONFIGURATION AND CONDUCTING AN EVALUATION TO ESTABLISH THE FREQUENCY FOR VERIFYING ARV STROKE LENGTH. ALSO, THE CALIBRATION DATA SHEETS WERE CORRECTED AND THE I&C PROGRAM FOR THE REVISION OF CALIBRATION DATA SHEETS HAD PREVIOUSLY BEEN REVISED TO REQUIRE THE SUPERVISOR, I&C ENGINEERING TO REVIEW ANY CHANGES TO DESIGN REQUIREMENTS.

[33] COMANCHE 1 DOCKET 50-445 LER 90-017
REACTOR TRIP DUE TO FEEDWATER CONTROL VALVE SOLENOID FAILURE.
EVENT DATE: 052790 REPORT DATE: 062690 NSSS: WE TYPE: PWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)

(NSIC 218774) ON 5/27/90, AT 0126 WHILE PERFORMING STEAM GENERATOR ATMOSPHERIC RELIEF VALVE (ARV) CAPACITY TESTING, A MAIN FEEDWATER FLOW CONTROL VALVE (FCV) FAILED CLOSED. THIS RESULTED IN REDUCED FEEDWATER FLOW AND DECREASING STEAM GENERATOR (SG) NO. 3 WATER LEVEL. THE OPERATOR CLOSED THE ARV, WHICH WAS OPEN FOR TEST PURPOSES, AND STARTED TO MANUALLY RAMP DOWN THE MAIN TURBINE TO REDUCE REACTOR POWER. THE OPERATOR THEN OPENED BY THE BYPASS FLOW CONTROL VALVE AROUND THE FAILED CLOSED FCV, BUT THE SG WATER LEVEL CONTINUED TO DECREASE. AT 0128, WHEN NO. 3 SG WATER LEVEL REACHED APPROXIMATELY 30% (AUTOMATIC REACTOR TRIP IS AT

28% SG WATER LEVEL), THE OPERATOR MANUALLY TRIPPED THE REACTOR. ALL OTHER PLANT SYSTEMS OPERATED PROPERLY. THE CAUSE OF THE EVENT WAS THE FAILURE OF A SOLENOID VALVE COIL, ASSOCIATED WITH NO. 3 SG FCV, DUE TO RAIN WATER INTRUSION (FCV'S ARE LOCATED OUTSIDE). A TEMPORARILY REMOVED COVER ALLOWED WATER TO ENTER A JUNCTION BOX THEN DRAIN VIA CONDUIT TO THE SOLENOID COIL HOUSING. CORRECTIVE ACTION INCLUDED THE REPLACEMENT OF THE FAILED SOLENOID COIL AND INSPECTION OF THE OTHER SOLENOIDS FOR WATER/MOISTURE INTRUSION. AN EVALUATION WILL DETERMINE IF ADDITIONAL CRITICAL COMPONENTS EXIST IN A SIMILAR CONFIGURATION. GUIDANCE FOR THE CONDUCT OF OUTDOOR MAINTENANCE ACTIVITIES WILL BE ADDRESSED PROGRAMMATICALLY.

[34] COMANCHE 1 DOCKET 50-445 LER 90-018
INADVERTENT AUTOMATIC START OF AUXILIARY FEEDWATER PUMP DUE TO PERSONNEL ERROR.
EVENT DATE: 061390 REPORT DATE: 070690 NSSS: WE TYPE: PWR

(NSIC 218807) ON 6/13/90, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 WAS IN MODE 5; TWO STEAM GENERATORS HAD BEEN DRAINED TO ACCOMMODATE CHEMISTRY CONTROL ACTIVITIES. DURING PERFORMANCE OF TRAIN "A" SAFEGUARDS SLAVE RELAY TESTING ACTIVITIES, OPERATIONS PERSONNEL DISCOVERED THAT THE SOLID STATE PROTECTION SYSTEM WAS NOT ALIGNED TO SUPPORT TEST PERFORMANCE. WHEN THE OPERATOR ATTEMPTED TO ESTABLISH THE REQUIRED ALIGNMENT, THE TRAIN "A" MOTOR DRIVEN AUXILIARY FEEDWATER PUMP STARTED AS A RESULT OF LO LO LEVEL IN THE DRAINED STEAM GENERATORS. CAUSES OF THE EVENT ARE PERSONNEL ERROR AND PROCEDURAL INADEQUACIES. CORRECTIVE ACTIONS INCLUDE EVENT REVIEW BY OPERATIONS PERSONNEL AND PROCEDURAL ENHANCEMENTS.

[35] CONNECTICUT YANKEE DOCKET 50-213 LER 90-004
FEEDWATER REGULATOR BYPASS CHECK VALVES FAILED SURVEILLANCE TEST.
EVENT DATE: 031690 REPORT DATE: 062990 NSSS: WE TYPE: PWR
VENDOR: POWELL, WILLIAM COMPANY, THE

(NSIC 218686) ON DECEMBER 7, 1989, AT 1500 HOURS WITH THE PLANT IN MODE 6 (REFUELING), THE FEEDWATER (FW) REGULATOR BYPASS LINE CHECK VALVES FW-CV-135-1,2,3,4 FAILED THE AS-FOUND REFUELING OUTAGE SURVEILLANCE LEAK TEST ACCEPTANCE CRITERIA OF 0.5 GPM. THE CAUSE IS ATTRIBUTED TO VALVE CHATTER CAUSED BY LEAKAGE OF DOWNSTREAM FW FLOW CONTROL BYPASS VALVES FW-HICV-1301-1,2,3,4. SHORT TERM CORRECTIVE ACTION WAS TO REPAIR THE CHECK VALVES AND THE FLOW CONTROL VALVES. THE VALVES WERE RETESTED SATISFACTORILY FOLLOWING REPAIRS. LONG TERM CORRECTIVE ACTION INCLUDES CONDUCTING A DESIGN REVIEW OF THE CHECK VALVES AND PERFORMING A SEAT LEAKAGE TEST DURING THE NEXT COLD SHUTDOWN. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(V)(D) SINCE THIS CONDITION ALONE COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF A SYSTEM NEEDED TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT.

[36] CONNECTICUT YANKEE DOCKET 50-213 LER 90-005
SPURIOUS ACTUATION OF THE CONTAINMENT ISOLATION ACTUATION SYSTEM.
EVENT DATE: 060690 REPORT DATE: 070390 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 218687) ON JUNE 6, 1990, AT 2350, WITH THE PLANT IN MODE 6 AND THE CORE OFFLOADED TO THE SPENT FUEL POOL, THE CONTAINMENT ISOLATION ACTUATION SYSTEM (CIAS) WAS SPURIOUSLY ACTUATED. THE PLANT RESPONDED AS EXPECTED TO THE ACTUATION. THE CIAS SYSTEM WAS OUT OF SERVICE FOR THE UPGRADE OF THE REACTOR PROTECTION SYSTEM (RPS) AT THE TIME OF THE EVENT. DUE TO THE PLANT CONFIGURATION AT THE TIME OF THE EVENT, THE ACTUATION HAD NO DELETERIOUS AFFECTS. THE ROOT CAUSE OF THE EVENT WAS FOUND TO BE IMPROPER DESIGN DOCUMENT CONTROL DURING INITIAL CONSTRUCTION THAT LED TO A FAULTY SPARE CONTACT BEING USED IN THE RPS UPGRADE PROJECT. THE DESIGN DOCUMENTS WILL BE CORRECTED. CONTROLS ON THE DESIGN PROCESS IN PLACE SINCE THE EARLY 1980'S WILL PREVENT RECURRENCE. THIS EVENT IS BEING REPORTED UNDER 10CFR50.73(A)(2)(IV) SINCE THIS CONDITION RESULTED IN AUTOMATIC ACTUATION OF AN ENGINEERED SAFETY FEATURE.

[37] CONNECTICUT YANKEE DOCKET 50-213 LER 90-006
FAILURE TO ESTABLISH FIRE WATCH DUE TO PERSONNEL ERROR.
EVENT DATE: 061190 REPORT DATE: 070990 NSSS: WE TYPE: PWR

(NSIC 218784) ON JUNE 11, 1990, AT 0130 HOURS, WITH THE PLANT SHUT DOWN IN MODE 6 (REFUELING), AN AUXILIARY OPERATOR, PERFORMING ROUTINE ROUNDS IN THE SCREENWELL BUILDING, IDENTIFIED AN ALARM CONDITION ON LOCAL FIRE DETECTION PANEL FDS-1. ALSO, UPON FURTHER INVESTIGATION, HE FOUND THAT THE FIRE DETECTOR ABOVE THE DIESEL DRIVEN FIRE PUMP WAS IN ALARM. SINCE THERE WAS NO EVIDENCE OF A FIRE, IT APPEARED THAT THE DETECTOR WAS FAULTY AND THEREFORE INOPERABLE. IT WAS DETERMINED THAT A FIRE WATCH HAD NOT BEEN ESTABLISHED WITHIN ONE HOUR AS REQUIRED BY THE TECHNICAL SPECIFICATIONS. THE FIRE WATCH WAS NOT ESTABLISHED WITHIN THE PRESCRIBED TIME BECAUSE OF PERSONNEL ERROR. IMMEDIATE CORRECTIVE ACTION WAS TO ESTABLISH A CONTINUOUS FIRE WATCH IN ACCORDANCE WITH THE TECHNICAL SPECIFICATIONS. THE CAUSE OF THE ALARM WAS FOULING OF THE DETECTOR. THE DETECTOR WAS REPLACED AND THE SYSTEM WAS RETURNED TO SERVICE ON JUNE 14, 1990. SUBSEQUENT CORRECTIVE ACTION WAS TO COUNSEL ALL OPERATORS ON THE REQUIREMENT FOR MAINTAINING A HIGH LEVEL OF ALERTNESS RELATIVE TO FIRE DETECTION SYSTEM OPERABILITY REQUIREMENTS. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B) SINCE IT INVOLVES A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[38] COOK 1 DOCKET 50-315 LER 90-004
INADVERTENT OPERATION OF THE WRONG CONTROL SWITCH DUE TO PERSONNEL ERROR RESULTED IN OPENING OF THE LOWER ICE CONDENSER INLET DOORS.
EVENT DATE: 050890 REPORT DATE: 060790 NSSS: WE TYPE: PWR

(NSIC 218573) ON 5/8/90, AT 1555, A REACTOR OPERATOR INADVERTENTLY STARTED A CONTAINMENT RECIRCULATION FAN (CEQ) INSTEAD OF THE INTENDED HYDROGEN RECOMBINER DURING A SURVEILLANCE TEST. THE FAN OPERATION CAUSED SUFFICIENT DIFFERENTIAL PRESSURE ACROSS THE LOWER ICE CONDENSER INLET DOORS TO OPEN THEM. SINCE THE ICE CONDENSER IS ONE OF THE ENGINEERED SAFETY FEATURES (ESF), THE INADVERTENT STARTING OF THE CEQ FAN, WITH SUBSEQUENT OPENING OF THE INLET DOORS, WAS CONSIDERED AN ESF ACTUATION. VARIOUS VENTILATION ALIGNMENTS WERE ATTEMPTED TO RECLOSE THE INLET DOORS, BUT WITHOUT SUCCESS. THE ICE CONDENSER WAS DECLARED INOPERABLE AT 2129 WHEN IT WAS DETERMINED THAT THE TS 3.6.5.1 MAXIMUM ICE BED TEMPERATURE OF 27F HAD BEEN EXCEEDED. POWER WAS DECREASED TO 8% AT 1125 ON 5/9/90, TO ALLOW LOWER CONTAINMENT ENTRY AND MANUAL CLOSURE OF THE LOWER ICE CONDENSER INLET DOORS. THE INLET DOORS WERE CLOSED AND DECLARED OPERABLE AT 1248. THE ICE BED TEMPERATURES WERE DETERMINED TO BE WITHIN THE TS LIMITS AT 1325 ON 5/10/90. THEREFORE, THE TS 3.6.5.1 REQUIREMENT TO RESTORE THE ICE BED TO OPERABLE STATUS WITHIN 48 HOURS WAS MET. THE REQUIRED WORK PRACTICES TO PREVENT RECURRENCE OF A SIMILAR EVENT WERE REVIEWED WITH THE INVOLVED REACTOR OPERATOR. MANAGEMENT'S EXPECTATI[illegible] CONCERNING WORK PRACTICES TO PREVENT SIMILAR EVENTS WILL BE COMMUNICATED TO [illegible] PERSONNEL BY 8/30/90.

[39] COOK 2 DOCKET 50-316 LER 88-003 REV 08
UPDATE ON REPETITIVE VIOLATION OF ESF INSTRUMENTATION LIMITING CONDITIONS FOR OPERATION TOLERANCES DUE TO HIGHLY RESTRICTIVE ALLOWABLE VALUES.
EVENT DATE: 031188 REPORT DATE: 062290 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 1 (PWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 218679) THIS REVISION IS BEING SUBMITTED TO REFLECT AN UPDATE ON THE RESULTS OF THE INCREASED FREQUENCY (MONTHLY) CALIBRATION CHECKS PERFORMED TO DATE. ON MARCH 11, 1988 AN EQUIPMENT TREND INVESTIGATION WAS BEING PERFORMED ON 4KV BUS LOSS OF VOLTAGE RELAYS AND THE 4KV BUS DEGRADED VOLTAGE RELAYS (EIIS/EK-27). THE "AS FOUND" CONDITION OF THESE RELAYS DURING PAST CALIBRATION CHECKS HAS GENERALLY BEEN FOUND TO BE BEYOND THE TECHNICAL SPECIFICATION (T.S.) ALLOWABLE VALUES. EACH RELAY WAS ADJUSTED TO WITHIN ALLOWABLE VALUES AT THE TIME IT WAS DISCOVERED OUT OF SPECIFICATION. ALL RELAYS WERE FUNCTIONAL AND WOULD HAVE PERFORMED THE ESF FUNCTION, ALTHOUGH AT A SLIGHTLY DIFFERENT VOLTAGE THAN SPECIFIED IN T.S. AN ENGINEERING REVIEW HAS DETERMINED A PLUS OR MINUS 3 PERCENT TOLERANCE (AS OPPOSED TO THE CURRENT 0.5 PERCENT) TO BE ACCEPTABLE FOR THE LOSS OF VOLTAGE APPLICATION. THE DEGRADED VOLTAGE APPLICATION WILL ACCEPT A PLUS OR

MINUS 1.5 PERCENT TOLERANCE AND WILL REQUIRE INSTALLATION OF MORE ACCURATE UNDERVOLTAGE RELAYS (DESIGN CHANGE CURRENTLY UNDERWAY). A T.S. CHANGE REQUEST HAS BEEN SUBMITTED. AS STATED IN THE ORIGINAL LER, WE HAVE INCREASED THE CALIBRATION FREQUENCY FROM EVERY EIGHTEEN MONTHS TO MONTHLY.

[40] COOK 2 DOCKET 50-316 LER 90-003
TRAIN "B" LOWER CONTAINMENT PURGE AND EXHAUST ISOLATION INOPERABLE DUE TO RADIATION MONITOR FLUSH ISOLATION VALVE NOT FULLY CLOSED.
EVENT DATE: 021490 REPORT DATE: 062590 NSSS: WE TYPE: PWR
VENDOR: EBERLINE INSTRUMENT CORP.

(NSIC 218706) ON FEBRUARY 14, 1990 IT WAS IDENTIFIED THAT THE VALVE WHICH ALLOWS PURGING OF THE LOWER CONTAINMENT TRAIN "B" SPECIAL PARTICULATE, IODINE, NOBLE GAS MONITOR (SPING) (MON) SAMPLE CHAMBER WITH AMBIENT (AUXILIARY BUILDING) AIR WAS OPEN DURING OPERATION OF THE MONITOR. IT WAS DETERMINED ON MAY 29, 1990 THAT THIS CONDITION WOULD HAVE PREVENTED THE REQUIRED CHANNELS (CHANNEL 1 - BETA PARTICULATE, CHANNEL 5 - LOW RANGE NOBLE GAS) FROM FULFILLING THE REQUIREMENTS OF TECHNICAL SPECIFICATION TABLE 3.3-3, ESF ACTUATION SYSTEM (ESF) INSTRUMENTATION FOR THE CONTAINMENT PURGE SYSTEM, TRAIN "B". THE THIRD REQUIRED CHANNEL FOR TRAIN "B", VRS-2201 (AREA MONITOR) WAS OPERABLE AND CAPABLE OF PROVIDING THE ACTUATION. ERS 2301, 2305, WHICH PROVIDE TRAIN "A" CONTAINMENT PURGE SYSTEM TRIP SIGNALS WAS OPERABLE AT ALL TIMES. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO A FAULTY LOGIC CIRCUIT IN THE SPING STARTUP PROCESS FROM A NO POWER CONDITION. WHEN POWER IS APPLIED TO THE MICROPROCESSOR, THE PURGE VALVE BEGINS TO OPEN BEFORE A LOGIC RELAY INTERRUPTS THE POWER TO THAT PARTICULAR CIRCUIT. TO PREVENT RECURRENCE, THE INSTRUMENT OPERATION PROCEDURE WAS REVISED TO REQUIRE A "FLUSH" COMMAND WHENEVER RETURNING A SPING TO SERVICE. THE "FLUSH" COMMAND CYCLES THE PURGE VALVE FULLY CLOSED.

[41] CRYSTAL RIVER 3 DOCKET 50-302 LER 89-041 REV 01
UPDATE ON RELAY FAILURE IN THE CONTROL ROD TRANSFER LOGIC AND ORDER OF PROCEDURE STEP SEQUENCING ALLOWS PARTIAL SIMULTANEOUS WITHDRAWAL OF THE SAFETY GROUPS.
EVENT DATE: 120889 REPORT DATE: 070990 NSSS: BW TYPE: PWR
VENDOR: ELECTRONIC CONTROL

(NSIC 218781) ON DECEMBER 8, 1989, A PARTIAL, SIMULTANEOUS WITHDRAWAL OF TWO CONTROL ROD SAFETY GROUPS, OCCURRED AT CRYSTAL RIVER UNIT 3. THIS EVENT OCCURRED IN MODE 3 (HOT STANDBY) WHILE PERFORMING A PLANT SHUTDOWN. OPERATORS WERE IN THE PROCESS OF SELECTING SAFETY GROUP 1 CONTROL WHEN SAFETY GROUP 3 CONTROL ALSO TRANSFERRED TO THE AUXILIARY POWER SUPPLY. INITIAL ATTEMPTS TO REMOVE CONTROL POWER FROM GROUP 3 WERE UNSUCCESSFUL. OPERATORS INSERTED RODS APPROXIMATELY 2% FROM FULL WITHDRAWN AND THEN WITHDREW RODS BACK TO FULLY WITHDRAWN. ELECTRICIANS INSTALLED A TEMPORARY JUMPER ACROSS PORTIONS OF THE TRANSFER LOGIC CIRCUIT ALLOWING OPERATORS TO REMOVE GROUP 3 CONTROL POWER. THE SHUTDOWN THEN CONTINUED NORMALLY. ALTHOUGH THE EVENT COULD NOT BE DUPLICATED, IT WAS MOST LIKELY CAUSED BY A COMBINATION OF A FAILED RELAY IN THE CONTROL ROD TRANSFER LOGIC AND THE ORDER OF STEP SEQUENCING IN THE CONTROL ROD TRANSFER PROCEDURE. THE FAULTY RELAY HAS BEEN REPLACED AND THE PROCEDURE HAS BEEN REVISED. ADDITIONAL PROCEDURE GUIDANCE HAS BEEN PROVIDED TO THE OPERATORS CONCERNING MULTIPLE GROUP MOVEMENT. ALL OPERATING CREWS HAVE BEEN INFORMED OF THIS EVENT.

[42] CRYSTAL RIVER 3 DOCKET 50-302 LER 90-009
POST MODIFICATION TEST NOT PERFORMED DUE TO LOST DOCUMENTATION.
EVENT DATE: 052390 REPORT DATE: 062290 NSSS: BW TYPE: PWR

(NSIC 218705) ON JUNE 5, 1990, CRYSTAL RIVER UNIT 3 (CR-3) WAS IN MODE 6 (REFUELING). THE AMERICAN NUCLEAR INSURERS' (ANI) RESIDENT INSPECTOR, WHILE PERFORMING A ROUTINE AUDIT, DISCOVERED THAT PAPERWORK FOR A MODIFICATION TO REPLACE FEEDWATER VENT VALVE (FWV-163) WAS STILL OPEN TWELVE MONTHS AFTER WORK WAS COMPLETED. AN INVESTIGATION DETERMINED THAT A BLANKET WORK REQUEST (WR) TO PERFORM POST MAINTENANCE (PM) TESTING DURING PLANT STARTUP FROM A PREVIOUS OUTAGE HAD BEEN LOST, RESULTING IN A REQUIRED INITIAL SERVICE LEAK CHECK NOT BEING PERFORMED. THE LOSS OF THE BLANKET WR WAS THE RESULT OF A PERSONNEL ERROR. THE

WELD FOR THE REPLACEMENT VALVE HAS BEEN SUBJECTED TO OPERATIONAL CONDITIONS AND HAS PERFORMED SATISFACTORILY FOR NINE MONTHS OF POWER OPERATION. THE FEEDWATER TRAIN TO THE 'B' STEAM GENERATOR AND BOTH TRAINS OF EMERGENCY FEEDWATER HAVE REMAINED OPERABLE AND CAPABLE OF PERFORMING THEIR DESIGN AND SAFETY FUNCTIONS. THEREFORE, THERE WERE NO SAFETY CONSEQUENCES AS A RESULT OF THIS ERROR. A WR HAS BEEN INITIATED TO PERFORM THE INITIAL SERVICE LEAK CHECK ON FWV-163. A REVIEW TO DETERMINE IF ANY OTHER PM TESTING WAS MISSED IS BEING CONDUCTED AND ANY OMITTED TESTING WILL BE PERFORMED BY OCTOBER 30, 1990. CR-3 MANAGEMENT WILL TAKE APPROPRIATE ACTION TO INSURE THAT WRS NOT BEING WORKED ARE PROPERLY STORED.

[43] FERMI 2 DOCKET 50-341 LER 88-034 REV 01
UPDATE ON ISOLATION OF REACTOR WATER CLEANUP SYSTEM DUE TO SUSPECTED RELAY FAILURE.
EVENT DATE: 083188 REPORT DATE: 062290 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 218764) THE REACTOR WATER CLEANUP SYSTEM (RWCU) OUTBOARD ISOLATION VALVE, G3352F004, CLOSED AND CAUSED THE RWCU PUMPS TO TRIP. PLANT PERSONNEL DISCOVERED THAT RELAY A71B-K27 WAS DE-ENERGIZED BUT NO REASON FOR THIS COULD BE DETERMINED. THE ISOLATION SIGNAL WAS RESET, G3352F004 WAS OPENED AND THE RWCU SYSTEM WAS RETURNED TO SERVICE. AN INVESTIGATION BY ENGINEERING PERSONNEL WAS INITIATED. THE CLOSURE OF G3352F004 WAS CAUSED BY CONTACTS 3-4 ON RELAY A71B-K27 CLOSING UPON DE-ENERGIZATION OF THE RELAY. THE APPARENT CAUSE OF THE DE-ENERGIZATION WAS LOSS OF CONTINUITY TO THE RELAY DUE TO DEPOSITS BUILT UP ON THE SURFACE OF THE 1-2 CONTACTS. TWO RELAYS (GE MODEL CR120A) IN THE VALVE ISOLATION CIRCUIT WERE REPLACED AND THE PREVIOUSLY INSTALLED RELAYS WERE EVALUATED. AS A RESULT OF THESE EVALUATIONS, DISCUSSIONS ARE ON-GOING WITH THE MANUFACTURER TO DETERMINE A FUTURE COURSE OF ACTION.

[44] FITZPATRICK DOCKET 50-333 LER 89-026 REV 01
UPDATE ON REACTOR SAFETY RELIEF VALVE PILOT ASSEMBLY SETPOINT DRIFT.
EVENT DATE: 122689 REPORT DATE: 070590 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 218783) FOLLOWING A REACTOR SCRAM ON 11/5/89, TWO SAFETY RELIEF VALVES (SRVS)(AD) WERE REMOVED FOR TESTING. THE VALVE TEST FACILITY PROVIDED WRITTEN NOTICE, RECEIVED BY THE AUTHORITY ON 12/20/89, THAT BOTH VALVES ACTUATED AT SETPOINTS WHICH DEVIATED FROM THE DESIGN POINT BY MORE THAN THE +/- 1% ALLOWED BY TECH SPECS. SRV "E" LIFTED EARLY AT -1.4%. SRV "F" LIFTED AT +4.7%. DISASSEMBLY AND EXAMINATION OF THE PILOT MECHANISMS FOUND STEAM CUTS ON THE PILOT VALVE DISC SEAT AND BELLOWS FOR SRV "F". DISC TO SEAT BONDING IS BELIEVED TO BE THE CAUSE OF THE HIGH INITIAL LIFT PRESSURE FOR SRV "F". NO DETERIORATION WAS NOTED ON SRV "E" COMPONENTS AND NO CAUSE FOR EARLY LIFTING OF THAT VALVE WAS DETERMINED. EVALUATION OF REACTOR PRESSURE RELIEF CAPABILITY SHOWS OPERATION WOULD BE ACCEPTABLE WITH 2 OF 11 SRVS INOPERABLE AND A SETPOINT TOLERANCE OF +/-3%. CORRECTIVE ACTIONS INCLUDED REPLACING FAILED SRVS WITH RECERTIFIED VALVES, CONTINUED PARTICIPATION IN THE BWR OWNER'S GROUP TO RESOLVE SRV ISSUES, AND SUBMISSION TO THE NRC OF PROPOSED CHANGES TO TECH SPECS TO TAKE CREDIT FOR EXCESS INSTALLED SRV CAPACITY. LER-85-009, 85-013, 87-004, 88-004, AND 88-010 ARE SIMILAR EVENTS INVOLVING SRV SETPOINT DRIFT.

[45] FITZPATRICK DOCKET 50-333 LER 90-006 REV 01
UPDATE ON EXTREMITY EXPOSURE RESULTING FROM CONTAMINATION EVENT.
EVENT DATE: 031290 REPORT DATE: 062990 NSSS: GE TYPE: BWR

(NSIC 218713) ON 3/12/90 IT WAS DETERMINED THAT ON 3/8/90 AT APPROXIMATELY 1600 HOURS DURING 100% STEADY STATE POWER OPERATION, A RADIATION PROTECTION TECHNICIAN (RPT) RECEIVED A SIGNIFICANT EXTREMITY DOSE FROM NA-24 CONTAMINATION OF THE LEFT HAND THUMB. THE RPT WAS RADIOLOGICALLY MONITORING WORKERS PERFORMING A FEEDWATER FLOW MEASUREMENT TEST. FOLLOWING THE TRANSFER OF THE 240 MCI RADIOTRACER SOLUTION, THE RPT INAPPROPRIATELY PICKED UP THE HIGHLY CONTAMINATED VIAL CAP WITH THE RIGHT HAND AND PLACED IT ON THE EMPTY VIAL. CONTAMINATION OF THE LEFT HAND THUMB OCCURRED WHILE REMOVING THE SURGEON GLOVES UPON EXITING THE WORK AREA.

THUMB DOSE WAS CALCULATED AT A DEPTH OF 64 MG/CM THIS AND THE PREVIOUS CURRENT QUARTER EXTREMITY DOSE WAS 17.33 REM. THE QUARTERLY OCCUPATIONAL EXTREMITY DOSE LIMIT IN 10 CFR 20.101 IS 18.75 REM. A 49.03 REM DOSE WOULD BE CALCULATED AT A DEPTH OF 7 MG/CM USING GENERIC NRC RECOMMENDATIONS; HOWEVER, IT WOULD BE INAPPROPRIATE IN THIS INSTANCE DUE TO THE PRESENCE OF A HEAVY KERATINIZED (I.E., CALLUS) LAYER ON THE RPT'S THUMB. DEFICIENCIES INCLUDED FAILURE TO ADEQUATELY ASSESS A POTENTIAL HAZARD AND PERFORM ONLY THE ASSIGNED TASK. CORRECTIVE ACTIONS INCLUDE UPGRADING THE ALARA REVIEW PROCESS, REVIEW RPT TRAINING, AND COUNSELING. INSPECTION REPORT 50-333/90-12 IS RELATED. LER 50-333/87-002 DESCRIBES A PREVIOUS EXTREMITY EXPOSURE INCIDENT.

[46]   FITZPATRICK   DOCKET 50-333   LER 90-017
INSTRUMENT SETPOINT DRIFT EXCEEDS TECHNICAL SPECIFICATION LIMIT FOR HIGH PRESSURE COOLANT INJECTION TURBINE TRIP ON HIGH REACTOR WATER LEVEL.
EVENT DATE: 052690   REPORT DATE: 062190   NSSS: GE   TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 218746) THE REACTOR WAS SHUTDOWN FOR REFUELING ON 3/31/90. AN INSTRUMENT SURVEILLANCE PROCEDURE, "REACTOR LEVEL (ECCS) TRANSMITTER CALIBRATION AND FUNCTIONAL TEST (ATTS)" WAS PERFORMED ON 5/26/90. THE TEST FOUND THAT THE HIGH REACTOR WATER LEVEL TRIP POINT ON A TRANSMITTER FOR THE HIGH PRESSURE COOLANT INJECTION (HPCI) (BJ) TURBINE WAS 0.2 INCHES (0.3 PERCENT OF RANGE) ABOVE THE SETPOINT VALUE REQUIRED BY TECHNICAL SPECIFICATIONS. THE SAFETY FUNCTION PREVENTS POTENTIAL WATER CARRYOVER DAMAGE TO TURBINES BY SHUTDOWN OF HPCI TO PREVENT EXCESSIVELY HIGH REACTOR WATER LEVEL. THE TRANSMITTER WAS RECALIBRATED AND SURVEILLANCE WAS TEMPORARILY INCREASED FROM ONCE PER CYCLE TO ONCE EVERY SIX MONTHS UNTIL THREE CONSECUTIVE AS-FOUND READINGS ARE WITHIN SPECIFICATIONS UNLESS A CONSISTENT NON-CONSERVATIVE TREND IN THE DIRECTION OF THE SETPOINT DRIFT INDICATES A NEED FOR EARLIER REPLACEMENT. THE UPWARD SETPOINT DRIFT OF 0.2 INCHES WAS NOT SIGNIFICANT TO SAFETY SINCE IT WOULD NOT HAVE RESULTED IN A MEASURABLE INCREASE IN MOISTURE CARRYOVER SHOULD A HIGH LEVEL SITUATION HAVE OCCURRED. MORE THAN FIVE FEET REMAINED BETWEEN THE TRIP SETPOINT AND THE STEAM LINES. RELATED LERS:85-014, 85-015, AND 86-008.

[47]   FT. CALHOUN 1   DOCKET 50-285   LER 90-010 REV 01
UPDATE ON UNPLANNED ATTEMPTED START OF EMERGENCY DIESEL GENERATOR.
EVENT DATE: 032790   REPORT DATE: 070390   NSSS: CE   TYPE: PWR

(NSIC 218703) ON MARCH 27, 1990, AT 1428, AN UNPLANNED ATTEMPTED START OF EMERGENCY DIESEL GENERATOR D-1 OCCURRED WHILE THE PLANT WAS IN REFUELING SHUTDOWN. OPERATIONS PERSONNEL WERE PERFORMING AN ELECTRICAL CHECKLIST AS PART OF POST-MAINTENANCE TESTING PRIOR TO RETURNING THE DIESEL GENERATOR TO SERVICE. THE DIESEL ATTEMPTED TO START WHEN THE MODE SELECTOR SWITCH WAS TAKEN TO "EMERGENCY STANDBY", PER THE PROCEDURE, AND THE START CIRCUITRY SENSED A LOW VOLTAGE ON A ASSOCIATED NON-VITAL BUS. THE NON-VITAL BUS HAD BEEN TAGGED OUT AND DE-ENERGIZED FOR SCHEDULED MAINTENANCE. THE DIESEL FAILED TO START DUE TO LOW AIR PRESSURE IN THE DIESEL AIR START SYSTEM WHICH HAD NOT YET BEEN COMPLETELY PRESSURIZED. AN UNPLANNED START OF AN EMERGENCY DIESEL GENERATOR IS CONSIDERED AN ACTUATION OF A EMERGENCY SAFEGUARDS FEATURE. THE UNSUCCESSFUL START IN THIS EVENT WAS CONSIDERED TO MEET THE REPORTABILITY CRITERIA. A FOUR-HOUR NOTIFICATION TO THE NRC WAS MADE AT 1[illegible] IN ACCORDANCE WITH 10 CFR 50.72(B)(2)(II). CORRECTIVE ACTIONS INCLUDED CHANGES TO THE OPERATING INSTRUCTIONS TO PROVIDE ADDITIONAL GUIDANCE PRIOR TO REALIGNING THE DIESEL FOR NORMAL OPERATION OR TESTING. THE REVISION INCLUDED A LIST OF THE AUTO-START SIGNALS THAT WILL INITIATE A DIESEL START. AS AN INTERIM MEASURE APPROPRIATE CAUTION TAGS WERE PLACED ON THE MODE SELECTOR SWITCHES.

[48]   FT. CALHOUN 1   DOCKET 50-285   LER 90-017
FAILURE TO PERFORM LOCAL PANEL SURVEILLANCE REQUIRED BY INAPPROPRIATE TECH SPEC.
EVENT DATE: 051790   REPORT DATE: 062790   NSSS: CE   TYPE: PWR

(NSIC 218704) FORT CALHOUN TECHNICAL SPECIFICATION 3.6(3)A SPECIFICALLY STATES FOR THE CONTAINMENT SPRAY, SAFETY INJECTION AND SHUTDOWN COOLING PUMPS:

"ALTERNATE MANUAL STARTING BETWEEN CONTROL ROOM CONSOLE AND THE LOCAL PANEL SHALL BE PRACTICED DURING REFUELING OUTAGES." THESE PUMPS HAVE NEVER HAD A REMOTE OPERATING PANEL SINCE INITIAL PLANT STARTUP, SO NO TESTING HAS BEEN CONDUCTED IN ACCORDANCE WITH THIS TECHNICAL SPECIFICATION. THIS CONDITION WAS VERIFIED ON MAY 17, 1990 IN RESPONSE TO A REQUEST FROM THE SENIOR RESIDENT INSPECTOR. AN ASSESSMENT WAS PERFORMED, INDICATING THAT THERE IS NO CURRENT DESIGN REQUIREMENT OR LICENSING BASIS FOR A LOCAL PANEL. THE LACK OF THE PANEL AND THE FAILURE TO PERFORM SURVEILLANCE TESTING HAVE HAD NO EFFECT ON THE OPERABILITY OF THE ASSOCIATED SYSTEMS OR ON PLANT SAFETY, AND NO LIMITING CONDITIONS FOR OPERATIONS IN SECTION 2.0 OF THE TECHNICAL SPECIFICATIONS HAVE BEEN VIOLATED. THIS REPORT IS BEING SUBMITTED VOLUNTARILY DUE TO POTENTIAL INDUSTRY AND NRC INTEREST. THIS CONDITION WAS THE RESULT OF INADEQUATE IDENTIFICATION, UPDATING, AND TRACKING OF ADMINISTRATIVE ERRORS IN THE TECHNICAL SPECIFICATIONS BY OMAHA PUBLIC POWER DISTRICT (OPPD) AND NRC. OPPD HAS IMPROVED THE PROCESSING AND TRACKING OF CHANGES TO THE TECHNICAL SPECIFICATIONS IN RECENT YEARS THROUGH ENHANCED ADMINISTRATIVE GUIDANCE AND DEDICATION OF RESOURCES.

[49] FT. CALHOUN 1 DOCKET 50-285 LER 90-018
ERROR IN ASI CHANNEL CALIBRATION DUE TO PROCEDURAL DEFICIENCIES.
EVENT DATE: 061290 REPORT DATE: 071290 NSSS: CE TYPE: PWR

(NSIC 218817) ON MAY 31, 1990 AT 1240 HOURS, THE CHANNEL "C" REACTOR PROTECTIVE SYSTEM (RPS) TRIP UNITS FOR AXIAL POWER DISTRIBUTION (APD) AND THERMAL MARGIN/LOW PRESSURE (TM/LP) WERE DETERMINED TO BE INOPERABLE. THE INOPERABILITY RESULTED FROM PROCEDURAL DEFICIENCIES FOR CHANNEL CALIBRATION IN SURVEILLANCE TEST RE-ST-NI-0001. CHANNEL "C" WAS SUBSEQUENTLY CORRECTLY CALIBRATED AND RETURNED TO OPERABILITY. FURTHER EVALUATION OF THIS EVENT ON JUNE 12, 1990, DETERMINED THAT THE PROCEDURAL DEFICIENCIES COULD HAVE LED TO INOPERABILITY OF THE APD AND TM/LP TRIP FUNCTIONS FOR ALL FOUR RPS CHANNELS, EVEN THOUGH ONLY THE "C" CHANNEL OPERABILITY WAS ACTUALLY AFFECTED. THIS EVENT IS THEREFORE REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(V). CORRECTIVE ACTIONS INCLUDE REVISING THE PROCEDURE TO ELIMINATE THE DEFICIENCIES REVIEWING OTHER RPS SURVEILLANCE TESTS TO ENSURE SIMILAR DEFICIENCIES DO NOT EXIST, AND IMPLEMENTING ANY RELEVANT IMPROVEMENTS IN THE PROCEDURE UPGRADE PROGRAM.

[50] FT. ST. VRAIN DOCKET 50-267 LER 89-018 REV 01
UPDATE ON CRACKING IN INCOLOY ALLOY 800 MAIN STEAM RINGHEADERS.
EVENT DATE: 082789 REPORT DATE: 053190 NSSS: GA TYPE: HTGR

(NSIC 218933) CAUSE - CREEP FAILURE. ON 8/25/89, PUBLIC SERVICE COMPANY OF COLORADO (PSC) DISCOVERED A LEAK IN THE MAIN STEAM RINGHEADER OF ONE OF THE LOOP I STEAM GENERATOR MODULES. UPON FURTHER INVESTIGATION, ON 8/27/89, 37 CRACK INDICATIONS WERE FOUND IN THE INCOLOY ALLOY 800 RINGHEADERS IN 8 OF THE 12 MODULES, AND THE CONDITION WAS DETERMINED TO HAVE GENERIC IMPLICATIONS. PSC'S METALLURGICAL EXAMINATION DETERMINED THAT THE CRACKING WAS CAUSED BY CREEP FAILURE, WITH THE PRINCIPLE FAILURE MECHANISM BEING A REDUCTION OF CREEP DUCTILITY OF THE MATERIAL. VARIOUS FACTORS COULD HAVE CONTRIBUTED TO THIS REDUCTION OF CREEP DUCTILITY, INCLUDING COLD WORKING WITHOUT SOLUTION ANNEALING, LOW TITANIUM/CARBON RATIO, HIGH TITANIUM PLUS ALUMINUM CONTENT, AND A COARSE-GRAINED MICROSTRUCTURE. BASED ON THE EXTENT OF THE RINGHEADER CRACKING AND OTHER OPERATIONAL CONSIDERATIONS, ON 8/29/89, PSC DECIDED TO PERMANENTLY SHUT DOWN FORT ST. VRAIN. WELD REPAIR ACTIVITIES HAVE BEEN PERFORMED TO THE EXTENT NECESSARY TO ENSURE DECAY HEAT REMOVAL CAPABILITY DURING SHUTDOWN AND DEFUELING CONDITIONS.

[51] FT. ST. VRAIN DOCKET 50-267 LER 90-001
NON-ESSENTIAL 4160/480 VOLT BUS 5 TRANSFORMER FAULT DUE TO INSULATION BREAKDOWN LEADING TO A FIRE.
EVENT DATE: 041090 REPORT DATE: 051090 NSSS: GA TYPE: HTGR

(NSIC 218934) CAUSE - INSULATION BREAKDOWN. AT 0541 HOURS ON 4/10/90, WITH THE PLANT PERMANENTLY SHUTDOWN FOR DEFUELING, NON-ESSENTIAL BUS 5, 4160/480 VOLT TRANSFORMER, N-9215, EXPERIENCED A FAULT. CONTROL ROOM OPERATORS WERE ALERTED TO

THE FAULT BY AN AUDIBLE NOISE AND ACTUATION OF NUMEROUS CONTROL ROOM TROUBLE ALARMS. ACTION WAS TAKEN FROM THE CONTROL ROOM TO DISCONNECT 480 VOLT BUS 5 FROM ITS POWER SUPPLY. OPERATIONS PERSONNEL DISPATCHED FROM THE CONTROL ROOM DISCOVERED A SMALL FIRE WITHIN THE BUS 5, 4160/480 VOLT TRANSFORMER CABINET. FIRE WAS EXTINGUISHED WITH TWO HAND HELD CO(2) EXTINGUISHERS AT 0550 HOURS. AN AUX. TENDER REMAINED IN THE AREA TO WATCH FOR REFLASH UNTIL A PERMANENT WATCH WAS ESTABLISHED. WITH THE LOSS OF NON-ESSENTIAL BUS 5, ELECTRICAL POWER WAS LOST TO SEVERAL NON-ESSENTIAL BUS LOADS INCLUDING PORTIONS OF THE REACTOR AND TURBINE BUILDING HVAC SYSTEMS, SOME PLANT LIGHTING, REACTOR AND TURBINE BUILDING OVERHEAD CRANES, 2 REACTOR AND TURBINE BLDG MOTOR CONTROL CENTERS (MCCS), AND LOOP II STARTUP BYPASS DRAG VALVE, PV-21130-1. LOSS OF POWER TO PV-21130-1 ALLOWED THE VALVE TO DRIFT CLOSED AND ISOLATE SECONDARY COOLANT FLOW. BUS 5 WAS RE-ENERGIZED AT 0930 HOURS. SECONDARY COOLANT FLOW WAS RE-ESTABLISHED AT 0934 HOURS. INSULATION BREAKDOWN ON ONE OF THE BUS 5 TRANSFORMER'S SECONDARY COILS HAS BEEN IDENTIFIED AS THE CAUSE OF THE FAULT. THE COILS OF ALL 3 PHASES ON THIS TRANSFOMER WERE RE-WOUND.

[52] GINNA DOCKET 50-244 LER 89-016 REV 02
UPDATE ON DUE TO A DESIGN DEFICIENCY THE FAILURE OF THE SI BLOCK/UNBLOCK SWITCH COULD RENDER SOME AUTOMATIC ACTUATION FEATURES OF BOTH TRAINS OF SI INOPERABLE.
EVENT DATE: 111789 REPORT DATE: 062890 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 218680) ON NOVEMBER 17, 1989, AN INITIAL ENGINEERING EVALUATION WAS COMPLETED, WHICH IDENTIFIED A POTENTIAL PROBLEM WITH THE SAFETY INJECTION (SI) BLOCK/UNBLOCK SWITCH. ON DECEMBER 20, 1989, AT 1630 EST, WITH THE REACTOR AT APPROXIMATELY 99% FULL POWER, THE PLANT OPERATING REVIEW COMMITTEE (PORC) REVIEWED THE ENGINEERING EVALUATION, AND MADE A FINAL DETERMINATION THAT A SINGLE FAILURE OF THE SI BLOCK/UNBLOCK SWITCH COULD RENDER SOME AUTOMATIC ACTUATION FEATURES OF BOTH TRAINS OF SI INOPERABLE. THE PORC, AFTER REVIEWING THE SITUATION THOROUGHLY, CONCLUDED THAT SUFFICIENT JUSTIFICATION EXISTED FOR CONTINUED SAFE PLANT OPERATION. THE UNDERLYING CAUSE OF THE EVENT WAS A DESIGN ERROR WHICH OCCURRED DURING THE ORIGINAL CONSTRUCTION OF THE R.E. GINNA NUCLEAR POWER PLANT. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO VERIFY THAT THE PLUNGER POSITION OF THE SI BLOCK/UNBLOCK SWITCH CONTACTS WERE IN THE PROPER POSITION.

[53] GINNA DOCKET 50-244 LER 90-006
ISOLATION OF WRONG PRESSURIZER PRESSURE TRANSMITTER (DUE TO PERSONNEL ERROR), DURING MAINTENANCE, CAUSES AN INADVERTENT SAFETY INJECTION ACTUATION.
EVENT DATE: 050590 REPORT DATE: 060490 NSSS: WE TYPE: PWR

(NSIC 218501) ON MAY 5, 1990, AT 1523 EDST, WITH THE REACTOR IN THE HOT SHUTDOWN CONDITION, DURING STARTUP FROM THE ANNUAL REFUELING AND MAINTENANCE OUTAGE, AN INADVERTENT SAFETY INJECTION (SI) ACTUATION OCCURRED. ALL SAFEGUARDS EQUIPMENT OPERATED AS DESIGNED, BUT NO INJECTION OCCURRED BECAUSE REACTOR COOLANT SYSTEM PRESSURE WAS BEING MAINTAINED ABOVE THE SI PUMP SHUT OFF HEAD. THE SI WAS DUE TO 2 OUT OF 3 LOW PRESSURIZER PRESSURE SI BISTABLES IN THE TRIP MODE. THE UNDERLYING CAUSES OF THE 2 LOW PRESSURIZER PRESSURE SI BISTABLES IN THE TRIP MODE WAS MAINTENANCE ON ONE TRANSMITTER AND THE INADVERTENT CLOSING OF THE WRONG ROOT VALVE TO THE OTHER TRANSMITTER. IMMEDIATE CORRECTIVE ACTION WAS TO STABILIZE THE PLANT, TERMINATE SI, AND OPEN THE WRONGLY CLOSED VALVE. SUBSEQUENT ACTION TO PREVENT RECURRENCE WAS TO COUNSEL PERSONNEL INVOLVED ON STRICT ADHERENCE TO PROCEDURE AND UNDERSTANDING THE CONSEQUENCES OF MISTAKES.

[54] GINNA DOCKET 50-244 LER 90-007
INADVERTENT CLOSURE OF "A" STEAM GENERATOR MAIN FEEDWATER REGULATING VALVE DUE TO CONTROLLER MALFUNCTION CAUSES A REACTOR TRIP ON LOW STEAM GENERATOR LEVEL.
EVENT DATE: 051090 REPORT DATE: 061190 NSSS: WE TYPE: PWR
VENDOR: FOXBORO CO., THE

(NSIC 218502) ON MAY 10, 1990, AT 0219 EDST, WITH THE REACTOR AT APPROXIMATELY 88% FULL POWER, A REACTOR TRIP OCCURRED FROM "A" STEAM GENERATOR (S/G) LOW LEVEL COINCIDENT WITH "A" S/G FEED FLOW/STEAM FLOW MISMATCH. THE TWO REACTOR TRI

BREAKERS OPENED AS REQUIRED AND ALL SHUTDOWN AND CONTROL RODS INSERTED AS DESIGNED. THE REACTOR TRIP WAS DUE TO A MALFUNCTIONING "A" S/G MAIN FEEDWATER REGULATING VALVE CONTROL SYSTEM. THE UNDERLYING CAUSE OF THE MALFUNCTIONING "A" S/G MAIN FEEDWATER REGULATING VALVE CONTROL SYSTEM WAS ATTRIBUTED TO THE SHORTING OF TWO HIGH GAIN AC AMPLIFIER TRANSISTORS IN THE FLOW CONTROLLER, DUE TO THE TRANSISTOR CANS TOUCHING EACH OTHER. IMMEDIATE CORRECTIVE ACTION WAS TO STABILIZE THE PLANT IN HOT SHUTDOWN. SUBSEQUENT ACTION WAS TO PHYSICALLY SEPARATE THE TWO TRANSISTOR CANS SO THAT THEY COULD NOT TOUCH EACH OTHER.

[55] GINNA DOCKET 50-244 LER 90-008
SAFEGUARDS BUSES DEGRADED VOLTAGE RELAYS MISCALIBRATED DUE TO PROCEDURE INADEQUACY CAUSES A CONDITION PROHIBITED BY PLANT TECHNICAL SPECIFICATIONS.
EVENT DATE: 052490 REPORT DATE: 062590 NSSS: WE TYPE: PWR
VENDOR: ITE IMPERIAL CORPORATION

(NSIC 218691) ON MAY 24, 1990 AT 1604 EDST WITH THE REACTOR AT APPROXIMATELY 98% FULL POWER, AN EVALUATION OF UNDERVOLTAGE RELAY TEST DATA REVEALED THAT FIVE (5) OF THE EIGHT (8) DEGRADED VOLTAGE RELAYS ON 480 VOLT SAFEGUARD BUSES WERE CALIBRATED SUCH THAT RELAY ACTUATION WOULD NOT OCCUR WITHIN THE LIMITS OF TECHNICAL SPECIFICATIONS. AS THIS WAS A SETPOINT PROBLEM WITH THE DEGRADED VOLTAGE RELAYS, NO IMMEDIATE PLANT OR OPERATOR RESPONSES WERE NECESSARY. THE UNDERLYING CAUSE OF THE EVENT WAS ATTRIBUTED TO THE WRONG SETPOINT AND SETPOINT TOLERANCE BEING SPECIFIED IN A CALIBRATION PROCEDURE. IMMEDIATE CORRECTIVE ACTION WAS TO CALIBRATE THE DEGRADED VOLTAGE RELAYS TO A SETPOINT ON THE CONSERVATIVE SIDE OF THE TOLERANCE. SUBSEQUENT TO THE CALIBRATION, THE DEGRADED VOLTAGE RELAYS WERE TESTED SATISFACTORILY AND RETURNED TO SERVICE.

[56] GRAND GULF 1 DOCKET 50-416 LER 90-006
EFFLUENT SAMPLE ANALYSIS EXCEEDS TIME LIMIT DUE TO PERSONNEL ERROR.
EVENT DATE: 050290 REPORT DATE: 060190 NSSS: GE TYPE: BWR

(NSIC 218367) ON MAY 2, 1990, A REVIEW OF A GASEOUS EFFLUENT SAMPLE ANALYSIS REVEALED THAT THE TURBINE BUILDING VENTILATION (TBV) EXHAUST SAMPLE OBTAINED ON APRIL 18 1990 HAD NOT BEEN ANALYZED WITHIN 48 HOURS AS REQUIRED BY NOTATION "C" OF TECHNICAL SPECIFICATION TABLE 4.11.2.1.2-1. THE RADWASTE BUILDING VENTILATION (RWBV) EXHAUST FILTER WAS ANALYZED TWICE; ONCE LABELED PROPERLY AS THE "RWBV" FILTER AND AGAIN INCORRECTLY LABELED AS THE "TBV" FILTER. AS A RESULT, THE TBV FILTER WAS NOT ANALYZED UNTIL DISCOVERY OF THE SITUATION ON MAY 2. THE CONDITIONS OF THE LIMITING CONDITION FOR OPERATION AND ITS ASSOCIATED ACTION STATEMENT WERE MET AT ALL TIMES. THE FAILURE TO ANALYZE THE TBV SAMPLE WITHIN 48 HOURS WAS CAUSED BY PERSONNEL ERROR DUE TO INATTENTION TO DETAIL AND A LACK OF A PROPER SELF-VERIFICATION. TRAINING AND COUNSELING OF APPROPRIATE CHEMISTRY PERSONNEL IS BEING CONDUCTED.

[57] GRAND GULF 1 DOCKET 50-416 LER 90-008
FIRE DOORS NOT SURVEILLED DUE TO INADEQUATE PROCEDURE.
EVENT DATE: 052190 REPORT DATE: 062090 NSSS: GE TYPE: BWR

(NSIC 218733) ON MAY 21, 1990, PLANT PERSONNEL DISCOVERED THAT A FIRE RATED DOOR REQUIRED BY TECHNICAL SPECIFICATION 3/4.7.7 WAS NOT DESIGNATED AS A TECHNICAL SPECIFICATION DOOR IN SURVEILLANCE PROCEDURES. UPON THIS FINDING, A COMPREHENSIVE REVIEW OF THE PROCEDURES WHICH IMPLEMENT BOTH TECHNICAL SPECIFICATION AND INSURANCE SURVEILLANCE REQUIREMENTS WAS CONDUCTED. THE REVIEW IDENTIFIED THREE FIRE RATED DOORS REQUIRED TO BE SURVEILLED BY TECHNICAL SPECIFICATION 4.7.7.2 WHICH WERE NOT INCLUDED IN ANY SURVEILLANCE PROCEDURES. UPON IDENTIFICATION OF THE DEFICIENCY, THE ACTIONS REQUIRED BY THE APPLICABLE TECHNICAL SPECIFICATION WERE IMPLEMENTED. THE APPLICABLE SURVEILLANCES WERE PERFORMED WHICH CONFIRMED OPERABILITY OF THE DOORS. THE REVIEW OF FIRE DOORS IN THE CONTROL BUILDING HAS BEEN COMPLETED. FURTHER REVIEWS OF FIRE DOORS IN OTHER SAFETY-RELATED AREAS ARE IN PROGRESS. THESE REVIEWS ARE SCHEDULED TO BE COMPLETED BY JULY 31, 1990.

[58] GRAND GULF 1 DOCKET 50-416 LER 90-009
LOW PRESSURE CORE SPRAY PUMP START DUE TO PERSONNEL ERROR.
EVENT DATE: 052690 REPORT DATE: 062590 NSSS: GE TYPE: BWR

(NSIC 218734) ON MAY 26, 1990, WHILE ATTEMPTING TO RACK-OUT THE LOW PRESSURE CORE SPRAY (LPCS) PUMP BREAKER, A NON-LICENSED OPERATOR PERFORMED A STEP OUT OF SEQUENCE WHICH CAUSED THE LPCS PUMP BREAKER TO CLOSE. THE ACTION OF CLOSING THE BREAKER DID NOT ACTUATE ANY ESF LOGIC, BUT ENERGIZED THE LPCS PUMP MOTOR AND INITIATED THE STANDBY SERVICE WATER (SSW) SYSTEM. THE OPERATOR IMMEDIATELY RECOGNIZED WHAT HAD HAPPENED AND PRESSED THE MANUAL TRIP BUTTON. THE SSW SYSTEM WAS THEN RETURNED TO ITS NORMAL LINEUP. THE OPERATOR HAD LIMITED EXPERIENCE IN THIS TYPE OF BREAKER MANIPULATION AND PERFORMED THE OPERATION ALONE WITHOUT A COPY OF THE GENERAL BREAKER OPERATING INSTRUCTION IN-HAND. THE OPERATOR INVOLVED WAS COUNSELED. SHIFT SUPERVISION WAS INFORMED OF THE INCIDENT AND OF PRECAUTIONS THAT COULD HAVE BEEN TAKEN TO COMPENSATE FOR THE OPERATORS LIMITED EXPERIENCE.

[59] HATCH 1 DOCKET 50-321 LER 90-010
PERSONNEL ERROR RESULTS IN UNPLANNED ACTUATION OF ENGINEERED SAFETY FEATURE.
EVENT DATE: 053090 REPORT DATE: 062690 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 218708) ON 6/30/90, AT APPROX. 0014 CDT, UNIT 1 WAS IN THE REFUEL MODE WITH REACTOR COOLANT PRESSURE ATMOSPHERIC AND REACTOR COOLANT TEMPERATURE AT APPROX. 160F. AT THAT TIME, I&C TECHNICIANS WERE CALIBRATING THE MAIN STEAM LINE RADIATION MONITORS (MSLRMS, MASTER PARTS LIST NUMBER 1D11-K603A/B/C/D, EIIS CODE IL) PER PROCEDURE 57SV-CAL-005-0S, "GE NUMAC LOGARITHMIC RADIATION MONITOR CALIBRATION," AND HAD THE "B" MSLRM REMOVED FROM SERVICE. LICENSED CONTROL ROOM OPERATIONS PERSONNEL THEN REQUESTED THE I&C TECHNICIANS TO REMOVE THE "A" MSLRM FROM SERVICE PRIOR TO RESETTING THE ISOLATION SIGNAL CAUSED BY THE "B" MSLRM. WHEN THE I&C TECHNICIANS REMOVED THE "A" MSLRM FROM SERVICE, LICENSED OPERATIONS PERSONNEL OBSERVED ANNUNCIATION THAT A FULL GROUP 1 ISOLATION SIGNAL HAD BEEN GENERATED. THE MAIN STEAM ISOLATION VALVES (MSIVS, MASTER PARTS LIST NUMBER 1B21-F022A/B/C/D, 1B21-F028A/B/C/D, EIIS CODE JM) WERE ALREADY CLOSED, AND THEREFORE DID NOT MOVE AS A RESULT OF THE ISOLATION SIGNAL. ALSO, IN ACCORDANCE WITH GROUP 1 ISOLATION LOGIC DESIGN, NO OTHER GROUP 1 VALVES CLOSED. THE CAUSE OF THIS EVENT IS PERSONNEL ERROR. PROPER ADMINISTRATIVE CONTROLS WERE NOT FOLLOWED TO TEMPORARILY REVISE THE PROCEDURE IN USE AT THE TIME. SPECIFICALLY, CONTRARY TO PROCEDURE PREREQUISITES, THE LICENSED SHIFT SUPERVISOR AND LICENSED PLANT OPERATORS REQUESTED I&C TECHNICIANS TO GENERATE A FULL GROUP 1 ISOLATION SIGNAL.

[60] HATCH 1 DOCKET 50-321 LER 90-011
PERSONNEL ERROR CAUSES UNPLANNED REACTOR PROTECTION SYSTEM ACTUATION WHEN SHUTDOWN.
EVENT DATE: 060190 REPORT DATE: 062290 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 218709) ON 6/1/90 AT APPROXIMATELY 1549 CDT, UNIT 1 WAS IN THE STARTUP MODE WITH ALL CONTROL RODS FULLY INSERTED AND THE MAIN STEAMLINE ISOLATION VALVES (MSIVS) CLOSED. AT THAT TIME, A FULL REACTOR PROTECTION SYSTEM (RPS) ACTUATION OCCURRED WHEN THE MODE SWITCH WAS MOVED INADVERTENTLY TO THE RUN POSITION RESULTING IN THE GENERATION OF A SCRAM SIGNAL ON MSIVS LESS THAN 90% OPEN WHILE IN THE RUN MODE. THE MODE SWITCH WAS MOVED INADVERTENTLY TOO FAR TO THE RUN POSITION BY A LICENSED PLANT OPERATOR AS HE WAS ADJUSTING IT IN AN ATTEMPT TO CLEAR THE "ROD SEQUENCE CONTROL SYSTEM MALFUNCTION" ANNUNCIATOR. THE SCRAM, WHICH CAUSED NO ROD MOVEMENT BECAUSE ALL RODS WERE ALREADY FULLY INSERTED, WAS RESET AT APPROXIMATELY 1550 CDT. THE ANNUNCIATOR WAS CLEARED WHEN A FAILED SEQUENCE LOGIC BOARD IN THE ROD SEQUENCE CONTROL SYSTEM WAS REPLACED. THE CAUSE OF THIS EVENT IS PERSONNEL ERROR BY A LICENSED PLANT OPERATOR. THE OPERATOR, IN AN ATTEMPT TO CLEAR AN ANNUNCIATOR, ADJUSTED THE MODE SWITCH TO ENSURE ALL POSITION DEPENDENT CONTACTS ON THE MODE SWITCH WERE IN THEIR PROPER POSITION. HE ERRONEOUSLY THOUGHT THE ANNUNCIATOR MIGHT HAVE RESULTED FROM SOME OF THE POSITION DEPENDENT CONTACTS NOT BEING IN THEIR PROPER POSITION. CORRECTIVE ACTIONS FOR

THIS EVENT INCLUDE COUNSELING THE OPERATOR AND COVERING THIS VENT IN THE OPERATIONAL EXPERIENCE ASSESSMENT REPORT ISSUED MONTHLY TO ALL LICENSED PERSONNEL.

[61] HATCH 1 DOCKET 50-321 LER 90-012
MANUAL SCRAM AND NOTIFICATION OF UNUSUAL EVENT DUE TO FIRE IN OFFGAS SYSTEM.
EVENT DATE: 061090 REPORT DATE: 071090 NSSS: GE TYPE: BWR

(NSIC 218813) ON 6/10/90 AT APPROX. 0826 CDT, UNIT 1 WAS IN THE RUN MODE AT AN APPROX. POWER LEVEL OF 609 CMWT. AT THAT TIME, A PROCEDURALLY CONTROLLED MANUAL SCRAM WAS DIRECTED BY MANAGEMENT TO ALLOW FOR TERMINATION OF OFFGAS SYSTEM FLOW TO FACILITATE FULL INVESTIGATION AND RESOLUTION OF OFFGAS OPERATIONAL PROBLEMS. THE GROUP 2 PRIMARY CONTAINMENT ISOLATION SYSTEM VALVES ISOLATED AND AN AUTOMATIC REACTOR PROTECTION SYSTEM ACTUATION WAS RECEIVED ON LOW LEVEL DUE TO THE EXPECTED VOID COLLAPSE AS A RESULT OF THE MANUAL SCRAM. LEVEL WAS RESTORED USING THE "B" REACTOR FEEDWATER PUMP. THE OFFGAS SYSTEM PROBLEMS RESULTED IN A REDUCTION OF HYDROGEN AND OXYGEN RECOMBINATION AND AN EVENTUAL HYDROGEN IGNITION IN THE CARBON ADSORBER BEDS. AT 1331 CDT, IT WAS CONFIRMED COMBUSTION WAS TAKING PLACE AND, AT 1338 CDT, A NOTIFICATION OF UNUSUAL EVENT (NUE) WAS DECLARED FOR A FIRE LASTING LONGER THAN 10 MINUTES (AFTER DISCOVERY). THE OFFGAS SYSTEM WAS PURGED WITH NITROGEN FOR SEVERAL DAYS. EXHAUSTIVE TESTING WAS COMPLETED AT APPROX. 1315 CDT ON 6/16/90 DEMONSTRATING THE FIRE WAS EXTINGUISHED AND THE NUE WAS TERMINATED AT 1325 EDT. THE CAUSE OF THE FIRE IN THE OFFGAS SYSTEM WAS COMPONENT MALFUNCTION DUE TO A COMBINATION OF COMPONENT FAILURE, DISCREPANCIES IN AS-BUILT EQUIPMENT CONFIGURATION, AND LESS THAN ADEQUATE SYSTEM OPERATING PROCEDURES.

[62] HATCH 2 DOCKET 50-366 LER 90-001 REV 01
UPDATE ON COMPONENT FAILURE AND INADEQUATE DESIGN CAUSE GROUP I ISOLATION AND SCRAM.
EVENT DATE: 011290 REPORT DATE: 060890 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 1 (BWR)
VENDOR: GENERAL ELECTRIC CO.
HANCOCK CO.

(NSIC 218530) ON 1/12/90 AT APPROXIMATELY 1610 CST, UNIT 2 WAS IN THE RUN MODE AT AN APPROXIMATE POWER OF 2436 CMWT (APPROXIMATELY 100% OF RATED THERMAL POWER). AT THAT TIME, THE REACTOR SCRAMMED BECAUSE THE MAIN STEAMLINE ISOLATION VALVES (MSIVS) WERE LESS THAN 90% OPEN. THE MSIVS HAD ISOLATED ON A GROUP 1 PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) SIGNAL WHICH RESULTED FROM A FALSE LOW CONDENSER VACUUM SIGNAL. HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM AUTOMATICALLY INITIATED AND INJECTED ON LOW REACTOR WATER LEVEL AS REQUIRED. FOLLOWING WATER LEVEL RECOVERY, HPCI INJECTION VALVE 2E41-F006 CLOSED AUTOMATICALLY ON HIGH WATER LEVEL; HOWEVER, IT COULD NOT BE RE-OPENED WHEN OPERATIONS PERSONNEL SUBSEQUENTLY ATTEMPTED TO START HPCI MANUALLY. THE REACTOR CORE ISOLATION COOLING SYSTEM AND TWO CONTROL ROD DRIVE SYSTEM PUMPS WERE USED TO CONTROL WATER LEVEL FOLLOWING THE FAILURE OF VALVE 2E41-F006 TO OPEN. THE ROOT CAUSES OF THE SCRAM ARE COMPONENT FAILURE AND THE CONFIGURATION OF THE CONDENSER VACUUM SENSING LINES AND INSTRUMENTS. THE DISC OF ROOT ISOLATION VALVE 2N61-F388D SEPARATED FROM ITS STEM ISOLATING THE COMMON SENSING LINE FOR VACUUM SWITCHES 2B21-N056C AND D. CONSEQUENTLY, THESE SWITCHES THEN SENSED A LOW CONDENSER VACUUM AND, BECAUSE THEY INPUT TO THE A AND B TRIP SYSTEMS RESPECTIVELY OF THE ISOLATION LOGIC, THE MSIVS ISOLATED. THE CAUSE OF VALVE 2E41-F006 FAILING TO OPEN IS COMPONENT FAILURE.

[63] HATCH 2 DOCKET 50-366 LER 90-004
PERSONNEL ERROR AND FSAR DEVIATION RESULT IN TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 052190 REPORT DATE: 061590 NSSS: GE TYPE: BWR

(NSIC 218531) ON 5/21/90, AT APPROXIMATELY 1200 CDT, UNIT 2 WAS IN THE RUN MODE AT APPROXIMATELY 2436 CMWT (APPROXIMATELY 100% OF RATED THERMAL POWER). AT THAT TIME, IT WAS CONFIRMED THAT PROCEDURE 62CI-OCB-031-OS, "AIMS GAMMA SPECTROMETER SETUP AND CALIBRATION," INCORRECTLY DIRECTED PERSONNEL TO PERIODICALLY OPEN BOTH POST ACCIDENT SAMPLING SYSTEM (PASS, EIIS CODE IP) ROOM AIRLOCK DOORS SIMULTANEOUSLY. THIS WAS INCLUDED IN A REVISION TO THE PROCEDURE WHICH BECAME

EFFECTIVE ON 9/11/89 AND WAS INTENDED TO PROVIDE A MEASURE OF PERSONNEL SAFETY WHILE WORKING WITH NITROGEN IN THE PASS ROOM. HOWEVER, AS CONFIRMED BY THE ARCHITECT ENGINEER (AE) ON 5/21/90, OPENING BOTH AIRLOCK DOORS TO THIS ROOM RESULTS IN A BREACH OF UNIT 2 SECONDARY CONTAINMENT (EIIS CODE NH). INVESTIGATION SHOWED THAT THE DIRECTION TO OPEN BOTH DOORS SIMULTANEOUSLY HAD BEEN ADDED TO THE PROCEDURE IN A WAY WHICH VIOLATED ESTABLISHED ADMINISTRATIVE CONTROLS AND CIRCUMVENTED THE SAFETY REVIEW PROCESS. AT THE TIME THE DEFICIENT CONDITION WAS CONFIRMED TO EXIST, CONSERVATIVE MEASURES HAD ALREADY BEEN TAKEN TO PREVENT SIMULTANEOUS OPENING OF THE DOORS. THE CAUSES OF THE EVENT ARE COGNITIVE PERSONNEL ERROR ON THE PART OF NON-LICENSED PERSONNEL AND A PERSONNEL AIRLOCK DESIGN WHICH WAS NOT IN COMPLIANCE WITH FSAR COMMITMENTS. CORRECTIVE ACTIONS INCLUDE COUNSELING OF INVOLVED PERSONNEL, REVISING PROCEDURE 62CI-OCB-031-OS, AND ISSUING A MANAGEMENT MEMO.

[64] HOPE CREEK 1 DOCKET 50-354 LER 90-008
REACTOR BUILDING EXHAUST RADIATION MONITOR SET IN EXCESS OF ALLOWABLE LIMITS DUE TO PERSONNEL ERROR.
EVENT DATE: 060490 REPORT DATE: 070590 NSSS: GE TYPE: BWR

(NSIC 218795) ON 6/4/90 AT 1108, A RADIATION PROTECTION SUPERVISOR REPORTED TO THE NUCLEAR SHIFT SUPERVISOR (NSS, SRO LICENSED) THAT, DURING REVIEW OF THE RADIATION MONITORING SYSTEM (RMS) DATA BASE, THE REACTOR BUILDING EXHAUST (RBE) RADIATION MONITOR TRIP SETPOINT HAD BEEN FOUND SET NON-CONSERVATIVELY HIGH (2X10-3 UCI/CC). TECHNICAL SPECIFICATION TABLE 3.3.?-2 REQUIRES THAT THE TRIP SETPOINT FOR THE SUBJECT MONITOR BE SET LESS THAN OR EQUAL TO 1X10-3 UCI/CC. UPON DISCOVERY, THE SETPOINT WAS IMMEDIATELY RESET TO WITHIN TECHNICAL SPECIFICATION REQUIRED PARAMETERS. FOLLOWUP INVESTIGATION DETERMINED THAT THE TRIP SETPOINT HAD BEEN INCORRECTLY ENTERED INTO THE RMS COMPUTER DURING THE PERFORMANCE OF AN I&C DEPARTMENT FUNCTIONAL TEST PROCEDURE WHICH FUNCTIONALLY CHECKS THE RBE RAD MONITOR AND REFUEL FLOOR EXHAUST (RFE) RADIATION MONITOR. THIS OCCURRED ON 5/29/90. THE I&C TECHNICIAN WHO PERFORMED THE PROCEDURE INCORRECTLY INPUT THE SETPOINT VALUE FOR THE RFE RADIATION MONITOR (2X10-3 UCI/CC) PUTTING THE RBE RADIATION MONITOR TRIP SETPOINT. THIS OCCURRED DUE TO IMPROPER RECORDING OF "AS FOUND" SETPOINT DATA WHEN PERFORMING THE PROCEDURE AND INADEQUATE VERIFICATION OF THE DATA WHEN THE PROCEDURE WAS COMPLETED. THE INDEPENDENT VERIFIER (ALSO AN I&C TECHNICIAN) DID NOT CATCH THE ERROR WHEN THE "AS FOUND" DATA WAS RECORDED OR DURING THE DATA VERIFICATION PROCESS AFTER INPUTTING THE SETPOINT.

[65] HOPE CREEK 1 DOCKET 50-354 LER 90-009
HIGH PRESSURE COOLANT INJECTION SYSTEM DECLARED INOPERABLE BASED ON RESULTS OF OIL SAMPLE ANALYSIS DUE TO DESIGN DEFICIENCY IN LUBE OIL RESERVOIR DRAIN ARRANGEMENT.
EVENT DATE: 060790 REPORT DATE: 070590 NSSS: GE TYPE: BWR

(NSIC 218796) ON 6/7/90 AT 1016, THE SYSTEM ENGINEER RESPONSIBLE FOR THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM REPORTED TO THE SENIOR NUCLEAR SHIFT SUPERVISOR (SNSS, SRO LICENSED) THAT HPCI OIL SAMPLES INDICATED HIGH WATER CONTENT AND TRACES OF SEDIMENT CONTAMINATION. THE SNSS DECLARED THE HPCI SYSTEM INOPERABLE, AND A WORK REQUEST WAS INITIATED TO CLEAN THE HPCI TURBINE LUBE OIL RESERVOIR AND CHANGE OUT THE HPCI TURBINE LUBE OIL. THE PRIMARY CAUSE OF THE OIL CONTAMINATION IS A DESIGN DEFICIENCY IN THE HPCI TURBINE LUBE OIL RESERVOIR THAT DOES NOT ALLOW FOR COMPLETE DRAINING RESERVOIR DURING OIL CHANGES. THE RESERVOIR IS NOT EQUIPPED WITH A DRAW LINE AT THE LOW POINT. AS SUCH, IT IS POSSIBLE FOR WATER AND SLUDGE COLLECT AT THE LOW POINT OVER TIME DUE TO CONDENSATION IN THE RESERVOIR HPCI ROOM TEMPERATURES CHANGE. CORRECTIVE ACTIONS INCLUDED CLEANING, FLUSHING, AND WIPE DOWN OF THE HPCI TURBINE LUBE OIL RESERVOIR, RESAMPLING THE LUBE OIL, AND REVIEWING THE HPCI LUBE OIL TRENDING PROGRAM. ADDITIONALLY, PRIOR TO THIS OCCURRENCE, SYSTEMS ENGINEERING HAD INITIATED A DESIGN CHANGE TO INSTALL A DRAIN LINE AT THE LOW POINT IN THE RESERVOIR TO FACILITATE PERIODIC DRAINING OF ACCUMULATED WATER AND SLUDGE.

[66]   LA SALLE 1   DOCKET 50-373   LER 90-008
MISSED TECHNICAL SPECIFICATION HOURLY FIRE WATCH DUE TO MISCOMMUNICATIONS.
EVENT DATE: 052590   REPORT DATE: 062590   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 2 (BWR)

(NSIC 218768) ON 5/25/90 WITH UNIT 1 IN THE RUN MODE AT 100% POWER AND UNIT 2 IN COLD SHUTDOWN WITH THE REACTOR VESSEL REASSEMBLY IN PROGRESS, THE REACTOR BUILDING REFUEL FLOOR HOURLY FIRE WATCH WAS NOT PERFORMED AT 2300 HOURS AS REQUIRED BY TECH SPECS (DUE TO FIRE DETECTION BEING INOPERABLE). THE MISSED FIRE WATCH OCCURRED DUE TO SEVERAL CAUSES WHICH WERE AS FOLLOW: (1) A MISCOMMUNICATION BETWEEN THE SECURITY PERSONNEL WHO NEEDED TO PERFORM THE FIRE WATCH AND THE RADIATION PROTECTION PERSONNEL WHO NEEDED TO AUTHORIZE SECURITY TO PERFORM THE FIRE WATCH IN A HIGH RADIATION AREA, (2) DUE TO SHIFT TURNOVER TAKING PLACE AT THE TIME OF THIS EVENT, NO PERSONNEL WERE LOCATED ON THE REFUEL FLOOR TO ALLOW SECURITY TO PHONE THEM FOR VERIFICATION THAT NO FIRES WERE PRESENT AND (3) INADEQUATE RADIATION CONTROL PRACTICES LEAD TO SECURITY NOT BEING ABLE TO PERFORM THE FIRE WATCH DUE TO THE SPREAD OF CONTAMINATION. THE FIRE WATCH WAS RE-ESTABLISHED AT 0013 HOURS ON 5/26/90, 1 HOUR AND 27 MINUTES FOLLOWING THE PREVIOUS FIRE WATCH AND ALL RESPONSIBLE DEPARTMENT PERSONNEL WILL BE TAILGATED ON THIS EVENT. THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL BECAUSE NO WORK WAS IN PROGRESS PRIOR TO THIS EVENT OR DURING THIS EVENT WHICH WOULD HAVE INCREASED THE POTENTIAL FOR A FIRE TO OCCUR ON THE REFUEL FLOOR, ONLY 27 MINUTES HAD ELAPSED SINCE THE FIRE WATCH WAS REQUIRED TO BE PERFORMED.

[67]   LACROSSE   DOCKET 50-409   LER 90-003
FAILURE TO PERFORM QA AUDITS DUE TO A SCHEDULING ERROR.
EVENT DATE: 060890   REPORT DATE: 070690   NSSS: AC   TYPE: BWR

(NSIC 218822) THE TECHNICAL SPECIFICATIONS FOR THE LA CROSSE BOILING WATER REACTOR (LACBWR) REQUIRE ROUTINE AUDITS OF THE VARIOUS UNIT ACTIVITIES AND PROGRAMS. THE SURVEILLANCE INTERVALS OF THE INDIVIDUAL AUDITS ARE INCLUDED IN THE SPECIFICATIONS. ON JUNE 8, 1990, DURING A DISCUSSION OF FIRE PROTECTION AUDIT REQUIREMENTS LISTED IN THE CURRENT TECHNICAL SPECIFICATIONS, IT WAS DISCOVERED BY THE FIRE PROTECTION SUPERVISOR THAT A REQUIRED TRIENNIAL FIRE PROTECTION AUDIT WAS NO LONGER SCHEDULED. A FURTHER INVESTIGATION OF AUDIT REQUIREMENTS VERSUS AUDIT PERFORMANCE REVEALED THAT FIVE REQUIRED AUDITS HAD BEEN MISSED, DUE TO A SCHEDULING ERROR, SINCE 1/1/89. THE QUALITY ASSURANCE DEPARTMENT WAS NOTIFIED.

[68]   LIMERICK 1   DOCKET 50-352   LER 90-013
THIS LER REPORTS THE AFFECTS OF UNDER RATED DC FUSES AND THE FAILURE TO MAINTAIN ADEQUATE ELECTRICAL ISOLATION BETWEEN CLASS 1E AND NON-CLASS 1E COMPONENTS.
EVENT DATE: 061190   REPORT DATE: 071290   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 218821) ON 6/11/90, BASED ON THE REVIEW OF THE DC ELECTRICAL DISTRIBUTION SYSTEM, IT WAS IDENTIFIED THAT THE UNIT 1 AND UNIT 2 DIVISION 1 AND DIVISION 2 DC DISTRIBUTION SYSTEMS HAD INADEQUATE ISOLATION CAPABILITY BETWEEN CLASS 1E AND NON-CLASS 1E COMPONENTS AND UNDER-RATED DC FUSES. UNITS 1 AND 2 DIVISIONS 1 AND 2 DC DISIRIBUTION SYSTEMS WERE DECLARED INOPERABLE UNTIL ELECTRICAL DISCONNECTS WERE OPENED TO ENSURE PROPER ELECTRICAL ISOLATION BETWEEN THE ASSOCIATED CLASS 1E AND NON-CLASS 1E COMPONENTS. A MODIFICATION WAS IMPLEMENTED THAT PROVIDES ADDITIONAL ISOLATION PROTECTION. FURTHER INVESTIGATION ON 6/13/90 IDENTIFIED THAT FIRE PROTECTION SAFE SHUTDOWN (SSD) METHODS 'B' (UNIT 1) OR 'C' (UNIT 2) COULD BE AFFECTED DUE TO POSTULATED FIRE INDUCED HIGH IMPEDANCE FAULTS RESULTING FROM THE UNDER-RATED DC FUSES FAILING TO ISOLATE HIGH OVERLOAD CURRENT CONDITIONS. IMMEDIATE CORRECTIVE ACTIONS WERE TAKEN TO ESTABLISH HOURLY FIRE WATCHES IN THE AFFECTED UNIT 2 FIRE AREAS UNTIL 6/26/90 WHEN A MODIFICATION WAS COMPLETED REPLACING UNDER-RATED FUSES. THE AFFECTED UNIT 1 FIRE AREA WAS NOT FIRE WATCHED SINCE UNIT 1 WAS IN COLD SHUTDOWN AT THE TIME. PROXIMATE CAUSES OF THESE CONDITIONS ARE ERRORS MADE DURING THE ORIGINAL DESIGN WHEN WE INCORRECTLY ASSUMED THAT DOUBLE FUSING WAS SUFFICIENT ISOLATION AND THAT THE DC FUSES HAD A +10% TOLERANCE.

[69] MCGUIRE 1 DOCKET 50-369 LER 90-013
SPENT FUEL POOL VENTILATION SYSTEMS WERE DECLARED INOPERABLE BECAUSE OF DESIGN DEFICIENCIES.
EVENT DATE: 032990 REPORT DATE: 062590 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)
VENDOR: CLARAGE FAN

(NSIC 218723) ON 3/29/90, PROBLEM INCIDENT REPORT (PIR) 2-M90-0094 WAS SUBMITTED TO DESIGN ENGINEERING BY PERFORMANCE PERSONNEL IDENTIFYING FLOW DISCREPANCIES ON THE UNIT 2 FUEL POOL VENTILATION (VF) SYSTEM. PERFORMANCE PERSONNEL OBTAINED DIFFERENT FLOW READINGS WHEN PERFORMING TWO METHODS FOR FLOW DETERMINATIONS. PERFORMANCE PERSONNEL NOTIFIED OPERATIONS (OPS) PERSONNEL OF THE DISCREPANCIES AND OPS PERSONNEL DECLARED THE UNIT 2 VF SYSTEM, TRAINS "A" AND "B" INOPERABLE BECAUSE OF FLOWS OUTSIDE OF THE REQUIRED TECH SPEC (TS) FLOW RATES. AT THIS TIME, UNIT 1 WAS IN MODE 6 (REFUELING) AND UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT 100% POWER. THE FLOW DISCREPANCIES LEADING UP TO THE PIR BEGAN ON 2/22/90 WHEN PERFORMANCE PERSONNEL WERE TAKING FLOW READINGS AS DIRECTED BY PROCEDURE PT/2/A/4450/09B, SPENT FUEL VENTILATION SYSTEM PERFORMANCE TEST. AFTER FURTHER DISCUSSION AND EVALUATION BY PERFORMANCE AND DESIGN ENGINEERING (DE) PERSONNEL, IT WAS DETERMINED THAT UNIT 1 VF SYSTEM WAS EXPERIENCING SIMILAR FLOW DISCREPANCIES. OPS PERSONNEL DECLARED UNIT 1 VF SYSTEM INOPERABLE ON 4/10/90. AT THIS TIME, UNIT 1 WAS IN MODE 5 (COLD SHUTDOWN) AND UNIT 2 WAS IN MODE 1 AT 100% POWER. AT THE TIME THIS EVENT WAS DETERMINED TO BE REPORTABLE, BOTH UNIT 1 AND UNIT 2 WERE IN MODE 1 AT 100% POWER. THIS EVENT IS ASSIGNED A CAUSE OF IMPROPER INSTALLATION AND A CONTRIBUTORY CAUSE OF DESIGN SELECTION DEFICIENCY.

[70] MCGUIRE 1 DOCKET 50-369 LER 90-012
LOOSE MATERIAL WAS LOCATED IN UPPER CONTAINMENT DURING UNIT OPERATION BECAUSE OF AN INAPPROPRIATE ACTION.
EVENT DATE: 052290 REPORT DATE: 062190 NSSS: WE TYPE: PWR

(NSIC 218722) LOOSE MATERIAL WAS DISCOVERED IN THE UNIT 1 UPPER CONTAINMENT ON MAY 22, 1990 AT 1100. UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT THE TIME OF THE EVENT DISCOVERY. UNIT 1 HAD ENTERED MODE 3 (HOT STANDBY) ON MAY 14, 1990 AT 1339. THE LOOSE MATERIAL PRESENT IN CONTAINMENT RESULTED IN A SURVEILLANCE REQUIREMENT OF TECHNICAL SPECIFICATION 3/4.5.2 NOT BEING MET. THE LOOSE MATERIAL WAS IMMEDIATELY REMOVED. SUBSEQUENTLY, AN INSPECTION WAS CONDUCTED OF THE UNIT 1 LOWER CONTAINMENT OUTSIDE OF THE CRANE WALL. NO LOOSE MATERIAL WAS FOUND. THIS EVENT IS ASSIGNED A CAUSE OF INAPPROPRIATE ACTION RESULTING FROM FAILURE TO FOLLOW PROCEDURE. THIS EVENT WILL BE COVERED WITH APPROPRIATE PERSONNEL. APPLICABLE PROCEDURES WILL BE REVIEWED AND CHANGED AS NECESSARY.

[71] MCGUIRE 1 DOCKET 50-369 LER 90-015
TECHNICAL SPECIFICATION 3.0.3 WAS ENTERED FOR INOPERABLE POWER RANGE NUCLEAR INSTRUMENTATION DURING POWER ESCALATION BECAUSE OF A MANAGEMENT DEFICIENCY.
EVENT DATE: 052490 REPORT DATE: 062590 NSSS: WE TYPE: PWR

(NSIC 218724) ON MAY 24, 1990, POWER ESCALATION WAS OCCURRING ON UNIT 1. AT 1023 THAT DAY, TECHNICAL SPECIFICATION 3.0.3 WAS ENTERED BECAUSE MORE THAN ONE POWER RANGE NUCLEAR INSTRUMENTATION (PRNI) CHANNEL EXCEEDED THE 5 PERCENT DEVIATION (NON-CONSERVATIVE) BETWEEN BEST ESTIMATE THERMAL POWER AND INDICATED EXCORE DETECTOR POWER. INSTRUMENTATION AND ELECTRICAL (IAE) PERSONNEL CALIBRATED THE PRNIS AS SPECIFIED BY PROCEDURE IP/0/A/3007/17, NIS POWER RANGE CALIBRATION TO BEST ESTIMATE THERMAL POWER. TECHNICAL SPECIFICATION 3.0.3 WAS EXITED AT 1037 WHEN CALIBRATION WAS COMPLETE ON THE PRNIS. UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 84 PERCENT AT THE TIME OF THE EVENT. THIS INCIDENT IS ASSIGNED A CAUSE OF MANAGEMENT DEFICIENCY RESULTING FROM INADEQUATE DIRECTION ON HOW TO MANAGE POWER MISMATCH DURING POWER MANEUVERS. OPERATIONS PERSONNEL WILL ASSUME THE RESPONSIBILITY FOR ADJUSTING THE POWER RANGE DETECTORS DURING POWER MANEUVERS, AT THE DISCRETION OF THE SHIFT SUPERVISOR. I/E PERSONNEL WILL CONTINUE TO CALIBRATE THE DETECTORS FOLLOWING REFUELING OUTAGES AND DURING NORMAL SURVEILLANCES. APPROPRIATE PROCEDURES AND TRAINING WILL BE DEVELOPED AND IMPLEMENTED TO ACCOMPLISH THIS.

[72] MCGUIRE 1 DOCKET 50-369 LER 90-014
BOTH TRAINS OF THE CONTROL ROOM VENTILATION SYSTEM WERE INOPERABLE BECAUSE OF A PROCEDURE DEFICIENCY.
EVENT DATE: 060390 REPORT DATE: 070390 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 218799) ON MAY 25, 1990 BETWEEN 1505 AND 1715, INSTRUMENTATION AND ELECTRICAL (IAE) PERSONNEL PERFORMED PROCEDURE IP/0/B/3006/09, RADIATION MONITORING SYSTEM RP-30A LOOP CALIBRATION, ON 1EMF-43A, CONTROL ROOM AIR RADIATION MONITOR FOR TRAIN "A". ON JUNE 3, 1990, BETWEEN 2041 AND 2217, IAE PERSONNEL PERFORMED THE SAME RADIATION MONITORING SYSTEM LOOP CALIBRATION PROCEDURE ON 1EMF-43B, CONTROL ROOM AIR RADIATION MONITOR FOR TRAIN "B". AFTER TESTING THE CONTROL ACTION FOR 1EMF-43B WHICH CLOSES THE CONTROL ROOM VENTILATION (VC) VALVES VC-9 THROUGH 12, UNIT 2 OUTSIDE AIR INTAKE ISOLATION VALVES, OPERATIONS PERSONNEL DISCOVERED THAT VALVES VC-1 THROUGH 4, UNIT 1 OUTSIDE AIR INTAKE ISOLATION VALVES, WERE ALSO CLOSED. PRIOR TO THE RADIATION MONITORING SYSTEM LOOP CALIBRATION PROCEDURE BEING PERFORMED ON 1EMF-43A, VALVES VC-1 THROUGH 4 AND VC-9 THROUGH 12 WERE VERIFIED TO BE OPEN. THEREFORE, IT IS BELIEVED THAT AFTER PERFORMANCE OF THE RADIATION MONITORING SYSTEM LOOP CALIBRATION PROCEDURE FOR 1EMF-43A VALVES VC-1 THROUGH 4 WERE NOT REOPENED. UNIT 1 AND UNIT 2 WERE IN MODE 1 (POWER OPERATION) AT THE TIME OF THIS EVENT. UNIT 1 VARIED IN POWER LEVEL DURING THE TIME THAT VALVES VC-1 THROUGH 4 WERE CLOSED. UNIT 2 WAS AT 100 PERCENT POWER DURING THAT TIME. THIS EVENT IS ASSIGNED A CAUSE OF PROCEDURE DEFICIENCY.

[73] MILLSTONE 1 DOCKET 50-245 LER 90-008
FUEL THERMAL LIMIT AVERAGE PLANAR LINEAR HEAT GENERATOR RATE EXCEEDED TECH SPEC LIMIT.
EVENT DATE: 053190 REPORT DATE: 062890 NSSS: GE TYPE: BWR

(NSIC 218752) ON 5/31/90, WHILE AT 100% POWER (525F, 1030 PSIG) IT WAS DETERMINED THAT THE FUEL THERMAL LIMIT APLHGR (AVERAGE PLANAR LINEAR HEAT GENERATOR RATE) HAD EXCEEDED THE TECH SPEC LIMIT. IMMEDIATE CORRECTIVE ACTION WAS TAKEN BY REACTOR ENGINEERING PERSONNEL TO REDUCE THE VALUE BELOW THE LIMIT. EVEN THOUGH THE CORRECTIVE ACTION WAS COMPLETED BEFORE THE LIMITING CONDITION FOR OPERATION EXPIRED, THIS EVENT IS BEING REPORTED AS REQUIRED BY TECH SPEC 3.11.D. THERE WERE NO SAFETY CONSEQUENCES ASSOCIATED WITH THIS EVENT.

[74] MILLSTONE 2 DOCKET 50-336 LER 90-006
MANUAL REACTOR TRIP ON STEAM GENERATOR 1 LOW LEVEL.
EVENT DATE: 050890 REPORT DATE: 060790 NSSS: CE TYPE: PWR
VENDOR: COPES-VULCAN, INC.

(NSIC 218570) ON 5/8/90 AT 00:49 HOURS WITH THE UNIT IN MODE 1 AT 100% POWER, PLANT OPERATORS MANUALLY TRIPPED THE REACTOR DUE TO DECREASING LEVELS IN THE #1 STEAM GENERATOR. THE SECONDARY PLANT OPERATOR OBSERVED A RAPIDLY DECREASING LEVEL IN THE #1 STEAM GENERATOR, WITH THE FEEDWATER REGULATING VALVE, 2-FW-51A, INDICATING FULL OPEN. WITH CONTROL OF MAIN FEED FLOW TO #1 STEAM GENERATOR APPARENTLY LOST AND SINCE STEAM GENERATOR LEVEL WAS DECREASING, THE UNIT WAS MANUALLY TRIPPED. OPERATORS THEN PERFORMED EOP 2525, "STANDARD POST TRIP ACTIONS" AND EOP 2526, "REACTOR TRIP RECOVERY". ALL EQUIPMENT RESPONDED AS EXPECTED AND THE UNIT WAS PLACED IN A STABLE CONDITION. MAIN FEED FLOW TO THE #1 STEAM GENERATOR WAS RESTORED AND CONTROLLED WITH THE FEEDWATER REGULATING BYPASS VALVE, 2-FW-41A. THE #1 FEEDWATER REGULATING VALVE, 2-FW-51A, WAS DISASSEMBLED AND INSPECTED. DURING THE DISASSEMBLY IT WAS DISCOVERED THAT THE STEM HAD SEPARATED FROM THE PLUG. THIS EVENT IS BEING REPORTED PURSUANT TO THE REQUIREMENTS OF PARAGRAPH 50.73(A)(2)(IV) DUE TO THE MANUAL REACTOR TRIP ON DECREASING STEAM GENERATOR LEVEL. SIMILAR LER'S: 87-009.

[75] MILLSTONE 3 DOCKET 50-423 LER 90-016
"B" STEAM GENERATOR LOW-LOW LEVEL ACTUATION DUE TO INADEQUATE GUIDANCE.
EVENT DATE: 051390 REPORT DATE: 061290 NSSS: WE TYPE: PWR

(NSIC 218562) ON 5/13/90, AT 0351 HOURS WITH THE PLANT SHUT DOWN IN MODE 5 (COLD SHUTDOWN) AT A TEMPERATURE OF 136 DEGREES FAHRENHEIT AND A PRESSURE OF 350 PSIA, A "B" STEAM GENERATOR (S/G) LOW-LOW LEVEL SIGNAL GENERATED A REACTOR TRIP SIGNAL, CAUSED AN AUTOMATIC START OF "A" AND "B" MOTOR DRIVEN AUXILIARY FEEDWATER (MDAFW) PUMPS, AND CLOSED "B" S/G BLOWDOWN ISOLATION VALVE. THE LOW-LOW LEVEL SIGNAL ALSO GENERATED SIGNALS FOR CLOSING "B" S/G SAMPLE VALVE, AND ALIGNING THE MDAFW PUMPS TO TRANSFER WATER FROM THE DEMINERALIZED WATER STORAGE TANK TO THE S/GS. ASIDE FROM THE "B" S/G BLOWDOWN ISOLATION VALVE, NO OTHER VALVES ACTUATED AS THEY WERE ALREADY IN THE REQUIRED POSITIONS. THE EVENT OCCURRED AS A RESULT OF IMPROPERLY ISOLATING THE LEVEL TRANSMITTERS FOR "B" S/G IN PREPAPATION FOR DRAINING THE GENERATOR TO REPAIR A LEAKING HAND HOLE COVER. THE ROOT CAUSE OF THE EVENT WAS INADEQUATE GUIDANCE TO THE OPERATORS ON THE AFFECTS OF ISOLATING THE TRANSMITTERS. TO PREVENT RECURRENCE, A PROCEDURE CHANGE HAS BEEN IMPLEMENTED WHICH PROVIDES GUIDANCE FOR ISOLATING THE LEVEL TRANSMITTERS.

[76] MILLSTONE 3 DOCKET 50-423 LER 90-017
LOSS OF BOTH TRAINS OF HIGH P[illegible]E SAFETY INJECTION DUE TO PERSONNEL ERROR.
EVENT DATE: 051890 REPORT [illegible]1890 NSSS: WE TYPE: PWR

(NSIC 218735) ON 5/18/90, A[illegible], WHILE SHUT DOWN IN MODE 3 (HOT STANDBY), AT 460F AND 900 PSIA, SAFET[illegible] COLD LEG MASTER ISOLATION VALVE 3SIH*MV8835 (MV8835) WAS CL[illegible]D TO FILL SAFETY INJECTION ACCUMULATORS. A LICENSED OPERATOR SPECIFICALLY ASSIGNED TO THE TASK INCORRECTLY FOLLOWED A PROCEDURE THAT WAS MEANT TO BE USED ONLY IF THE REACTOR IS SHUT DOWN WITH [illegible]PERATURE LESS THAN 350F. AT 1820 HOURS, ON 5/18/90, THE ACCUMULATOR FILL OPE[illegible]ON WAS COMPLETED, BUT THE OPERATOR FAILED TO REOPEN MV8835 AS REQUIRED BY THE [illegible]OCEDURE. THE ERROR WAS DISCOVERED AT 2217 HOURS, 5/18/90, BY THE REACTOR OPERATOR WHILE REVIEWING THE ESF STATUS ANNUNCIATORS. AT THIS TIME, MV8835 WAS REOPENED. BOTH HIGH PRESSURE SAFETY INJECTION TRAINS WERE INOPERABLE FOR 4 HOURS, 12 MINUTES. THE ROOT CAUSE WAS A COGNITIVE FAILURE ON THE PART OF THE LICENSED OPERATOR ASSIGNED TO THE EVOLUTION. THE OPERATOR FAILED TO OBSERVE A PROCEDURE NOTE WHICH STATES THE PROCEDURE CAN ONLY BE UTILIZED WHILE THE PLANT IS SHUT DOWN WITH TEMPERATURE LESS THAN 350F. THE OPERATOR ALSO DID NOT FOLLOW THE PROCEDURE WHEN HE FAILED TO REOPEN MV8835 AFTER FILLING THE ACCUMULATORS. CORRECTIVE ACTIONS INCLUDE COUNSELING OF THE INDIVIDUALS INVOLVED, DISSEMINATING INFORMATION ON THE USE OF DEDICATED OPERATORS AND ENSURING TECH SPEC ISSUES ARE COMMUNICATED TO THE APPROPRIATE LEVELS OF SUPERVISION.

[77] MILLSTONE 3 DOCKET 50-423 LER 90-018
IMPROPERLY ESTABLISHED FIRE WATCH DUE TO MISCOMMUNICATION.
EVENT DATE: 060190 REPORT DATE: 070290 NSSS: WE TYPE: PWR

(NSIC 218804) ON 6/2/90, AT 1245 HOURS, WITH THE PLANT AT 100% POWER (MODE 1), THE SHIFT SUPERVISOR (SS) DISCOVERED THAT AN HOURLY FIRE WATCH PATROL HAD NOT BEEN PROPERLY ESTABLISHED IN THE ENGINEERED SAFETY FEATURES (ESF) SUMP AREA AFTER ASSOCIATION FIRE RATED ASSEMBLIES HAD BEEN DECLARED INOPERABLE. THE DURATION OF THE EVENT WAS APPROXIMATELY 26 HOURS. ON 6/1/90 AT APPROXIMATELY 1100 HOURS, FOUR FIRE STOP AND SEAL PENETRATIONS WERE DECLARED INOPERABLE IN THE ENGINEERED SAFETY FEATURES BUILDING (ESF) SUMP AND THE "A" TRAIN CONTAINMENT RECIRCULATION SYSTEM (RSS) PIPE TUNNEL AREAS IN ASSOCIATION WITH A TECH SPEC SURVEILLANCE. AN HOURLY FIREWATCH PATROL WAS ESTABLISHED IN THE 4 FT. 6 IN. ELEVATION OF THE ESF BUT DID NOT ENCOMPASS THE ESF SUMP AREA. THE ROOT CAUSE OF THE EVENT WAS MISCOMMUNICATION BETWEEN SHIFT PERSONNEL WHICH RESULTED IN AN HOURLY FIREWATCH PATROL SIGNATURE SHEET BEING PLACED IN AN INCORRECT LOCATION. IMMEDIATE CORRECTIVE ACTION WAS TO ESTABLISH AN HOURLY FIREWATCH PATROL IN THE ESF SUMP AREA. THE SS HAS BEEN COUNSELED IN THE IMPORTANCE OF VERIFYING COMMUNICATIONS WITH SHIFT PERSONNEL. A PROCEDURE CHANGE WAS MADE TO REQUIRE TIMELY VERIFICATION OF FIRE WATCH PATROL BOUNDARIES.

[78] MILLSTONE 3 DOCKET 50-423 LER 90-019
REACTOR TRIP DUE TO DROPPED ROD DUE TO BROKEN CABLE TO STATIONARY GRIPPER.
EVENT DATE: 060690 REPORT DATE: 070390 NSSS: WE TYPE: PWR

(NSIC 218805) ON 6/6/90, AT 0618 HOURS WITH THE PLANT IN MODE 1 AT 100% POWER, 587F AND 2250 PSIA, AN AUTOMATIC REACTOR TRIP FROM A NEGATIVE FLUX RATE SIGNAL OCCURRED DUE TO A DROPPED CONTROL ROD. THE CAUSE OF THIS EVENT WAS A BROKEN CONNECTION IN THE STATIONARY GRIPPER COIL POWER CABLE FOR ROD G13. THIS SINGLE DROPPED ROD RESULTED IN A NEGATIVE FLUX RATE SIGNAL ON TWO POWER RANGE DETECTORS, THEREBY RESULTING IN A REACTOR TRIP SIGNAL. THE ROOT CAUSE OF THE BROKEN CONNECTION COULD NOT BE IMMEDIATELY DETERMINED. INDEPENDENT EVALUATION BY A MATERIAL TESTING FACILITY IS IN PROGRESS TO ASCERTAIN THE FAILURE MECHANISM. WHEN THE ROOT CAUSE IS POSITIVELY DETERMINED, A SUPPLEMENTAL REPORT WILL BE ISSUED. AS IMMEDIATE CORRECTIVE ACTION CONTROL ROOM OPERATORS PERFORMED THE ACTIONS REQUIRED BY THE APPLICABLE EMERGENCY OPERATING PROCEDURE. THE BROKEN CONNECTOR WAS REPLACED. A FUNCTIONAL TEST WAS PERFORMED BY FULLY WITHDRAWING AND THEN INSERTING THE AFFECTED ROD. POTENTIAL LONG TERM CORRECTIVE ACTIONS WILL BE EVALUATED WHEN THE ROOT CAUSE ANALYSIS IS COMPLETE.

[79] MONTICELLO DOCKET 50-263 LER 90-004
SPURIOUS SIGNAL FROM SPENT FUEL POOL RADIATION MONITOR CAUSES STANDBY GAS TREATMENT INITIATION AND REACTOR BUILDING ISOLATION.
EVENT DATE: 060690 REPORT DATE: 070690 NSSS: GE TYPE: BWR

(NSIC 218786) ON JUNE 8, 1990, WITH THE PLANT OPERATING AT 100% POWER, A PARTIAL GROUP II ISOLATION, STANDBY GAS TREATMENT SYSTEM INITIATION, AND REACTOR BUILDING VENTILATION ISOLATION OCCURRED. THIS WAS THE RESULT OF A SPURIOUS SIGNAL FROM CHANNEL "B" OF THE SPENT FUEL POOL RADIATION MONITOR. UPON RECEIPT OF THE ISOLATION, THE MONITOR WAS CHECKED AND SHOWED RADIATION LEVELS TO BE NORMAL. OTHER RADIATION MONITORS IN CLOSE PROXIMITY TO THE SPENT FUEL POOL ALSO SHOWED RADIATION LEVELS TO BE NORMAL. A RADIATION SURVEY PERFORMED AROUND THE SPENT FUEL POOL AND DIRECTLY AROUND THE "B" CHANNEL DETECTOR FOUND ALL RADIATION LEVELS TO BE NORMAL. FOLLOWING VERIFICATION THAT RADIATION LEVELS WERE NORMAL THE TRIP WAS RESET AND THE GROUP II VALVES, STANDBY GAS TREATMENT SYSTEM AND REACTOR BUILDING VENTILATION WERE RETURNED TO NORMAL. THE "B" CHANNEL OF THE SPENT FUEL POOL RADIATION MONITOR WAS CALIBRATION TESTED AND FUNCTIONALLY TESTED TO VERIFY PROPER OPERATION. AN INVESTIGATION REVEALED NO CAUSE FOR THE SPURIOUS SIGNAL.

[80] NINE MILE POINT 1 DOCKET 50-220 LER 88-001 REV 02
UPDATE ON TECHNICAL SPECIFICATION VIOLATION DUE TO ISI PROGRAM DEFICIENCIES.
EVENT DATE: 011588 REPORT DATE: 062890 NSSS: GE TYPE: BWR

(NSIC 218678) ON JANUARY 15, 1988, WITH NINE MILE POINT UNIT 1 (NMP1) AT 0% POWER AND THE MODE SWITCH IN REFUEL, A REVIEW OF THE FIRST TEN YEAR INSERVICE INSPECTION (ISI) INTERVAL WAS COMPLETED. THIS REVIEW IDENTIFIED SEVERAL INSPECTION DEFICIENCIES REQUIRING RESOLUTION, I.E. FAILURE TO COMPLETE THE FIRST TEN YEAR INTERVAL INSPECTION REQUIREMENTS AND FAILURE TO PROPERLY DISPOSITION DEFICIENCY/CORRECTIVE ACTION (DCA) NOTICES AND OTHER EXAMINATION RESULTS. THE FIRST TEN YEAR INTERVAL WAS SCHEDULED FOR COMPLETION DURING THE 1986 REFUELING OUTAGE. BY NOT COMPLETING AND RESOLVING ALL THE INSPECTION REQUIREMENTS REQUIRED BY SECTION XI OF THE ASME BOILER AND PRESSURE VESSEL CODE, A VIOLATION OF TECHNICAL SPECIFICATION 3.2.6 HAS OCCURRED. THE ROOT CAUSE OF THE EVENT IS MANAGEMENT INEFFECTIVENESS IN IMPLEMENTING THE ISI PROGRAM PLAN. INITIAL CORRECTIVE ACTION WAS TO DETERMINE WHAT EXAMINATIONS WERE OMITTED AND TO ENSURE THAT ANY NECESSARY REPAIRS BE PERFORMED. THESE EXAMINATIONS WERE COMPLETED DURING THE PRESENT REFUELING OUTAGE. ALSO, THE OUTSTANDING DCA'S AND INSPECTION REPORTS WERE PROPERLY DISPOSITIONED. THE LONG-TERM CORRECTIVE ACTION IS TO REALIGN THE ORGANIZATIONAL RESPONSIBILITIES SO THAT CLEAR ACCOUNTABILITY IS MAINTAINED. NECESSARY ADMINISTRATIVE PROCEDURES AND PROGRAM PLAN CHANGES HAVE BEEN MADE TO IMPLEMENT THE NEW ORGANIZATIONAL STRUCTURE.

[81] NINE MILE POINT 1 DOCKET 50-220 LER 88-009 REV 01
UPDATE ON FIRE BARRIER PENETRATIONS NON-FUNCTIONAL DUE TO BEING BREACHED.
EVENT DATE: 032688 REPORT DATE: 060890 NSSS: GE TYPE: BWR

(NSIC 218491) THIS SUPPLEMENT IS A COMPLETE REVISION OF LER 88-09. ON MARCH 26, 1988, WHILE IN A REFUELING OUTAGE WITH THE CORE OFF-LOADED, SIX POTENTIALLY

NON-FUNCTIONAL TECHNICAL SPECIFICATION FIRE BARRIER PENETRATIONS WERE DISCOVERED. THESE CONFIGURATIONS WERE DETERMINED TO HAVE INADEQUATE OR INCOMPLETE SUPPORTING DOCUMENTATION. IN MAY OF 1988, NIAGARA MOHAWK POWER CORPORATION (NMPC) INITIATED A 100% VISUAL RE-EXAMINATION OF THE REQUIRED FIRE BARRIER PENETRATIONS. A TOTAL OF THIRTEEN (13) NON-FUNCTIONAL PENETRATION SEALS WERE IDENTIFIED. THE ROOT CAUSE WAS PERSONNEL ERROR DUE TO A LACK OF UNDERSTANDING OF THE FIRE BARRIER COMMITMENTS. A CONTRIBUTING CAUSE WAS A LACK OF REQUIRED DOCUMENTATION AND INADEQUATE SURVEILLANCE PROCEDURES. CORRECTIVE ACTIONS INCLUDED THE ASSIGNMENT OF A FIRE PROTECTION PROGRAM MANAGER, THE DEVELOPMENT OF OVERALL FIRE PROTECTION PROGRAM AND PROCEDURES, THE UPDATE PENETRATION BASELINE DESIGN DATABASE FOR PROGRAM ALLOWABLES, AND REVISION TO THE ENGINEERING, SITE, AND SURVEILLANCE PROCEDURES.

[82] NINE MILE POINT 1 DOCKET 50-220 LER 89-010 REV 01
UPDATE ON DESIGN DEFICIENCY RESULTING IN POSSIBLE FAILURE OF SAFETY SYSTEM TO PERFORM ITS INTENDED FUNCTION.
EVENT DATE: 063089 REPORT DATE: 062790 NSSS: GE TYPE: BWR

(NSIC 218747) THIS SUPPLEMENT IS BEING SUBMITTED TO ADDRESS COMMITMENTS MADE IN LER 89-10 REVISION 00. ON 6/30/89, AT 0959 DURING SURVEILLANCE TESTING OF THE SERVICE WATER EFFLUENT RADIATION MONITOR, IT WAS NOTED THAT CERTAIN EQUIPMENT FAILURES WOULD NOT RESULT IN DOWNSCALE INDICATION AND, THEREFORE, ANNUNCIATION IN THE CONTROL ROOM. SUBSEQUENT REVIEW OF OTHER RADIATION MONITORING SYSTEMS RESULTED IN THE DISCOVERY OF A SIMILAR DEFICIENCY IN THE RADWASTE DISCHARGE MONITOR. THE ROOT CAUSE FOR BOTH OF THESE EVENTS IS AN INADEQUATE DESIGN REVIEW OF VENDOR SUPPLIED INFORMATION WHEN THE MONITORS WERE INSTALLED. CORRECTIVE ACTIONS TAKEN FOR THE SERVICE WATER EFFLUENT MONITOR INCLUDE A COMPENSATORY SURVEILLANCE TO VERIFY OPERABILITY AND PERFORMANCE OF A MODIFICATION TO PROVIDE ANNUNCIATION CONSISTENT WITH THE TECH SPEC INTENT. CORRECTIVE ACTIONS FOR THE RADWASTE DISCHARGE MONITOR WAS TO PERFORM A MODIFICATION TO ENSURE ANNUNCIATION.

[83] NINE MILE POINT 1 DOCKET 50-220 LER 89-015 REV 01
UPDATE ON EMERGENCY DIESEL FAILURE MODES NOT IDENTIFIED DURING APPENDIX "R" REVIEW.
EVENT DATE: 111089 REPORT DATE: 060890 NSSS: GE TYPE: BWR

(NSIC 218495) ON NOVEMBER 10, 1989, THE FIRE PROTECTION GROUP IDENTIFIED TWO FAILURE MODES NOT PREVIOUSLY ANALYZED IN THE NINE MILE POINT UNIT 1 (NMP1) APPENDIX "R" REVIEW SAFE SHUTDOWN ANALYSIS. THESE TWO FAILURE MODES COULD HAVE POTENTIALLY RENDERED BOTH EMERGENCY DIESEL GENERATORS (EDGS) INOPERABLE SUBSEQUENT TO AN APPENDIX "R" TYPE EVENT. AT THE TIME OF THE DISCOVERY, THE REACTOR WAS IN "COLD SHUTDOWN" WITH THE REACTOR MODE SWITCH IN THE "SHUTDOWN" POSITION AND THE REACTOR CORE OFF-LOADED. THE ANALYSIS DETERMINED THAT FAILURE TO PERFORM DETAILED FAILURE MODES AND EFFECT ANALYSIS ON THE COLD SHUTDOWN SYSTEMS, AS DONE, FOR THE HOT SHUTDOWN SYSTEMS, WAS THE ROOT CAUSE OF THE EVENT. THIS IS A PROGRAMMATIC DEFICIENCY, HOWEVER, THE IDENTIFICATION AND CORRECTION OF THIS CONDITION WAS DUE TO NIAGARA MOHAWK'S CONTINUOUS EFFORT TO ENHANCE THE APPENDIX "R" PROGRAM. CORRECTIVE ACTIONS WERE TO REVISE THE APPLICABLE DAMAGE REPAIR PROCEDURES, PERFORM A DETAILED FAILURE MODES EFFECTS ANALYSIS (FMEA) ON THE DIESEL GENERATORS AND RE-EVALUATE THE NECESSITY OF COMPLETING A DETAILED FMEA FOR OTHER "COLD SHUTDOWN" SYSTEMS.

[84] NINE MILE POINT 1 DOCKET 50-220 LER 90-008
REACTOR BUILDING EMERGENCY VENTILATION INITIATION DUE TO PERSONNEL ERROR.
EVENT DATE: 052390 REPORT DATE: 062290 NSSS: GE TYPE: BWR

(NSIC 218[illegible]1) ON 5/23/90, AT 1313 HOURS, NINE MILE POINT UNIT 1 (NMP1) EXPERIENCED AN ACTUATION OF AN ENGINEERING SAFETY FEATURE (ESF). SPECIFICALLY, INITIATION OF REACTOR BUILDING EMERGENCY VENTILATION (RBEV) AND ISOLATION OF REACTOR BUILDING NORMAL VENTILATION. AT THE TIME OF THE EVENT, THE PLANT WAS IN AN EXTENDED REFUELING OUTAGE WITH THE CORE LOADED AND THE MODE SWITCH IN THE "REFUEL" POSITION. THE ROOT CAUSE OF THE EVENT WAS DUE TO PERSONNEL ERROR IN THAT CARE WAS NOT EXERCISED WHEN WORKING IN CLOSE PROXIMITY TO CONTROL DEVICES.

THE IMMEDIATE CORRECTIVE ACTIONS CONSISTED OF VERIFYING THE RBEV INITIATION, RESETTING CONTROL LOGIC AND ALARMS, RETURNING THE REACTOR BUILDING NORMAL VENTILATION TO SERVICE AND RESTORING THE RBEV TO STANDBY. OTHER CORRECTIVE ACTIONS CONSISTED OF COUNSELING THE RESPONSIBLE TECHNICIAN AND DEVELOPMENT OF A LESSONS LEARNED TRANSMITTAL (LLT). LONG TERM CORRECTIVE ACTION WAS TO INCORPORATE THE ROOT CAUSE AND CONTRIBUTING FACTORS INTO THE INDEPENDENT SAFETY ENGINEERING GROUPS (ISEG) EVALUATION OF RECURRING RBEV INITIATIONS.

[85] NINE MILE POINT 1 DOCKET 50-220 LER 90-009
REACTOR WATER CLEANUP SYSTEM ISOLATION DUE TO A SPURIOUS SIGNAL FROM A HIGH AREA TEMPERATURE MONITOR.
EVENT DATE: 052490 REPORT DATE: 062590 NSSS: GE TYPE: BWR

(NSIC 218688) ON MAY 24, 1990, AT 2219 HOURS WITH THE MODE SWITCH IN THE "SHUTDOWN" POSITION AND NINE MILE POINT UNIT 1 (NMP1) IN AN EXTENDED REFUELING OUTAGE WITH THE CORE LOADED, THE REACTOR WATER CLEANUP SYSTEM ISOLATED DUE TO A SPURIOUS SIGNAL FROM A REACTOR WATER CLEANUP SYSTEM (RWCU) HIGH AREA TEMPERATURE MONITOR. AT THE TIME OF THE EVENT, THE REACTOR VESSEL WATER TEMPERATURE WAS AT APPROXIMATELY 144 DEGREES FAHRENHEIT. THE CAUSE OF THE EVENT WAS THE SPURIOUS TRIP OF REACTOR WATER CLEANUP AREA HIGH TEMPERATURE DETECTOR 09 (TE 33-88). THE ROOT CAUSE FOR THIS SPURIOUS TRIP CANNOT BE DETERMINED. IMMEDIATE CORRECTIVE ACTIONS WERE: TO DISPATCH OPERATORS TO THE AUXILIARY CONTROL ROOM TO VERIFY AREA TEMPERATURE ON REMOTE PANEL INDICATOR; RESETTING TEMPERATURE DETECTOR, AND SENDING OPERATORS TO CLEANUP SYSTEM AREA WHERE THE HIGH TEMPERATURE SIGNAL ORIGINATED TO OBSERVE ANY ABNORMAL CONDITIONS. ADDITIONAL CORRECTIVE ACTIONS INCLUDED RETURNING RWCU TO NORMAL OPERATION AND ISSUING A WORK REQUEST TO TROUBLESHOOT THE CAUSE OF THE SPURIOUS SIGNAL.

[86] NINE MILE POINT 1 DOCKET 50-220 LER 90-010
TWO REACTOR SCRAMS DUE TO PERSONNEL ERROR AND EQUIPMENT DEFICIENCY.
EVENT DATE: 052590 REPORT DATE: 062590 NSSS: GE TYPE: BWR

(NSIC 218689) ON MAY 25, 1990, NINE MILE POINT UNIT 1 EXPERIENCED TWO SCRAMS AND A REACTOR BUILDING EMERGENCY VENTILATION SYSTEM (RBEVS) INITIATION WHILE IN THE COLD SHUTDOWN CONDITION WITH THE CORE LOADED AND THE MODE SWITCH IN THE REFUEL POSITION. THE FIRST SCRAM AND ASSOCIATED RBEVS INITIATION OCCURRED WHEN REACTOR PROTECTION SYSTEM (RPS) BUS 12 EXPERIENCED A LOSS OF POWER WHILE BEING MANUALLY TRANSFERRED FROM INSTRUMENT AND CONTROLS (I&C) BUS 130A TO MOTOR GENERATOR (MG) SET 172. THE SECOND SCRAM OCCURRED 51 SECONDS LATER DUE TO A SCRAM DISCHARGE VOLUME HIGH WATER LEVEL TRIP WHEN THE OPERATOR FAILED TO BYPASS THIS SCRAM FUNCTION PRIOR TO RESETTING THE FIRST SCRAM. PRELIMINARY ASSESSMENT INDICATES THAT BOTH OPERATOR PERFORMANCE AND MG SET DEFICIENCIES WERE CONTRIBUTING CAUSES TO THIS FIRST EVENT. THE ROOT CAUSE FOR THE FIRST EVENT IS CONTINUING TO BE INVESTIGATED. THE ROOT CAUSE FOR THE SECOND EVENT IS FAILURE TO FOLLOW PROCEDURES. IMMEDIATE CORRECTIVE ACTION FOR THE FIRST EVENT WAS TO ASSIGN A SYSTEM ENGINEER TO BE PRESENT DURING FUTURE BUS TRANSFERS. ADDITIONAL CORRECTIVE ACTIONS INCLUDE REVISING THE OPERATING PROCEDURE AND COMPLETING A FORMAL ROOT CAUSE INVESTIGATION. CORRECTIVE ACTIONS TAKEN FOR THE SECOND EVENT INCLUDE GENERATION OF A LESSONS LEARNED TRANSMITTAL AND COUNSELING OF CONTROL ROOM PERSONNEL CONCERNING THEIR POST-EVENT RESPONSE AND SUBSEQUENT ACTIONS.

[87] NINE MILE POINT 1 DOCKET 50-220 LER 90-011
REACTOR SCRAM AND REACTOR BUILDING EMERGENCY VENTILATION SYSTEM INITIATION DURING SURVEILLANCE TEST DUE TO DESIGN DEFICIENCY.
EVENT DATE: 053090 REPORT DATE: 062890 NSSS: GE TYPE: BWR

(NSIC 218785) ON MAY 30, 1990, NINE MILE POINT UNIT ONE (NMP1) WAS IN A REFUELING OUTAGE WITH FUEL IN THE VESSEL AND THE MODE SWITCH IN THE REFUEL POSITION. AT APPROXIMATELY 2220 HOURS, WHILE PERFORMING SURVEILLANCE TEST N1-ST-R2, "LOSS OF COOLANT ACCIDENT AND EMERGENCY DIESEL GENERATOR SIMULATED AUTOMATIC INITIATION TEST", A FULL SCRAM SIGNAL WAS RECEIVED AND A REACTOR BUILDING EMERGENCY VENTILATION SYSTEM (RBEVS) INITIATION OCCURRED DUE TO A LOSS OF POWER TO REACTOR PROTECTION SYSTEM (RPS) BUS 12. THE CAUSE OF THIS EVENT IS THE MANUAL OPENING OF

MOTOR GENERATOR SET 172 (MG 172) DRIVE MOTOR CONTACTOR CAUSING LOSS OF POWER TO RPS BUS 12. THE MOST PROBABLE ROOT CAUSE FOR THIS EVENT IS A DESIGN DEFICIENCY IN THE TRANSFER LOGIC FOR MG SETS FROM AC TO DC POWER. THE IMMEDIATE CORRECTIVE ACTIONS TAKEN INCLUDED RESETTING THE SCRAM, PLACING RPS BUS 12 ON INSTRUMENTATION AND CONTROL BUS 130A, SECURING RBEVS AND RESTORING NORMAL REACTOR BUILDING VENTILATION. ADDITIONALLY, PLANT MODIFICATION N1-90-005, "MG SETS SPEED CONTROL", HAS BEEN COMPLETED, CULMINATING IN THE SUCCESSFUL COMPLETION OF N1-ST-R2 ON JUNE 16, 1990.

[88] NINE MILE POINT 1 DOCKET 50-220 LER 90-012
ENGINEERED SAFETY FEATURE ACTUATION DURING SURVEILLANCE TESTING DUE TO AN INADEQUATE PROCEDURE.
EVENT DATE: 053190 REPORT DATE: 062890 NSSS: GE TYPE: BWR

(NSIC 218690) AT 0454, ON MAY 31, 1990, WITH THE MODE SWITCH IN REFUEL AND NINE MILE POINT UNIT 1 (NMP1) IN THE COLD SHUTDOWN CONDITION, A MAIN STEAM ISOLATION VALVE (MSIV) ISOLATION RESULTED IN THE AUTOMATIC INITIATION OF A REACTOR SCRAM. THE MSIV ISOLATION WAS CAUSED BY THE IMPROPER SIMULATION OF THE CONDENSER VACUUM INPUT SIGNAL TO THE REACTOR PROTECTION SYSTEM (RPS) DURING SURVEILLANCE TESTING. THIS RESULTED IN THE RPS SENSING CONDENSER VACUUM AND REACTOR PRESSURE CONDITIONS WHICH SATISFIED THE LOGIC TO INITIATE AN MSIV ISOLATION. THE ROOT CAUSE OF THIS EVENT WAS AN INADEQUATE PROCEDURE. IMMEDIATE CORRECTIVE ACTIONS WERE TO RESTORE THE RPS TO A NORMAL CONFIGURATION AND RESET THE SCRAM. SUBSEQUENT CORRECTIVE ACTION WAS TO MAKE A TEMPORARY CHANGE TO THE ASSOCIATED PROCEDURE TO ENSURE RPS INPUT SIGNALS ARE PROPERLY SIMULATED. FINAL CORRECTIVE ACTIONS WILL BE TO REVISE THE PROCEDURE TO UTILIZE A MORE RELIABLE SIMULATION METHOD, AND TO ISSUE A LESSONS LEARNED TRANSMITTAL TO THE NINE MILE POINT OPERATIONS AND MAINTENANCE DEPARTMENTS.

[89] NINE MILE POINT 2 DOCKET 50-410 LER 89-029 REV 01
UPDATE ON TECHNICAL SPECIFICATION VIOLATION, INCORRECT SETTING OF SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS DUE TO ENGINEERING CHANGE.
EVENT DATE: 092089 REPORT DATE: 061490 NSSS: GE TYPE: BWR

(NSIC 218750) ON 9/20/89 AT 0445 HOURS IT WAS DISCOVERED THAT NINE MILE POINT UNIT 2 HAD BEEN IN A CONDITION NOT ALLOWED BY TECH SPECS (TS). SPECIFICALLY, THERE WAS A FAILURE TO COMPLY WITH THE REQUIREMENT THAT THE SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS HAVE AN OPENING SETPOINT TO BE LESS THAN OR EQUAL TO 0.25 POUNDS PER SQUARE INCH DELTA (PSID). AT THE TIME OF THE DETERMINATION THE REACTOR WAS IN COLD SHUTDOWN (OPERATIONAL CONDITION 4) AT AMBIENT PRESSURE WITH THE REACTOR COOLANT WATER TEMPERATURE ABOUT 115F. THE IMMEDIATE CAUSE OF THE CONDITION WAS AN ENGINEERING CHANGE ISSUED IN AUGUST 1986. THIS CHANGE ALLOWED THE MAINTENANCE SETPOINT FOR THE SUPPRESSION POOL/DRYWELL VACUUM BREAKERS TO BE SET NON-CONSERVATIVE TO THE TS REQUIREMENT. THE ROOT CAUSE FOR THIS CONDITION HAS BEEN DETERMINED TO BE POOR WORK PRACTICES AND INADEQUATE CHANGE MANAGEMENT BY THE CONTRACTOR DURING THE CONSTRUCTION PHASE OF THE PLANT. CORRECTIVE ACTION WAS TO CANCEL THE ENGINEERING AND DESIGN CHANGE AND CORRECT THE MAINTENANCE PROCEDURE. ALL SUPPRESSION POOL/DRYWELL VACUUM BREAKERS WERE IMMEDIATELY CHECKED AND ADJUSTED AS REQUIRED PER THE MAINTENANCE PROCEDURE TO WITHIN THE TS LIMIT.

[90] NINE MILE POINT 2 DOCKET 50-410 LER 90-009
MANUAL REACTOR SCRAM DUE TO LOSS OF CONDENSER VACUUM CAUSED BY INSTRUMENT AIR LINE BREAK.
EVENT DATE: 051490 REPORT DATE: 061390 NSSS: GE TYPE: BWR

(NSIC 218539) ON MAY 14, 1990, AT APPROXIMATELY 2052 HOURS, WITH THE REACTOR MODE SWITCH IN THE "RUN" POSITION AND THE REACTOR OPERATING AT 100% RATED THERMAL POWER, NINE MILE POINT UNIT 2 (NMP2) WAS EXPERIENCING NUMEROUS OFF-NORMAL PLANT CONDITIONS. OPERATIONS PERSONNEL WERE RESPONDING TO SEVERAL ABNORMAL OFFGAS SYSTEM (OFG) INDICATIONS AND ALARMS, AND MAIN CONDENSER VACUUM WAS DECREASING. AT APPROXIMATELY 2058 HOURS, OPERATORS INITIATED REACTOR POWER REDUCTION IN RESPONSE TO DECREASING CONDENSER VACUUM. AT 2119 HOURS WITH REACTOR POWER AT

APPROXIMATELY 45% RATED THERMAL POWER, THE REACTOR MODE SWITCH WAS PLACED IN THE "SHUTDOWN" POSITION, INITIATING A REACTOR SCRAM. THE CAUSE OF THIS EVENT WAS THE PARTIAL LOSS OF THE INSTRUMENT AIR SYSTEM (IAS) DUE TO PIPE FAILURE INDUCED BY STRESS-CORROSION CRACKING. IMMEDIATE CORRECTIVE ACTIONS INCLUDED: PERFORMING A WALKDOWN OF THE SUSPECTED BRANCH OF THE IAS TO LOCATE SOURCE OF AIR LOSS; GENERATING A WORK REQUEST TO REPLACE THE FAILED AIR LINE; AND REQUESTING A FAILURE ANALYSIS BE PERFORMED TO IDENTIFY CAUSE OF PIPE FAILURE.

[91] NINE MILE POINT 2 DOCKET 50-410 LER 90-010
SURVEILLANCE TESTS NOT PERFORMED DUE TO POOR WORK PRACTICES AND INADEQUATE WRITTEN COMMUNICATIONS.
EVENT DATE: 051790 REPORT DATE: 061890 NSSS: GE TYPE: BWR

(NSIC 216772) ON 5/17/90, AT 1538 HOURS WITH THE REACTOR AT 0% POWER AND IN COLD SHUTDOWN WITH THE MODE SWITCH IN THE "SHUTDOWN" POSITION (OPERATIONAL CONDITION 4), IT WAS DISCOVERED THAT NINE MILE POINT UNIT 2 WAS NOT IN COMPLIANCE WITH TECH SPEC (TS) SECTION 4.6.1.2.D. IT WAS DETERMINED THAT THE LEAK RATE SURVEILLANCE REQUIREMENT (TYPE B TEST) FOR THE FIVE TRAVERSING INCORE PROBE (TIP) PRIMARY CONTAINMENT PENETRATION FLANGES WAS NOT SATISFIED. IN ADDITION, ONE OF THE TIP PENETRATION ASSEMBLIES INCORPORATES A FLANGE JOINT WITH O-RING SEAL THAT ALSO HAD NOT BEEN TESTED. THE ROOT CAUSE OF BOTH MISSED SURVEILLANCE TESTS WAS POOR WORK PRACTICES. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDE: (1) REVISED THE TYPE "B" SURVEILLANCE PROCEDURE. (2) PERFORMED THE TYPE "B" SURVEILLANCE TEST SATISFACTORILY ON ALL FLANGES AS REQUIRED. (3) UPGRADED THE COMPUTERIZED SURVEILLANCE TRACKING SYSTEM. (4) PERFORMED A REVIEW OF TYPE "B" LEAK RATE TEST PROGRAM TO DETERMINE OTHER UNIQUE FLANGES AND/OR PENETRATIONS NEED TO BE LEAK RATE TESTED. (5) THE UPDATED SAFETY ANALYSIS REPORT WILL BE REVISED.

[92] NINE MILE POINT 2 DOCKET 50-410 LER 90-011
SHUTDOWN COOLING SYSTEM ISOLATION DUE TO INADEQUATE PROCEDURE DEVELOPMENT.
EVENT DATE: 051890 REPORT DATE: 061890 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 218730) ON MAY 18, 1990, AT APPROXIMATELY 1015 HOURS, THE SHUTDOWN COOLING (SDC) SYSTEM ISOLATED DURING THE PERFORMANCE OF THE REACTOR PROTECTION SYSTEM (RPS) ELECTRICAL PROTECTION ASSEMBLIES (EPA) FUNCTIONAL TEST PROCEDURE. WHEN DIVISION I POWER SUPPLY EPA-1 (LOOP A) WAS TRIPPED PER SURVEILLANCE, THE DIVISION II (LOOP B) INJECTION VALVE, RESIDUAL HEAT REMOVAL SYSTEM MOTOR OPERATED VALVE 2RHS*MOV40B, CLOSED UNEXPECTEDLY. SUBSEQUENTLY, SDC LOOP B PUMP 2RHS*P1B WAS MANUALLY TRIPPED. AT THE TIME OF THE EVENT, NINE MILE POINT UNIT 2 (NMP2) WAS AT 0 PERCENT POWER WITH ALL RODS INSERTED AND THE MODE SWITCH IN "SHUTDOWN" POSITION (MODE 4). THE IMMEDIATE CAUSE OF THIS EVENT WAS A PROCEDURE DEFICIENCY. THE ROOT CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE INADEQUATE PROCEDURE DEVELOPMENT AND REVIEW. IMMEDIATE CORRECTIVE ACTIONS WERE TO MONITOR COOLANT TEMPERATURES AND TO MANUALLY RESTORE NORMAL SDC BY 1035 HOURS. ADDITIONAL CORRECTIVE ACTION INCLUDES ISSUING PROCEDURAL CHANGES.

[93] NINE MILE POINT 2 DOCKET 50-410 LER 90-012
VIOLATION OF TECHNICAL SPECIFICATIONS DUE TO PERSONNEL ERROR.
EVENT DATE: 052990 REPORT DATE: 062890 NSSS: GE TYPE: BWR

(NSIC 218731) ON MAY 29, 1990, AT 1700 HOURS, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED A TECHNICAL SPECIFICATION VIOLATION FOR FAILURE TO COMPLETE TECHNICAL SPECIFICATION REQUIRED ACTION FOR A CONTROL ROD WITH AN INOPERABLE CONTROL ROD POSITION INDICATION. AT THE TIME OF THE EVENT, THE MODE SWITCH WAS IN "RUN" AND THE REACTOR WAS AT 31% RATED THERMAL POWER. THE ROOT CAUSE FOR THIS EVENT WAS DETERMINED TO BE PERSONNEL ERROR. THE CORRECTIVE ACTIONS TAKEN FOR THIS EVENT INCLUDED: THE AFFECTED CONTROL ROD WAS RESTORED TO OPERABLE STATUS; THE LICENSED REACTOR OPERATOR AND REACTOR ANALYST SUPERVISOR INVOLVED COMPLETED REMEDIAL TRAINING ON CONTROL ROD OPERABILITY, VERBAL COMMUNICATIONS AND PERSONNEL RESPONSIBILITIES DURING CONTROL ROD MOVEMENT; PRIOR TO RESUMING POWER ASCENSION, THE OPERATIONS SUPERINTENDENT AND ASSISTANT OPERATIONS SUPERINTENDENT REVIEWED THE EVENT AND CORRECTIVE ACTIONS WITH ALL OF THE OPERATING SHIFTS; A LESSONS

LEARNED TRANSMITTAL (LLT) WAS DEVELOPED; AND OPERATING PROCEDURE N2-OP-95A, "ROD WORTH MINIMIZER", SECTION H.1.0 WAS REVISED.

[94] NORTH ANNA 1 DOCKET 50-338 LER 90-007
SERVICE WATER PUMP HOUSE TORNADO MISSILE SHIELD BLOCKS NOT IN PLACE DUE TO INADEQUATE ADMINISTRATIVE CONTROL.
EVENT DATE: 052390 REPORT DATE: 062290 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)

(NSIC 218714) AT 1430 HOURS ON MAY 23, 1990, WITH BOTH UNITS AT 100% POWER (MODE 1), PLANT PERSONNEL DISCOVERED THAT TWO SETS OF CONCRETE ROOF BLOCKS ON THE SERVICE WATER PUMP HOUSE WERE NOT IN THEIR REQUIRED SAFETY POSITION. THESE BLOCKS ARE PART OF THE SERVICE WATER PUMP HOUSE MISSILE PROTECTION BOUNDARY, BUT ARE CAPABLE OF BEING PHYSICALLY REMOVED TO FACILITATE MAINTENANCE ON EQUIPMENT INSIDE THE STRUCTURE. THE SERVICE WATER PUMP HOUSE WAS MISSING AT LEAST ONE SET OF BLOCKS FROM OCTOBER 30, 1989 TO MAY 23, 1990. THIS EVENT CONSTITUTES A CONDITION THAT ALONE COULD HAVE PREVENTED THE FULFILLMENT OF A SAFETY FUNCTION NEEDED TO REMOVE RESIDUAL HEAT AND MITIGATE THE CONSEQUENCES OF AN ACCIDENT. THIS EVENT IS REPORTABLE PURSUANT TO 10 CFR 50.73 (A)(2)(V)(B&D). A FOUR HOUR REPORT WAS MADE TO THE NRC ON MAY 23, 1990 IN ACCORDANCE WITH 10 CFR 50.72 (B)(2)(III)(B&D). DURING THE PERIOD THAT ROOF BLOCKS WERE NOT IN PLACE, WIND VELOCITIES WERE NEVER HIGH ENOUGH TO CREATE PROJECTILES THAT COULD HAVE AFFECTED EQUIPMENT OPERABILITY IN THE SERVICE WATER PUMP HOUSE. IN ADDITION, DURING THIS TIME, A QUARTER INCH STEEL PLATE SHIELD WAS POSITIONED OVER THE ROOF HOLES. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED AT ANY TIME DURING THIS EVENT.

[95] NORTH ANNA 1 DOCKET 50-338 LER 90-008
SERVICE WATER SYSTEM SUPPORT DESIGN ERROR.
EVENT DATE: 052490 REPORT DATE: 061890 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)

(NSIC 218715) AT 1714 HOURS ON MAY 24, 1990, WITH UNITS 1 AND 2 AT 100% POWER (MODE 1), ENGINEERING PERSONNEL DISCOVERED A DESIGN ERROR WHERE THE SERVICE WATER SYSTEM PRESSURES EXCEEDED THE DESIGN VALUES FOR THE ANCHORED PIPING SUPPORTS ON THE SUPPLY AND RETURN LINES FOR THE RECIRCULATION SPRAY HEAT EXCHANGERS. UNDER MAXIMUM PRESSURE CONDITIONS, THE INTEGRITY OF THE PIPING SYSTEM COULD BE COMPROMISED. THIS EVENT CONSTITUTES A CONDITION THAT ALONE COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION AND IS REPORTABLE PURSUANT TO 10 CFR 50.73 (A)(2)(V)(D). A JUSTIFICATION FOR CONTINUED OPERATION (JCO) WAS DEVELOPED BASED ON AN EVALUATION OF THE PIPING SUPPORT SYSTEM AT ELEVATED PRESSURES. BASED ON AN EVALUATION OF THE SYSTEM STRESSES USING ASME SECTION III, APPENDIX F, IT WAS DETERMINED THAT THE PRESSURE BOUNDARY OF THE SERVICE WATER SYSTEM PIPING WOULD BE MAINTAINED. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED AT ANY TIME DURING THIS EVENT.

[96] OCONEE 1 DOCKET 50-269 LER 90-007
ACTUATION OF EMERGENCY SAFEGUARDS SYSTEM DUE TO DEFECTIVE PROCEDURE, LACK OF PROCEDURAL PRECAUTIONS.
EVENT DATE: 051690 REPORT DATE: 061490 NSSS: BW TYPE: PWR

(NSIC 218510) ON MAY 16, 1990, AT 1617 HOURS, WITH UNIT 1 IN A REFUELING OUTAGE THE ENGINEERED SAFEGUARDS (ES) SYSTEM, CHANNELS 1 THROUGH 6, ACTUATED UNEXPECTED AS A RESULT OF A PNEUMATIC PRESSURE TEST THAT WAS BEING PERFORMED DURING THE IMPLEMENTATION OF A NUCLEAR STATION MODIFICATION (NSM). THE NSM REQUIRED THE REPLACEMENT OF REACTOR BUILDING (RB) PRESSURE SWITCHES AND THE PRESSURE TEST OF THEIR ASSOCIATED TUBING. THE PRESSURE SWITCHES SUPPLY SIGNALS TO ES CHANNELS 7 & 8 WHEN HIGH RB PRESSURE IS SENSED. THE NSM PROCEDURE ISOLATED THE PRESSURE SWITCHES FROM THEIR ES CHANNELS BUT DID NOT ISOLATE THE PRESSURE TRANSMITTERS THAT ALSO EXISTED WITHIN THE PRESSURE BOUNDARY OF THE TEST. WHEN THE BOUNDARY WAS PRESSURIZED THE PRESSURE TRANSMITTERS, WHICH ALSO MONITOR RB PRESSURE, SUPPLIED A SIGNAL THAT ACTUATED ES CHANNELS 1 THROUGH 6. DUE TO THE REFUELING OUTAGE SOME ES COMPONENTS WERE OUT OF SERVICE, HOWEVER, THOSE COMPONENTS THAT WERE IN SERVICE OPERATED AS EXPECTED. THE REACTOR OPERATORS CONTROLLED THE

COMPONENTS AND RETURNED THEM TO NORMAL STATUS. THE CORRECTIVE ACTION REVISED THE PROCEDURE TO PROVIDE FOR PRESSURE TRANSMITTER ISOLATION PRIOR TO RESUMING THE TEST. THE ROOT CAUSE OF THIS EVENT WAS DEFECTIVE PROCEDURE, LACK OF PROCEDURAL PRECAUTIONS WITH A CONTRIBUTING CAUSE OF INAPPROPRIATE ACTION, LACK OF ATTENTION TO DETAIL.

[97] OCONEE 1 DOCKET 50-269 LER 90-009
INAPPROPRIATE OPERATOR ACTIONS TO CONTROL AND MAINTAIN MINIMUM LEVEL IN EMERGENCY FEEDWATER INVENTORY TANK RESULTED IN A TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 060490 REPORT DATE: 070390 NSSS: BW TYPE: PWR

(NSIC 218787) ON 6/4/90, AT APPROX. 0909 HOURS, A TECH SPEC VIOLATION OCCURRED DURING UNIT 1 START-UP ACTIVITIES WHILE AT HOT SHUTDOWN CONDITIONS. DURING ATTEMPTS TO DE-OXYGENATE THE FEEDWATER (FDW), SO THAT FINAL FDW WOULD MEET CHEMISTRY SPECIFICATIONS FOR FEEDING THE STEAM GENERATORS, THE UPPER SURGE TANK (UST) WAS NOT CONTROLLED AT A LEVEL OF SIX FEET OR GREATER AS REQUIRED TO PROVIDE AN ADEQUATE WATER SOURCE FOR THE EMERGENCY FEEDWATER PUMPS. THE SITUATION, WHICH LED TO THIS VIOLATION, DEVELOPED OVER A PERIOD OF 1.5 HOURS AND WAS CAUSED BY OPERATOR ACTIONS THAT WERE DIRECTED AT LOWERING THE WATER TEMPERATURE IN THE UST DURING CLEAN-UP OPERATIONS. OPERATOR ACTIONS TO MONITOR LEVEL AND CONTROL MAKEUP TO THE UST DURING THEIR EFFORTS TO REDUCE THE TEMPERATURE AND COMPLY WITH A NEW TEMPERATURE LIMIT WERE NOT ADEQUATE. ALSO, PROPER RESPONSE TO COMPUTER ALARMS WAS NOT MADE. IMMEDIATE CORRECTIVE ACTIONS TO LINE-UP SUFFICIENT MAKEUP TO THE UST WERE DELAYED DUE TO A MISPLACED PROCEDURE, BUT BY 0950 HOURS, WERE EFFECTIVE IN RESTORING REQUIRED TANK LEVEL. THE ROOT CAUSE OF THIS EVENT IS CLASSIFIED INAPPROPRIATE ACTION, NO ACTION TAKEN WHEN REQUIRED, BECAUSE THE NEED WAS NOT RECOGNIZED. TWO CONTRIBUTING CAUSES, MANAGEMENT DEFICIENCY, POOR MANAGEMENT INTERFACE AND DEFECTIVE PROCEDURE, INCOMPLETE INFORMATION ARE ALSO ASSIGNED TO THIS INCIDENT.

[98] PALISADES DOCKET 50-255 LER 90-010
TECHNICAL SPECIFICATION REQUIRED SAMPLING OF PRIMARY COOLANT FOR IODINE ACTIVITY DELAYED DUE TO PERSONNEL ERROR.
EVENT DATE: 052090 REPORT DATE: 061990 NSSS: CE TYPE: PWR

(NSIC 218695) AT 1800 HOURS ON MAY 20, 1990 THE PLANT WAS OPERATING AT APPROXIMATELY 35% POWER, WITH THE PRIMARY COOLANT SYSTEM (PCS) AT 538 DEGREES F AND 2060 PSIA. TECHNICAL SPECIFICATION 4.2, TABLE 4.2.1 REQUIRES AN ISOTOPIC ANALYSIS OF THE PCS FOR IODINE WITHIN TWO TO SIX HOURS AFTER A THERMAL POWER INCREASE OF GREATER THAN 15% IN A ONE HOUR PERIOD. THERMAL POWER WAS INCREASED FROM 3.3% TO 23.6% BETWEEN 1100 HOURS AND 1200 HOURS ON MAY 20, 1990; HOWEVER, A PCS SAMPLE WAS NOT OBTAINED WITHIN THE REQUIRED INTERVAL. UPON DISCOVERY OF THE MISSED SAMPLE, A PCS SAMPLE WAS OBTAINED FOR ISOTOPIC ANALYSIS AT 0910 HOURS ON MAY 21, 1990. THIS EVENT WAS CAUSED BY A PERSONNEL ERROR INVOLVING A CHEMISTRY TECHNICIAN, AND WAS CONTRIBUTED TO BY INADEQUATE COMMUNICATION. AS CORRECTIVE ACTION, PROCEDURE GOP-5 WILL BE REVISED TO MORE CLEARLY STATE THAT CHEMISTRY TECHNICIANS ARE TO BE INFORMED OF THE HOURLY RATE OF POWER CHANGE AND EXPECTED POWER LEVEL WHEN REQUESTING TECHNICAL SPECIFICATION REQUIRED SAMPLING THAT IS POWER DEPENDENT. ADDITIONALLY, INFORMATION WILL BE PROVIDED IN THE TRAINING PROGRAM FOR CHEMISTRY TECHNICIANS REGARDING THE TYPES OF PLANT EVOLUTIONS THAT CAN POTENTIALLY IMPACT THEIR WORK ACTIVITIES, AND THE NEED TO INQUIRE ABOUT POWER LEVEL CHANGES WHEN CONTACTED BY OPERATIONS PERSONNEL TO PERFORM SAMPLING. THIS EVENT DID NOT INVOLVE FAILURE OF A COMPONENT OR SYSTEM.

[99] PEACH BOTTOM 2 DOCKET 50-277 LER 89-028 REV 01
UPDATE ON STANDBY GAS TREATMENT SYSTEM HEATER CONTROL RELAYS INSTALLED WITHOUT ENVIRONMENTAL QUALIFICATION DUE TO UNKNOWN CAUSE.
EVENT DATE: 110889 REPORT DATE: 062590 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 218681) ON NOVEMBER 8, 1989, FOLLOWING A REVIEW OF SAFETY RELATED RELAY APPLICATIONS IT WAS DETERMINED THAT THE STANDBY GAS TREATMENT SYSTEM (SBGT) HEATER CONTROL RELAYS WERE NOT QUALIFIED FOR THE ANALYZED POST LOSS OF COOLANT

ACCIDENT (LOCA) RADIATION ENVIRONMENT. AT 1335 HOURS, THE SBGT WAS DECLARED INOPERABLE AND A UNIT 2 PLANT SHUTDOWN WAS INITIATED IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS. BY 0100 HOURS, ON NOVEMBER 9, TEMPORARY RADIATION SHIELDING WAS INSTALLED AROUND THE RELAYS, THE SBGT DECLARED OPERABLE AND UNIT 2 RESTORED TO FULL POWER. THE RELAYS WERE SUBSEQUENTLY RELOCATED TO AN AREA IN WHICH THE CALCULATED POST LOCA RADIATION LEVELS DO NOT EXCEED THEIR 2.0E5 RAD RATING. THE ROOT CAUSE OF THIS EVENT WAS A LACK OF PROCEDURAL GUIDANCE TO CONTROL THE ENVIRONMENTAL QUALIFICATION ASPECTS OF THE MODIFICATION DESIGN PROCESS (CIRCA 1980). AN ASSESSMENT OF THE PEACH BOTTOM EQ PROGRAM WILL BE PERFORMED TO ENSURE THAT REQUIRED COMPONENTS HAVE BEEN INCLUDED. NO PREVIOUS SIMILAR LERS WERE IDENTIFIED.

[100] PEACH BOTTOM 2 DOCKET 50-277 LER 90-013
FAILURE TO PERFORM A TECHNICAL SPECIFICATION SURVEILLANCE DUE TO A PROCEDURAL DEFICIENCY.
EVENT DATE: 113089 REPORT DATE: 062790 NSSS: GE TYPE: BWR

(NSIC 218737) ON 5/30/90, DURING A TEST REVISION OF THE DRYWELL FIRE DETECTION INSTRUMENTATION CIRCUITS SURVEILLANCE TEST (ST), IT WAS IDENTIFIED THAT THE TESTS WERE NOT WRITTEN TO ALLOW TESTING IN ACCORDANCE WITH TECHNICAL SPECIFICATION (TECH SPEC) 4.14.C.1.C AND HAD NOT BEEN PERFORMED IN THE REQUIRED PERIODICITY. ST 15.4B-2 AND ST 15.5B-2 ARE REQUIRED TO BE PERFORMED EVERY 6 MONTHS. A REVIEW OF THE LAST PERFORMANCES INDICATED THAT THE UNIT 2 TESTS WENT OUT OF SURVEILLANCE ON 11/30/89. THE ROOT CAUSE WAS AN INCORRECT STANDARD PRACTICE OF ONLY PERFORMING THE TEST WHEN SHUTDOWN. CONSEQUENTLY, THE SURVEILLANCE TEST WAS NOT COMPLETED WITHIN ITS SPECIFIED INTERVAL. A REVIEW INDICATED THAT THE UNIT 2 AND UNIT 3 TESTS WERE IN SURVEILLANCE AT THE TIME OF DISCOVERY. THERE WERE NO ACTUAL SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT. THE TESTS HAVE BEEN REVISED TO CLARIFY THE TESTING REQUIREMENTS OF TECH SPEC 4.14.C.1.C AND OTHER SMOKE AND HEAT DETECTOR TESTS HAVE BEEN REVIEWED AND REVISED AS REQUIRED. THERE WERE TWO PREVIOUS SIMILAR EVENTS IDENTIFIED. A TASK FORCE HAS BEEN ESTABLISHED TO PERFORM A ROOT CAUSE ANALYSIS ON THE GENERIC ISSUE OF MISSED STS AND ASSESS THE CORRECTIVE ACTIONS TAKEN AS A RESULT OF PREVIOUS EVENTS.

[101] PERRY 1 DOCKET 50-440 LER 90-009
INOPERABILITY OF BOTH LOOPS OF CONTAINMENT SPRAY AND SUPPRESSION POOL COOLING DUE TO EQUIPMENT FAILURE DURING SURVEILLANCE TESTING.
EVENT DATE: 051790 REPORT DATE: 061890 NSSS: GE TYPE: BWR
VENDOR: LIMITORQUE CORP.

(NSIC 218739) ON 5/17/90 AT 1930, THE "B" RHR SYSTEM HEAT EXCHANGER BYPASS VALVE FAILED TO REPOSITION ON DEMAND, RENDERING THIS TRAIN OF RHR SYSTEM INOPERABLE FOR THE CONTAINMENT SPRAY AND SUPPRESSION POOL COOLING MODES OF OPERATION. AT THIS TIME, THE "A" TRAIN OF THE RHR SYSTEM WAS ALSO OUT OF SERVICE DUE TO SURVEILLANCE TESTING. OPERATORS THEN TERMINATED THE SURVEILLANCE AND RESTORED OPERABILITY OF THE "A" TRAIN OF THE RHR SYSTEM AT 1943. ROOT CAUSE OF THIS EVENT IS COMPONENT FAILURE. THE VALVE STEM NUT FAILED DUE TO EXCESS WEAR, POSSIBLY CAUSED BY INADEQUATE STEM LUBRICATION. TWO DAYS PREVIOUS TO THE EVENT, THE NORMAL PREVENTIVE MAINTENANCE TASK TO INSPECT AND LUBRICATE THE STEM WAS COMPLETED. DURING THIS ACTIVITY IT WAS OBSERVED THAT THE STEM LUBRICATION WAS MINIMAL. DURING STEM NUT REPLACEMENT, VALVE INSPECTION, RETEST, AND MOTOR OPERATOR VALVE ANALYSIS AND TESTING SYSTEM TESTING, NO OTHER VALVE PROBLEMS WERE IDENTIFIED. CORRECTIVE ACTIONS TAKEN AT THE TIME OF THE EVENT INCLUDED REPLACING THE STEM NUT, CONDUCTING A VALVE INSPECTION TO DETERMINE THE CAUSE OF FAILURE AND PERFORMING THE TECH SPEC OPERABILITY SURVEILLANCE. A MONITORING SCHEDULE HAS BEEN DEVELOPED TO VISUALLY INSPECT THE VALVE FOR DEGRADATION OF THE STEM NUT ON A PERIODIC BASIS, UNTIL SUCH TIME THAT SYSTEM ENGINEERING PERSONNEL CONFIRM THAT THE PROBLEM IS NOT RECURRING.

[102] PERRY 1 DOCKET 50-440 LER 90-010
INADEQUATE PROCEDURES IN CONJUNCTION WITH ROD CONTROL AND INFORMATION SYSTEM MALFUNCTION RESULT IN TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 052190 REPORT DATE: 062090 NSSS: GE TYPE: BWR

VENDOR: GENERAL ELECTRIC CO.

(NSIC 218740) ON MAY 2, 1990 AT 1950, PLANT OPERATORS FAILED TO IMPLEMENT TECHNICAL SPECIFICATION ACTION REQUIREMENTS FOR INOPERABLE CONTROL ROD SCRAM ACCUMULATORS. A FAILURE OF THE ROD CONTROL AND INFORMATION SYSTEM (RC&IS) RESULTED IN THE INABILITY TO RECEIVE UPDATED LEAK DETECTOR AND PRESSURE DETECTOR INFORMATION REQUIRED FOR ACCUMULATOR OPERABILITY. WITH MORE THAN ONE CONTROL ROD SCRAM ACCUMULATOR INOPERABLE, TECHNICAL SPECIFICATIONS REQUIRE THE ASSOCIATED CONTROL RODS TO BE DECLARED INOPERABLE, AND IMMEDIATE VERIFICATION OF CONTROL ROD DRIVE PUMP OPERATION. APPROXIMATELY TWO HOURS LATER, OPERATORS REALIZED THAT THIRTY-TWO CONTROL ROD SCRAM ACCUMULATORS WERE INOPERABLE AND IMPLEMENTED THE REQUIRED ACTIONS. THE CAUSE OF THIS EVENT WAS INADEQUATE PROCEDURES. ALTHOUGH EQUIPMENT MALFUNCTION INITIATED THIS EVENT, LACK OF PROCEDURAL GUIDANCE FOR OPERATOR RESPONSE TO THIS PARTICULAR TYPE OF FAILURE RESULTED IN A TECHNICAL SPECIFICATION VIOLATION. TO PREVENT RECURRENCE, OPERATIONS PROCEDURES HAVE BEEN REVISED TO INCLUDE MORE DETAILED GUIDANCE TO ASSIST OPERATORS IN DETERMINING CONTROL ROD AND CONTROL ROD SCRAM ACCUMULATOR OPERABILITY. AN INVESTIGATION INTO POSSIBLE TECHNICAL SPECIFICATION CHANGES FOR IMPROVED CLARIFICATION IS ALSO BEING CONDUCTED.

[103] PERRY 1 DOCKET 50-440 LER 90-011
PROCEDURAL DEFICIENCY RESULTS IN ECCS SYSTEM INOPERABILITY IN EXCESS OF TECHNICAL SPECIFICATION LIMITATIONS AND ENTRY INTO TECHNICAL SPECIFICATION 3.0.3.
EVENT DATE: 052190 REPORT DATE: 062090 NSSS: GE TYPE: BWR

(NSIC 218741) ON MAY 21, 1990, IT WAS DETERMINED THAT ON THREE PREVIOUS OCCASIONS DIVISIONAL LOW PRESSURE COOLANT INJECTION (LPCI) AND LOW PRESSURE CORE SPRAY (LPCS) SYSTEMS HAD BEEN INOPERABLE FOR A PERIOD OF TIME EXCEEDING TECHNICAL SPECIFICATION (3.5.1.A) ACTION LIMITS. IN ADDITION, IT WAS ALSO DETERMINED THAT ON TWO OF THESE OCCASIONS EMERGENCY CORE COOLING SYSTEMS (ECCS) WERE INOPERABLE IN COMBINATIONS NOT ADDRESSED IN TECHNICAL SPECIFICATIONS (TS), THEREBY REQUIRING ENTRY INTO TS 3.0.3. THE ROOT CAUSE OF THESE EVENTS WAS A PROCEDURE DEFICIENCY. SYSTEM OPERATING INSTRUCTION (SOI-P47),"CONTROL COMPLEX CHILLED WATER SYSTEM", ALLOWED A LINEUP DURING MAINTENANCE ACTIVITIES WHICH PRECLUDED ADEQUATE FLOW CONDITIONS FOR THE EMERGENCY CLOSED COOLING (ECC) SYSTEM DURING A LOSS OF COOLANT ACCIDENT (LOCA) EVENT. IN ORDER TO PREVENT RECURRENCE, SOI-P47 WAS MODIFIED TO ENSURE THAT THE APPLICABLE CHILLER BYPASS VALVE WOULD REMAIN OPEN IN THE EVENT OF A LOCA WHILE THE CHILLER IS ISOLATED. ADDITIONALLY, ALL SYSTEM OPERATING PROCEDURES WERE REVIEWED TO ENSURE MINIMUM FLOW PUMP REQUIREMENTS ARE MET DURING ABNORMAL LINEUPS; NO ADDITIONAL PROCEDURAL DEFICIENCIES WERE DISCOVERED. THIS EVENT WILL BE REVIEWED WITH ALL LICENSED OPERATORS AS PART OF LICENSED OPERATOR REQUALIFICATION TRAINING.

[104] PERRY 1 DOCKET 50-440 LER 90-012
INOPERABILITY OF CONTROL ROOM EMERGENCY RECIRCULATION SYSTEM DUE TO PERSONNEL ERROR RESULTS IN ENTRY INTO TECHNICAL SPECIFICATION 3.0.3.
EVENT DATE: 060790 REPORT DATE: 070690 NSSS: GE TYPE: BWR
VENDOR: CARRIER AIR CONDITIONING CO.

(NSIC 218806) FROM 0400 ON 6/5/90 UNTIL 0035 ON 6/8/90 BOTH TRAINS OF THE CONTROL ROOM HEATING, VENTILATION AND AIR CONDITIONING (CRHVAC) SYSTEM WERE INOPERABLE FOR THE EMERGENCY RECIRCULATION MODE DUE TO THE INOPERABILITY OF BOTH SUPPORTING CHILLERS. THIS CONDITION VIOLATED TECH SPEC 3.7.2., PLACING THE PLANT IN TECH SPEC 3.0.3 AND POTENTIALLY PREVENTED A SYSTEM RESPONSIBLE FOR CONTROL ROOM HABITABILITY FROM FULFILLING ITS SAFETY FUNCTION. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR, INATTENTION TO DETAIL. TWO RECORDED PARAMETERS WERE NOT NOTED AS BEING OUTSIDE OF ADMINISTRATIVE LIMITS BY THE OPERATORS TAKING THE READINGS OR BY THE SUPERVISORS REVIEWING THIS INFORMATION. A WORK REQUEST TO CORRECT THE EQUIPMENT PROBLEM WAS INAPPROPRIATELY CANCELLED DUE TO INSUFFICIENT RESEARCH INTO THE PROBLEM. CONTRIBUTING FACTORS TO THIS EVENT WERE POOR GUIDE VANE LINKAGE DESIGN AND INEFFECTIVE CORRECTIVE ACTIONS TAKEN AS A RESULT OF A SIMILAR EVENT DOCUMENTED IN LER 80-19. THE CORRECTIVE ACTIONS TO PREVENT RECURRENCE INCLUDED MANAGEMENT PROVIDING DIRECTION CONCERNING THE IMPORTANCE OF TAKING AND REVIEWING PLANT OPERATING DATA. RESPONSIBLE SYSTEM ENGINEERING WILL BE INSTRUCTED ON THE

NEED TO THOROUGHLY RESEARCH WORK ORDERS. THE "A" CHILLER WAS REPAIRED AND THE "C" CHILLER SHAFT WILL BE MODIFIED IN THE SAME MANNER.

[105] PILGRIM 1 DOCKET 50-293 LER 90-009
AUTOMATIC CLOSING OF THE OUTBOARD REACTOR WATER CLEANUP SYSTEM ISOLATION VALVES.
EVENT DATE: 060690 REPORT DATE: 070590 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 218789) ON 6/6/90 AT 1655 HOURS, A PORTION OF THE PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM (PCIS) ACTUATED. THE ACTUATION RESULTED IN THE AUTOMATIC CLOSING OF THE OUTBOARD PRIMARY CONTAINMENT SYSTEM GROUP 6 (I.E. REACTOR WATER CLEANUP SYSTEM) ISOLATION VALVES AND THE AUTOMATIC TRIP OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM PUMP "B". THE OPERATION OF THE RWCU SYSTEM WAS TEMPORARILY INTERRUPTED. THE INBOARD RWCU SYSTEM ISOLATION VALVE REMAINED OPEN AS EXPECTED. THE CAUSE OF THE ACTUATION WAS THE FAILURE OF A LOGIC RELAY COIL. WHEN THE COIL FAILED EXCESS CURRENT IN THE CIRCUIT ENERGIZING THE COIL CAUSED THE CIRCUIT'S FUSE TO BLOW AND DE-ENERGIZED THE CIRCUIT. THE ACTUATION WAS THE EXPECTED RESULT OF THE CIRCUIT BECOMING DE-ENERGIZED. WHILE THE EXACT CAUSE OF THE COIL FAILURE IS UNKNOWN IT IS BELIEVED TO BE A RANDOM ISOLATED OCCURRENCE. ADDITIONAL INVESTIGATION IS BEING CONDUCTED TO DETERMINE EXACT CAUSE ADDITIONAL CORRECTIVE ACTION WILL BE DICTATED BY RESULTS OF THE INVESTIGATION. THE RELAY IS A TYPE CR120A RELAY (120 VAC COIL) MANUFACTURED BY THE GENERAL ELECTRIC COMPANY. THE RELAY COIL AND FUSE WERE REPLACED. THE RWCU SYSTEM WAS RETURNED TO SERVICE ON JUNE 6, 1990 AT APPROXIMATELY 2200 HOURS. THE EVENT OCCURRED DURING POWER OPERATION WITH THE REACTOR MODE SELECTOR SWITCH IN THE RUN POSITION. THE REACTOR VESSEL (RV) PRESSURE WAS 1034 PSIG WITH THE RV WATER TEMPERATURE AT APPROXIMATELY 548F.

[106] POINT BEACH 1 DOCKET 50-266 LER 90-005
STEAM GENERATOR LOW-LOW LEVEL REACTOR TRIP DURING COLD SHUTDOWN.
EVENT DATE: 051090 REPORT DATE: 060890 NSSS: WE TYPE: PWR

(NSIC 218509) ON MAY 10, 1990 AT 1608, THE PBNP UNIT 1 REACTOR PROTECTION SYSTEM (RPS) GENERATED A REACTOR TRIP SIGNAL UPON RECEIPT OF A 2/3 TRIP LOGIC FOR LOW-LOW STEAM GENERATOR WATER LEVEL. WHEN THE EVENT OCCURRED, UNIT 1 WAS NEARING THE END OF A REFUELING OUTAGE AND WAS IN A COLD SHUTDOWN (CSD) CONDITION. THE REACTOR TRIP BREAKERS WERE OPEN AT THE TIME THE TRIP SIGNAL WAS GENERATED. PBNP PERSONNEL WERE MAKING PREPARATIONS FOR COLD ROD DROP TESTING. DUMMY TEST SIGNALS HAD BEEN INSERTED INTO 4 OF THE 6 STEAM GENERATOR NARROW RANGE LEVEL CHANNELS IN ORDER TO SIMULATE STEAM GENERATOR LEVELS ABOVE THE LOW-LOW LEVEL TRIP SETPOINT. THE DUMMY TEST SIGNALS WERE BEING SUPPLIED BY A TEST CART WHICH WAS POWERED BY A LIGHTING RECEPTACLE THROUGH AN EXTENSION CORD. A TECHNICIAN WORKING IN THE AREA DISCONNECTED THE EXTENSION CORD FROM THE CART. WITH THE LOSS OF THE TEST SIGNALS, THE RPS RECEIVED LOW-LOW STEAM GENERATOR LEVEL SIGNALS, AND GENERATED THE TRIP FUNCTION. BECAUSE THE REACTOR WAS IN A CSD CONDITION WITH REACTOR TRIP BREAKERS OPEN, AND THE REACTOR PROTECTION SYSTEM FUNCTIONED AS DESIGNED, THERE WERE NO SAFETY IMPLICATIONS. A MODIFICATION TO THE TEST EQUIPMENT IS IN PROGRESS WHICH WILL MAKE DISCONNECTION ERRORS IN THIS APPLICATION HIGHLY UNLIKELY.

[107] PRAIRIE ISLAND 1 DOCKET 50-282 LER 90-006
AUTO-START OF SPENT FUEL POOL SPECIAL VENTILATION SYSTEM DUE TO PROCEDURE INADEQUACY.
EVENT DATE: 051790 REPORT DATE: 061590 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)

(NSIC 218514) ON MAY 17, 1990, BOTH UNITS WERE OPERATING AT 100% POWER. WHEN AN I&C TECHNICIAN HAD COMPLETED WORK ON RADIATION MONITOR CHANNEL R-25, HE RESTORED THE CHANNEL TO NORMAL AND CALLED FOR A RADIATION PROTECTION SPECIALIST TO TEST THE MONITOR WITH A TEST SOURCE. CHANNEL R-25 WAS PLACED IN RESET TO PREVENT ACTUATION OF THE SPENT FUEL POOL SPECIAL VENTILATION SYSTEM AND THE MONITOR WAS THEN TESTED. WHEN MONITOR R-25 WAS TESTED WITH THE TEST SOURCE NEARBY REDUNDANT MONITOR R-31 RESPONDED TO THE RADIATION SOURCE AND ACTUATED NO. 122 SPENT FUEL POOL SPECIAL VENTILATION SYSTEM. THE CONTROL ROOM OPERATORS AND THE TECHNICIANS

REALIZED IMMEDIATELY WHAT HAD HAPPENED; THE TEST SOURCE WAS REMOVED, THE MONITOR WAS RESET, SPENT FUEL POOL SPECIAL VENTILATION SYSTEM WAS SHUT DOWN AND THE SPENT FUEL POOL NORMAL VENTILATION SYSTEM WAS RESTORED. CAUSE OF THE EVENT WAS PROCEDURE INADEQUACY. THE AUTHOR OF THE WORK PROCEDURE HAD CONSIDERED THE EFFECT OF THE MONITOR BUGGING OPERATION, BUT HAD WRITTEN THE CAUTION AS A NOTE INSTEAD OF A PROCEDURAL SIGNOFF STEP. AS A RESULT, THE TECHNICIAN DID NOT DISABLE THE REDUNDANT MONITOR BEFORE THE BUGGING OPERATION.

[108] QUAD CITIES 1 DOCKET 50-254 LER 90-001 REV 01
UPDATE ON SEVEN PATHWAYS WERE NOT INCLUDED IN THE TYPE "B" AND "C" LOCAL LEAK RATE TESTING PROGRAM DUE TO A RECENT INTERPRETATION OF 10CFR50 APPENDIX J.
EVENT DATE: 120889 REPORT DATE: 060190 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 218505) ON DECEMBER 8, 1989, AT 1130 HOURS, UNIT ONE WAS AT 95% RATED CORE THERMAL POWER. A STUDY TO IMPROVE THE TYPE "B" AND "C" LOCAL LEAK RATE TESTING (LLRT) PROGRAM AT QUAD CITIES STATION CONCLUDED THAT SEVEN PATHWAYS SHOULD BE ADDED TO THE LLRT PROGRAM. THESE TEST VOLUMES INCLUDED: THE REACTOR BUILDING CLOSED COOLING (RBCCW) INLET/OUTLET; THE CORE SPRAY DISCHARGE LINES; THE INSTRUMENT AIR TO THE DRYWELL; THE SERVICE AIR TO THE DRYWELL; THE STANDBY LIQUID CONTROL SYSTEM; THE CLEAN DEMINERALIZED WATER TO THE DRYWELL; AND THE DRYWELL AIR SAMPLING LINES. THESE PATHWAYS WERE EXCLUDED FROM THE STATION'S LLRT PROGRAM DUE TO AN INTERPRETATION OF 10CFR50 APPENDIX J WHICH DID NOT CONSIDER THESE TEST VOLUMES APPLICABLE TO THE TYPE "C" TESTING REQUIREMENTS. DUE TO A RECENT INTERPRETATION OF 10CFR50 APPENDIX J WITH RESPECT TO LICENSING DESIGN CRITERIA, THESE VOLUMES SHALL BE ADDED TO THE STATION'S TYPE "B" AND "C" LLRT PROGRAM. MODIFICATION OF THE SYSTEMS WILL BE PERFORMED AS NECESSARY TO INSTALL THE REQUIRED VENTS AND TEST TAPS TO PERFORM THE TYPE "C" TESTING. THIS REPORT IS BEING SUBMITTED AS A VOLUNTARY REPORT.

[109] QUAD CITIES 1 DOCKET 50-254 LER 89-024 REV 01
UPDATE ON TURBINE TRIP DUE TO HIGH REACTOR WATER LEVEL SIGNAL CAUSED BY UNDETECTED TRIPPED LEVEL SWITCH.
EVENT DATE: 121489 REPORT DATE: 060190 NSSS: GE TYPE: BWR

(NSIC 218496) ON DECEMBER 14, 1989 AT APPROXIMATELY 0315 HOURS, UNIT ONE WAS OPERATING IN THE RUN MODE AT 35 PERCENT OF RATED CORE THERMAL POWER. THE MAIN TURBINE UNEXPECTEDLY TRIPPED FOLLOWING THE ISOLATION OF REACTOR WATER LEVEL SWITCH (LITS) 1-263-59A. THE CAUSE OF THE TRIP WAS A RESULT OF THE "B" CHANNEL LITS HAVING BEEN PREVIOUSLY REPLACED WITH A SWITCH THAT OPERATED THE REVERSE OF WHAT WAS REQUIRED. THUS, WHEN THE "A" CHANNEL LITS WAS ISOLATED, THE TURBINE TRIP LOGIC WAS COMPLETED. THE ALARM THAT SHOULD HAVE BEEN ANNUNCIATED DUE TO THE CONDITION OF "B" LITS, WAS FOUND TO HAVE BEEN INACTIVATED AS A RESULT OF ITS SIGNAL LEADS BEING INADVERTENTLY LEFT DETERMINATED DURING A MODIFICATION INSTALLATION. CORRECTIVE ACTIONS INCLUDED REPLACING THE "B" LITS SWITCH, RETERMINATING THE ALARM LEADS, CHECKING THE OTHER UNIT, TESTING, AND A PROCEDURE REVISION. FURTHER CORRECTIVE ACTIONS ARE IN THE PROCESS. THIS REPORT IS BEING SUBMITTED AS A VOLUNTARY REPORT.

[110] QUAD CITIES 1 DOCKET 50-254 LER 90-010
CLEAN-UP ISOLATION ON HIGH TEMPERATURE DUE TO CHECK VALVES LEAKING.
EVENT DATE: 052290 REPORT DATE: 062190 NSSS: GE TYPE: BWR
VENDOR: CRANE VALVE CO.

(NSIC 218694) ON MAY 22, 1990 AT 1242 HOURS, UNIT ONE WAS IN THE RUN MODE AT 97 PERCENT OF RATED CORE THERMAL POWER. WHILE RETURNING THE REACTOR WATER CLEAN-UP (RWCU) SYSTEM TO SERVICE, A NON-REGENERATIVE HEAT EXCHANGER (NRHX) HIGH TEMPERATURE ALARM WAS RECEIVED. ALTHOUGH NOT AN ENGINEERED SAFETY FEATURE (ESF), GROUP III ISOLATION, THIS RESULTED IN A CHALLENGE TO THE ESF LOGIC AND A SYSTEM ISOLATION. AN EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE NOTIFICATION WAS MADE AT 1405 HOURS IN ACCORDANCE WITH 10CFR50.72(B)(2)(II). THE CAUSE WAS DETERMINED TO BE DUE TO SYSTEM CHECK VALVES LEAKING. WORK REQUESTS WILL BE WRITTEN TO INSPECT AND REPAIR THE RWCU RECIRC PUMPS DISCHARGE CHECK VALVES, 1-1201-87A/87B, AND THE

FEEDWATER RETURN CHECK VALVE, 1-1201-81. THIS REPORT IS BEING SUBMITTED TO COMPLY WITH THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV).

[111] QUAD CITIES 1 DOCKET 50-254 LER 90-011
DIESEL FIRE PUMP OUT OF SERVICE LONGER THAN SEVEN DAYS TO INSTALL NEW SUCTION LINE.
EVENT DATE: 061190 REPORT DATE: 071190 NSSS: GE TYPE: BWR

(NSIC 218816) ON 6/11/90 AT 1500 HOURS, QUAD CITIES UNIT ONE AND UNIT TWO WERE IN THE RUN MODE AT 100 PERCENT OF RATED CORE THERMAL POWER. AT THIS TIME, THE 1/2A DIESEL FIRE PUMP (FP) EXCEEDED THE SEVEN DAY REPORTING CRITERIA DUE TO ITS BEING TAKEN OUT OF SERVICE (OOS) FOR MAINTENANCE ON JUNE 4, 1990. THE OOS WAS NEEDED SO THE RELIABILITY OF THE PUMP COULD BE INCREASED DURING LOW WATER LEVEL CONDITIONS. A FIVE FOOT SUCTION EXTENSION WAS TO BE ADDED. THE 1/2B DIESEL FIRE PUMP REMAINED IN SERVICE DURING THE DURATION THAT THE 1/2A DIESEL FIRE PUMP WAS OUT OF SERVICE AND REDUNDANT FIRE SUPPRESSION WATER WAS AVAILABLE THROUGH THE SERVICE WATER SYSTEM. FOLLOWING COMPLETION OF THE ADDITICN OF THE FIVE FOOT SUCTION EXTENSION, THE 1/2A DIESEL FIRE PUMP WAS SUCCESSFULLY TESTED ON JUNE 29, 1990 AND RETURNED TO SERVICE. THIS REPORT IS BEING SUBMITTED TO COMPLY WITH TECHNICAL SPECIFICATION 3.12.B.2.

[112] QUAD CITIES 2 DOCKET 50-265 LER 90-007
PARTIAL LOSS OF ROD POSITION INDICATION SYSTEM DUE TO BLOWN FUSE.
EVENT DATE: 053190 REPORT DATE: 070290 NSSS: GE TYPE: BWR

(NSIC 218698) ON MAY 31, 1990 AT 1615 HOURS, UNIT TWO WAS IN THE RUN MODE AT 100 PERCENT OF RATED CORE THERMAL POWER. AT THIS TIME, THE FOLLOWING ALARMS WERE RECEIVED ON THE 902-5 PANEL: A-3, ROD DRIFT; B-3, ROD WORTH MIN. BLOCK; E-3, ROD OVERTRAVEL; AND G-5, RPIS INOPERATIVE. ROD POSITION INDICATION (RPIS) HAD BEEN LOST ON THE LOWER HALF OF THE FULL CORE DISPLAY AND 4 ROD DISPLAY. AN EQUIPMENT OPERATOR (EO) WAS DISPATCHED TO THE 902-27 AND 28 PANELS TO INVESTIGATE. THE CAUSE OF THE EVENT WAS FOUND TO BE DUE TO A BLOWN POWER SUPPLY FUSE IN THE 902-27 PANEL. IMMEDIATE CORRECTIVE ACTION INVOLVED THE UNIT OPERATOR MONITORING AVERAGE AND LOCAL POWER RANGE MONITORS (APRM) (LPRM) TO ENSURE STEADY STATE POWER LEVELS WERE MAINTAINED. THE BLOWN FUSE WAS REPLACED AT 1643 HOURS WHICH RESTORED FULL ROD POSITION INDICATION. FURTHER CORRECTIVE ACTION WILL INCLUDE SUBMITTING A TECHNICAL SPECIFICATION CHANGE AND UPGRADING AND EXISTING PROCEDURE. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(I)(B).

[113] QUAD CITIES 2 DOCKET 50-265 LER 90-008
FAILURE OF THE AUTOMATIC FUNCTION OF THE HIGH PRESSURE COOLANT INJECTION FLOW CONTROLLER DUE TO UNKNOWN CAUSES.
EVENT DATE: 060290 REPORT DATE: 070290 NSSS: GE TYPE: BWR
VENDOR: YOKOGAWA CORPORATION OF AMERICA

(NSIC 218699) AT 1723 HOURS ON JUNE 2, 1990, UNIT TWO WAS IN THE RUN MODE AT 100 PERCENT POWER. THE HIGH PRESSURE COOLANT INJECTION (HPCI) PUMP FLOW INDICATING CONTROLLER (FIC), 2-2340-1, WAS FOUND IN THE MANUAL MODE AND THE RED FAIL LAMP ON THE FIC WAS LIT. HPCI WAS DECLARED INOPERABLE AND AN OUTAGE REPORT WAS INITIATED. THE CAUSE OF THE FAILURE COULD NOT BE DETERMINED. THE POWER SUPPLY FUSE WAS REMOVED AND REINSTALLED, THEN THE FIC RETURNED TO NORMAL. OPERATING PERSONNEL BEGAN OPERABILITY TESTING OF HPCI, BUT STOPPED AT 2105 HOURS DUE TO A STEAM LEAK AT THE FLANGED CONNECTION OF THE TURBINE STOP VALVE-BELOW SEAT DRAIN ORIFICE, 2-2301-98. THE GASKET WAS REPLACED AT THE FLANGE CONNECTION. AT 0510 HOURS ON JUNE 3, 1990, THE HPCI OPERABILITY TEST WAS SUCCESSFULLY COMPLETED. NRC NOTIFICATION OF THE EVENT VIA THE EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE WAS COMPLETED AT 2118 HOURS ON JUNE 2, 1990, TO COMPLY WITH THE REQUIREMENTS OF 10CFR50.72(B)(2)(III).

[114] RIVERBEND 1 DOCKET 50-458 LER 90-022
FAILURE TO DECLARE THE HIGH PRESSURE CORE SPRAY DIESEL GENERATOR INOPERABLE AS REQUIRED BY TECHNICAL SPECIFICATION 3.5.1.
EVENT DATE: 051890 REPORT DATE: 061890 NSSS: GE TYPE: BWR

(NSIC 218742) ON 5/18/90 WITH THE UNIT AT 100 PERCENT POWER (OPERATIONAL CONDITION 1), THE HPCS SYSTEM WAS DECLARED INOPERABLE AND A 14 DAY SHUTDOWN LIMITING CONDITION FOR OPERATION (LCO) WAS ENTERED DUE TO AN INOPERATIVE SUPPRESSION POOL LEVEL TRANSMITTER. HOWEVER, TECHNICAL SPECIFICATION (TS) 3.5.1, ACTION C.1 ALSO REQUIRES DECLARING THE HPCS DIESEL GENERATOR INOPERABLE AND THIS REQUIREMENT WAS OVERLOOKED. THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(I)(B) AS OPERATION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. THE ROOT CAUSE OF THIS EVENT WAS THE FAILURE TO RECOGNIZE THE REQUIREMENT TO DECLARE THE HPCS DIESEL GENERATOR INOPERABLE. CORRECTIVE ACTIONS INCLUDED TRAINING OF ALL LICENSED REACTOR OPERATORS (ROS) AND SENIOR REACTOR OPERATORS (SROS) ON THIS INCIDENT WITH A REVIEW OF TECHNICAL SPECIFICATION REQUIREMENTS. FURTHERMORE, THE PLANT MANAGER HAS ISSUED A MEMORANDUM EMPHASIZING THE IMPORTANCE OF MEETING ALL REGULATORY REQUIREMENTS. DURING THIS EVENT, THE HPCS SYSTEM WOULD STILL HAVE BEEN AVAILABLE TO FULFILL ITS SAFETY FUNCTION AND ALL REDUNDANT BACKUP SYSTEMS WERE FULLY OPERABLE. THEREFORE, THIS EVENT DID NOT ADVERSELY AFFECT THE HEALTH AND SAFETY OF THE PUBLIC.

[115] RIVERBEND 1 DOCKET 50-458 LER 90-023
STANDBY GAS TREATMENT AND ANNULUS MIXING SYSTEMS NOT INITIATED DUE TO FAILED RADIATION MONITOR SAMPLE PUMP AND FLOW SWITCH.
EVENT DATE: 052390 REPORT DATE: 062290 NSSS: GE TYPE: BWR
VENDOR: STANDARD SWITCH

(NSIC 218743) AT APPROXIMATELY 0730 ON 05/23/90, WITH THE PLANT AT 100 PERCENT POWER (OPERATIONAL CONDITION 1), THE AUXILIARY BUILDING OPERATOR DISCOVERED THAT THE ANNULUS VENTILATION RADIATION MONITOR, 1RMS*RE11A WAS INOPERATIVE. THE OPERATOR REPORTED THIS DISCOVERY TO CONTROL ROOM PERSONNEL WHO OBSERVED THAT THE CONTROL ROOM INDICATION OF 1RMS*RE11A ERRONEOUSLY INDICATED THAT IT WAS OPERABLE. FOLLOWING THIS DISCOVERY, THE OPERATING LOGS WERE REVIEWED AND IT WAS DETERMINED THAT THE ANNULUS MIXING AND STANDBY GAS TREATMENT (SGTS) SYSTEMS HAD NOT BEEN INITIATED WITHIN THE TIME PERIOD (1 HOUR) REQUIRED BY TECHNICAL SPECIFICATION (TS) 3.3.2, TABLE 3.3.2.1-1.2.D, ACTION 29. THEREFORE, THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(I)(B) AS OPERATION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. THE ROOT CAUSE OF THIS EVENT WAS THE FAILURE OF THE RADIATION MONITOR SAMPLE PUMP AND THE FAILURE OF THE SAMPLE PUMP FLOW SWITCH TO ACTUATE. THE FAILURE OF THE FLOW SWITCH DENIED THE CONTROL ROOM OPERATORS THE INDICATION NEEDED TO CONFIRM THE OPERABILITY STATUS OF THE SAMPLE PUMP. DURING THE PERIOD OF TIME THAT 1RMS*RE11A WAS OUT OF SERVICE, A REDUNDANT RADIATION MONITOR WAS AVAILABLE. THEREFORE, THIS EVENT DID NOT ADVERSELY AFFECT THE HEALTH AND SAFETY OF THE PUBLIC.

[116] ROBINSON 2 DOCKET 50-261 LER 90-007
REACTOR TRIP DUE TO FAILURE OF FEEDWATER REGULATING VALVE.
EVENT DATE: 051790 REPORT DATE: 061890 NSSS: WE TYPE: PWR
VENDOR: COPES-VULCAN, INC.

(NSIC 218697) ON MAY 17, 1990, AT 0606 HOURS. AN AUTOMATIC REACTOR TRIP WAS RECEIVED FROM A STEAM FLOW-FEEDWATER FLOW MISMATCH COINCIDENT WITH A LOW LEVEL IN STEAM GENERATOR (SG) "B". THE CAUSE OF THE EVENT WAS AN EQUIPMENT FAILURE IN THAT THE SG "B" FEEDWATER REGULATING VALVE (FRV) MALFUNCTIONED IN A MANNER WHICH IMPEDED FLOW TO SG "B". DISASSEMBLY AND INSPECTION REVEALED THAT A SPRING PIN HAD SHEARED WHICH ALLOWED THE VALVE PLUG TO UNTHREAD FROM THE VALVE STEM. A POSSIBLE CONTRIBUTING FACTOR WAS A PROCEDURAL DEFICIENCY IN THAT THE VENDOR RECOMMENDED STEM-TO-PLUG TORQUE REQUIREMENT WAS NOT PROVIDED WITHIN THE FRV CORRECTIVE MAINTENANCE PROCEDURE. THE STEM AND PLUG WERE TORQUED TO THE REQUIRED VALUE AND A NEW STEM AND LONGER SPRING PIN WERE INSTALLED. THE FRVS FOR SGS "A" AND "C" WERE ALSO DISASSEMBLED AND INSPECTED, WITH PRECAUTIONARY REPLACEMENT OF BOTH SPRING PINS, AND REPLACEMENT OF THE VALVE STEM FOR THE SG "C" FRV. A PROCEDURE REVISION HAS BEEN MADE TO INCORPORATE THE RECOMMENDED TORQUE VALUE.

THIS EVENT WAS REPORTED VIA THE EMERGENCY NOTIFICATION SYSTEM AT 0717 HOURS PURSUANT TO 10CFR50.72 (B)(2)(II). THIS LICENSEE EVENT REPORT IS SUBMITTED PURSUANT TO 10CFR50.73 (A)(2)(IV).

[117] SALEM 1 DOCKET 50-272 LER 89-031 REV 01
UPDATE ON WASTE GAS SYSTEM OXYGEN GREATER THAN 2% OXYGEN FOR MORE THAN 48 HOURS.
EVENT DATE: 101989 REPORT DATE: 062690 NSSS: WE TYPE: PWR

(NSIC 218748) OXYGEN CONCENTRATION WITHIN THE WASTE GAS HOLDUP (WGH) SYSTEM WAS GREATER THAN 2% FOR MORE THAN 48 HOURS BETWEEN 10/17/89 AND 10/19/89 AND BETWEEN 6/3/90 AND 6/6/90 CONTRARY TO THE REQUIREMENTS OF TECH SPEC ACTION STATEMENT 3.11.2.5.A. THE MAXIMUM OXYGEN CONCENTRATION OBSERVED IN ANY WASTE GAS DECAY TANK (WGDT) DURING THE OCTOBER 1989 EVENT WAS 3.2% (IN NO. 11 WGDT) AND 3.0% DURING THE JUNE 1990 EVENT (ALSO IN NO. 11 WGDT). NO WGDT CONTAINED A CONCENTRATION OF OXYGEN ABOVE THE 2% LIMIT FOR MORE THAN APPROXIMATELY 30 HOURS. EFFORTS TO REDUCE THE OXYGEN CONCENTRATION WERE IMMEDIATELY IMPLEMENTED UPON IDENTIFICATION OF THE HIGH CONCENTRATIONS IN THE WGDT'S. IT HAS BEEN CONCLUDED THAT OXYGEN IS ENTERING THE WASTE GAS HOLDUP SYSTEM VIA THE CVCS HOLDUP TANKS. SAMPLES OF NO. 13 CVCS HOLDUP TANK COVER GAS HAD OXYGEN CONCENTRATIONS AS HIGH AS 3.5% ON 10/28/89. SAMPLES OF THE OTHER TANK COVER GAS SOURCES, TAKEN 11/12/89 THROUGH 11/14/89, (OTHER THAN NO. 11 CVCS HOLDUP TANK) TO THE WASTE GAS HOLDUP SYSTEM INDICATED CONCENTRATIONS OF LESS THAN 0.1%. INVESTIGATION AS TO THE SOURCE OF OXYGEN TO THE CVCS HOLDUP TANKS AND THUS ULTIMATELY THE ROOT CAUSE OF THIS EVENT IS CONTINUING. ASSOCIATED TRANSMITTERS AND REGULATING VALVES HAVE BEEN RECALIBRATED. WORK ORDERS HAVE BEEN ISSUED TO SUPPORT THE INVESTIGATION.

[118] SALEM 1 DOCKET 50-272 LER 90-018
CONTROL ROOM VENTILATION SWITCH (1R1B) DUE TO EQUIPMENT CONCERNS.
EVENT DATE: 052890 REPORT DATE: 062790 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)
VENDOR: LFE CORP.

(NSIC 218755) ON 5/28/90 AT 0834 HOURS, WITH THE PLANT IN MODE 5 (COLD SHUTDOWN), THE CONTROL ROOM AIR INTAKE RADIATION MONITORING SYSTEM (RMS) MONITOR (1R1B) ALARM CIRCUITRY FAILED. THIS RESULTED IN THE AUTOMATIC SWITCHING OF THE CONTROL ROOM VENTILATION FROM NORMAL OPERATION TO THE ACCIDENT MODE OF OPERATION (100% RECIRCULATION) FOR BOTH SALEM UNIT 1 AND SALEM UNIT 2 (BY DESIGN). THE SWITCHING OF THE CONTROL ROOM VENTILATION SYSTEM TO THE EMERGENCY MODE OF OPERATION IS AN ENGINEERED SAFETY FEATURE (ESF). THE SWITCHING OF THE VENTILATION RESULTED FROM A SPURIOUS HIGH CHANNEL SPIKE. THE 1R1B CHANNEL DETECTOR IS AN LFE-TRAPELO MD12C(V-11) GEIGER-MUELLER TUBE. A CHANNEL FUNCTIONAL TEST WAS SUCCESSFULLY COMPLETED ON 6/11/90. NO SPECIFIC PROBLEM WAS IDENTIFIED BY MAINTENANCE PERSONNEL DURING THEIR INVESTIGATION OF THIS EVENT. SINCE THE SPURIOUS CHANNEL SPIKES HAVE OCCURRED. A REVIEW OF THE WORK ORDER HISTORY OF THE 1R1B RMS CHANNEL WAS CONDUCTED. NO SPECIFIC FAILURE TREND WAS IDENTIFIED; HOWEVER, INVESTIGATION TO DETERMINE THE ROOT CAUSE OF THE SPURIOUS HIGH CHANNEL SPIKE IS CONTINUING.

[119] SALEM 1 DOCKET 50-272 LER 90-019
MAIN STEAMLINE ISOLATION DUE TO EQUIPMENT DESIGN/EQUIPMENT CONCERNS.
EVENT DATE: 060390 REPORT DATE: 070390 NSSS: WE TYPE: PWR
VENDOR: ROSEMOUNT, INC.

(NSIC 218756) ON 6/3/90 AT 0712 HOURS, A MAIN STEAMLINE ISOLATION ACTUATION OCCURRED. AT THE TIME OF THE EVENT, THE UNIT WAS IN MODE 4 AND HEATING UP IN PREPARATION FOR STARTUP. THE MAIN STEAMLINE ISOLATION SIGNAL WAS GENERATED UPON COMPLETION OF THE HIGH STEAMLINE FLOW COINCIDENT WITH LOW STEAMLINE PRESSURE LOGIC (BY DESIGN). IN MODE 4, THE BISTABLES FOR LOW STEAMLINE PRESSURE ARE TRIPPED DUE TO PLANT CONDITIONS PROVIDING HALF OF THE LOGIC SIGNAL REQUIRED FOR MAIN STEAMLINE ISOLATION. THE HIGH STEAMLINE FLOW LOGIC WAS COMPLETED WHEN THE BISTABLES FOR NO. 11 AND NO. 12 STEAM GENERATOR (S/G) HIGH STEAMLINE FLOW TRIPPED. MAIN STEAMLINE ISOLATION IS AN ENGINEERED SAFETY FEATURE (ESF). THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO POSSIBLE EQUIPMENT/DESIGN CONCERNS ASSOCIATED WITH THE MAIN STEAMLINE FLOW TRANSMITTER SENSING LINES. THE

INITIAL INVESTIGATION BY MAINTENANCE PERSONNEL, TO DETERMINE WHY THE NOS. 11 AND 12 S/G HIGH STEAMLINE FLOW BISTABLES TRIPPED, DID NOT IDENTIFY ANY FAILED COMPONENTS. A CHANNEL FUNCTIONAL TEST, OF THE SUBJECT STEAM FLOW CHANNELS, WAS SUCCESSFULLY COMPLETED. STEAMLINE FLOW MEASURING DESIGN CONCERNS HAVE PREVIOUSLY BEEN IDENTIFIED VIA LER 272/88-017-01 (I.E., STEAMLINE FLOW MEASUREMENT DRIFT CONCERN). ENGINEERING BELIEVES THAT THE DRIFT CONCERN AND THIS RECENT EVENT APPEAR TO HAVE A RELATED CAUSE.

[120] SALEM 1 DOCKET 50-272 LER 90-020
TECH SPEC TABLE 3.3-1 NONCOMPLIANCE; PERMISSIVE P-6 SET INCORRECTLY DUE TO INADEQUATE PROCEDURAL REVIEW.
EVENT DATE: 060890 REPORT DATE: 070990 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 218788) ON 6/8/90, TECH SPEC REVIEW PROJECT PERSONNEL IDENTIFIED THAT THE MAINTENANCE-I&C PROCEDURES FOR SETTING THE INTERMEDIATE RANGE EXCORE NUCLEAR INSTRUMENTATION SYSTEM PERMISSIVE P-6, FOR BOTH SALEM UNIT 1 AND SALEM UNIT 2, IS NOT CORRECT. PERMISSIVE P-6 AUTOMATICALLY ENERGIZES THE SOURCE RANGE CHANNELS (AND THE SOURCE RANGE REACTOR TRIP FUNCTION) WHEN THE NEUTRON FLUX LEVEL DECREASES TO BELOW THE RESET VALUE. TECH SPEC TABLE 3.3-1 (FOR BOTH UNITS) REQUIRES THE P-6 TO BE RESET WHEN BOTH INTERMEDIATE RANGE NEUTRON FLUX CHANNELS INDICATE < 6X10(-11) AMPS. CONTRARY TO THIS REQUIREMENT, THE PROCEDURES REQUIRED SETTING THE P-6 RESET VALUE AT 5X10(-11) AMPS. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INADEQUATE PROCEDURAL REVIEW. REVIEW OF PRIOR MAINTENANCE-I&C PROCEDURE REVISIONS HAS SHOWN THAT THE RESET VALUE HAS ALWAYS BEEN PROCEDURALLY REQUIRED TO BE 5X10(-11) AMPS INSTEAD OF 6X10(-11) AMPS. APPARENTLY, THE ORIGINAL MAINTENANCE-I&C PROCEDURES AND SUBSEQUENT REVISIONS WERE NOT ADEQUATELY REVIEWED TO ENSURE COMPLIANCE WITH ALL APPLICABLE TECH SPEC REQUIREMENTS. THE MAINTENANCE-I&C PROCEDURES, WHICH CONTAIN THE REQUIREMENT TO SET THE P-6 PERMISSIVE RESET VALUE (BOTH UNITS) HAVE BEEN REVISED TO SET THE RESET VALUE TO 7X10(-11) AMPS. THE TECH SPEC REVIEW PROJECT IS CONTINUING.

[121] SALEM 2 DOCKET 50-311 LER 90-018
CONTAINMENT VENTILATION ISOLATION DUE TO EQUIPMENT/DESIGN CONCERNS.
EVENT DATE: 051190 REPORT DATE: 060890 NSSS: WE TYPE: PWR
VENDOR: I-T-E CIRCUIT BREAKER
VICTOREEN INSTRUMENT DIVISION

(NSIC 218578) ON 5/11/90 AT 0334 HOURS, THE 2HL LIGHTING TRANSFORMER FAILED. THIS CAUSED AN ELECTRICAL TRANSIENT IN THE RADIATION MONITORING SYSTEM (RMS) RESULTING IN A CONTAINMENT PURGE/PRESSURE-VACUUM RELIEF SYSTEM (CP/P-VRS) ISOLATION SIGNAL. AT THE TIME OF THE EVENT, A CONTAINMENT PURGE WAS IN PROGRESS. THE CP/P-VRS ISOLATION VALVES CLOSED AS DESIGNED. THE ROOT CAUSE OF THE ACTUATION OF CP/P-VRS HAS BEEN ATTRIBUTED TO DESIGN/EQUIPMENT CONCERNS. THE VICTOREEN RMS EQUIPMENT IS SUSCEPTIBLE TO ELECTRICAL TRANSIENTS. THE 2HL LIGHTING TRANSFORMER WAS DRAWING POWER FROM THE NO. 21 STATION POWER TRANSFORMER (SPT) WHICH WAS ALSO SUPPLYING POWER TO THE VITAL BUSSES AT THE TIME OF THE 2HL LIGHTING TRANSFORMER FAILURE. POWER TO THE RMS IS SUPPLIED BY THE VITAL BUSSES WHICH SUBSEQUENTLY SAW THE RESULTING ELECTRICAL TRANSIENT CAUSED BY THE 2HL LIGHTING TRANSFORMER FAILURE. AT THIS TIME, THE CAUSE OF THE 2HL LIGHTING TRANSFORMER FAILURE HAS BEEN ATTRIBUTED TO AN EQUIPMENT FAILURE (AGE DEGRADATION). A FAILURE OF A HIGH VOLTAGE WINDING OCCURRED, ON THE 2HL LIGHTING TRANSFORMER, RESULTING IN HIGH PHASE TO PHASE CURRENTS AS SEEN BY THE 2HL LIGHTING TRANSFORMER BREAKER (2H3D) CAUSING THE BREAKER TO TRIP. THE 2HL LIGHTING TRANSFORMER HAS BEEN REPLACED. ENGINEERING HAS INVESTIGATED THE CONCERNS WITH THE UNIT 2 RMS CHANNELS.

[122] SALEM 2 DOCKET 50-311 LER 90-019
CONTAINMENT VENTILATION ISOLATION DUE TO EQUIPMENT/DESIGN CONCERNS.
EVENT DATE: 051190 REPORT DATE: 060890 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 218577) ON 5/11/90 AT 1435 HOURS, THE 2R12A (CONTAINMENT RADIOACTIVE NOBLE

GAS MONITOR) RADIATION MONTORING SYSTEM (RMS) CHANNEL SPIKED HIGH. THIS RESULTED IN AN ENGINEERED SAFETY FEATURE (ESF) SIGNAL ACTUATION FOR CONTAINMENT PURGE/PRESSURE-VACUUM RELIEF SYSTEM (CP/P-VRS) ISOLATION. THE CHANNEL WAS DECLARED INOPERABLE AND TECH SPEC 3.3.3.1 TABLE 3.3-6 ACTION 26 WAS ENTERED. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO EQUIPMENT/DESIGN CONCERNS. THE VICTOREEN RMS EQUIPMENT IS SUSCEPTIBLE TO ELECTRICAL TRANSIENTS. AT THE TIME OF THE EVENT, THE CHANNEL FUNCTIONAL PROCEDURE FOR THE 2R11A RMS CHANNEL (CONTAINMENT RADIOACTIVE PARTICULATE MONITOR) WAS IN PROGRESS. AS PART OF THE PROCEDURE, THE POWER SUPPLY TO THE CHANNEL IS SHUT OFF TO VERIFY THE BATTERY PACK RETAINS THE REQUIRED SETPOINTS UPON LOSS OF AC POWER. WHEN THE POWER SUPPLY WAS TURNED OFF, AN ELECTRICAL TRANSIENT OCCURRED RESULTING IN THE CHANNEL SPIKE. ENGINEERING HAS INVESTIGATED THE CONCERNS WITH THE UNIT 2 RMS CHANNELS. IT IS ANTICIPATED THAT SEVERAL SYSTEM DESIGN MODIFICATIONS WILL ELIMINATE THE SPURIOUS ESF ACTUATION SIGNALS. ONE OF THESE DESIGN MODIFICATIONS IS THE INSTALLATION OF AN UNINTERRUPTABLE POWER SUPPLY. THE PLANS FOR COMPLETION OF THESE MODIFICATIONS ARE INCLUDED IN THE CURRENT PSE&G LIVING ENGINEERING PLAN FOR THE RMS SYSTEM.

[123] SALEM 2 DOCKET 50-311 LER 90-020
CONTAINMENT VENTILATION ISOLATION DUE TO EQUIPMENT/DESIGN CONCERNS.
EVENT DATE: 051290 REPORT DATE: 060890 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 218576) ON 5/12/90 AT 0640 HOURS, A CONTAINMENT PURGE/PRESSURE-VACUUM RELIEF SYSTEM (CP/P-VRS) ISOLATION SIGNAL WAS RECEIVED FROM THE RADIATION MONITORING SYSTEM (RMS). AT THE TIME OF THE EVENT, A CONTAINMENT PURGE WAS IN PROGRESS. THE CP/P-VRS ISOLATION VALVES CLOSED AS DESIGNED. THE ROOT CAUSE OF CP/P-VRS ISOLATION HAS BEEN ATTRIBUTED TO DESIGN/EQUIPMENT CONCERNS. THE VICTOREEN RMS EQUIPMENT IS SUSCEPTIBLE TO ELECTRICAL TRANSIENTS. AT THE TIME OF THE EVENT, THE 2A VITAL BUS WAS BEING RETURNED TO AN OPERABLE STATUS. DURING THE TAGGING RELEASE, THE NO. 21 STATION POWER TRANSFORMER (SPT) OUTPUT WAS SENSED AS AN UNDERVOLTAGE CONDITION RESULTING IN THE AUTOMATIC TRANSFER OF THE 2B AND 2C 4KV VITAL BUSES FROM NO. 21 SPT TO NO. 22 SPT OUTPUT. INVESTIGATIONS REVEALED THAT OPERATIONS PERSONNEL OPENED THE NO. 21 SPT 4KV POTENTIAL TRANSFORMERS (PT) & PRIMARY FUSES DRAWER WHICH PROVIDES INFEED BREAKER CONTROL TO THE 2A, 2B, AND 2C VITAL BUSSES. THE TAGGING RELEASE DID NOT REQUIRE THE CHECK OF THIS PARTICULAR DRAWER. WHEN THE VITAL BUSSES TRANSFERRED SPTS, A MOMENTARY LOSS OF POWER ON THE BUS OCCURRED RESULTING IN THE CP/P-VRS ISOLATION SIGNALS. ENGINEERING HAS INVESTIGATED THE CONCERNS WITH THE UNIT 2 RMS CHANNELS. IT IS ANTICIPATED THAT SEVERAL SYSTEM DESIGN MODIFICATIONS WILL ELIMINATE THE SPURIOUS ESF ACTUATION SIGNALS.

[124] SALEM 2 DOCKET 50-311 LER 90-021
CONTAINMENT VENTILATION ISOLATION DUE TO RADIATION MONITORING SYSTEM EQUIPMENT/DESIGN CONCERNS.
EVENT DATE: 051590 REPORT DATE: 061390 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 218575) ON 5/15/90 AT 1509 HRS, 2R12A (CONTAINMENT RADIOACTIVE NOBLE GAS MONITOR) RADIATION MONITORING SYSTEM (RMS) CHANNEL FAILED (OUTPUT WENT TO 0). THIS RESULTED IN AN ENGINEERED SAFETY FEATURE (ESF) SIGNAL ACTUATION FOR CONTAINMENT PURGE/PRESSURE-VACUUM RELIEF SYSTEM (CP/P-VRS) (BF) ISOLATION. THE CHANNEL WAS DECLARED INOPERABLE AND TECH SPEC 3.3.3.1 TABLE 3.3-6 ACTION 26 WAS ENTERED. THE CHANNEL ESF FUNCTION WAS NOT PLACED IN "BLOCK", ANOTHER 2R12A CHANNEL FAILURE (OUTPUT WENT TO 0) OCCURRED ON 5/17/90 AT 0327 HOURS RESULTING IN A SECOND CP/P-VRS. THE ROOT CAUSE OF THE 2R12A CHANNEL FAILURE EVENTS HAVE BEEN ATTRIBUTED TO EQUIPMENT/DESIGN CONCERNS. AS ADDRESSED BY PRIOR LERS, THE VICTOREEN RMS EQUIPMENT HAS FAILED FOR VARIOUS REASONS (E.G., HIGH SUSCEPTIBILITY TO ELECTRICAL TRANSIENTS AS ADDRESSED BY LER 311/90-019-00). TROUBLESHOOTING OF THIS EVENT REVEALED A PROBLEM WITH AN EDGE CONNECTOR LOCATED ON THE BACKPLANE. A CONTACT HAD BEEN BENT DURING INSTALLATION OF A CIRCUIT CARD, CREATING A MARGINAL ELECTRICAL CONNECTION THAT EVENTUALLY FAILED. THE DAMAGED BACKPLANE CONNECTOR WAS REPAIRED AND A CHANNEL FUNCTIONAL TEST WAS SUCCESSFULLY COMPLETED. ON 5/18/90 AT 0125 HOURS, THE CHANNEL WAS DECLARED OPERABLE AND TECH SPEC TABLE 3.3-6 ACTION 26

WAS EXITED. AS INDICATED IN LER 311/90-019-00, ENGINEERING HAS INVESTIGATED THE CONCERNS WITH THE UNIT 2 RMS CHANNELS.

[125] SALEM 2 DOCKET 50-311 LER 90-023
2A 4KV VITAL BUS DEENERGIZED DUE TO PERSONNEL ERROR.
EVENT DATE: 051690 REPORT DATE: 061390 NSSS: WE TYPE: PWR

(NSIC 218574) ON 5/16/90, DURING PERFORMANCE OF A 2A SAFEGUARD EQUIPMENT CONTROL (SEC) 18 MONTH SURVEILLANCE, THE 2A 4KV VITAL BUS DEENERGIZED. THE ASSOCIATED DIESEL GENERATOR (D/G) DID NOT START SINCE IT WAS CLEARED AND TAGGED IN SUPPORT OF ENGINE MAINTENANCE. THE LOSS OF VOLTAGE TO A VITAL BUS IS AN ENGINEERED SAFETY FEATURE (ESF). THE ROOT CAUSE OF THE ESF ACTUATION HAS BEEN ATTRIBUTED TO PERSONNEL ERROR DUE TO INATTENTION TO DETAIL. DURING THE PERFORMANCE OF THE SEC SURVEILLANCE, THE MAINTENANCE TECHNICIAN INADVERTENTLY CONNECTED HIS TEST EQUIPMENT (A VISICORDER) TO THE LEADS HE HAD LIFTED INSTEAD OF TO THE ASSOCIATED RELAY CONTACTS. THIS CAUSED A SHORT TO GROUND RESULTING IN THE OPENING OF THE INFEED BREAKER FOR THE 2A 4KV VITAL BUS. A CONTRIBUTING FACTOR TO THIS EVENT WAS HUMAN FACTORS DEFICIENCIES CONTAINED IN THE SURVEILLANCE PROCEDURE. THE PROCEDURE BEING USED, TO SUPPORT THE 2A SEC SURVEILLANCE, WAS PROCEDURE M3[illegible]. THE PROCEDURE TENDED TO BE CONFUSING IN THAT OF THE 17 TESTS, ONLY 2 REQUIRE THE VISICORDER TO BE HOOKED TO THE RELAY AND NOT THE LIFTED LEADS. THESE 2 PROCEDURAL STEPS DID IDENTIFY THIS REQUIREMENT; ALTHOUGH NOT QUITE AS CLEARLY AS IT COULD HAVE BEEN (I.E., A PRECAUTIONARY OR WARNING TYPE OF "NOTE"). ALSO, THESE 2 STEPS ARE IN THE MIDDLE OF THE PROCEDURE (I.E., STEPS 9.7 AND 9.8).

[126] SALEM 2 DOCKET 50-311 LER 90-024
TECH SPEC NON-COMPLIANCE IN SUPPORT OF RADIOACTIVE LIQUID RELEASE DUE TO PERSONNEL ERROR.
EVENT DATE: 052090 REPORT DATE: 061990 NSSS: WE TYPE: PWR

(NSIC 218759) ON 5/20/90 AT 2123 HOURS, RELEASE OF THE NO. 22 CVCS MONITOR TANK (WD) RADIOACTIVE LIQUID CONTENTS WAS INITIATED. THE RELEASE WAS TERMINATED ON 5/21/90 AT 0144 HOURS. DURING THE POST-RELEASE FLUSH, A T-HANDLE INSTRUMENT ISOLATION VALVE WAS DISCOVERED CLOSED RESULTING IN THE ISOLATION OF THE 2R18 LIQUID RADWASTE EFFLUENT LINE MONITOR. HAD IT BEEN REALIZED PRIOR TO THE RELEASE THAT THE 2R18 MONITOR WAS "INOPERABLE", EITHER THE MONITOR WOULD HAVE BEEN MADE OPERABLE OR TECH SPEC 3.3.3.8 TABLE 3.3-12 ACTION 26 SAMPLE REQUIREMENTS WOULD HAVE BEEN MET. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO PERSONNEL ERROR. THE ISOLATION OF THE 2R18 INSTRUMENT ISOLATION VALVE OCCURRED DUE TO AN ERROR BY CHEMISTRY DEPARTMENT PERSONNEL AFTER PERFORMING A CHEMICAL ADDITION WHERE THIS VALVE WAS REQUIRED TO BE ISOLATED. CONTRIBUTING TO THIS EVENT WAS PROCEDURAL INADEQUACY. THE VALVE ALIGNMENT VERIFICATION STEPS, IN THE OPERATIONS PROCEDURE USED TO CONTROL RADIOACTIVE LIQUID RELEASES DOES NOT IDENTIFY OR REQUIRE VERIFICATION THAT THE INSTRUMENT ISOLATION VALVE IS OPEN PRIOR TO A RELEASE. THE CHEMISTRY DEPARTMENT PERSONNEL INVOLVED IN THIS EVENT HAVE BEEN HELD ACCOUNTABLE. A CRITIQUE OF THIS EVENT WAS CONDUCTED WITH ALL CHEMISTRY DEPARTMENT PERSONNEL. THE INSTRUMENT ISOLATION VALVE HAS BEEN NUMBERED, 22WR170.

[127] SALEM 2 DOCKET 50-311 LER 90-022
CONTAINMENT VENTILATION ISOLATION DUE TO EQUIPMENT/DESIGN CONCERNS.
EVENT DATE: 052290 REPORT DATE: 061390 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 218520) ON MAY 22, 1990 AT 0446 HOURS, THE 2R12A (CONTAINMENT RADIOACTIVE NOBLE GAS MONITOR) RADIATION MONITORING SYSTEM (RMS) CHANNEL FAILED LOW (OUTPUT WENT TO ZERO). THIS RESULTED IN AN ENGINEERED SAFETY FEATURE (ESF) SIGNAL ACTUATION FOR CONTAINMENT PURGE/PRESSURE-VACUUM RELIEF SYSTEM (CP/P-VRS) ISOLATION. THE CHANNEL WAS DECLARED INOPERABLE. TECHNICAL SPECIFICATION 3.3.3.1 TABLE 3.3-6 ACTION 26 WAS NOT ENTERED AT THAT TIME SINCE THE 2R41C CHANNEL WAS OPERABLE (IT IS THE ACCEPTED ALTERNATE CHANNEL PER TECHNICAL SPECIFICATIONS). THE ROOT CAUSE OF THE 2R12A CHANNEL FAILURE EVENTS HAVE BEEN ATTRIBUTED TO EQUIPMENT/DESIGN CONCERNS. AS ADDRESSED BY PRIOR LERS, THE VICTOREEN RMS EQUIPMENT HAS FAILED FOR VARIOUS REASONS (E.G., HIGH SUSCEPTIBILITY TO ELECTRICAL

TRANSIENTS AS ADDRESSED BY LER 311/90-019-00). INVESTIGATION OF THIS EVENT DID NOT IDENTIFY A SPECIFIC CAUSE. TROUBLESHOOTING OF THE CHANNEL INCLUDED CHECKING OF ALL RAM AND PROM CHIPS, REINSERTION OF THE SETPOINTS, CLEANING OF THE MODULES AND CHANNEL BACKPLANE, AND A BATTERY CHECK. NO SPECIFIC PROBLEMS OR CONCERNS WERE IDENTIFIED. UPON SUCCESSFUL COMPLETION OF A CHANNEL FUNCTIONAL SURVEILLANCE TEST, THE 2R12A CHANNEL WAS DECLARED OPERABLE ON MAY 29, 1990 AT 2250 HOURS. AS INDICATED IN LER 311/90-019-00, ENGINEERING HAS INVESTIGATED THE CONCERNS WITH THE UNIT 2 RMS CHANNELS.

[128] SALEM 2 DOCKET 50-311 LER 90-025
ENVIRONMENTAL QUALIFICATION OF MS PANELS NOT MET DUE TO INADEQUATE DESIGN REVIEW.
EVENT DATE: 052590 REPORT DATE: 062290 NSSS: WE TYPE: PWR

(NSIC 218760) AN ACTION REQUEST FORM WAS INITIATED ON 5/11/90 BY THE MAINTENANCE PLANNING DEPARTMENT QUESTIONING WHETHER THE UNIT 2 MAIN STEAM MS169 VENT VALVE CONTROL PANELS REQUIRE VENT OPENINGS. THE AR NOTED THAT THE CORRESPONDING UNIT 1 MS169 VENT VALVE CONTROL PANELS HAD VENT OPENINGS. IN RESPONSE TO THE AR, SYSTEM ENGINEERING IDENTIFIED, ON 5/25/90, THAT THE UNIT 2 MAIN STEAM MS169 VENT VALVE CONTROL PANELS DO NOT MEET REQUIRED ENVIRONMENTAL QUALIFICATIONS. THE PANELS DO NOT HAVE REQUIRED VENT OPENINGS WHICH WOULD PREVENT COLLAPSE OF THE PANELS IN THE EVENT OF A MAIN STEAMLINE BREAK IN THE PENETRATION. THE PANELS CONTAIN THE SOLENOID VALVES WHICH OPERATE THE MAIN STEAM ISOLATION VALVES (MSIV) VENT VALVES (MS169 AND MS171). THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INADEQUATE DESIGN REVIEW. IN 1979/80, THE UNIT 1 MS169 VENT VALVE CONTROL PANELS WERE MODIFIED TO HAVE INSTALLED VENTILATION OPENINGS. THE UNIT 2 VENT VALVE CONTROL PANELS WERE NOT IDENTIFIED TO BE MODIFIED. THE UNIT 2 MS169 VENT VALVE CONTROL PANELS HAVE HAD VENT OPENINGS CUT INTO THE DOORS. ENGINEERING REVIEWED THE UNIT 1 AND UNIT 2 DRAWINGS WHICH IDENTIFY ANY PANELS REQUIRING VENTS. THE UNIT 1 AND UNIT 2 MAIN STEAM MS169 VENT VALVE CONTROL PANELS WERE THE ONLY PANELS WHICH WERE NOT IDENTIFIED AS REQUIRING VENTS.

[129] SALEM 2 DOCKET 50-311 LER 90-026
CONTAINMENT VENTILATION ISOLATION DUE TO EQUIPMENT/DESIGN CONCERNS.
EVENT DATE: 060990 REPORT DATE: 070990 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 218812) ON 6/9/90 AT 2330 HOURS, THE 2R12B (CONTAINMENT RADIOACTIVE IODINE MONITOR) RADIATION MONITORING SYSTEM (RMS) (IL) CHANNEL FAILED LOW. THIS RESULTED IN AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION SIGNAL FOR CONTAINMENT PURGE/PRESSURE-VACUUM RELIEF (CP/P-VR) SYSTEM ISOLATION. THE CHANNEL WAS DECLARED INOPERABLE. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO DESIGN/EQUIPMENT CONCERNS. THE TYPE DETECTOR SYSTEM USED FOR THE SALEM UNIT 2 RMS CHANNELS IS MANUFACTURED BY VICTOREEN. PERIODIC PROBLEMS WITH THIS SYSTEM HAVE BEEN EXPERIENCED AS INDICATED IN PRIOR LERS (E.G., 311/90-010-00). TROUBLESHOOTING BY MAINTENANCE PERSONNEL IDENTIFIED A FAILED SCALAR, FAILED DISCRIMINATOR POTENTIOMETER, AND A FAILED HIGH VOLTAGE POTENTIOMETER. THE FAILED COMPONENTS WERE REPLACED. SUBSEQUENTLY, THE 2R12B CHANNEL WAS SUCCESSFULLY FUNCTIONALLY TESTED AND THE CHANNEL WAS DECLARED OPERABLE ON 7/6/90. AS INDICATED IN PRIOR LERS, ENGINEERING HAS INVESTIGATED THE CONCERNS WITH THE UNIT 2 RMS CHANNELS. IT IS ANTICIPATED THAT SEVERAL SYSTEM DESIGN MODIFICATIONS WILL ELIMINATE THE SPURIOUS ESF ACTUATION SIGNALS. THE PLANS FOR COMPLETION OF THESE MODIFICATIONS ARE INCLUDED IN THE CURRENT PSE&G LIVING ENGINEERING PLAN FOR THE RMS SYSTEM.

[130] SAN ONOFRE 1 DOCKET 50-206 LER 89-011 REV 02
UPDATE ON SAFETY INJECTION ALIGNMENT DELAY CONTRARY TO THE SAFETY ANALYSIS.
EVENT DATE: 032389 REPORT DATE: 070690 NSSS: WE TYPE: PWR
VENDOR: AGASTAT RELAY CO.

(NSIC 218779) ON 3/21/89 A WIRING ERROR WAS DISCOVERED ON THE MAIN FEEDWATER PUMP (MFP) MINIMUM FLOW VALVES WHICH RESULTED IN THE VALVES OPERATING OTHER THAN ASSUMED IN THE SAFETY ANALYSIS. THIS SINGLE ISSUE WAS EVALUATED AND IT WAS CONCLUDED THAT ANY DEGRADATION IN SAFETY INJECTION SYSTEM (SI) PERFORMANCE DUE TO THIS DISCREPANCY WAS EASILY ACCOMMODATED BY THE MARGIN AVAILABLE IN THE ANALYSIS.

ON 3/23/89, WITH UNIT 1 IN COLD SHUTDOWN, IT WAS DETERMINED THAT THIS WIRING ERROR, IN CONJUNCTION WITH OTHER VARIANCES IN ACTUAL SI OPERATION FROM THAT ASSUMED IN THE SAFETY ANALYSIS, COULD RESULT IN SAFETY ANALYSIS ACCEPTANCE CRITERIA BEING EXCEEDED. SI SYSTEM OPERATION WAS THEN EXTENSIVELY REVIEWED AND COMPARED TO ASSUMPTIONS IN THE SAFETY ANALYSIS. THIS REVIEW REVEALED OTHER SAFETY ANALYSIS ASSUMPTIONS WHICH REQUIRED MODIFICATION. THESE INCLUDED CHANGES IN ASSUMED VALVE OPERATING TIMES TO CONFORM TO INSERVICE TEST PROGRAM VALUES, INCLUSION OF OPERATING PRACTICES NOT ACCOUNTED FOR IN THE ANALYSIS, AND ADDITION OF UNCERTAINTY IN AGASTAT RELAY TIMING TO CONSERVATIVELY ACCOUNT FOR THE LATEST POSSIBLE COMPONENT OPERATION. THE CAUSE OF THIS CONDITION IS RELATED TO PAST WEAKNESSES IN THE TECHNICAL AND ENGINEERING SUPPORT FOR SAN ONOFRE WHICH RESULTED IN INCONSISTENCIES BETWEEN THE SAFETY ANALYSIS AND THE ACTUAL PLANT CONFIGURATION.

[131] SAN ONOFRE 1 DOCKET 50-206 LER 89-022 REV 02
UPDATE ON DESIGN BASIS OF THE OVERPRESSURE MITIGATION SYSTEM NOT MET.
EVENT DATE: 091489 REPORT DATE: 070390 NSSS: WE TYPE: PWR

(NSIC 218780) ON 9/14/89, AT 0845, WITH UNIT 1 AT 91% POWER, AN ENGINEERING REVIEW OF THE REACTOR COOLANT SYSTEM (RCS) OVERPRESSURE MITIGATION SYSTEM (OMS) DETERMINED THAT TECH SPEC (TS) 3.20, AND ITS ADMINSTRATIVE CONTROLS, WHICH PERMIT OPERATION WITH OMS OUT OF SERVICE WITH A PRESSURIZER LEVEL < 50% AND RCS PRESSURES < 400 PSIG, ARE NON-CONSERVATIVE. THE POTENTIAL TO EXCEED 10CFR50, APP. G RCS PRESSURE LIMITS AT LOW TEMPERATURE HAS EXISTED AND UNIT 1 HAS BEEN OPERATED WITH OMS OUT OF SERVICE WHEN OMS PROTECTION WAS REQUIRED. REVISION 0 OF THIS LER INDICATED THAT AN IN-DEPTH ENGINEERING REVIEW OF THE OMS WAS CONTINUING. ON 5/25/90, THIS REVIEW DETERMINED THAT THE OMS DOES NOT MEET ITS DESIGN BASIS REQUIREMENTS FOR THE FOLLOWING REASONS: 1) THE SYSTEM IS SUSCEPTIBLE TO POTENTIAL SINGLE FAILURE SCENARIOS WHICH COULD DISABLE ONE OR BOTH POWER OPERATED RELIEF VALVES (PORVS) RESULTING IN THE INABILITY OF THE OMS TO PROVIDE A RELIEF PATH IF REQUIRED, AND 2) ONE PORV HAS INSUFFICIENT DISCHARGE FLOW CAPACITY TO PROTECT AGAINST AND POSSIBLE LOW TEMPERATURE OVERPRESSURE TRANSIENTS. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO PAST WEAKNESSES IN SCE'S ENGINEERING AND TECHNICAL SUPPORT TO SAN ONOFRE, WHICH IS DESCRIBED IN DETAIL IN THE 10/3/88 SUBMITTAL TO THE NRC ADDRESSING THIS SUBJECT. THE CORRECTIVE ACTIONS IDENTIFIED IN THAT SUBMITTAL ARE ALSO APPLICABLE TO THE CAUSES OF THIS CONDITION.

[132] SAN ONOFRE 1 DOCKET 50-206 LER 90-011
MANUAL REACTOR TRIP FOLLOWING A LOSS OF FEEDWATER TO ONE STEAM GENERATOR AS THE RESULT OF UNIDENTIFIED SYSTEM INTERACTIONS.
EVENT DATE: 051590 REPORT DATE: 061490 NSSS: WE TYPE: PWR

(NSIC 218499) ON 5/15/90, AT 0243 WITH UNIT 1 AT 91.5% POWER, THE REACTOR WAS MANUALLY TRIPPED DUE TO A LOW AND DECREASING LEVEL IN STEAM GENERATOR (SG) "C" RESULTING FROM A LOSS OF FEEDWATER FLOW. THE LOSS OF FEEDWATER FLOW OCCURRED DURING MAINTENANCE ON AN AUX. FEEDWATER (AFW) PUMP WHILE DE-TERMINATING A VALVE CONTROL CIRCUIT WHICH RESULTED IN AN INADVERTENT SHORT CIRCUIT TO GROUND. SHORT CIRCUIT INITIATED AUTOMATIC TRANSFER OF A VITAL BUS AND A MOMENTARY POWER INTERRUPTION TO THE SG "C" HIGH LEVEL ACTUATION WHICH, IN TURN, INITIATED CLOSURE OF SG "C" MAIN FCV. HIGH LEVEL ACTUATION CIRCUITRY RESET WHEN IT WAS RE-ENERGIZED AT THE COMPLETION OF THE TRANSFER. CONTROL ROOM OPERATOR THEN RESET THE FCV CONTROLS IN ORDER TO GAIN MANUAL CONTROL OF THE FCV IN ACCORDANCE WITH PROCEDURES. HOWEVER, THE CONTROL OPERATOR WAS UNABLE TO RE-ESTABLISH FEEDWATER FLOW BEFORE SG "C" REACHED THE LEVEL AT WHICH PROCEDURES REQUIRE THE REACTOR TO BE TRIPPED. CLOSURE OF FCV RESULTED FROM A RELAY/BISTABLE RACE WHICH OCCURRED WHEN SG "C" HIGH LEVEL FCV CLOSURE CIRCUITRY WAS RE-ENERGIZED. THE POTENTIAL FOR THIS INTERACTION WAS NOT PREVIOUSLY RECOGNIZED. CORRECTIVE ACTIONS INCLUDE: 1) TEMPORARILY DISABLING THE SG HIGH LEVEL CLOSURE OF THE FCV AND 2) TESTING OF THE VITAL BUSES WHICH ARE SUSCEPTIBLE TO BRIEF POWER INTERRUPTIONS REVEALED NO OTHER UNACCEPTABLE INTERACTIONS.

[133] SAN ONOFRE 1 DOCKET 50-206 LER 90-012
VOLUNTARY ENTRY INTO TECH SPEC 3.0.3 DURING DC GROUND TROUBLESHOOTING.
EVENT DATE: 053090 REPORT DATE: 062990 NSSS: WE TYPE: PWR

(NSIC 218685) AT 1540 ON MAY 30, 1990, WITH REACTOR POWER AT 92%, A GROUND ALARM ON DC BUS NO. 1 WAS RECEIVED. THE ABNORMAL OPERATIONS INSTRUCTION (AOI) FOR GROUND ISOLATION WAS ENTERED AND AT 1959, THE GROUND WAS DETERMINED TO BE ASSOCIATED WITH THE THERMAL BARRIER PUMP AND THE AOI WAS EXITED. DURING "THE GROUND ISOLATION PROCESS, IT WAS NECESSARY ON SIX OCCASIONS (ONCE EACH), FROM 1936 TO 1957, TO OPEN THE DC FEEDER BREAKERS TO VARIOUS OTHER COMPONENTS WHICH CONSTITUTED AN ENTRY INTO TECHNICAL SPECIFICATION (TS) 3.0.3. ON EACH OF THESE OCCASIONS, THE BREAKER WAS ONLY OPENED BRIEFLY (APPROXIMATELY 2-3 SECONDS) IN ACCORDANCE WITH ADMINISTRATIVE CONTROLS. CONSEQUENTLY, THE TOTAL TIME THAT THE COMPONENTS WERE INOPERABLE WAS LESS THAN THE ONE HOUR PERMITTED BY TS 3.0.3. ON JUNE 1, 1990, DURING FURTHER INVESTIGATION OF THE GROUND, A JUNCTION BOX WHICH HOUSES POWER SUPPLY CABLES FOR THE THERMAL BARRIER PUMP WAS IDENTIFIED TO CONTAIN WATER. AT 1246 ON JUNE 1, 1990, THE PUMP WAS DECLARED INOPERABLE FOR THE PERFORMANCE OF MAINTENANCE. VISUAL INSPECTION OF THE JUNCTION BOX REVEALED A VERTICAL CONDUIT LOCATED ON THE TOP OF THE BOX WHICH ALLOWED WATER ACCESS DURING PERIODS OF RAIN. APPROXIMATELY TWO DAYS PRIOR TO THE APPEARANCE OF THE GROUND ON DC BUS NO. 1, HEAVY RAINS WERE EXPERIENCED AT SAN ONOFRE. FOLLOWING REMOVAL OF THE WATER AND DRYING OF THE ELECTRICAL CONNECTIONS, THE DC BUS NO. 1 GROUND ALARM CLEARED.

[134] SAN ONOFRE 1 DOCKET 50-206 LER 90-013
VOLUNTARY ENTRY INTO TECHNICAL SPECIFICATION 3.0.3 DUE TO HYDRAZINE TANK LEVEL INDICATOR FAILURE.
EVENT DATE: 060990 REPORT DATE: 070990 NSSS: WE TYPE: PWR
VENDOR: SIGMA INSTRUMENTS, INC.

(NSIC 218814) AT 0200 ON 6/9/90, WITH UNIT 1 AT 92% POWER, A TRAIN "A" HYDRAZINE STORAGE TANK (HST) LOW LEVEL ALARM WAS RECEIVED IN THE CONTROL ROOM. AFTER VERIFYING ADEQUATE HST LEVEL VIA THE TRAIN "B" INDICATION, THE TRAIN "A" HYDRAZINE PUMP G-200A WAS DECLARED INOPERABLE SINCE ITS AUTO-START CAPABILITY IS BLOCKED BY THE HST LOW LEVEL SIGNAL. THE TECH SPEC (TS) 72-HOUR ACTION STATEMENT WAS ENTERED FOR ONE INOPERABLE HYDRAZINE PUMP. AT 1443 ON 6/9/90, PRIOR TO PERFORMANCE OF TROUBLESHOOTING AND MAINTENANCE ON THE TRAIN "A" LEVEL INDICATOR, THE TRAIN "B" HYDRAZINE PUMP WAS TESTED IN ACCORDANCE WITH TS 3.3.1.C. SINCE PERFORMANCE OF THE TEST RENDERED THE PUMP INOPERABLE, TS 3.0.3 WAS ENTERED BECAUSE BOTH HYDRAZINE PUMPS WERE INOPERABLE. AT 1519, FOLLOWING SATISFACTORY TESTING OF THE TRAIN "B" PUMP, THE PUMP WAS RETURNED TO SERVICE AND TS 3.0.3 WAS EXITED. THE CAUSE OF THE TRAIN "A" HST LEVEL INDICATOR FAILING LOW WAS FAILURE OF THE POWER SUPPLY CIRCUIT FOR LEVEL INDICATOR SWITCH LIS-500A. THE VOLTAGE REGULATOR HAD FAILED WHICH CAUSED THE POWER SUPPLY CIRCUIT TO FAIL AND THE INDICATOR SWITCH TO ACTIVATE THE LOW LEVEL BLOCK OF HYDRAZINE PUMP G-200A. THE INOPERABLE SWITCH WAS REPLACED AND THE CIRCUIT TESTED SATISFACTORILY. THE CAUSE OF THE TS 3.0.3 ENTRY IS THE ABSENCE OF APPROPRIATE ACTION STATEMENTS. SIMILAR LERS: 206/90-005, 89-018, AND 89-024.

[135] SAN ONOFRE 2 DOCKET 50-361 LER 90-004
MISSED HOURLY FIREWATCH INSPECTION DUE TO A FAILED DOOR LOCK.
EVENT DATE: 050690 REPORT DATE: 060490 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 218527) AT 1321 ON 2/9/90, AN HOURLY FIREWATCH WAS POSTED FOR ROOM 219 AS A COMPENSATORY MEASURE IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS (TS) 3.3.3.7, "FIRE DETECTION INSTRUMENTATION," DUE TO AN INOPERABLE EARLY WARNING DETECTOR FOR THE TRAIN "A" CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM (CREACUS) [VI] CHARCOAL ABSORBER [ADS] LOCATED IN ROOM 219. ROOM 219 CONTAINS OTHER SAFETY-RELATED EQUIPMENT AND IS KEY LOCKED FOR SECURITY PURPOSES. AT 1629 ON 5/6/90, WITH UNIT 2 AT 100% POWER AND UNIT 3 IN MODE 6, SEVERAL ATTEMPTS BY SECURITY TO UNLOCK THE DOOR TO PERMIT ENTRY INTO ROOM 219 FOR THE SCHEDULED INSPECTION WERE UNSUCCESSFUL DUE TO A LOCK FAILURE. THE KEY COULD BE INSERTED INTO THE LOCK BUT COULD NOT BE ROTATED. A LOCKSMITH WAS DISPATCHED TO THE AREA AND THE LOCK REPAIRED AT 1800

HOURS, PERMITTING THE FIREWATCH INSPECTION. AS A RESULT OF THE INACCESSIBILITY OF ROOM 219 FROM 1629 TO 1800, A SINGLE HOURLY INSPECTION REQUIRED BY TS 3.3.3.7 WAS NOT ACCOMPLISHED. THE CAUSE OF THIS EVENT WAS A STICKING TUMBLER IN THE LOCK MECHANISM WHICH PREVENTED THE KEY FROM TURNING. THE LOCKSMITH LUBRICATED THE LOCK WHICH FREED THE TUMBLER AND ALLOWED THE KEY TO ROTATE. IN ADDITION TO SCE'S EXISTING WEEKLY PREVENTIVE MAINTENANCE PROGRAM FOR SECURITY LOCKS, LUBRICANT WILL BE MADE READILY AVAILABLE TO SECURITY OFFICERS TO FACILITATE FREEING STICKY LOCK MECHANISMS IN A TIMELY MANNER IN THE FUTURE.

[136] SAN ONOFRE 2 DOCKET 50-361 LER 90-005
CONTAINMENT PURGE ISOLATION SYSTEM ACTUATION DUE TO TECHNICIAN ERROR DURING MONTHLY SURVEILLANCE.
EVENT DATE: 051490 REPORT DATE: 061290 NSSS: CE TYPE: PWR

(NSIC 218528) AT 1528 ON 5/14/90, WITH UNIT 2 AT 100% POWER, CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) TRAIN "A" ACTUATED DURING THE PERFORMANCE OF THE CHANNEL FUNCTIONAL SURVEILLANCE ON THE REDUNDANT TRAIN "B". AT 1545, AFTER DETERMINATION THAT CONTAINMENT RADIATION LEVELS WERE NORMAL AND THAT THE CPIS HAD BEEN INADVERTENTLY ACTUATED, CPIS TRAIN "A" WAS RESET AND CONTAINMENT VENTILATION WAS RETURNED TO NORMAL. THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE RADIATION LEVELS REMAINED NORMAL AND ALL CPIS TRAIN "A" COMPONENTS FUNCTIONED AS DESIGNED. THE TRAIN "A" CPIS WAS THE RESULT OF A SIMULATED HIGH RADIATION SIGNAL BEING INPUT INTO THE AREA RADIATION MONITOR (ARM) 2RIT7856. THE TECHNICIAN PERFORMING THE SURVEILLANCE ON TRAIN "B" ERRED IN THAT HE INCORRECTLY SELECTED THE TRAIN "A" ARM RATHER THAN THE TRAIN "B" ARM (2RIT7857). INATTENTION TO DETAIL ON THE PART OF THE TECHNICIAN PERFORMING THE 31-DAY SURVEILLANCE RESULTED IN THE CPIS TRAIN "A" ACTUATION. THIS INCIDENT HAS BEEN DISCUSSED WITH ALL THE RADIATION INSTRUMENT TECHNICIANS WITH EMPHASIS PLACED ON THE NECESSITY FOR "ATTENTION TO DETAIL" AND APPROPRIATE DISCIPLINARY ACTION WAS TAKEN FOR THE TECHNICIANS INVOLVED IN THE EVENT. TO PRECLUDE RECURRENCE, IMPROVED HIGHLY VISIBLE SIGNAGE HAS BEEN INSTALLED AND THE SURVEILLANCE PROCEDURE HAS BEEN REVISED TO INCLUDE INDEPENDENT VERIFICATION OF THE CORRECT MONITOR.

[137] SAN ONOFRE 2 DOCKET 50-361 LER 90-003
TOXIC GAS ISOLATION SYSTEM ACTUATION DUE TO PERSONNEL ERROR.
EVENT DATE: 060390 REPORT DATE: 062790 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 218720) AT 1634 ON 6/3/90, WITH UNIT 2 AT 75% POWER AND UNIT 3 IN MODE 5, A TOXIC GAS ISOLATION SYSTEM (TGIS) TRAIN "B" ACTUATION OCCURRED WHEN AN OPERATOR DEPRESSED THE RESET PUSHBUTTON SWITCH (PB) FOR THE TRAIN "B" CHLORINE ANALYZER. THE OPERATOR HAD BEEN DISPATCHED TO RESET TGIS TRAIN "A", WHICH HAD ACTUATED AS EXPECTED DURING A PLANNED POWER SUPPLY TRANSFER. DUE TO INADEQUATE ATTENTION TO DETAIL, HOWEVER, THE OPERATOR DEPRESSED THE RESET PB FOR TRAIN "B" RATHER THAN FOR TRAIN "A", RESULTING IN THE UNPLANNED TGIS TRAIN "B" ACTUATION. TGIS TRAINS "A" AND "B" WERE RESET AND VENTILATION LINEUPS RETURNED TO NORMAL BY 1700. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE TGIS TRAIN "A" AND TRAIN "B" COMPONENTS OPERATED AS DESIGNED, AND AN ABNORMAL CHLORINE CONCENTRATION WAS NOT DETECTED. DEPRESSING THE TGIS TRAIN "B" CHLORINE ANALYZER RESET PB RESULTED IN THE MOMENTARY DE-ENERGIZATION OF THE TGIS TRAIN "B" ACTUATION RELAY, CAUSING THE ACTUATION. OPERATIONS PERSONNEL WERE GENERALLY NOT AWARE OF THIS SYSTEM CHARACTERISTIC. WITH REGARD TO RESETTING THE INCORRECT TRAIN OF TGIS, APPROPRIATE DISCIPLINARY ACTION HAS BEEN ADMINISTERED TO THE OPERATOR INVOLVED WITH THIS EVENT, EMPHASIZING THE IMPORTANCE OF ATTENTION TO DETAIL. THIS EVENT WILL BE REVIEWED WITH UNITS 2 AND 3 OPERATIONS PERSONNEL. A HUMAN PERFORMANCE EVALUATION WILL ALSO BE PERFORMED OF THIS EVENT TO DETERMINE FURTHER CORRECTIVE ACTIONS, AS APPROPRIATE.

[138] SAN ONOFRE 3 DOCKET 50-362 LER 90-007
ENVIRONMENTAL QUALIFICATION DISCREPANCIES IDENTIFIED WHICH IS A CONDITION PROHIBITED BY TECH SPECS.
EVENT DATE: 050390 REPORT DATE: 061990 NSSS: CE TYPE: PWR

(NSIC 218721) WITH UNIT 3 SHUTDOWN FOR REFUELING, AN EXTENSIVE VALIDATION EFFORT OF ENVIRONMENTALLY QUALIFIED (EQ) EQUIPMENT LOCATED IN UNIT 3 CONTAINMENT WAS PERFORMED AS A CORRECTIVE ACTION DISCUSSED IN LER 89-012-01 (DOCKET NO. 50-361). THIS EFFORT RESULTED IN THE IDENTIFICATION OF SEVERAL DISCREPANCIES. ON 5/20/90, IT WAS CONFIRMED THAT SOME OF THESE DISCREPANCIES WERE NONCONFORMING, AND THEREFORE, MAY BE REPORTABLE AS A CONDITION PROHIBITED BY PLANT TECHNICAL SPECIFICATIONS. SPECIFICALLY, A NON-EQ PRESSURE TRANSMITTER (PT) WAS INSTALLED AS AN EQ WIDE RANGE PRESSURIZER (PZR) PRESSURE TRANSMITTER, A NON-EQ TERMINAL BLOCK (BLK) WAS INSTALLED ON A SHUTDOWN COOLING SYSTEM (BP) SUCTION ISOLATION VALVE (ISV), AND HEAT SHRINK INSULATION (ISL) IS MISSING ON THE CABLE CONNECTORS (CON) FOR ALL FOUR PRESSURIZER SAFETY VALVE (RV) POSITION SENSORS. ACTIONS WILL BE TAKEN TO CORRECT THE ABOVE NOTED DISCREPANCIES IN ACCORDANCE WITH EQ REQUIREMENTS PRIOR TO RETURNING THE UNIT TO SERVICE. OUR PRELIMINARY INVESTIGATION INDICATES THAT THESE EQ DISCREPANCIES ARE A RESULT OF BOTH ORIGINAL INSTALLATION ACTIVITIES AND MAINTENANCE ACTIVITIES. A FORMAL ROOT CAUSE EVALUATION WILL BE PERFORMED AND CORRECTIVE ACTIONS DEVELOPED. A REVISION TO THIS LER WILL BE SUBMITTED, WHICH WILL PROVIDE ADDITIONAL INFORMATION ABOUT OUR FINDINGS, CAUSES AND CORRECTIVE ACTION TO PREVENT RECURRENCE.

[139] SAN ONOFRE 3 DOCKET 50-362 LER 90-005
STEAM GENERATOR FEEDWATER SPARGER (FEEDRING) DAMAGE.
EVENT DATE: 051090 REPORT DATE: 070390 NSSS: CE TYPE: PWR

(NSIC 218797) ON 5/10/90, WITH UNIT 3 IN MODE 6, DURING A ROUTINE INSPECTION OF THE TUBESHEET OF THE STEAM GENERATORS (SGS) [SG], METAL DEBRIS WAS FOUND ON THE SECOND SIDE OF BOTH UNIT 3 SGS. THE SOURCES OF THE DEBRIS WERE DETERMINED TO BE FROM BOTH THE FEEDRING AT ITS INTERSECTION WITH THE FEEDWATER INLET DISTRIBUTION BOX AND THE "T" VENT ASSEMBLY ATTACHED TO EACH FEEDWATER INLET DISTRIBUTION BOX. THE PRELIMINARY ROOT CAUSE FOR DEGRADATION OF THE FEEDRING IS STRESS CONCENTRATION AT THE WELDS ASSOCIATED WITH THE TRANSITION PIECE BETWEEN THE FEEDRING AND THE DISTRIBUTION BOX, CAUSING CRACKING OF THE WELDS OVER A PERIOD OF SEVERAL YEARS, AND EROSION AND ERROSION/CORROSION OF THE CRACKED PIECES DURING POWER OPERATION. THE PRELIMINARY ROOT CAUSE FOR THE "T" VENT ASSEMBLY FAILURE IS EROSION AND ERROSION/CORROSION DUE TO HIGH VELOCITY WATER. THE ROOT CAUSE INVESTIGATION OF THIS EVENT IS CONTINUING AND A SUPPLEMENT TO THIS REPORT WILL BE SUBMITTED UPON ITS COMPLETION. THE FEEDRING DESIGN WAS UPGRADED SO THAT THE JUNCTION BETWEEN THE FEEDRING AND DISTRIBUTION BOX CAN WITHSTAND GREATER STRESSES. THE UPGRADED DESIGN INCLUDES: 1) USE OF SCHEDULE 120 PIPE TO REPLACE THE TRANSITION PIECE (FORMERLY SCHEDULE 40 PIPE), AND 2) IMPLEMENTATION OF AN IMPROVED WELD JOINT DESIGN AND WELD PRACTICES. THE FEEDRING SUPPORTS WERE ALSO UPGRADED TO REDUCE THE STRESS CONCENTRATION AT THE FEEDRING/DISTRIBUTION BOX JUNCTION.

[140] SAN ONOFRE 3 DOCKET 50-362 LER 90-006
FUEL MOVEMENT WITH AN INOPERABLE SOURCE RANGE MONITOR.
EVENT DATE: 051090 REPORT DATE: 061190 NSSS: CE TYPE: PWR

(NSIC 218529) ON 5/10/90, WITH UNIT 3 IN MODE 6, CONTRARY TO THE REQUIREMENTS OF TECH SPECS (TS) 3.9.2, CORE ALTERATIONS WERE INITIATED WITH ONE SOURCE RANGE DETECTOR OPERABLE. TS 3.9.2. REQUIRES THAT TWO SOURCE RANGE DETECTORS BE OPERABLE FOR FUEL MOVEMENT. EXCORE SAFETY LOG CHANNELS "C" AND "D" WERE BEING USED AS SOURCE RANGE MONITORS. AT 0740 ON 5/11/90, A CONTROL ROOM ENGINEER OBSERVED THAT CHANNEL "C" DID NOT MEET THE ACCEPTANCE CRITERIA FOR SIGNAL-TO-NOISE-RATIO (SNR) AND FUEL MOVEMENT WAS SECURED. INVESTIGATION REVEALED THAT A HIGH NOISE LEVEL IN CHANNEL "C" HAD RESULTED IN A SNR NOT MEETING THE OPERABILITY ACCEPTANCE CRITERIA SPECIFIED IN PROCEDURE S023-X-7, "NUCLEAR FUEL MOVEMENT." AT 0640 ON 5/12/90, FOLLOWING VERIFICATION THAT EXCORE CHANNEL "A" MET THE OPERABILITY ACCEPTANCE CRITERIA SPECIFIED IN PROCEDURE S023-X-7, CORE RELOAD WAS RESUMED. THE ROOT CAUSE OF THIS EVENT WAS THE FAILURE TO DETERMINE THE INOPERABILITY OF EXCORE CHANNEL "C" PRIOR TO CORE RELOAD DUE TO PERSONNEL ERROR. CONTRIBUTING CAUSES INCLUDED PROCEDURAL DEFICIENCIES AND TRAINING WEAKNESSES. CORRECTIVE ACTIONS INCLUDE APPROPRIATE DISCIPLINARY ACTION, PROCEDURE REVISIONS, AND REVIEWING THE EVENT WITH APPROPRIATE PERSONNEL. AT THE TIME THAT CORE ALTERATIONS WERE SECURED, 17 FUEL ASSEMBLIES WERE LOADED IN THE

CORE. THE CORE CONFIGURATION (CORE RELOAD PATTERN) OF THESE FUEL ASSEMBLIES WAS SUCH THAT A CRITICALITY CONDITION WAS NOT POSSIBLE.

[141] SAN ONOFRE 3 DOCKET 50-362 LER 90-009
STEAM GENERATOR MINIMUM PRESSURIZATION TEMPERATURE.
EVENT DATE: 070590 REPORT DATE: 070690 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 2 (PWR)

(NSIC 218798) ON 7/5/90, DURING AN ENGINEERING REVIEW OF RECENT SECONDARY PRESSURE TEST RESULTS, SCE IDENTIFIED CORRESPONDENCE FROM THE NSSS SUPPLIER ISSUED IN 198[illegible] WHICH RECOMMENDS THAT MINIMUM PRESSURIZATION TEMPERATURE FOR THE SAN ONOFRE UNIT 3 STEAM GENERATORS SHOULD BE 90F, INSTEAD OF 70F AS CURRENTLY PROVIDED IN TECH SPEC 3.7.2 (STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION). (SAN ONOFRE UNIT 2 DOES NOT APPEAR TO HAVE BEEN AFFECTED BY THIS RECOMMENDATION, BUT THIS IS UNDER FURTHER REVIEW.) OPERATING PROCEDURES ARE CONSISTENT WITH THE EXISTING TECH SPEC LIMIT, AND INITIAL REVIEW OF UNIT 3 OPERATING HISTORY HAS IDENTIFIED 3 OCCASIONS WHEN A STEAM GENERATOR HAS BEEN PRESSURIZED ABOVE 200 PSIG WHEN COOLANT TEMPERATURE WAS BETWEEN 70 AND 90F. PENDING FURTHER EVALUATION OF THIS SITUATION, SCE HAS CONSERVATIVELY APPLIED THE PRINCIPLES OF ACTION "B" OF TECH SPEC 3.7.2 TO THE LIMITING CASE OF THESE 3 OCCASIONS, IN THAT WE HAVE OBTAINED A PRELIMINARY ENGINEERING EVALUATION FROM THE NSSS SUPPLIER INDICATING THAT THE STEAM GENERATORS REMAIN ACCEPTABLE FOR CONTINUED OPERATION. PENDING RESOLUTION OF THE CORRECT MINIMUM PRESSURIZATION TEMPERATURE IN TECH SPEC 3.7.2, AND REVISION OF THE TECH SPECS IF NECESSARY, SCE WILL IMPLEMENT ADMINISTRATIVE CONTROLS TO ENSURE THE UNITS 2 AND 3 STEAM GENERATORS ARE NOT PRESSURIZED IN EXCESS OF 200 PSIG AT TEMPERATURES BELOW 90F.

[142] SEQUOYAH 1 DOCKET 50-327 LER 90-010
LIMITING CONDITION FOR OPERATION 3.0.3 ENTERED WHEN ONE MAIN STEAM LINE ISOLATION VALVE (MSIV) FAILED TO CLOSE WITH ANOTHER MSIV INOPERABLE FOR CORRECTIVE MAINTENANCE.
EVENT DATE: 052690 REPORT DATE: 062590 NSSS: WE TYPE: PWR

(NSIC 218711) ON MAY 26, 1990, AT 0100 EASTERN DAYLIGHT TIME (EDT) WITH UNIT 1 IN MODE 3, LIMITING CONDITION FOR OPERATION (LCO) 3.0.3 WAS ENTERED BECAUSE MAIN STEAM ISOLATION VALVE (MSIV) FAILED TO CLOSE WHEN ANOTHER MSIV WAS INOPERABLE FOR MAINTENANCE ACTIVITIES. A WORK REQUEST (WR) TO ADJUST THE PACKING ON MSIV LOOP 1 VALVE 1-FCV-1-004 HAD BEEN PERFORMED. DURING THE PERFORMANCE OF THE WR, MSIV LOOP 4 VALVE 1-FCV-1-029 WAS ERRONEOUSLY STROKED ON TWO SEPARATE OCCASIONS. DURING THE SECOND STROKING ATTEMPT, VALVE 1-FCV-1-029 FAILED TO CLOSE AND WAS DECLARED INOPERABLE. A WR WAS IMMEDIATELY WRITTEN TO REPAIR 1-FCV-1-029. MAINTENANCE AND POST-MAINTENANCE TESTING (PMT) WAS COMPLETED ON VALVE 1-FCV-1-004, AND LCO 3.0.3 WAS EXITED AT 031[illegible] EDT. VALVE 1-FCV-1-029 WAS RETURNED OPERABLE AT 0426 EDT AFTER THE VALVE WAS STROKE TESTED. THE VALVE WAS FOUND TO HAVE DIRT AND GRIME ON THE STEM AND VALVE GUIDES AND REQUIRED ONLY CLEANING AND RELUBRICATING TO RESTORE OPERABILITY. THE REMAINING TWO MSIVS WERE INSPECTED AND TESTED TO ENSURE OPERABILITY OF THE VALVES AS A PRECAUTIONARY MEASURE.

[143] SEQUOYAH 1 DOCKET 50-327 LER 90-009
AUTOMATIC START OF THE AUXILIARY FEEDWATER SYSTEM AS A RESULT OF A UNIT OPERATOR FAILING TO ADHERE TO PROCEDURE PRECAUTIONS.
EVENT DATE: 052790 REPORT DATE: 062190 NSSS: WE TYPE: PWR

(NSIC 218710) MAY 27, 1990, AT 1756 EASTERN DAYLIGHT TIME WITH UNIT 1 IN MODE 3 (0 PERCENT POWER, 2220 PSIG, AND 547 DEGREES F), AN AUTOMATIC START OF THE UNIT 1 AUXILIARY FEEDWATER OCCURRED. TESTING OF THE UNIT 1 "B" MAIN FEEDWATER PUMP (MFP) WAS IN PROGRESS WITH THE TRIP CIRCUIT ENERGIZED WHEN THE "A" MFP TRIP CIRCUIT WAS ENERGIZED IN PREPARATION FOR PUTTING THE "A" MFP ON TURNING GEAR IN ANTICIPATION OF CHANGING MODES. THE COMBINATION OF TWO MFP TRIP SIGNALS COMPLETES A LOGIC TO START THE AUXILIARY FEEDWATER PUMPS, WHICH IS AN ESF. ONCE THE TRIP SIGNAL IS ENERGIZED ON A MFP, IT HAS TO BE MANUALLY RESET TO BE CLEARED. IMMEDIATE CORRECTIVE ACTION WAS TAKEN BY RESETTING THE "B" MFP TRIP CIRCUIT, STOPPING THE AUXILIARY FEED PUMPS AND RETURNING THEM TO THEIR NORMAL

CONFIGURATION. THE SYSTEMS AND COMPONENTS REQUIRED TO ACTUATE AS A RESULT OF THE ESF FUNCTIONED AS REQUIRED. THE ROOT CAUSE OF THE EVENT HAS BEEN ATTRIBUTED TO PERSONNEL ERROR AND ATTENTION TO DETAIL IN THAT THE UNIT OPERATOR FAILED TO ADHERE TO THE PRECAUTIONS AS OUTLINED IN PLANT PROCEDURES.

[144] SEQUOYAH 1 DOCKET 50-327 LER 90-013
MAIN CONTROL ROOM FIRE DETECTORS INOPERABLE AS A RESULT OF INADEQUATE DESIGN REVIEW.
EVENT DATE: 060190 REPORT DATE: 070290 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 218712) ON JUNE 1, 1990, AT 0930 EASTERN DAYLIGHT TIME WITH UNIT 1 IN MODE 2, AND UNIT 2 IN MODE 1, IT WAS DISCOVERED THAT THE REMOVAL OF THE LIGHTING DIFFUSER PANELS IN THE MAIN CONTROL ROOM AFFECTED THE AIR FLOW PATTERN SUCH THAT THE IONIZATION-TYPE SMOKE DETECTORS ARE EFFECTIVELY INOPERABLE. LIMITING CONDITION FOR OPERATION (LCO) 3.3.3.8 WAS ENTERED AT 0930 EDT ON JUNE 1, 1990, AND AN HOURLY FIRE WATCH WAS ESTABLISHED. THE LIGHTING DIFFUSERS WERE REMOVED IN DECEMBER 1989 TO BE CLEANED AND A TEMPORARY ALTERATION CHANGE FORM (TACF) WAS INITIATED TO LEAVE THEM OUT BECAUSE OF THE IMPROVED LIGHTING EFFECT. A PERMANENT DESIGN CHANGE WAS REQUESTED IN JANUARY 1990, WHICH RESULTED IN DESIGN CHANGE NOTICE 2178 BEING ISSUED ON MARCH 23, 1990. THE ROOT CAUSE OF THE EVENT IS INADEQUATE DOCUMENTATION OF INITIAL DETECTOR DESIGN ASSUMPTIONS AND INADEQUATE DISCIPLINE DESIGN REVIEW WHEN THE DECISION WAS MADE TO REMOVE THE DIFFUSER PERMANENTLY. CORRECTIVE ACTIONS INCLUDE CLARIFICATION OF DISCIPLINE FIRE PROTECTION REVIEW RESPONSIBILITIES AND EVALUATION OF OPTIONS TO RETURN THE DETECTORS OPERABLE.

[145] SEQUOYAH 1 DOCKET 50-327 LER 90-012
REACTOR TRIP CAUSED BY LOW-LOW STEAM GENERATOR LEVEL RESULTING FROM INADEQUATE COMMUNICATION BETWEEN CONTROL ROOM OPERATORS AND TURBINE BUILDING AUXILIARY UNIT OPERATORS.
EVENT DATE: 060290 REPORT DATE: 070290 NSSS: WE TYPE: PWR

(NSIC 218792) ON JUNE 2, 1990, WITH UNITS 1 AND 2 AT APPROXIMATELY 11 AND 100 PERCENT POWER RESPECTIVELY, A REACTOR TRIP OCCURRED ON UNIT 1 ABOUT 17 MINUTES AFTER A GENERATOR/TURBINE TRIP HAD OCCURRED AS A RESULT OF ELECTRICAL PROBLEMS. CONTROL ROOM OPERATORS REDUCED REACTOR POWER AND ANNOUNCED THE TURBINE TRIP ON THE PLANT PUBLIC ACCESS (PA) SYSTEM. THE PLANT WAS STABILIZING AS REACTOR POWER REACHED APPROXIMATELY 15 PERCENT WHEN MAIN FEEDWATER (MFW) FLOW WAS LOST. TWO AUXILIARY UNIT OPERATORS (AUOS) HAD MISHEARD THE PA ANNOUNCEMENT AS "UNIT TRIP" RATHER THAN "TURBINE TRIP" AND HAD ISOLATED THE STEAM SUPPLIES TO THE MFW PUMPS. OPERATORS STARTED THE AUXILIARY FEEDWATER PUMPS WHILE CONTINUING TO REDUCE REACTOR POWER, BUT THE REACTOR TRIPPED ON LOW-LOW STEAM GENERATOR LEVEL. THE ROOT CAUSE OF THE REACTOR TRIP HAS BEEN ATTRIBUTED TO INADEQUATE COMMUNICATION BETWEEN CONTROL ROOM OPERATORS AND AUOS. AS CORRECTIVE ACTION, OPERATIONS MANAGEMENT HAS ISSUED A NIGHT ORDER CLARIFYING WHAT SPECIFIC ACTIONS SHOULD BE TAKEN BY AUOS FOLLOWING A REACTOR OR TURBINE TRIP ONLY WITH UNIT OPERATOR GUIDANCE. A RELATED WEAKNESS IN AUO TRAINING REGARDING ACTIONS TO BE TAKEN FOLLOWING A TURBINE TRIP OR REACTOR TRIP WAS IDENTIFIED AND CORRECTED DURING THE INVESTIGATION OF THIS EVENT.

[146] SEQUOYAH 1 DOCKET 50-327 LER 90-011
NUCLEAR INSTRUMENTATION SYSTEM INTERMEDIATE AND POWER RANGE CHANNELS WERE NOT CONSERVATIVELY CALIBRATED BECAUSE OF MISINTERPRETATION OF VENDOR INFORMATION.
EVENT DATE: 060890 REPORT DATE: 070990 NSSS: WE TYPE: PWR
VENDOR: GAMMA-METRICS INC.

(NSIC 218819) ON 6/8/90, AT 0100 EASTERN DAYLIGHT TIME (EDT) WITH UNIT 1 IN MODE 1, IT WAS DETERMINED THAT UNIT 1 HAD OPERATED IN NONCOMPLIANCE WITH TECH SPEC 2.2.1, "LIMITING SAFETY SYSTEM SETTINGS." THE ACTUAL NUCLEAR INSTRUMENTATION SYSTEM (NIS) POWER RANGE (PR) DETECTOR CURRENTS WERE DETERMINED TO HAVE BEEN 20 TO 31 PERCENT LOWER THAN PREDICTED, WHICH WOULD SHIFT THE PR TRIP SETPOINT ACTUATION 20 TO 31 PERCENT HIGHER THAN EXPECTED. THIS NONCONSERVATIVE

CALIBRATION OF THE PR CHANNELS WAS PRESENT FROM 5/31/90 (UNIT 1 CYCLE 5 CRITICALITY), TO 6/6/90. THE INTERMEDIATE RANGE (IR) CHANNELS HAD BEEN NONCONSERVATIVELY CALIBRATED IN THE SAME MANNER FROM 5/31/90 TO 6/1/90. THE NONCONSERVATIVE CALIBRATION RESULTED FROM A MISINTERPRETATION OF VENDOR INFORMATION WHEN CALCULATING EXPECTED NIS IR AND PR DETECTOR CURRENTS FOR CYCLE 5 OPERATION. ALTHOUGH THE NONCONSERVATIVE CALIBRATION RESULTED IN IR AND PR SETPOINTS BEING OUTSIDE OF THEIR RESPECTIVE TS ALLOWABLE VALUES, THE PLANT REMAINED WITHIN THE UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) ACCIDENT ANALYSES LIMITS. THE PRESTARTUP NIS CALIBRATION PROCEDURES WILL BE REVISED TO REQUIRE THE USE OF THE CORRECT PREDICTION METHODOLOGY. THE NIS CORRECTION PROCEDURES WILL BE REVISED TO PROVIDE GUIDANCE FOR PERFORMING EVALUATIONS OF OBSERVED NIS DEVIATIONS IN NIS DETECTOR INDICATIONS.

[147] SHEARON HARRIS 1 DOCKET 50-400 LER 90-014
TECHNICAL SPECIFICATION VIOLATION DUE TO EXCEEDED RELEASE RATES CAUSED BY PERSONNEL ERROR.
EVENT DATE: 051390 REPORT DATE: 061590 NSSS: WE TYPE: PWR

(NSIC 218538) THE PLANT WAS OPERATING IN MODE 1, POWER OPERATION, AT 100 PERCENT REACTOR POWER ON MAY 16 1990. AT 0800 IT WAS DISCOVERED THAT TWO LIQUID RADIOACTIVE RELEASES INITIATED ON MAY 13, 1990, FROM THE WASTE EVAPORATOR CONDENSATE TANKS WERE IN EXCESS OF THE FLOW RATE SPECIFIED ON THE RELEASE PERMITS. THE PERMITS SPECIFIED A RELEASE FLOW RATE OF 14.6 GALLONS PER MINUTE GPM) MAXIMUM, WHICH WAS LESS THAN THE MAXIMUM PUMP CAPACITY OF 35 GPM. THE TWO SEPARATE RELEASES MADE WERE AT 29.9 GPM AND 29.1 GPM. THE RELEASES WERE INITIATED BY TWO SEPARATE OPERATORS. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR. THE RELEASE PERMIT CLEARLY SPECIFIED [illegible] FLOW RATE TO BE USED. HOWEVER, BOTH OPERATORS FAILED TO RECOGNIZE THE [illegible] FLOW RATE OF 14.6 GPM FOR THE RELEASE. CORRECTIVE ACTIONS INCLU[illegible]ING THE OPERATOR TO TRANSCRIBE THE MAXIMUM RELEASE RATE FROM THE DISC[illegible]MIT TO THE DISCHARGE LOG, REQUIRING THE COGNIZANT RADWASTE SHIFT FOREM[illegible]IFY THE MAXIMUM RELEASE RATE PRIOR TO THE DISCHARGE, AND COUNSELING OF TH[illegible]IATE PERSONNEL. THERE WERE NO SAFETY CONSEQUENCES AS A RESULT OF THIS EVEN[illegible]T- RELEASE CALCULATIONS, USING ACTUAL COOLING TOWER BLOWDOWN DILUTION RATES DETERMINED THAT NO 10CFR20 OR 10CF[illegible] LIMITS WERE EXCEEDED. THIS EVENT IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A) (2) (I) (B) AS A TECHNICAL SPECIFICATION VIOLATION.

[148] SHEARON HARRIS 1 DOCKET 50-400 LER 90-015
BOTH EMERGENCY LOAD SEQUENCERS SUBJECT TO COMMON MODE FAILURE DUE TO IMPROPER APPLICATION OF RELAYS NOT ACCOUNTING FOR DC INDUCTIVE LOAD RATING.
EVENT DATE: 052490 REPORT DATE: 062590 NSSS: WE TYPE: PWR
VENDOR: AGASTAT RELAY CO.
MICRO SWITCH
POTTER & BRUMFIELD

(NSIC 218729) ON MAY 24, 1990, WITH THE PLANT IN A SCHEDULED OUTAGE CONDUCTING REPAIRS ON VALVES INSIDE CONTAINMENT TO ENSURE CONTINUOUS AVAILABILITY DURING THE SUMMER MONTHS, IT WAS CONCLUDED THAT SEVERAL AGASTAT MICROSWITCH AND POTTER-BRUMFIELD RELAY CONTACTS USED IN THE EMERGENCY LOAD SEQUENCERS WOULD BE SUBJECTED TO EXCESSIVE INDUCTIVE LOADS DURING CERTAIN EMERGENCY SCENARIOS. THE ACTUAL INDUCTIVE CURRENTS WERE DEMONSTRATED BY TESTING TO CAUSE EARLY FAILURE OF THE CONTACTS. AN EVALUATION CONCLUDED THAT THE FAILURE OF THE MICROSWITCH CONTACTS COULD INHIBIT PROPER SEQUENCER RESPONSE TO SAFETY INJECTION AND LOSS OF OFFSITE POWER SIGNALS, IF THESE OCCURRED AT DIFFERENT TIMES. THE PLANT DECLARED THE SEQUENCERS INOPERABLE AT 1130 AND BROUGHT THE UNIT FROM HOT SHUTDOWN INTO COLD SHUTDOWN AT 1738 ON MAY 24. FURTHER QUALIFICATION TESTING OF THESE CONTACTS DEMONSTRATED ACCEPTABLE OPERATION BY MODIFYING THE CIRCUIT TO PLACE TWO CONTACTS IN SERIES, REDUCING THE INDUCTIVE LOAD ON EACH INDIVIDUAL CONTACT. THE AFFECTED CIRCUITS WERE MODIFIED AND TESTING WAS COMPLETED ON MAY 29, AND THE SEQUENCERS WERE THEN DECLARED OPERABLE. THE PLANT WAS RETURNED TO SERVICE ON MAY 31. A DEFICIENCY IN THE SEQUENCER DESIGN WAS THE CAUSE [illegible] THIS CONDITION. THE CONTACT RATING SPECIFIED BY THE MICROSWITCH MANUFACTURER [illegible] EXCEEDED IN THE DESIGN.

[149] SHEARON HARRIS 1 DOCKET 50-400 LER 90-016
INVALID TESTING OF FUEL HANDLING BUILDING EMERGENCY EXHAUST FILTRATION UNIT DUE TO FLOWRATE MEASUREMENT INACCURACIES.
EVENT DATE: 052990 REPORT DATE: 062690 NSSS: WE TYPE: PWR

(NSIC 218771) ON 5/29/90, AN INVESTIGATION INTO A FLOWRATE INDICATION DISCREPANCY BETWEEN THE MAIN CONTROL BOARD (MCB) INSTRUMENTATION AND TEST MEASUREMENTS USED FOR THE "B" TRAIN FUEL HANDLING BUILDING EMERGENCY EXHAUST FILTRATION UNIT CONCLUDED THAT A PREVIOUS TEST WAS CONDUCTED WITH FLOWRATES OUTSIDE OF TECH SPEC (TS) 4.9.12 REQUIREMENTS. A FLOWRATE OF 6600 CUBIC FEET PER MINUTE (CFM) +/- 10% IS SPECIFIED, AND THIS FLOWRATE IS REQUIRED BY TS TO BE DETERMINED USING DUCT TRAVERSE AIR FLOW MEASUREMENTS. DURING TESTING ON 3/30/90, A FLOWRATE OF 6340 CFM WAS MEASURED IN THE DUCT, WHILE THE MCB INSTRUMENTATION INDICATED APPROXIMATELY 75_0 CFM. FURTHER TESTING AND EVALUATION SHOWED THAT THE LOCATION OF THE TEST MEASUREMENT WAS A TURBULENT AIRFLOW REGION WITH SOME BACKFLOW. ACTUAL FLOWRATES MEASURED IN A MORE LAMINAR FLOW REGION CONFIRMED THE ACCURACY OF THE MCB INDICATION. THE FILTRATION UNIT HAD NOT BEEN DECLARED OPERABLE FOLLOWING TESTING PENDING RESOLUTION OF THE FLOWRATE DISCREPANCY. WHEN THE DEFICIENCY WAS IDENTIFIED IN THE TEST MEASUREMENT LOCATION, THE TEST PROCEDURE WAS RERUN AND COMPLETED SATISFACTORILY ON 4/27. A REVIEW OF PREVIOUS TEST RESULTS DETERMINED THAT THE TESTING PERFORMED ON 6/29/88, WAS MOST LIKELY CONDUCTED AT ACTUAL FLOWRATES BELOW THE 6600 CFM +/- 10% CRITERIA, WHICH DOES NOT COMPLY WITH TS REQUIREMENTS.

[150] SOUTH TEXAS 1 DOCKET 50-498 LER 90-009
INADEQUATE SURVEILLANCE PERFORMED ON A CONTROL ROOM TOXIC GAS ANALYZER.
EVENT DATE: 051490 REPORT DATE: 061390 NSSS: WE TYPE: PWR

(NSIC 218545) ON MAY 14, 1990, UNIT 1 WAS IN MODE 5. AT 0400 HOURS, ON MAY 14, 1990, A NON-LICENSED OPERATOR DISCOVERED THAT THE PRINTER FOR TOXIC GAS ANALYZER XE-9325 WAS NOT UPDATING. AT 0900 HOURS, OPERATIONS PERSONNEL DETERMINED THAT TECHNICAL SPECIFICATION REQUIRED SURVEILLANCES FOR TOXIC GAS MONITOR XE-9325 HAD BEEN PERFORMED INCORRECTLY FROM 2000 HOURS ON MAY 11, 1990, UNTIL 0400 HOURS ON MAY 14, 1990. THE PRINTER ASSOCIATED WITH ANALYZER XE-9325 FAILED TO UPDATE WHICH RESULTED IN NON-LICENSED OPERATORS RECORDING ERRONEOUS DATA FOR APPROXIMATELY 56 HOURS. THE CAUSES OF THE EVENT WERE LESS THAN ADEQUATE TRAINING ON THE PERFORMANCE OF THE CHANNEL CHECK AND LESS THAN ADEQUATE DETAIL IN THE OPERATOR LOGGING PROCEDURE. CORRECTIVE ACTIONS INCLUDE THE TRAINING OF LICENSED AND NON-LICENSED OPERATORS AND THE REVISION OF THE OPERATOR LOGGING PROCEDURES.

[151] SOUTH TEXAS 1 DOCKET 50-498 LER 90-010
FAILURE TO PERFORM A SEALED SOURCE SURVEILLANCE WITHIN THE REQUIRED TECHNICAL SPECIFICATION INTERVAL.
EVENT DATE: 051590 REPORT DATE: 061490 NSSS: WE TYPE: PWR

(NSIC 218546) ON MAY 15, 1990, IT WAS DETERMINED DURING A REVIEW OF RADIOACTIVE SOURCE INVENTORY AND LEAK TEST RECORDS THAT THE LEAK AND CONTAMINATION TEST INTERVAL FOR SOME SOURCES WERE GREATER THAN THE TECHNICAL SPECIFICATION ALLOWABLE AS REQUIRED BY TECHNICAL SPECIFICATION 4.7.10.2.A. CAUSES OF THIS EVENT WERE INSUFFICIENT DETAIL IN THE SURVEILLANCE PROGRAM PROCEDURE AND THAT TRAINING WAS NOT PROVIDED TO THE RESPONSIBLE INDIVIDUAL. CORRECTIVE ACTIONS INCLUDE A PROGRAM REVISION TO CLARIFY THE MEANING OF TEST COMPLETION, A SPECIAL INSTRUCTION SESSION FOR THE INDIVIDUAL RESPONSIBLE FOR THE SEALED SOURCE SURVEILLANCE, INSTRUCTION FOR INDIVIDUALS RESPONSIBLE FOR COORDINATION OF SURVEILLANCES WITHIN THEIR DIVISIONS, REVISIONS TO A PLANT PROCEDURE AND SURVEILLANCE SCHEDULING DATABASE TO ALLOW FOR SURVEILLANCES TO BE PERFORMED ON A FIXED CALENDAR BASIS, A REVIEW OF SURVEILLANCES TO DETERMINE OTHER SURVEILLANCES WHICH COULD BE AFFECTED BY PERFORMANCE OVER A PERIOD OF TIME, AND ADMINISTRATIVE CONTROLS WILL BE ESTABLISHED, AS NECESSARY, TO ENSURE THAT OTHER SURVEILLANCES IDENTIFIED WILL NOT EXCEED THE GRACE PERIOD FOR DISCRETE CONSTITUENT TESTS.

[152] SOUTH TEXAS 2 DOCKET 50-499 LER 90-008
A TECHNICAL SPECIFICATION REQUIRED SHUTDOWN DUE TO PRIMARY COOLANT SYSTEM LEAKAGE.
EVENT DATE: 050890 REPORT DATE: 060890 NSSS: WE TYPE: PWR

(NSIC 218547) ON MAY 8, 1990, UNIT 2 WAS IN MODE 1 AT 100 PERCENT POWER. AT APPROXIMATELY 0030 HOURS, IT WAS DETERMINED THAT A SMALL PRESSURE BOUNDARY LEAK (APPROXIMATELY 10 ML/MINUTE) EXISTED ON THE ASME CLASS 2 STEAM GENERATOR "C" LOWER HEAD DRAIN LINE. A NOTIFICATION OF UNUSUAL EVENT WAS DECLARED AND TECHNICAL SPECIFICATION REQUIRED SHUTDOWN WAS INITIATED TO MODE 5. THE CAUSE OF THIS EVENT WAS HIGH CYCLE FATIGUE FAILURE OF THE WELD ON THE UPSTREAM SIDE OF THE DRAIN VALVE. THE DRAIN VALVES ON BOTH UNITS' STEAM GENERATORS HAVE BEEN REPLACED WITH THREADED AND WELDED CAPS WHICH ARE NOT EXPECTED TO DEVELOP HIGH ALTERNATING STRESSES. THE PERMANENT DESIGN OF THIS DRAIN VALVE PIPING IS BEING REVIEWED TO ASSURE NO CYCLIC FATIGUE STRESS CONCERNS EXIST. AN EVALUATION OF THE HIGH CYCLE FATIGUE EXPERIENCED ON THE STEAM GENERATOR DRAIN LINE WILL BE PERFORMED FOR POTENTIAL GENERIC IMPLICATIONS.

[153] SOUTH TEXAS 2 DOCKET 50-499 LER 90-009
DISCOVERY OF INCORRECT WIRING IN THE SOLID STATE PROTECTION SYSTEM.
EVENT DATE: 051290 REPORT DATE: 061390 NSSS: WE TYPE: PWR

(NSIC 218548) ON MAY 12, 1990, UNIT 2 WAS IN MODE 5 FOR A FORCED OUTAGE. DURING THE PERFORMANCE OF CORRECTIVE MAINTENANCE TO INVESTIGATE A FLASHING "TRAIN C SI BLOCKED" STATUS LAMP, AN EXTRA WIRE WAS DISCOVERED IN THE SOLID STATE PROTECTION SYSTEM (SSPS). EVALUATION OF THIS WIRE DETERMINED THAT IT WOULD NOT HAVE PREVENTED FULFILLMENT OF A SAFETY FUNCTION; HOWEVER, DUE TO THE POTENTIAL SIGNIFICANCE, HL&P HAS CHOSEN TO REPORT THIS EVENT AS A VOLUNTARY LER. IT WAS SUBSEQUENTLY DETERMINED THAT THE EXTRA WIRE WAS INADVERTENTLY INSTALLED BY WESTINGHOUSE DURING ASSEMBLY OR CONTINUITY TESTING OF THE SSPS CABINETS. THE WIRE WAS REMOVED AND THE SYSTEM SUCCESSFULLY TESTED. THE REMAINING SSPS LOGIC TRAIN CABINETS ON BOTH UNITS HAVE BEEN INSPECTED TO VERIFY THE ABSENCE OF THIS WIRING ERROR.

[154] SOUTH TEXAS 2 DOCKET 50-499 LER 90-010
INADVERTENT ENGINEERED SAFETY FEATURE ACTUATION DUE TO INCORRECT CONNECTION OF TEST EQUIPMENT.
EVENT DATE: 051590 REPORT DATE: 061490 NSSS: WE TYPE: PWR

(NSIC 218549) ON MAY 15, 1990, UNIT 2 WAS IN MODE 5 FOR A FORCED OUTAGE. AT 1413 HOURS, AN UNPLANNED ACTUATION OF THE ENGINEERED SAFETY FEATURES TRAIN "C" STANDBY DIESEL GENERATOR OCCURRED WHEN A MAINTENANCE ELECTRICIAN CONNECTED TEST EQUIPMENT LEADS TO THE WRONG UNDERVOLTAGE RELAY DURING A SURVEILLANCE TEST. THE CAUSE OF THIS EVENT WAS FAILURE OF THE MAINTENANCE ELECTRICIAN TO ENSURE THAT HE WAS CONNECTING THE TEST EQUIPMENT TO THE PROPER RELAY. THE PROCEDURE STEPS SPECIFIED THE CORRECT RELAY AND THE RELAY HE CONNECTED THE LEADS TO WAS CLEARLY LABELED. A TRAINING BULLETIN WILL BE ISSUED TO MAINTENANCE PERSONNEL REGARDING THIS EVENT AND THE NEED TO PERFORM PREINSTALLATION VERIFICATION OF TEST EQUIPMENT.

[155] ST. LUCIE 1 DOCKET 50-335 LER 90-007
MANUAL REACTOR TRIP FOLLOWING SEVERE LEAKAGE OF MAIN TURBINE DIGITAL ELECTRO-HYDRAULIC CONTROL FLUID DUE TO THE INSTALLATION OF IMPROPERLY SIZED O-RINGS.
EVENT DATE: 052490 REPORT DATE: 062390 NSSS: CE TYPE: PWR
VENDOR: MOOG INC.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 218762) ON 5/24/90 ST. LUCIE UNIT 1 WAS IN MODE 1 AT 91% POWER PERFORMING MAIN TURBINE MAINTENANCE ON A HYDRAULIC CONTROL VALVE (MOOG) IN THE DIGITAL ELECTRO-HYDRAULIC (DEH) CONTROL SYSTEM ON THE #3 GOVERNOR VALVE. WHEN DEH FLUID WAS RE-ESTABLISHED TO THE VALVE, AN EXCESSIVE UNISOLABLE LEAK DEVELOPED. THE ASSISTANT NUCLEAR PLANT SUPERVISOR INSTRUCTED THE REACTOR CONTROL OPERATORS TO MANUALLY TRIP THE REACTOR AND TURBINE AT 0452. THE STANDARD POST TRIP ACTIONS WERE PERFORMED AND THE UNIT WAS STABILIZED IN MODE 3, HOT STANDBY. THE INITIATOR

OF THIS EVENT WAS THE SEVERE LEAKAGE OF DEH FLUID FROM THE ASSOCIATED MOOG VALVE OF THE #3 GOVERNOR VALVE. AN INVESTIGATION REVEALED THE O-RINGS INSIDE THE BASE OF THE MOOG VALVE WHICH SEAL THIS VALVE TO THE #3 GOVERNOR VALVE MOUNTING PLATE HAD RUPTURED. THE CAUSE OF THE RUPTURE WAS DUE TO THE INSTALLATION OF IMPROPERLY SIZED O-RINGS. THE PARTS LIST USED TO OBTAIN REPLACEMENT O-RINGS FOR MAINTENANCE HAD THE INCORRECT PART NUMBER LISTED FOR THESE O-RINGS. THE ROOT CAUSE FOR THIS EVENT IS INADEQUATE TECH MANUALS AND DRAWINGS FOR THE DEH SYSTEM WHICH LACK THE PART NUMBERS FOR REPLACEMENT PARTS. AN OFFICIAL PARTS LIST WAS REQUESTED FROM THE VENDOR FOR ALL O-RINGS USED ON THE TURBINE/GENERATOR SET, AS WELL AS ALL OTHER SUPPORTING WESTINGHOUSE SYSTEMS.

[156] ST. LUCIE 1 DOCKET 50-335 LER 90-008
REACTOR SHUTDOWN DUE TO UNRECOVERABLE DROPPED CONTROL ELEMENT ASSEMBLY CAUSED BY AN IMPROPERLY INSTALLED FUSE.
EVENT DATE: 061490 REPORT DATE: 070990 NSSS: CE TYPE: PWR

(NSIC 218820) AT 0110 ON 6/14/90, ST. LUCIE UNIT 1 WAS OPERATING AT 100% POWER WHEN UTILITY LICENSED OPERATORS RECEIVED INDICATIONS OF A DROPPED CONTROL ELEMENT ASSEMBLY: CEA #8. OPERATORS MATCHED TURBINE LOAD TO REACTOR POWER AT 84%. A REACTOR ENGINEER WAS CALLED IN TO MONITOR CORE PARAMETERS. FOLLOWING AN OPERABILITY CHECK, THE CEA WAS WITHDRAWN AT 0157. THE CEA DROPPED FULLY INTO THE CORE AGAIN AT 0332, AND THE UNIT WAS STABILIZED AT 70% POWER. UTILITY MAINTENANCE PERSONNEL (I&C) WERE CALLED OUT TO ASSIST. ON TWO LATER OCCASIONS, 0557 AND 0653, CEA #8 DROPPED FULLY INTO THE CORE AFTER HAVING BEEN FULLY WITHDRAWN AFTER SPECIFIC MAINTENANCE REPAIRS HAD BEEN ACCOMPLISHED. AT 0653, DURING THE LAST DROP OF CEA #8, A UNIT SHUTDOWN FROM 45% POWER WAS COMMENCED IN ACCORDANCE WITH THE PLANT'S TECH SPECS. AN UNUSUAL EVENT WAS DECLARED AT 0657, DUE TO A TECH SPEC REQUIRED SHUTDOWN, AND THE EVENT WAS TERMINATED AT THE COMPLETION OF THE SHUTDOWN AT 0803. THE UNIT WAS BROUGHT BACK ON LINE AT 2010 ON 6/14/90. THE CAUSE OF THIS EVENT WAS DUE TO A PERSONNEL ERROR IN THE INSTALLATION OF A POWER SUPPLY FUSE IN THE CEA + 12V DC LOGIC CIRCUIT WHICH WAS NOT LOCKED IN PLACE AND CAUSED INTERMITTENT POWER LOSES. A CONTRIBUTING FACTOR IS THAT THE LOCATION OF THE FUSE MAKES IT DIFFICULT TO DISTINGUISH IF THE FUSE IS LOCKED IN PLACE OR JUST PUSHED IN.

[157] SUMMER 1 DOCKET 50-395 LER 90-003
COMPUTER SOFTWARE ERROR CAUSED NONCONSERVATIVE RADIATION MONITOR SETPOINTS.
EVENT DATE: 040690 REPORT DATE: 050790 NSSS: WE TYPE: PWR

(NSIC 218770) ON 4/6/90, SOUTH CAROLINA ELECTRIC & GAS COMPANY (SCE&G) IDENTIFIED A COMPUTER SOFTWARE ERROR WHICH RESULTED IN A NONCOMPLIANCE WITH TECH SPECS 3.3.3.1, "RADIATION MONITORING INSTRUMENTATION" AND 3.3.3.9, "RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION." THE TECH SPECS REQUIRE THE ALARM/TRIP SETPOINTS FOR THE REACTOR BUILDING PURGE EXHAUST RADIATION MONITOR (RM-A4) TO BE SET IN ACCORDANCE WITH THE OFFSITE DOSE CALCULATION MANUAL (ODCM) WHEN PURGE EXHAUST OPERATIONS ARE IN PROGRESS THE CALCULATIONS IN THE ODCM ARE DESIGNED TO ENSURE THAT THE MONITOR WILL ISOLATE THE PURGE PRIOR TO EXCEEDING THE INSTANTANEOUS RELEASE LIMITS OF TECH SPEC 3.11.2.1, "DOSE RATE." AN ERROR IN THE COMPUTER SOFTWARE WHICH PERFORMS THE CALCULATION FOR THE SETPOINT CAUSED A DEFAULT TO THE MAXIMUM INDICATION RANGE OF 1E6 CPM WHENEVER ACTIVITY WAS PRESENT IN THE RELEASE. THE INVESTIGATION DETERMINED THAT THIS NONCONSERVATIVE CALCULATION HAS BEEN UTILIZED BY SCE&G SINCE THE INITIAL STARTUP OF THE PLANT IN OCTOBER OF 1982. THIS EVENT WAS DUE TO PERSONNEL ERROR BY THE SOFTWARE DEVELOPER AND SUBSEQUENTLY IN THE INDEPENDENT VERIFICATION PROCESS OF THE SOFTWARE. IMMEDIATE CORRECTIVE ACTIONS BY SCE&G INVOLVED A REVIEW OF ALL RELEASES FROM THE REACTOR BUILDING PURGE (1982 TO THE PRESENT), READJUSTMENT OF SETPOINTS, AND ESTABLISHMENT OF MANUAL CALCULATIONS FOR SETPOINTS.

[158] SURRY 1 DOCKET 50-280 LER 89-003 REV 01
UPDATE ON DEGRADED INSIDE RECIRCULATION SPRAY PUMP MOTOR POWER FEEDER CABLES AND MOTOR LEADS.
EVENT DATE: 012689 REPORT DATE: 062190 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

VENDOR: CONTINENTAL ELECTRIC CO, INC.
GENERAL ELECTRIC CO.

(NSIC 218682) ON JANUARY 20, 1989, UNIT 1 AND 2 WERE IN COLD SHUTDOWN. DURING THE INSTALLATION OF THE UNIT 1 "B" INSIDE RECIRCULATION SPRAY PUMP (IRSP) MOTOR, DAMAGE TO THE INSULATION OF ONE OF THE MOTOR POWER FEEDER CABLES WAS OBSERVED. THE DAMAGED AREA WAS LOCATED INSIDE THE FLEXIBLE CONDUIT CONNECTING THE MOTOR LEADS AT THE JUNCTION BOX TO THE RIGID CONDUIT CONTAINING THE FIELD RUN CABLE LEADS. AN INSPECTION WAS CONDUCTED ON THE UNIT 1 "A" AND UNIT 2 "A" AND "B" MOTORS FEEDER CABLES AND ADDITIONAL DAMAGE TO THE BRAIDED OUTER JACKET OF THE FEEDER CABLE WAS NOTED WITH SOME MINOR FLAWS AND SCRATCHES ON THE INSULATION. ON JANUARY 26, 1989, A REPORT ON THESE FINDINGS WAS MADE TO THE NRC. SUBSEQUENT INSPECTIONS OF THE PUMP MOTORS IDENTIFIED ADDITIONAL CONCERNS RELATED TO MOTOR LEAD SEALING AREAS AND LUG CRIMPING WHICH WERE EVALUATED AND CORRECTED BY THE MANUFACTURER. ON JANUARY 29, A 10CFR50.72 FOUR HOUR REPORT WAS MADE TO THE NRC IDENTIFYING THE CONDITION NOTED IN THE PUMP'S MOTOR LEAD SEALING AREA AND DEFICIENCIES IN THE MOTOR LEAD LUGS RESULTING IN THE UNIT 1,"B" MOTOR AND THE UNIT 2 "B" MOTOR BEING EVALUATED AS HAVING BEEN INOPERABLE.

[159] SURRY 2 DOCKET 50-281 LER 90-003
MANUAL REACTOR TRIP DUE TO FAILURE OF "A" MAIN FEEDWATER REGULATING VALVE.
EVENT DATE: 053190 REPORT DATE: 062890 NSSS: WE TYPE: PWR
VENDOR: BAILEY METER COMPANY

(NSIC 218702) ON MAY 31, 1990 AT 2005 HOURS WITH UNIT 2 AT 100% POWER, A MANUAL REACTOR TRIP WAS INITIATED BY THE UNIT 2 LICENSED CONTROL ROOM OPERATOR. A MALFUNCTION OF THE "A" MAIN FEEDWATER REGULATING VALVE (MFRV) POSITIONER CAUSED THE VALVE TO CLOSE, DECREASING FEEDWATER FLOW TO THE "A" STEAM GENERATOR (S/G) TO NEAR ZERO. OPERATORS PERFORMED THE APPROPRIATE PLANT PROCEDURES AND QUICKLY STABILIZED THE PLANT FOLLOWING THE TRIP. SAFETY SYSTEMS FUNCTIONED AS DESIGNED WITH THE EXCEPTION THAT ONE INDIVIDUAL ROD POSITION INDICATOR (IRPI) ROD BOTTOM BISTABLE LIGHT FAILED TO ILLUMINATE AND SEVERAL OTHERS DID NOT ILLUMINATE IMMEDIATELY. THE FAILURE OF THE MFRV WAS CAUSED BY BLOCKADE OF THE POSITIONER AIR SUPPLY INLET FILTER/ORIFICE ASSEMBLY. THE FAILURE OF THE ROD BOTTOM LIGHT TO ILLUMINATE WAS CAUSED BY A FAULTY LIGHT BULB. THE DELAYED ILLUMINATION OF THE OTHER ROD BOTTOM LIGHTS WAS THE RESULT OF A SEMI-VITAL BUS VOLTAGE FLUCTUATION. THE MAINTENANCE PROCEDURE WILL BE REVISED TO REQUIRE REPLACEMENT OF THE FILTER/ORIFICE ASSEMBLIES DURING EACH REFUELING. OPERATION OF THE LOAD TAP CHANGER FOR THE "C" RESERVE STATION SERVICE TRANSFORMER (RSST) IS UNDER EVALUATION. A FOUR HOUR NON-EMERGENCY REPORT WAS MADE TO THE NUCLEAR REGULATORY COMMISSION PER 10CFR 50.72.B.2.II.

[160] SUSQUEHANNA 2 DOCKET 50-388 LER 90-003 REV 01
UPDATE ON AUTOMATIC DEPRESSURIZATION SYSTEM DECLARED INOPERABLE WHEN CONTAINMENT INSTRUMENT GAS HEADER PRESSURE DROPPED BELOW 135 PSIG DUE TO A PRESSURE RELIEF VALVE LIFTING.
EVENT DATE: 022890 REPORT DATE: 062990 NSSS: GE TYPE: BWR
VENDOR: LONERGAN, J.E., CO.

(NSIC 218726) AT 1110 HOURS ON FEBRUARY 28, 1990, WITH UNIT 2 OPERATING IN CONDITION 1 AT 100% POWER, THE AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) WAS DECLARED WHEN CONTAINMENT INSTRUMENT GAS (CIG) SYSTEM'S HEADER PRESSURE DROPPED BELOW 135 PSIG DUE TO THE UNEXPECTED OPENING OF A PRESSURE RELIEF VALVE. THE CIG SYSTEM AUTOMATICALLY TRANSFERRED TO ITS BACKUP NITROGEN SUPPLIES. THE "A" HEADER PRESSURE CONTINUED TO DECREASE. THE "B" HEADER WAS BEING CONTAINED AT 160 PSIG BY THE BACKUP SUPPLIES. INVESTIGATIONS BY PLANT PERSONNEL FOUND THAT PRESSURE RELIEF VALVE, PSV-22643, LOCATED ON THE "A" HEADER, WAS STUCK OPEN. THE VALVE WAS MANUALLY RE-SEATED AND HEADER PRESSURE RETURNED TO NORMAL. AT 1151 HOURS LCO 3.5.1 ACTION D WAS EXITED WHEN ADS WAS DECLARED OPERABLE. THE CAUSE OF THIS EVENT WAS THE MIS-OPERATION OF PSV-22643. THE VALVE WAS REMOVED FROM THE SYSTEM FOR TESTING AND INSPECTION. NEITHER ACTIVITY IDENTIFIED A SPECIFIC CAUSE FOR THE MIS-OPERATION. AS A PRUDENT MEASURE THE VALVE'S INTERNAL COMPONENTS WERE REPLACED. THE VALVE WAS TESTED AND RE-INSTALLED. THIS EVENT WAS DETERMINED REPORTABLE PER 10CFR50.73(A)(2)(V) IN THAT THE ADS WAS UNABLE TO PERFORM ITS

COMPLETE SAFETY FUNCTION. THERE WERE NO SAFETY CONSEQUENCES OR COMPROMISE TO PUBLIC HEALTH OR SAFETY AS A RESULT OF THIS EVENT.

[161] SUSQUEHANNA 2 DOCKET 50-388 LER 90-005
AUTOMATIC REACTOR SHUTDOWN DUE TO HIGH VESSEL WATER LEVEL CAUSED BY FEEDWATER LEVEL TRANSMITTER FAILURE.
EVENT DATE: 052890 REPORT DATE: 062790 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 218769) AT 0256 HOURS ON 5/28/90, AN AUTOMATIC ACTUATION OF THE RPS OCCURRED ON UNIT 2. THE UNIT HAD BEEN OPERATING AT 100% RATED POWER PRIOR TO THE ACTUATION. FEEDWATER CONTROL SYSTEM TRANSMITTER PDT-C32-2N004B FAILED DOWNSCALE WHICH RESULTED IN THE FEEDWATER MASTER CONTROLLER INCREASING TO 100% DEMAND. REACTOR WATER LEVEL INCREASED FROM THE STEADY STATE LEVEL OF +35 INCHES TO +54 INCHES. WHEN VESSEL LEVEL REACHED +54 INCHES, THE MAIN AND FEED PUMP TURBINES TRIPPED. TRIPPING OF THE MAIN TURBINE RESULTED IN A FAST CLOSURE OF THE TURBINE STOP AND CONTROL VALVES. FAST CLOSURE OF THE TURBINE STOP AND CONTROL VALVES RESULTED IN A REACTOR SCRAM. THE LOWEST VESSEL LEVEL OBSERVED DURING THE TRANSIENT WAS APPROXIMATELY -26 INCHES. THE RCIC SYSTEM WAS MANUALLY INITIATED TO RESTORE REACTOR VESSEL LEVEL TO AN ACCEPTABLE VALUE. A COMPONENT FAILURE IN THE AMPLIFIER CIRCUIT CARD FOR FEEDWATER CONTROL SYSTEM TRANSMITTER PDT-C32-2N004B CAUSED THE TRANSMITTER TO FAIL DOWNSCALE WHICH IN TURN RESULTED IN THE FEEDWATER MASTER CONTROLLER INCREASING TO 100% DEMAND. THE EVENT HAS BEEN DETERMINED TO BE REPORTABLE PER 10CFR50.73(A)(2)(IV), IN THAT AN AUTOMATIC ACTUATION OF THE RPS OCCURRED WHEN THE FEEDWATER LEVEL TRANSMITTER FAILED. THE FAILED AMPLIFIER CIRCUIT CARD IS BEING SENT TO ROSEMOUNT TO PERFORM A FAILURE ANALYSIS TO DETERMINE EXACT CAUSE OF FAILURE.

[162] SUSQUEHANNA 2 DOCKET 50-388 LER 90-006
RESIDUAL HEAT REMOVAL PUMP MOTOR OIL COOLER FAILURE WITH POSSIBLE GENERIC IMPLICATIONS.
EVENT DATE: 052990 REPORT DATE: 070690 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 1 (BWR)
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 218800) ON MAY 29, 1990, WITH UNIT 2 IN COLD SHUTDOWN (CONDITION 4) AND UNIT 1 AT 100% POWER, IT WAS DISCOVERED DURING ROUTINE OPERATOR ROUNDS THAT THE UNIT 2 "C" RHR PUMP MOTOR OIL COOLER HAD DEVELOPED A TUBE LEAK. THE LEAK CAUSED THE RESERVOIR TO OVERFLOW AND THE OIL/WATER MIXTURE FLOWED INTO THE MONITOR INTERNALS. INITIAL ANALYSIS INDICATED THAT THE CAUSE MIGHT HAVE BEEN MICROBIOLOGICALLY INFLUENCED CORROSION (MIC) AND RAPID PENETRATION OF THE TUBE WALL WAS POSSIBLE. BASED ON POTENTIAL FOR COMMON MODE FAILURE, IT WAS DECIDED TO SHUT DOWN UNIT 1 AND NOT RESTART UNIT 2 UNTIL ALL RHR MONITOR OIL COOLERS HAD BEEN REPLACED. ADDITIONAL ANALYSIS OF THE TUBING AND CORROSION DEPOSITS BY CORROSION EXPERTS CONCLUDED THAT THE ROOT CAUSE OF THE FAILURE WAS DUE TO UNDER-DEPOSIT CORROSION ACCELERATED BY THE PRESENCE OF MANGANESE, AND NOT MICROBIOLOGICALLY INFLUENCED CORROSION AS ORIGINALLY SUSPECTED. THIS EVENT WAS INITIALLY DETERMINED TO BE REPORTABLE PER 10CFR50.72(B)(2)(III) AS A CONDITION THAT ALONE COULD HAVE PREVENTED A SAFETY SYSTEM FROM FULFILLING ITS FUNCTION. AFTER ADDITIONAL EXAMINATION AND ANALYSIS, IT IS NOW BELIEVED THAT NEITHER UNIT WAS AT ANY SIGNIFICANT RISK BASED ON THE CONDITION OF THE OTHER RHR PUMP OIL COOLERS. THE DECISION TO SHUTDOWN UNIT 1 AND NOT RE-START UNIT 2 UNTIL THE MOTOR OIL COOLERS WERE REPLACED WAS PRUDENT BASED ON THE DATA AVAILABLE AT THE TIME.

[163] THREE MILE ISLAND 1 DOCKET 50-289 LER 90-003 REV 01
UPDATE ON ISOLATION OF TURBINE BYPASS VALVES DURING PLANT HEATUP DUE TO PERSONNEL ERROR.
EVENT DATE: 022790 REPORT DATE: 070990 NSSS: BW TYPE: PWR

(NSIC 218818) TMI-1 WAS IN A HEATUP MODE ON 2/27/90, WITH THE REACTOR IN A SHUTDOWN CONDITION. THE REACTOR COOLANT SYSTEM (RCS) TEMPERATURE WAS GREATER THAN 250F WHICH, PURSUANT TO TECH SPEC 3.4.1.1.C, REQUIRES THAT 4 OF THE 6 TURBINE BYPASS VALVES BE OPERABLE. AT APPROX. 0400 HOURS, OPERATORS DISCOVERED

THAT MAIN STEAM VALVES MS-V-8A/B WERE CLOSED WHICH ISOLATED THE TURBINE BYPASS VALVES. THUS, THIS EVENT IS REPORTABLE PER 10CFR 50.73(A)(2)(I)(B) DUE TO A CONDITION PROHIBITED BY THE PLANT'S TECH SPECS. VALVES MS-V-8A/B WERE SUBSEQUENTLY OPENED. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. DURING THE MAIN STEAM SYSTEM VALVE ALIGNMENT BEING PERFORMED ON FEBRUARY 17, 1990, OPERATIONS PERSONNEL FOUND MS-V-8A/B IN THE CLOSED POSITION CONTRARY TO THE REQUIREMENTS OF THE VALVE ALIGNMENT PROCEDURE (I.E., PROCEDURE 1106-14). AN ON-DUTY SHIFT SUPERVISOR BELIEVED THAT MS-V-8A/B VALVES WERE NOT PERMITTED TO BE OPENED DUE TO SYSTEM CONFIGURATION TO SUPPORT HEAT SINK PROTECTION SYSTEM (HSPS) TESTING. IT WAS LATER DETERMINED THAT MS-V-8A/B WERE NOT INCLUDED IN THE VALVE LINEUP FOR HSPS TESTING. ADDITIONALLY, PERSONNEL PERFORMING AND REVIEWING PROCEDURE 1106-14 (I.E. AUX. OPERATORS AND SENIOR LICENSED OPERATORS, RESPECTIVELY) FAILED TO IDENTIFY THE CLOSED POSITION OF MS-V-8A/B AS A DEFICIENCY IN ACCORDANCE WITH ADMINISTRATIVE REQUIREMENTS.

[164] TROJAN DOCKET 50-344 LER 90-020
PARTICULATE CHANNEL OF CONTAINMENT RADIATION MONITOR INOPERABLE DUE TO LOSS OF FILTER PAPER AND FAILURE OF FILTER PAPER ALARM.
EVENT DATE: 013090 REPORT DATE: 062290 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 218719) ON 1/30/90, THE TROJAN NUCLEAR PLANT WAS OPERATING AT 100% POWER. AT APPROX. 1855, DURING THE PERFORMANCE OF A PERIODIC OPERATING TEST ON PROCESS RADIATION MONITOR (PRM) 2, A CHECK OF PRM 1 WAS PERFORMED TO OBSERVE THE FILTER PAPER INSTALLATION. IT WAS DISCOVERED THAT THE FILTER PAPER IN THE RADIOACTIVE AIRBORNE PARTICULATE CHANNEL OF PRM 1 WAS DEPLETED. NO ALARMS HAD BEEN RECEIVED AT THE CONTROLLER FOR THIS CONDITION. THE MONITOR WAS DECLARED INOPERABLE AND A NEW SUPPLY OF FILTER PAPER WAS INSTALLED. AN ALARM WAS NOT RECEIVED DUE TO A BROKEN MICRO-SWITCH ON THE PAPER OUT ALARM AND A BROKEN WIRE ON THE PAPER TEAR ALARM. NO PROCEDURES EXISTED TO DETERMINE OPERABILITY OF THE ALARMS ON A PERIODIC BASIS. THE BROKEN MICRO-SWITCH HAS BEEN BYPASSED AND THE PAPER TEAR ALARM HAS BEEN RESTORED. A PERIODIC CHECK HAS BEEN IMPLEMENTED TO MONITOR THE FILTER PAPER SUPPLY. ON ?/16/90, DURING AN EVALUATION OF THE ABOVE EVENT IT WAS DETERMINED THAT THE FILTER PAPER SPEED WAS SET AT 2.5 INCHES PER HOUR INSTEAD OF THE REQUIRED 1 INCH PER HOUR. IMMEDIATE ACTIONS WERE TAKEN TO RESTORE THE FILTER PAPER SPEED TO 1 INCH PER HOUR. THE CAUSE WAS DUE TO INADEQUATE PROCEDURES FOR RESTORING POWER TO THE MONITOR CONTROLLER. THE OPERATING INSTRUCTION HAS BEEN REVISED TO INCLUDE PROVISIONS FOR RESETTING FILTER PAPER SPEED WHEN REENERGIZING THE MONITOR.

[165] TROJAN DOCKET 50-344 LER 90-013 REV 01
UPDATE ON INADEQUATE IMPLEMENTATION OF ADMINISTRATIVE CONTROLS AND PERSONNEL ERROR RESULT IN CLOSING OF A PLANT DOOR AND DISABLING THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM.
EVENT DATE: 042290 REPORT DATE: 061590 NSSS: WE TYPE: PWR

(NSIC 218716) ON 4/22/90, THE TROJAN NUCLEAR PLANT WAS IN THE 1990 REFUELING OUTAGE. AT 1145, THE CONTROL ROOM WAS NOTIFIED THAT PLANT DOOR 25 WAS CLOSED. DOOR 25 OPENS TO THE OUTSIDE AT THE END OF A CORRIDOR BETWEEN THE CONTROL BUILDING AND THE SECURITY DOORS TO THE AUX. BUILDING. THIS DOOR IS REQUIRED TO BE OPEN TO ASSURE THAT THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM IS OPERABLE AND ABLE TO MAINTAIN AT LEAST 0.125 INCHES, WATER GAUGE, PRESSURE WITHIN THE CONTROL ROOM. THE VENTILATION SYSTEM WAS CONSIDERED TO BE INOPERABLE WHILE DOOR 25 WAS CLOSED. TROJAN TECH SPECS REQUIRE SUSPENSION OF ALL OPERATIONS INVOLVING CORE ALTERATIONS OR POSITIVE REACTIVITY CHANGES IF BOTH TRAINS OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM ARE INOPERABLE. CORE REFUELING WAS IN PROGRESS WHEN DOOR 25 WAS FOUND CLOSED. INADEQUATE IMPLEMENTATION OF ADMINISTRATIVE CONTROLS AND PERSONNEL ERROR WERE DETERMINED TO BE THE CAUSE OF THIS EVENT. TESTS CONDUCTED FOLLOWING THE DISCOVERY THAT DOOR 25 WAS CLOSED SHOWED THAT POSITIVE PRESSURE COULD BE MAINTAINED IN THE CONTROL ROOM. THE DOOR HAS BEEN SECURED OPEN WITH A CHAIN AND LOCK. PERSONNEL HAVE BEEN INSTRUCTED ON THE NEED TO COMPLY WITH ALL DOOR POSTINGS.

[166] TROJAN DOCKET 50-344 LER 90-016
PERSONNEL ERROR LEADS TO INADEQUATE PROCEDURE AND INCOMPLETE INSTRUMENT CALIBRATION.
EVENT DATE: 051490 REPORT DATE: 061390 NSSS: WE TYPE: PWR

(NSIC 218766) ON 5/14/90, THE TROJAN NUCLEAR PLANT WAS SHUTDOWN FOR THE 1990 REFUELING OUTAGE. DURING A REVIEW OF REACTOR TRIP SYSTEM (RTS) AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) INSTRUMENT SETPOINTS, IT WAS DISCOVERED THAT THE PROCEDURES USED TO PERFORM THE CHANNEL CALIBRATION FOR MAIN STEAM SYSTEM FLOW INSTRUMENTS DID NOT CONTAIN ALL OF THE ELEMENTS REQUIRED TO ENSURE THE INSTRUMENT LOOP ACCURACY ASSUMED IN THE PLANT SETPOINT ANALYSES. THE MISSING ELEMENT WAS BENCHMARKING THE OUTPUT OF THE STEAM FLOW INSTRUMENTS SO THEIR OUTPUT CORRESPONDS WITH THE FEEDWATER FLOW INSTRUMENTS' OUTPUT WHEN THE PLANT IS OPERATING, AT POWER, UNDER STEADY STATE CONDITIONS. THIS TYPE OF CALIBRATION IS REQUIRED BECAUSE THE ARRANGEMENT OF THE STEAM FLOW INSTRUMENTS PROVIDES NO METHOD FOR ACCURATELY PREDICTING DIFFERENTIAL PRESSURE FOR A GIVEN FLOW. THE MAIN STEAM FLOW INSTRUMENTS SUPPLY INPUT TO THE RTS STEAM FLOW/FEEDWATER FLOW MISMATCH AND THE ESFAS HIGH STEAM FLOW TRIPS. THIS EVENT WAS THE RESULT OF PERSONNEL ERROR IN FAILURE TO ENSURE THAT APPLICABLE PROCEDURES INCLUDED A REQUIREMENT TO PERFORM THE BENCHMARK CALIBRATION. PLANT PROCEDURES WILL BE CHANGED TO INCORPORATE THE REQUIREMENTS TO PERFORM THE BENCHMARK CALIBRATION. THE BENCHMARK CALIBRATIONS WILL BE PERFORMED DURING PLANT STARTUP FROM THE CURRENT REFUELING OUTAGE, IF REQUIRED.

[167] TROJAN DOCKET 50-344 LER 90-015
INADEQUATE ORIGINAL DESIGN OF CONTROL ROOM EMERGENCY VENTILATION SYSTEM COOLERS RESULTS IN PLANT OPERATION IN AN UNANALYZED CONDITION.
EVENT DATE: 051690 REPORT DATE: 070290 NSSS: WE TYPE: PWR

(NSIC 218717) ON 5/16/90, THE TROJAN NUCLEAR PLANT WAS IN THE 1990 REFUELING OUTAGE. DURING A DESIGN REVIEW OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM, IT WAS DISCOVERED THAT THE SYSTEM'S CALCULATED COOLING CAPACITY WAS NOT ADEQUATE, AND THAT CONTROL ROOM TEMPERATURE COULD EXCEED THE DESIGN LIMIT OF 110F DURING A DESIGN BASIS ACCIDENT WHEN OFFSITE POWER REMAINED AVAILABLE. THE CONTROL ROOM HEAT LOAD IS HIGHER WHEN OFFSITE POWER IS AVAILABLE THAN WHEN IT IS LOST BECAUSE MORE EQUIPMENT AND LIGHTING IN THE CONTROL ROOM REMAIN ENERGIZED. THIS CONDITION WAS THE RESULT OF AN INADEQUATE ORIGINAL DESIGN. CORRECTIVE ACTIONS: IN 1988, A SUPPLEMENTAL COOLING SYSTEM WAS INSTALLED TO HELP THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM MAINTAIN LOWER CONTROL ROOM TEMPERATURES. THE SUPPLEMENTAL COOLING SYSTEM WAS UPGRADED TO MEET SAFETY RELATED, SEISMIC CATEGORY I CRITERIA AND IS NOW CONSIDERED A REQUIRED PORTION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM. THE DESIGN CALCULATION RELATED TO THE CONTROL ROOM HEAT LOADS HAS BEEN REVISED TO REFLECT THE WORST-CASE HEAT LOAD CONDITIONS UNDER WHICH THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM MUST OPERATE (I.E. WHEN OFFSITE POWER REMAINS AVAILABLE). FINAL CONTROL ROOM TEMPERATURE CALCULATIONS WILL BE PERFORMED AND THE RESULTS WILL BE REPORTED IN A SUPPLEMENT TO THIS LICENSEE EVENT REPORT.

[168] TROJAN DOCKET 50-344 LER 90-018
LACK OF A PERIODIC COOLER INSPECTION AND CLEANING PROGRAM RESULTS IN EXCESSIVE COOLER BLOCKAGE AND OPERATION IN AN UNANALYZED CONDITION.
EVENT DATE: 052590 REPORT DATE: 062690 NSSS: WE TYPE: PWR

(NSIC 218718) ON APRIL 27, 1990 THE TROJAN NUCLEAR PLANT WAS IN A REFUELING OUTAGE. INSPECTIONS OF THE "A" TRAIN CONTAINMENT SPRAY AND CENTRIFUGAL CHARGING PUMP ROOM COOLERS WERE IN PROGRESS. THE INSPECTIONS REVEALED THAT UP TO 13 PERCENT OF THE CONTAINMENT SPRAY PUMP ROOM COOLER'S TUBES WERE BLOCKED BY SILT AND CLAM SHELLS. THE SIZE OF THIS COOLER PROVIDED SUFFICIENT MARGIN FOR IT TO BE CONSIDERED OPERABLE. AS A PRECAUTIONARY MEASURE, IT WAS DECIDED TO INSPECT 23 ADDITIONAL ROOM COOLERS WITH LESS AVAILABLE MARGIN. ON MAY 25, 1990, EVALUATION OF THE ADDITIONAL INSPECTIONS SHOWED THAT THE "A" TRAIN ELECTRICAL SWITCHGEAR ROOM COOLERS DID NOT HAVE SUFFICIENT CAPACITY TO MAINTAIN ROOM TEMPERATURES WITHIN DESIGN LIMITS UNDER ORIGINALLY ANALYZED DESIGN CONDITIONS. THE CAUSE OF THE TUBE BLOCKAGE WAS THE LACK OF A PERIODIC ROOM COOLER INSPECTION AND CLEANING

PROGRAM AT TROJAN. ALL SERVICE WATER SUPPLIED ENGINEERED SAFETY FEATURES EQUIPMENT ROOM COOLERS, EXCEPT THREE WHICH HAVE A LARGE DESIGN MARGIN AND TWO IN AN AREA WHERE ALTERNATE COOLING IS AVAILABLE, WERE INSPECTED AND CLEANED DURING THE REFUELING OUTAGE. THE REMAINING FIVE COOLERS WILL BE INSPECTED PRIOR TO RESTART FROM THE 1991 REFUELING OUTAGE. A COOLER PREVENTATIVE MAINTENANCE AND INSPECTION PLAN IS BEING DEVELOPED.

[169] TROJAN DOCKET 50-344 LER 90-017
LACK OF PROPER DESIGN INTERFACE DURING ORIGINAL PLANT DESIGN COULD CAUSE COMMON MODE FAILURE OF (REACTOR TRIP OR ESFAS FUNCTION.
EVENT DATE: 053190 REPORT DATE: 070290 NSSS: WE TYPE: PWR

(NSIC 218765) ON 5/18/90 A DESIGN BASIS WALKDOWN OF THE RPS IDENTIFIED THAT CIRCUITRY FOR THE UNDERVOLTAGE (UV) AND UNDERFREQUENCY (UF) REACTOR TRIPS, LOCATED INSIDE NON-CLASS 1E SWITCHGEAR (12 KILOVOLT (KV)) WHICH POWERS THE REACTOR COOLANT PUMPS, DID NOT APPEAR TO MEET SEPARATION CRITERIA STATED IN THE FSAR. IT WAS DETERMINED ON MAY 31 THROUGH FURTHER EVALUATION THAT THE UV AND UF REACTOR TRIP CIRCUITRY INSIDE THE 12 KV SWITCHGEAR WAS NON-CLASS 1E. THE UV AND UF REACTOR TRIPS ARE CREDITED IN THE SAFETY ANALYSIS AS THE PRIMARY PROTECTION FOR A COMPLETE LOSS OF FORCED REACTOR COOLANT FLOW ACCIDENT. FURTHERMORE, IT WAS DETERMINED THAT POWER INPUTS USED FOR THE UV AND UF INPUT CHANNELS ALSO POWER THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) OUTPUT RELAYS FOR EACH TRAIN. A FAILURE MODE WHICH CAUSES A GROUND OF THE UV OR UF NON-CLASS 1E CIRCUITRY COULD AFFECT THE ESFAS RELAY POWER SUPPLY. THE ISOLATION SCHEME PROVIDED FOR THESE CIRCUITS CONSISTS OF OVERCURRENT DEVICES. IT WAS DISCOVERED DURING THE INVESTIGATION THAT THE COORDINATION WOULD NOT NECESSARILY GUARANTEE ISOLATION OF THE NON-CLASS 1E CIRCUITRY PRIOR TO LOSS OF POWER TO THE ESFAS OUTPUT RELAYS. THE REASON THAT THE CIRCUITS WHICH SENSE UV AND UF CONDITIONS ON THE REACTOR COOLANT PUMP BUSES ARE NON-CLASS 1E IS LACK OF PROPER DESIGN INTERFACE BETWEEN THE NSSS VENDOR AND THE ARCHITECT-ENGINEER.

[170] TROJAN DOCKET 50-344 LER 90-021
INCOMPLETE SURVEILLANCE TEST ON EDG DECOUPLE CIRCUITRY DUE TO AN INADEQUATE SURVEILLANCE PROCEDURE.
EVENT DATE: 060690 REPORT DATE: 070690 NSSS: WE TYPE: PWR
VENDOR: AMERACE CORP.

(NSIC 218793) ON JUNE 6, 1990, THE TROJAN NUCLEAR PLANT WAS IN COLD SHUTDOWN, MODE 5, WITH A "B" TRAIN OUTAGE IN PROGRESS. A MAINTENANCE REQUEST WAS IN PROGRESS TO CONDUCT SPEED DROOP PERFORMANCE CHECKS OF THE 'B' EMERGENCY DIESEL GENERATOR (EDG) ELECTRIC GOVERNOR SPEED DROOP CIRCUIT WHICH IS A PART OF THE DECOUPLE CIRCUIT. THE DECOUPLE CIRCUIT ISOLATES REMOTE CONTROL IN THE CONTROL ROOM FROM LOCAL CONTROL AT THE DIESEL ENGINE PANEL IN THE EVENT OF A FIRE IN THE CONTROL ROOM OR CABLE SPREADING ROOM. ONE OF TWO PARALLEL RELAYS IN THE DECOUPLE CIRCUIT WAS FOUND TO BE OPERATING INTERMITTENTLY THE RELAY WAS REPLACED AND POST MAINTENANCE TESTING WAS INITIATED TO CONFIRM OPERABILITY. IT WAS IDENTIFIED THAT THE PERIODIC OPERATING TEST WHICH IS USED TO SATISFY THE TECHNICAL SPECIFICATION SURVEILLANCE WAS INADEQUATE IN THAT IT FUNCTIONALLY TESTED ONLY ONE OF THE TWO PARALLEL RELAYS. THE SURVEILLANCE PROCEDURE HAS BEEN REVISED TO ADEQUATELY TEST THE CIRCUITRY AND A REVIEW OF OTHER DECOUPLE CIRCUITS IS BEING PERFORMED THERE WAS NO IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC.

[171] TROJAN DOCKET 50-344 LER 90-023
INADEQUATE TEMPORARY PROCEDURE REVISION LEADS TO FAILURE TO DOCUMENT VERIFICATION OF REACTOR COOLANT FLOW DURING BORON DILUTION.
EVENT DATE: 060890 REPORT DATE: 070990 NSSS: WE TYPE: PWR

(NSIC 218803) ON 6/8/90, THE TROJAN NUCLEAR PLANT WAS IN THE 1990 REFUELING OUTAGE. BOTH TRAINS OF THE RESIDUAL HEAT REMOVAL SYSTEM WERE IN OPERATION. AT 1857, OPERATIONS TO FILL THE REACTOR COOLANT SYSTEM AND REDUCE ITS BORON CONCENTRATION WERE STARTED. THE SYSTEM FILL AND BORON CONCENTRATION REDUCTION WERE BEING PERFORMED IN ACCORDANCE WITH A TEMPORARY REVISION TO PLANT OPERATING INSTRUCTION 3-1, REACTOR COOLANT SYSTEM VENT AND FILL. THE REACTOR COOLANT

SYSTEM FILL WAS COMPLETED AT 2340 AND THE BORON CONCENTRATION HAD BEEN REDUCED FROM AN INITIAL VALUE OF 2193 PPM TO 1738 PPM. ON 6/15/90, A MEMBER OF THE PLANT SYSTEMS ENGINEERING STAFF WAS REVIEWING THE TEMPORARY REVISION TO OPERATING INSTRUCTION 3-1 AND DISCOVERED IT DID NOT HAVE REQUIREMENTS TO VERIFY THE RATE OF COOLANT FLOW IN THE REACTOR COOLANT SYSTEM PRIOR TO STARTING, OR DURING THE BORON DILUTION PROCESS. THIS EVENT WAS THE RESULT OF AN INADEQUATE AND IMPROPER REVISION TO THE PROCEDURE. THE PROCEDURE REVISION HAS BEEN WITHDRAWN. OTHER PROCEDURES WILL BE REVIEWED AND REVISED, AS APPROPRIATE, TO FULLY ADDRESS THE COOLANT FLOW SURVEILLANCE REQUIREMENTS. A PERMANENT CAUTION TAG HAS BEEN PLACED ON THE MAKEUP CONTROL SWITCH TO REMIND PLANT OPERATIONS STAFF OF THE FLOW SURVEILLANCE REQUIREMENTS.

[172] TURKEY POINT 3 DOCKET 50-250 LER 90-001 REV 01
UPDATE ON LIQUID EFFLUENT PROCESS RADIATION MONITOR R-18 INOPERABLE DURING A LIQUID RELEASE DUE TO A CONTROL CIRCUIT MALFUNCTION.
EVENT DATE: 011290 REPORT DATE: 061590 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)
VENDOR: NUCLEAR RESEARCH CORP.

(NSIC 218753) AT 1043, ON 1/12/90, WITH UNIT 3 IN MODE 1 AT 100% POWER AND UNIT 4 IN MODE 4 (HOT SHUTDOWN), A REACTOR CONTROL OPERATOR (RCO) NOTED AT THE END OF A RELEASE THAT THE LIQUID EFFLUENT PROCESS RADIATION MONITOR (PRM R-18) CHANNEL HAD FAILED. THE RATEMETER DID NOT HAVE DISPLAY INDICATION, THE ASSOCIATED CHART RECORDER WAS OFF SCALE HIGH, NO ALARM CONDITIONS EXISTED, AND AUTOMATIC CLOSURE OF CONTROL VALVE RCV-018 HAD NOT OCCURRED. A SIMILAR EVENT OCCURRED ON 12/22/89. THE PRM R-18 VENDOR BELIEVES AN INTERMITTENT FAILURE OF THE +5 VOLT LOW VOLTAGE POWER SUPPLY WITHIN THE RADIATION MONITOR CABINET CAUSED THESE EVENTS. FAILURE OF THE + 5 VOLT POWER SUPPLY WOULD AFFECT THE ABILITY TO CAUSE AUTOMATIC CLOSURE OF THE LIQUID EFFLUENT RELEASE PATH CONTROL VALVE UPON REACHING THE PREDETERMINED ALARM SETPOINT. COGNITIVE ERROR BY LICENSED UTILITY PERSONNEL CONTRIBUTED TO THESE EVENTS. FAILURE TO FREQUENTLY MONITOR THE PRM R-18 CHANNEL DURING A RELEASE IS NOT IN ACCORDANCE WITH OPERATING PROCEDURE OP 5163.2; HOWEVER, FPL BELIEVES THAT THE PRM R-18 CHANNEL ALARM SETPOINT WAS NOT REACHED DURING THE RELEASES. A REVIEW OF TECH SPEC TABLE 3.9-2, ITEM 1.A CONCLUDED THAT BOTH EVENTS ARE REPORTABLE. THE LOW VOLTAGE POWER SUPPLY, PROCESSOR CARD, TIMER CARD, AND LIMIT SWITCH CARD WERE REPLACED.

[173] TURKEY POINT 3 DOCKET 50-250 LER 90-009
BREATHING AIR CONTAINMENT ISOLATION VALVE CV-3-6165 FOUND PINNED OPEN WHILE IN MODE 3 (HOT STANDBY).
EVENT DATE: 051990 REPORT DATE: 061190 NSSS: WE TYPE: PWR

(NSIC 218503) AT 0610, ON MAY 19, 1990, WITH UNIT 3 IN MODE 3 (HOT STANDBY), BREATHING AIR SYSTEM PHASE "A" CONTAINMENT ISOLATION VALVE CV-3-6165 WAS DISCOVERED TO BE IN THE PINNED OPEN POSITION. VALVE CV-3-6165 MUST BE PINNED CLOSED AND LOCKED PRIOR TO ENTERING MODE 4 TO MEET THE REQUIREMENTS OF TECHNICAL SPECIFICATIONS 3.0.4 (OPERATIONAL MODE CHANGES), 3.[illegible] (CONTAINMENT INTEGRITY), AND 3.3.3 (CONTAINMENT ISOLATION VALVES). UNIT 3 ENTERED MODE 4 AT 0535, ON MAY 18, 1990. VALVE CV-3-6165 WAS PINNED CLOSED AND LOCKED AT 0627, ON MAY 19, 1990. THE ROOT CAUSE FOR FAILURE TO VERIFY VALVE CV-3-6165 IN THE CORRECT POSITION PRIOR TO ENTERING MODE 4 IS INADEQUATE PROCEDURAL CONTROLS. OPERATING SURVEILLANCE PROCEDURE 3-OSP-053.4, "CONTAINMENT INTEGRITY PENETRATION ALIGNMENT VERIFICATION," SPECIFIED A NORMAL VALVE POSITION OF "OPERABLE", WHICH WOULD BE EXPECTED FOR VALVES RECEIVING A CONTAINMENT ISOLATION SIGNAL. HOWEVER, TO PROTECT BREATHING AIR USERS, VALVE CV-3-6165 IS PINNED OPEN WHEN THE BREATHING AIR SYSTEM IS IN USE TO PREVENT INADVERTENT VALVE CLOSURE. ON-THE-SPOT-CHANGES (OTSCS) HAVE BEEN PREPARED FOR PROCEDURES 3/4-OSP-053.4 SPECIFYING THE CORRECT NORMAL VALVE POSITION FOR VALVES CV-*-6165.

[174] TURKEY POINT 3 DOCKET 50-250 LER 90-011
HI-HI STEAM GENERATOR WATER LEVEL TURBINE TRIP AND SUBSEQUENT REACTOR TRIP DUE TO FAILURE OF A SWITCH IN A FEEDWATER VALVE CONTROLLER HAND/AUTO STATION.
EVENT DATE: 061090 REPORT DATE: 062990 NSSS: WE TYPE: PWR

VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 218815) ON JUNE 9, 1990, AT 0648 EDT, WITH UNIT 3 IN MODE 1 (POWER OPERATION) AT 26 PERCENT POWER AND UNIT 4 IN MODE 1 AT 100 PERCENT POWER, UNIT 3 EXPERIENCED A HI-HI STEAM GENERATOR WATER LEVEL TURBINE TRIP AND SUBSEQUENT REACTOR TRIP. ALL SAFETY SYSTEMS PERFORMED AS DESIGNED. AFTER THE TRIP, THE OPERATORS STABILIZED THE UNIT IN MODE 3 (HOT STANDBY) BY USING APPLICABLE PROCEDURES. THE CAUSE OF THIS EVENT WAS A MALFUNCTION OF THE 3C FEEDWATER REGULATOR HAND/AUTO STATION OPEN PUSHBUTTON SWITCH FOR VALVE CONTROLLER FC-3-498F. THE SWITCH IS A MOMENTARY ACTION SWITCH DESIGNED TO SPRING BACK TO THE NO CONTACT POSITION UPON RELEASE. THE SWITCH WAS FOUND SPRUNG BACK TO THE NO CONTACT POSITION, BUT THE SWITCH CONTACT WAS STILL MADE. THIS CAUSED A FULL OPEN DEMAND CONDITION CAUSING THE 3C FEEDWATER REGULATING VALVE TO FULLY OPEN. THE FAILED 3C FEEDWATER REGULATING VALVE HAND/AUTO STATION AND THE HAND/AUTO STATION FOR THE 3B FEEDWATER REGULATING VALVE WERE REPLACED WITH HAND/AUTO STATIONS HAVING NEW STYLE SWITCHES. THE HAND/AUTO STATION FOR THE 3A FEEDWATER REGULATING VALVE HAD BEEN REPLACED IN JUNE, 1989. ON JUNE 9, 1990, AT 0716 EDT, THE NRC WAS NOTIFIED OF THIS EVENT IN ACCORDANCE WITH 10 CFR 50.72(B)(2)(II).

[175] TURKEY POINT 3 DOCKET 50-250 LER 90-012
A POSTULATED FAILURE OF A SINGLE MANUAL RESET PUSHBUTTON COULD RENDER BOTH TRAINS OF CONTAINMENT SPRAY INOPERABLE.
EVENT DATE: 061390 REPORT DATE: 070290 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 218693) AT 1500, ON JUNE 13, 1990, WITH UNITS 3 AND 4 IN MODE 1 AT 50 PERCENT POWER AND 100 PERCENT POWER, RESPECTIVELY, FPL DETERMINED THAT USE OF A SINGLE RESET PUSHBUTTON FOR BOTH TRAINS OF CONTAINMENT SPRAY (CS) WAS OUTSIDE THE DESIGN BASIS OF THE PLANTS. A POSTULATED FAILURE OF THE PLUNGER ON THE FIRST CONTACT BLOCK OF THE RESET PUSHBUTTON COULD BLOCK BOTH REDUNDANT CS TRAINS FROM AUTOMATICALLY ACTUATING UPON RECEIPT OF CONCURRENT HIGH AND HIGH-HIGH CONTAINMENT PRESSURE SIGNALS. AT 1547, FPL NOTIFIED THE NRC OPERATIONS CENTER OF A SIGNIFICANT EVENT IN ACCORDANCE WITH 10CFR50.72(B)(2)(III)(D). THE CAUSE FOR THE CONDITION IS AN INADEQUATE DESIGN. DURING CONSTRUCTION OF TURKEY POINT UNITS 3 AND 4, WESTINGHOUSE TOOK CREDIT FOR THE INHERENT DESIGN OF THE CS RESET PUSHBUTTON TO PROVIDE CHANNEL INDEPENDENCE. HOWEVER, A SINGLE FAILURE OF THE CS RESET PUSHBUTTON WAS NOT ANALYZED FOR ADVERSE IMPACT ON DESIGN REDUNDANCY REQUIREMENTS. THE CS RESET PUSHBUTTONS ON UNITS 3 AND 4 WERE VISUALLY INSPECTED. BOTH RESET PUSHBUTTONS WERE VERIFIED TO BE PROPERLY RESET. MANUAL ACTUATION OF THE CS PUMPS AND ASSOCIATED DISCHARGE VALVES ARE NOT AFFECTED BY THE POSTULATED FAILURE OF THE RESET PUSHBUTTON. EMERGENCY OPERATING PROCEDURES ADDRESS MANUAL INITIATION OF CS. FPL ENGINEERING IS EVALUATING THE USE OF A SINGLE SWITCH FOR BOTH TRAINS OF A SAFETY FUNCTION IN OTHER APPLICATIONS.

[176] TURKEY POINT 4 DOCKET 50-251 LER 90-004
MANUAL REACTOR TRIP DURING PERFORMANCE OF AN OPERATIONAL SURVEILLANCE PROCEDURE DUE TO PERSONNEL ERROR.
EVENT DATE: 052690 REPORT DATE: 061190 NSSS: WE TYPE: PWR

(NSIC 218504) ON MAY 26, 1990, AT 0556 EDT, DURING START-UP OF BOTH UNITS, WITH UNIT 3 IN MODE 3 (HOT STANDBY) AND UNIT 4 IN MODE 2 (START-UP) AT APPROXIMATELY ONE (1) PERCENT POWER, A LICENSING TRAINEE UNDER THE DIRECTION OF THE UNIT 4 REACTOR OPERATOR MANUALLY TRIPPED THE UNIT 4 REACTOR. THIS OCCURRED DURING THE RESTORATION PHASE OF THE TURBINE VALVE TEST ON UNIT 4. THE UNIT 4 REACTOR WAS MANUALLY TRIPPED IN AN ERRONEOUS RESPONSE TO STEP 7.2.59 OF PROCEDURE 4-OSP-089 "MAIN TURBINE VALVES OPERABILITY TEST." SUBSEQUENT TO THE TRIP, THE OPERATORS VERIFIED THE UNIT TO BE IN A STABLE CONDITION BY USING APPLICABLE EMERGENCY OPERATING PROCEDURES. THIS EVENT WAS CAUSED BY COGNITIVE PERSONNEL ERROR ON THE PART OF PLANT LICENSED OPERATORS. TO PRECLUDE RECURRENCE OF THIS EVENT, PROCEDURES 3-OSP-089 AND 4-OSP-089 HAVE BEEN REVISED TO CLARIFY THE INTENT OF STEP 7.2.59. IN ADDITION, THIS EVENT IS BEING REVIEWED WITH APPLICABLE OPERATIONS PERSONNEL. ON MAY 26, 1990, AT 0634 EDT, THE NRC WAS NOTIFIED OF THIS EVENT IN ACCORDANCE WITH 10 CFR 50.72(B)(2)(II)

[177] TURKEY POINT 4 DOCKET 50-251 LER 90-003
INTAKE COOLING WATER (ICW) FLOW TO THE COMPONENT COOLING WATER (CCW) HEAT EXCHANGERS DECREASED BELOW 15,400 GPM WHILE CLEANING THE 4A ICW/CCW BASKET STRAINER.
EVENT DATE: 060590 REPORT DATE: 070290 NSSS: WE TYPE: PWR

(NSIC 218754) AT 1130, ON 6/5/90, WITH UNIT 4 IN MODE 1 AT 100% POWER, THE INTAKE COOLING WATER (ICW) FLOW RATE TO THE COMPONENT COOLING WATER (CCW) HEAT EXCHANGERS WAS VERIFIED TO BE BELOW THE CAUTION LEVEL ICW FLOW RATE OF 15,400 GPM SPECIFIED IN OPERATING PROCEDURE 4-OP-019, "INTAKE COOLING WATER SYSTEM." WITH THE 4A ICW HEADER OUT OF SERVICE TO MANUALLY CLEAN THE 4A ICW/CCW BASKET STRAINER, THIS CONDITION WAS CONSIDERED TO PLACE BOTH ICW HEADERS IN AN INOPERABLE STATUS. UNIT 4 ENTERED TS 3.0.1 AT THIS TIME. AT 1149, THE 4A ICW/CCW BASKET STRAINER WAS RETURNED TO SERVICE AND THE ICW FLOW RATE TO ENSURE REMOVAL OF THE DESIGN BASIS CCW SYSTEM HEAT LOAD. AT 1223, FPL NOTIFIED THE NRC OPERATIONS CENTER OF A SIGNIFICANT EVENT IN ACCORDANCE WITH 10CFR50.72(B)(1)(II)(B). THE ICW FLOW RATE HAD DECREASED TO 11,900 GPM. AN ENGINEERING CALCULATION, BASED ON THE PLANT CONDITIONS THAT EXISTED AT THE TIME, VERIFIED THAT THE ICW FLOW RATE OF 11,900 GPM WOULD ENSURE REMOVAL OF THE DESIGN BASIS CCW SYSTEM HEAT LOAD. THE ICW FLOW RATE WAS NOT BELOW THAT REQUIRED TO REMOVE DESIGN BASIS HEAT LOADS AND THEREFORE, THE 4B ICW HEADER WAS NOT INOPERABLE. AT 1520, ON 6/27/90, FPL RETRACTED THE SIGNIFICANT EVENT NOTIFICATION.

[178] VERMONT YANKEE DOCKET 50-271 LER 90-008
FAILURE TO MEET SEPARATION CRITERIA FOR POWER CABLES TO REGULATORY GUIDE 1.97 INSTRUMENTATION LOOPS.
EVENT DATE: 052990 REPORT DATE: 062990 NSSS: GE TYPE: BWR

(NSIC 218700) SUBMISSION ON MAY 29, 1990, AT APPROXIMATE1Y 1645 HOURS WITH THE REACTOR OPERATING AT 100% POWER, IT WAS IDENTIFIED THAT FOUR CABLES PROVIDING POWER TO CERTAIN POST-ACCIDENT MONITORING INSTRUMENT LOOPS (REGULATORY GUIDE 1.97) WERE NOT ROUTED IN ACCORDANCE WITH REQUIRED SEPARATION CRITERIA. IT WAS DETERMINED THAT THE FOUR CABLES ARE ROUTED IN CABLE TRAYS USED TO CARRY THE CABLES OF THE OPPOSITE DIVISION POWER CABLES. IMMEDIATELY FOLLOWING THE IDENTIFICATION OF THIS EVENT, A JUSTIFICATION FOR CONTINUED OPERATION (JCO) WAS PRESENTED AND ACCEPTED BY THE PLANT OPERATIONS REVIEW COMMITTEE (PORC). VERMONT YANKEE MADE THE DETERMINATION THAT ALTHOUGH THE SEPARATION CRITERIA WAS NOT FULLY MET, ALL INSTRUMENT LOOPS WERE CONSIDERED OPERATIONAL BASED UPON THEIR CONDITION OF BEING NATIONAL ALONG WITH AN EVALUATION OF THE DEFICIENCY RELATIVE TO THE DESIGN BASIS ACCIDENT ANALYSIS. ON MAY 30, 1990, THE NRC APPEARED TO DISAGREE WITH THE VERMONT YANKEE POSITION. THIS RESULTED IN THE FOUR INSTRUMENTATION LOOPS BEING CONSERVATIVELY PLACED IN A 30-DAY LIMITING CONDITION FOR OPERATION (LCO). THIS EVENT WAS IDENTIFIED AS A RESULT OF LER 89-09, IN WHICH VERMONT YANKEE COMMITTED TO VIEW AND CLARIFY THE PLANT SEPARATION CRITERIA ESPECIALLY AS IT RELATES TO INSTRUMENTATION. THE EFFORT TO REVISE THE CRITERIA IS ONGOING.

[179] VERMONT YANKEE DOCKET 50-271 LER 90-009
INADVERTENT REACTOR SCRAM DUE TO A SHORT CIRCUIT ON THE VITAL AC BUS AS A RESULT OF PERSONNEL ERROR.
EVENT DATE: 060190 REPORT DATE: 062990 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 218701) ON 06/01/90, AT 1353 HOURS, WITH THE REACTOR AT 100% POWER, A CONTRACT ELECTRICIAN WORKING ON A NORMAL/EMERGENCY LIGHTING PANEL ALLOWED A GROUND WIRE TO COME IN CONTACT WITH A LIVE BUS. THE RESULTING SHORT CIRCUIT CAUSED THE VITAL AC MOTOR GENERATOR SET (EIIS=EF) TO LOSE THE FIELD EXCITATION AND SUBSEQUENTLY A LOSS OF GENERATOR OUTPUT. ON THE LOSS OF GENERATOR OUTPUT, THE VITAL AC BUS TRANSFERRED TO ITS ALTERNATE SOURCE. THIS CAUSED A PRESSURE TRANSIENT IN THE REACTOR COOLANT SYSTEM DUE TO THE TRANSFER FROM THE ELECTRIC PRESSURE REGULATOR TO THE MECHANICAL PRESSURE REGULATOR, RESULTING IN A REACTOR SCRAM. ON 06/03/90, AT 0103 HOURS, THE REACTOR MODE SWITCH WAS RETURNED TO THE RUN POSITION AND THE IN GENERATOR PHASED TO THE GRID. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR DUE TO A LACK OF MENTAL ATTENTION. CORRECTIVE ACTIONS

WILL INCLUDE ELECTRICAL CONTRACTOR RETRAINING AND INCREASED EMPHASIS SAFETY AND ATTENTION TO DETAIL. AN EVALUATION WILL BE PERFORMED ON THE VITAL AC MG SET TO DETERMINE IF COLLAPSE OF THE GENERATOR FIELD AS A RESULT OF THE FAULT WAS THE APPROPRIATE EQUIPMENT RESPONSE. A PLANT OPERATIONAL REVIEW SUB-COMMITTEE WILL EVALUATE THE NEED FOR A FORMAL GUIDELINE GOVERNING WORK IN ENERGIZED ELECTRICAL EQUIPMENT. NO OTHER INCIDENTS INVOLVING A FAULT ON THE VITAL AC CAUSING A REACTOR SCRAM HAVE OCCURRED IN THE LAST FIVE YEARS.

[180] VOGTLE 1 DOCKET 50-424 LER 90-006 REV 01
UPDATE ON LOSS OF OFFSITE POWER LEADS TO SITE AREA EMERGENCY.
EVENT DATE: 032090 REPORT DATE: 062990 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: VOGTLE 2 (PWR)
VENDOR: CALCON

(NSIC 218736) ON 3-20-90, UNIT 1 WAS IN A REFUELING OUTAGE AND UNIT 2 WAS OPERATING AT 100% POWER. AT 0820 CST, THE DRIVER OF A FUEL TRUCK IN THE SWITCHYARD BACKED INTO A SUPPORT FOR THE PHASE "C" INSULATOR FOR THE UNIT 1 RESERVE AUXILIARY TRANSFORMER (RAT) 1A. THE INSULATOR AND LINE FELL CAUSING A PHASE TO GROUND FAULT. BOTH UNIT 1 RAT 1A AND UNIT 2 RAT 2B HIGH SIDE AND LOW SIDE BREAKERS TRIPPED, CAUSING A LOSS OF OFFSITE POWER CONDITION (LOSP). UNIT 1 DIESEL GENERATOR (DG) 1A AND UNIT 2 DG2B STARTED, BUT DG1A TRIPPED, CAUSING A LOSS OF RESIDUAL HEAT REMOVAL (RHR) TO THE REACTOR CORE SINCE THE UNIT 1 TRAIN "B" RAT AND DG WERE OUT OF SERVICE FOR MAINTENANCE. A SITE AREA EMERGENCY (SAE) WAS DECLARED AND THE SITE EMERGENCY PLAN WAS IMPLEMENTED. THE REACTOR COOLANT SYSTEM HEATED UP TO 136 DEGREES F FROM 90 DEGREES F BEFORE THE DG WAS EMERGENCY STARTED AT 0856 CST AND RHR WAS RESTORED. THE INITIAL NOTIFICATIONS WERE NOT MADE WITHIN THE REQUIRED 15 MINUTES DUE TO THE LOSS OF POWER TO THE EMERGENCY NOTIFICATION NETWORK (ENN). AT 0915 CST, THE SAE WAS DOWNGRADED TO AN ALERT AFTER ONSITE POWER WAS RESTORED. THE DIRECT CAUSE OF THIS SERIES OF EVENTS WAS A COGNITIVE PERSONNEL ERROR. THE TRUCK DRIVER FAILED TO USE PROPER BACKING PROCEDURES AND HIT A SUPPORT, CAUSING THE PHASE TO GROUND FAULT AND LOSP. THE MOST PROBABLE CAUSE OF THE DG1A TRIP WAS THE INTERMITTENT ACTUATION OF THE DG JACKET WATER TEMPERATURE SWITCHES.

[181] VOGTLE 1 DOCKET 50-424 LER 90-012
INADEQUATE TESTING LEADS TO INADEQUATELY PERFORMED SURVEILLANCE.
EVENT DATE: 052990 REPORT DATE: 062890 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: VOGTLE 2 (PWR)

(NSIC 218737) ON 5-29-90, A SYSTEM ENGINEER WAS PERFORMING ESFAS RESPONSE TIME SUMMATIONS BASED ON RESULTS FROM TESTS COMPLETED DURING THE SPRING 1990 REFUELING OUTAGE. THE SUMMATION PROCEDURE REQUIRES THAT THE RESPONSE TIME FOR THE CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS) BE DETERMINED. IT WAS FOUND THAT NO RESPONSE TIME HAD BEEN MEASURED FOR EITHER OF THE UNITS CONTROL BUILDING CONTROL ROOM (CBCR) FANS. FURTHER REVIEW REVEALED THAT SEQUENCER DELAY TIMES (ALSO A PART OF THE OVERALL CREFS RESPONSE TIME) HAD NOT BEEN TAKEN INTO ACCOUNT IN THE ORIGINAL PERFORMANCE OF THIS SURVEILLANCE ON EITHER UNIT. THE SHIFT SUPERVISORS WERE NOTIFIED OF THIS CONDITION AND LIMITING CONDITIONS FOR OPERATION (LCOS) WERE INITIATED FOR BOTH UNITS 1 AND 2 WHICH REQUIRED STARTING TWO TRAINS OF THE CREFS IN THE EMERGENCY MODE. THE CAUSE OF THIS EVENT WAS AN INADEQUATE REVIEW OF THE BASIS FOR THE CONTROL ROOM EMERGENCY FILTRATION SYSTEM RESPONSE TIME WHICH RESULTED IN INADEQUATE PROCEDURES. THE REVIEW FAILED TO ACCOUNT FOR SEQUENCER DELAY TIMES AS REQUIRED BY THE TECHNICAL SPECIFICATIONS. THE FOUR AFFECTED FANS IN UNIT 1 AND UNIT 2 HAVE BEEN TESTED AND THEIR RESPONSE TIMES WERE WITHIN EXPECTED VALUES. SEQUENCER DELAY TIMES HAVE BEEN MEASURED ON BOTH TRAINS FOR UNIT 1 AND UNIT 2 AND THE SUMMED RESPONSE TIMES ARE WITHIN ALLOWABLE VALUES.

[182] VOGTLE 1 DOCKET 50-424 LER 90-013
IMPROPER APPLICATION OF GRACE PERIOD TO CONTAINMENT AIR LOCK SURVEILLANCE.
EVENT DATE: 060190 REPORT DATE: 062990 NSSS: WE TYPE: PWR

(NSIC 218738) ON 6-1-90, A MAINTENANCE ENGINEER DISCOVERED THAT THE CONTAINMENT AIR LOCK LEAKAGE SURVEILLANCE INTERVAL HAD EXCEEDED THE 6 MONTH TEST INTERVAL

SPECIFIED IN TECHNICAL SPECIFICATION (TS) 4.6.1.3.B ON THREE PREVIOUS OCCASIONS. ON EACH OF THESE OCCASIONS, A GRACE PERIOD HAD BEEN APPLIED TO THE SURVEILLANCE INTERVAL; HOWEVER, A FOOTNOTE TO TS 4.6.1.3.B INDICATES THAT THE GRACE PERIOD PROVISIONS OF TS 4.0.2 ARE NOT APPLICABLE. THEREFORE, THE PLANT HAD OPERATED IN A CONDITION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. NO IMMEDIATE ACTION WAS REQUIRED DUE TO THIS DISCOVERY SINCE A GRACE PERIOD WAS NOT CURRENTLY BEING APPLIED TO THE SURVEILLANCE. THE CAUSE OF THIS EVENT WAS COGNITIVE PERSONNEL ERROR INVOLVING A FAILURE TO INCORPORATE THE FOOTNOTE OF TS 4.6.1.3.B INTO THE ORIGINAL SCHEDULING OF THE CONTAINMENT AIR LOCK LEAKAGE SURVEILLANCE. CORRECTIVE ACTION HAS BEEN TAKEN TO REVISE THE PLANT SURVEILLANCE TRACKING PROGRAM TO INCORPORATE THE REQUIREMENTS OF THE FOOTNOTE.

[183] VOGTLE 2 DOCKET 50-425 LER 90-001 REV 01
UPDATE ON MISLEADING TASK SHEET LEADS TO INADEQUATE TECHNICAL SPECIFICATION SURVEILLANCES.
EVENT DATE: 010390 REPORT DATE: 061590 NSSS: WE TYPE: PWR

(NSIC 218542) ON 1-3-90, A SURVEILLANCE TO VERIFY CONTAINMENT INTEGRITY WAS COMPLETED AND REVIEWED. THE SURVEILLANCE VERIFIED THAT VALVES 21204U4293 AND 21204U4324 WERE CLOSED AND SECURED. ON 2-1-90, THE SURVEILLANCE WAS REPEATED AND VALVES 21204U4293 AND 21204U4324 WERE AGAIN VERIFIED TO BE CLOSED AND SECURED. ON 2-28-90, THE SURVEILLANCE WAS AGAIN PERFORMED. DURING THE REVIEW BY THE SHIFT SUPERVISOR (SS), HE NOTED THAT FOR THE PREVIOUS MONTH'S SURVEILLANCE, ONLY 2 OF THE 41 VALVES AND FLANGES LISTED IN THE ASSOCIATED PROCEDURE WERE ADDRESSED. HE INITIATED AN INVESTIGATION WHICH DETERMINED THAT ALL 41 LINE ITEMS SHOULD HAVE BEEN VERIFIED ON 1-3-90 AND 2-1-90 AS REQUIRED BY TECH SPEC (TS) 4.6.1.1.A. THIS SPECIFICATION REQUIRES THAT CONTAINMENT PENETRATIONS WHICH ARE NOT CLOSED BY AUTOMATIC ISOLATION VALVES BE VERIFIED CLOSED AND SECURED AT LEAST ONCE PER 31 DAYS. THEREFORE, THE SURVEILLANCES PERFORMED ON 1-3-90 AND 2-1-90 FAILED TO MEET THE REQUIREMENTS OF TS 4.6.1.1.A. THE PRINCIPAL REASON FOR THE MISSED SURVEILLANCES WAS THE FORMAT OF THE SURVEILLANCE TASK SHEETS (STS'S) WHICH RESULTED IN COGNITIVE PERSONNEL ERRORS ON BEHALF OF THE PERSONNEL INVOLVED SINCE THEY WERE LED TO BELIEVE THAT ONLY TWO VALVES WERE REQUIRED TO BE SURVIELLED. THE STS'S ASSOCIATED WITH VALVES 21204U4293 AND 21204U4324 HAVE BEEN REVISED TO DELETE THESE VALVE NUMBERS AND DIRECT PERSONNEL TO SEE THE APPROPRIATE PROCEDURE.

[184] WATERFORD 3 DOCKET 50-382 LER 90-006
SHUTDOWN COOLING SYSTEM RELIEF VALVE SETPOINT NOT SET IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 121884 REPORT DATE: 062790 NSSS: CE TYPE: PWR

(NSIC 218725) ON MAY 31, 1990, A REVIEW OF TECHNICAL SPECIFICATION (TS) SURVEILLANCE PROCEDURES REVEALED THAT THE LIFT PRESSURE OF THE SHUTDOWN COOLING (SDC) SYSTEM RELIEF VALVE SI-406A WAS 438 PSIA, WHICH EXCEEDED THE MAXIMUM TS ALLOWABLE SETPOINT PRESSURE OF 430 PSIA. A FURTHER REVIEW OF TS SURVEILLANCE PROCEDURES REVEALED THAT BOTH SDC SYSTEM RELIEF VALVES (SI-406A & SI-406B) HAD BEEN SET ABOVE THE MAXIMUM ALLOWED PRESSURE ON TWO PREVIOUS OCCASIONS. THIS EVENT IS THEREFORE REPORTABLE AS PLANT OPERATION PROHIBITED BY TS. THE ROOT CAUSE OF THIS EVENT WAS AN INADEQUATE MAINTENANCE PROCEDURE. THE ALLOWABLE LIFT SETPOINT TOLERANCE OF +/- 2% WAS INCORRECTLY APPLIED TO THE MAXIMUM ALLOWED SETPOINT PRESSURE (430 PSIA) WHICH ALLOWED THE TS MAXIMUM SETPOINT TO BE EXCEEDED. PLANT OPERATION WITH SDC SYSTEM RELIEFS SET 2% ABOVE 430 PSIA WOULD STILL HAVE ENSURED THAT REACTOR COOLANT SYSTEM (RCS) PRESSURE LIMITS WERE MET DURING THE LIMITING PRESSURE TRANSIENT; THEREFORE, THIS EVENT DID NOT RESULT IN AN INCREASED RISK TO THE HEALTH AND SAFETY OF THE PUBLIC OR PLANT PERSONNEL.

[185] WATERFORD 3 DOCKET 50-382 LER 89-006 REV 01
UPDATE ON SAFETY CLASS BREAK REQUIREMENTS NOT MET DUE TO INADEQUATE ADMINISTRATIVE CONTROLS.
EVENT DATE: 010587 REPORT DATE: 061490 NSSS: CE TYPE: PWR

(NSIC 218498) AT 1300 HOURS ON MARCH 27, 1989, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% POWER WHEN EVENT ANALYSIS & REPORTING PERSONNEL

DISCOVERED THAT TWO MANUAL ISOLATION VALVES WHICH SUPPLY COMPONENT COOLING WATER (CCW) TO THE POST ACCIDENT SAMPLING SYSTEM (PASS) COOLERS WERE SHOWN TO BE NORMALLY OPEN ON DESIGN DRAWINGS. EACH VALVE IS LOCATED AT A SAFETY CLASS 3 (SC3) TO NON-NUCLEAR SAFETY (NNS) CLASS BREAK IN THE CCW SYSTEM. BECAUSE THESE VALVES DO NOT HAVE REMOTE OPERATORS, THEY ARE REQUIRED TO BE NORMALLY CLOSED FOR LEAK ISOLATION IN THE EVENT OF A NNS PIPING BREAK. THIS EVENT IS REPORTABLE AS A CONDITION OUTSIDE THE PLANT DESIGN BASIS. THE ROOT CAUSE OF THIS EVENT WAS A PROGRAMMATIC BREAKDOWN IN ADMINISTRATIVE CONTROLS. A 10CFR50.59 EVALUATION WAS REVIEWED BY THE PLANT OPERATIONS REVIEW COMMITTEE. A CONTRIBUTING CAUSE WAS INADEQUATE TRAINING BECAUSE ENGINEERS INVOLVED IN THE REVIEW PROCESS OVERLOOKED THE SAFETY CLASS BREAK REQUIREMENTS. THE VALVES HAVE BEEN CLOSED AND DESIGN DRAWINGS AND PROCEDURES ARE BEING REVISED. A FINAL ENGINEERING EVALUATION HAS DETERMINED THAT COOLING WATER WOULD LAST AT LEAST 1.53 HOURS ASSUMING A NNS PIPING BREAK WITH BOTH VALVES LEFT OPEN. THIS WOULD HAVE ALLOWED SUFFICIENT TIME FOR A LEAK TO BE DETECTED AND ISOLATED. THUS, THIS EVENT DID NOT THREATEN THE HEALTH OR SAFETY OF THE PUBLIC OR PLANT PERSONNEL.

[186] WATERFORD 3 DOCKET 50-382 LER 90-001 REV 01
UPDATE ON CONTROLLED VENTILATION AREA SYSTEM INOPERABLE DUE TO INADEQUATE ADMINISTRATIVE CONTROLS.
EVENT DATE: 022390 REPORT DATE: 061590 NSSS: CE TYPE: PWR

(NSIC 218535) AT APPROXIMATELY 1330 HOURS ON FEBRUARY 23, 1990, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% POWER WHEN ENGINEERING PERSONNEL DISCOVERED CONTROLLED VENTILATION AREA SYSTEM (CVAS) AIRLOCK DOORS D-170 AND D-171 OPEN. THE DOORS WERE OPEN TO EXTEND A SERVICE AIR LINE INTO SAFEGUARDS PUMP ROOM "A" FOR FIRE SEAL REPAIRS. WITH BOTH DOOR 170 AND 171 OPEN, THE CVAS WOULD HAVE BEEN UNABLE TO FULLY PERFORM ITS DESIGN SAFETY FUNCTION OF ESTABLISHING -0.25 INCHES WATER GAGE (WG) NEGATIVE PRESSURE IN THE 60 SECONDS FOLLOWING A SAFETY INJECTION ACTUATION SIGNAL (SIAS). THE ROOT CAUSE OF THIS EVENT IS INADEQUATE ADMINISTRATIVE CONTROLS OVER AIRLOCK DOORS. INSUFFICIENT FORMAL CONTROLS OVER AIRLOCK DOORS ALLOWED DOORS 170 AND 171 TO BE PROPPED OPEN SIMULTANEOUSLY. PROCEDURAL INSTRUCTIONS HAVE BEEN DEVELOPED TO GOVERN THE CONTROL OF AIR LOCK DOORS AND PREVENT RECURRENCE OF THIS EVENT. BECAUSE THE CVAS WAS CAPABLE OF ALIGNING AND INITIATING FILTRATION OF EXHAUST VENTILATION FLOW IN THE CONSERVATIVE DIRECTION (INTO SAFEGUARDS) ROOM "A" AND OUT THROUGH THE CVAS SYSTEM), THIS EVENT DID NOT THREATEN THE HEALTH AND SAFETY OF THE GENERAL PUBLIC OR PLANT PERSONNEL.

[187] WOLF CREEK 1 DOCKET 50-482 LER 90-008
FUEL BUILDING EXHAUST RADIATION MONITOR CAUSES ENGINEERED SAFETY FEATURES EQUIPMENT ACTUATION AS A RESULT OF LOSS OF POWER.
EVENT DATE: 050490 REPORT DATE: 060490 NSSS: WE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 218553) ON 5/4/90, AT APPROXIMATELY 0849 CDT, A FUEL BUILDING VENTILATION ISOLATION SIGNAL (FBIS) AND A CONTROL ROOM VENTILATION ISOLATION SIGNAL (CRVIS) WERE RECEIVED FROM FUEL BUILDING EXHAUST RADIATION MONITOR GG RE-27. ALL REQUIRED ENGINEERED SAFETY FEATURES EQUIPMENT RESPONDED PROPERLY. GG RE-27 WAS PLACED IN BYPASS FOR TROUBLESHOOTING AND THE AFFECTED VENTILATION SYSTEMS WERE RESTORED TO NORMAL CONFIGURATION AT APPROXIMATELY 1040 CDT. DURING TROUBLESHOOTING, A BURNED OUT OPERATE LAMP AND A BLOWN FUSE IN THE POWER SUPPLY TO THE RM-80 MICROPROCESSOR ASSOCIATED WITH GG RE-27 WERE DISCOVERED. WHEN THE FUSE BLEW, THE MONITOR EXPERIENCED A LOSS OF POWER AND INITIATED A FBIS AND CRVIS AS DESIGNED. IT IS BELIEVED THAT A VOLTAGE TRANSIENT ASSOCIATED WITH THE LAMP BURNING OUT CAUSED THE BLOWN FUSE. THE FUSE AND LAMP WERE REPLACED, AND GG RE-27 WAS RETURNED TO SERVICE ON 5/5/90, AT APPROXIMATELY 0811 CDT.

[188] WOLF CREEK 1 DOCKET 50-482 LER 90-010
TECHNICAL SPECIFICATION VIOLATION DUE TO SIMULTANEOUS INOPERABILITY OF TWO AUXILIARY FEEDWATER PUMPS BECAUSE OF SUPPORT SYSTEM INOPERABILITY.
EVENT DATE: 051690 REPORT DATE: 061590 NSSS: WE TYPE: PWR

(NSIC 218552) ON 5/16/90, AT APPROXIMATELY 1545 CDT, AN ENGINEER DISCOVERED THAT THE INSPECTION ACCESS DOOR FOR THE MOTOR DRIVEN AUXILIARY FEEDWATER PUMP "B" (AFP) ROOM COOLER HAD BEEN REMOVED. THE SHIFT SUPERVISOR REINSTALLED THE DOOR AT APPROXIMATELY 1630 CDT. IT WAS CONSERVATIVELY DETERMINED THAT THIS CONDITION RENDERED THE ROOM COOLER INOPERABLE AND CONSEQUENTLY CAUSED AFP "B" TO BE INOPERABLE. DURING THIS TIME, EMERGENCY DIESEL GENERATOR "A" WAS OUT OF SERVICE FOR MAINTENANCE, RENDERING AFP "A" INOPERABLE. BECAUSE THE TECHNICAL SPECIFICIATION (TS) ACTION STATEMENT FOR TWO INOPERABLE AFP'S WAS NOT SATISISFIED, THE EVENT REPRESENTS A CONDITION PROHIBITED BY THE PLANT'S TS. IT WAS DETERMINED THAT THE DOOR HAD BEEN REMOVED AFTER 0800 CDT ON 5/16/90, BY MAINTENANCE PERSONNEL DURING A TROUBLESHOOTING INSPECTION. MAINTENANCE AND OPERATIONS PERSONNEL WERE UNAWARE THAT REMOVAL OF THE DOOR RENDERED THE ROOM COOLER AND CONSEQUENTLY THE AFP INOPERABLE. MAINTENANCE AND OPERATIONS PERSONNEL HAVE BEEN INFORMED THAT ACCESS DOORS MUST BE INSTALLED TO ENSURE OPERABILITY OF ROOM COOLERS THROUGHOUT THE PLANT.

[189] WOLF CREEK 1 DOCKET 50-482 LER 90-012
REACTOR TRIP CAUSED BY STEAM GENERATOR ATMOSPHERIC RELIEF VALVE REMAINING OPEN.
EVENT DATE: 051790 REPORT DATE: 061890 NSSS: WE TYPE: PWR
VENDOR: MASONEILAN INTERNATIONAL, INC.

(NSIC 218778) ON 5/17/90, AT 2305 CDT, WITH THE UNIT AT 0.5% POWER, A REACTOR TRIP SIGNAL, MAIN TURBINE (TA-TRB) TRIP SIGNAL, AUXILIARY FEEDWATER ACTUATION SIGNAL (AFAS), FEEDWATER ISOLATION SIGNAL (FWIS) AND STEAM GENERATOR BLOWDOWN AND SAMPLE ISOLATION SIGNAL (SGBSIS) OCCURRED AS A RESULT OF A LOW-LOW WATER LEVEL IN STEAM GENERATOR 'C'. PRIOR TO THIS EVENT, THE UNIT HAD BEEN TAKEN OFF LINE AND THE STEAM GENERATOR ATMOSPHERIC RELIEF VALVES (ARV'S) WERE BEING USED TO MAINTAIN REACTOR COOLANT SYSTEM TEMPERATURE. AT APPROXIMATELY 2300 CDT, IT WAS DISCOVERED THAT THE 'C' ARV WAS STUCK OPEN. EFFORTS TO CLOSE THE ARV FROM THE CONTROL ROOM WERE UNSUCCESSFUL, AND OPERATORS WERE DISPATCHED TO MANUALLY ISOLATE THE VALVE. THIS VALVE WAS ISOLATED AT 2310 CDT, AND PLANT CONDITIONS WERE STABILIZED. DURING SUBSEQUENT TROUBLESHOOTING, A CURRENT-TO-PNEUMATIC CONVERTER IN THE ARV POSITIONER CIRCUITRY WAS REPLACED. THE ARV WAS VERIFIED TO BE OPERATING PROPERLY AND WAS RESTORED TO SERVICE AT 0716 CDT ON 5/18/90. IT IS BELIEVED THAT THE FAILURE OCCURRED AS A RESULT OF A PRESSURE REGULATING BALL IN THE AIR BLEED OFF LINE OF THE CONVERTER BEING RESTRICTED FROM FREE MOVEMENT. THIS CONDITION CAUSED THE ARV TO REMAIN IN THE OPEN POSITION.

[190] WOLF CREEK 1 DOCKET 50-482 LER 90-013
REACTOR TRIP AND MAIN TURBINE TRIP CAUSED BY HIGH MOISTURE SEPARATOR REHEATER LEVEL.
EVENT DATE: 051990 REPORT DATE: 061890 NSSS: WE TYPE: PWR

(NSIC 218550) ON 5/19/90, AT 2353 CDT, A MAIN TURBINE (TA-TRB) TRIP OCCURRED AS A RESULT OF HIGH-HIGH MOISTURE SEPARATOR REHEATER LEVEL. BECAUSE THE UNIT WAS OPERATING AT GREATER THAN 50% POWER, THE MAIN TURBINE TRIP CAUSED A REACTOR TRIP. AS EXPECTED, AN AUXILIARY FEEDWATER ACTUATION SIGNAL, A FEEDWATER ISOLATION SIGNAL AND A STEAM GENERATOR BLOWDOWN AND SAMPLE ISOLATION SIGNAL ALSO OCCURRED. ALL REACTOR PROTECTION SYSTEM AND ENGINEERED SAFETY FEATURES EQUIPMENT FUNCTIONED PROPERLY. FOLLOWING THIS EVENT, EXTENSIVE TROUBLESHOOTING ACTIVITIES WERE CONDUCTED. A LEVEL SWITCH ON MOISTURE SEPARATOR DRAIN TANK (MSDT) 'A' WAS FOUND TO INITIALLY BE STUCK THUS PREVENTING A MAIN CONTROL ROOM ALARM ON HIGH LEVEL. THE SWITCH WAS REPLACED. THE REMAINDER OF THE LEVEL CONTROL CIRCUITRY FOR MSDT 'A' WAS FOUND TO BE OPERATING PROPERLY. NO ABNORMALITIES WERE IDENTIFIED IN THE DUMP VALVE TO THE CONDENSER, THE NORMAL LEVEL CONTROL VALVE, OR ITS UPSTREAM CHECK VALVE. NO SIGNIFICANT ABNORMALITIES COULD BE IDENTIFIED DURING THE TROUBLESHOOTING. THE UNIT WAS RESTARTED ON 5/20/90, AND LEVELS WERE CLOSELY MONITORED. NO FURTHER DIFFICULTIES WERE ENCOUNTERED IN THE LEVEL CONTROL SYSTEM.

[191] WPPSS 2 DOCKET 50-397 LER 89-040 REV 01
UPDATE ON STANDBY GAS TREATMENT SYSTEM CAPABILITY NOT WITHIN LICENSE BASIS CONSIDERATION FOR SECONDARY CONTAINMENT PERFORMANCE UNDER CERTAIN CONDITIONS DUE TO DESIGN.

EVENT DATE: 091989 REPORT DATE: 061990 NSSS: GE TYPE: BWR

(NSIC 218684) ON SEPTEMBER 19, 1989 IT WAS DETERMINED BY ENGINEERING ANALYSIS THAT UNDER CERTAIN METEOROLOGICAL CONDITIONS, MODERATE WIND AND LOW TEMPERATURE), COINCIDENT WITH A DBA LOCA AND ASSUMED FAILURE OF ONE TRAIN OF THE STANDBY GAS TREATMENT (SGT) SYSTEM, A SITUATION WOULD BE CREATED THAT IS NOT WITHIN THE LICENSING BASIS CONSIDERATION FOR SECONDARY CONTAINMENT PERFORMANCE. THE ENGINEERING ANALYSIS WAS PERFORMED AS A FURTHER CORRECTIVE ACTION FOR LER 88-023. THE WNP-2 FSAR STATES THAT THE SECONDARY CONTAINMENT WILL BE MAINTAINED AT MINIMUM DIFFERENTIAL PRESSURE OF -0.25" W.G. FOLLOWING A POSTULATED LOCA, AND THAT THIS DIFFERENTIAL WILL BE ESTABLISHED WITHIN TWO MINUTES FOLLOWING THE ACCIDENT. RECENT ANALYSIS, BASED UPON STANDBY GAS TREATMENT, SECONDARY CONTAINMENT, STANDBY SERVICE WATER AND WEATHER MODELING, SHOWS THAT DURING POST-LOCA, OR ADVERSE WEATHER, DIFFERENTIAL PRESSURE OF THE SECONDARY CONTAINMENT MAY NOT ALWAYS MEET THE FSAR COMMITMENTS. CERTAIN COMBINATIONS OF POST-LOCA SINGLE ACTIVE FAILURES AND WINTER CONDITIONS ADVERSELY AFFECT SECONDARY CONTAINMENT AND, AS A RESULT, INCREASE SECONDARY CONTAINMENT LEAKAGE.

[192] WPPSS 2 DOCKET 50-397 LER 90-011
HIGH PRESSURE CORE SPRAY SYSTEM PUMP SUCTION VALVE SWITCHOVER ACTUATION DUE TO UNKNOWN CAUSE.
EVENT DATE: 052190 REPORT DATE: 061990 NSSS: GE TYPE: BWR

(NSIC 218727) ON MAY 21, 1990 AT 1525 HOURS, A HIGH PRESSURE CORE SPRAY (HPCS) SYSTEM PUMP SUCTION VALVE SWITCHOVER ACTUATION OCCURRED DURING THE ANNUAL MAINTENANCE AND REFUELING OUTAGE. PLANT CONFIGURATION AT THE TIME WAS SUCH THAT HPCS SUCTION WAS LINED UP TO THE CONDENSATE STORAGE TANKS (THE NORMAL LINE-UP, WITH HPCS-V-1 OPEN), AND THE SUPPRESSION POOL SUCTION VALVE (HPCS-V-15) WAS CLOSED (REFERENCE FIGURE 1). THE SWITCHOVER ACTUATION WAS THE AUTOMATIC CLOSURE OF HPCS-V-1 AND THE OPENING OF HPCS-V-15. AT THE TIME OF THE SUCTION VALVE TRANSFER, PLANT CONTROL ROOM OPERATORS RECEIVED A SUCTION SWITCHOVER SUPPRESSION POOL HIGH LEVEL ANNUNCIATOR, WHICH IMMEDIATELY CLEARED BEFORE THE OPERATORS HAD A CHANCE TO RESET. ALTHOUGH THE SWITCHOVER WAS UNEXPECTED, THE CLOSURE OF HPCS-V-1 AND THE OPENING OF HPCS-V-15 WAS BY PLANT DESIGN. AFTER VERIFYING THAT NO ACTUAL HIGH SUPPRESSION POOL OR LOW CONDENSATE STORAGE TANK LEVEL CONDITION EXISTED, PLANT CONTROL ROOM OPERATORS RESTORED THE SYSTEM TO THE PRE-EVENT LINE-UP (HPCS-V-15 WAS CLOSED AND HPCS-V-1 WAS RE-OPENED), AS AN IMMEDIATE CORRECTIVE ACTION AT 1527 HOURS.

[193] WPPSS 2 DOCKET 50-397 LER 90-012
FIRE IN DIVISION ONE DIESEL GENERATOR CAUSED BY GENERATOR THRUST BEARING FAILURE.
EVENT DATE: 052790 REPORT DATE: 062590 NSSS: GE TYPE: BWR
VENDOR: STEWART & STEVENSON SERVICES, INC.

(NSIC 218728) ON MAY 27, 1990, AT 1742 HOURS, THE DIVISION 1 EMERGENCY DIESEL GENERATOR (DG-GEN-DG1) FAILED APPROXIMATELY SIX HOURS INTO A 24 HOUR FULL LOAD RUN. THE FAILURE OF THE DIESEL GENERATOR SLIP RING END BEARING RESULTED IN A SMALL FIRE IN THE AREA OF THE BEARING ITSELF WHICH WAS QUICKLY EXTINGUISHED. AN EMERGENCY CLASSIFICATION OF UNUSUAL EVENT WAS DECLARED BY THE SHIFT MANAGER AND THE EVENT WAS REPORTED TO THE NRC BETHESDA OPERATIONS CENTER. THERE WERE FOUR ROOT CAUSES DETERMINED TO BE RESPONSIBLE FOR OCCURRENCE OF THIS EVENT: 1) EQUIPMENT - MANUFACTURING ERROR - NOT MADE PER DESIGN - AN EXTRA O-RING GROOVE WAS FOUND MACHINED INTO THE GENERATOR THRUST BEARING BRACKET WHICH PREVENTED THE BEARING OIL RESERVOIR FROM OBTAINING A TIGHT SEAL. THIS RESULTED IN OIL LEAKAGE FROM THE RESERVOIR, OIL STARVATION OF THE THRUST BEARING AND EVENTUAL BEARING FAILURE 2) EQUIPMENT - DESIGN DEFICIENCY - SPECIFICATION LESS THAN ADEQUATE - THE OIL LEVEL BAND FOR THE OIL RESERVOIRS IS TOO NARROW TO ALLOW PRACTICAL MAINTENANCE OF LEVEL; 3) EQUIPMENT - DESIGN DEFICIENCY - LOSS OF MONITORING ALERTNESS - THE TEMPERATURE AND VIBRATION ALARM INSTRUMENTATION AND THE OIL RESERVOIR SIGHT GLASS WERE EVALUATED AS INADEQUATE TO PROVIDE THE OPERATORS WITN SUFFICIENT WARNING TO RESPOND TO THIS TYPE OF EVENT; AND 4) PERSONNEL MANAGEMENT PROGRAMS - FAILURE TO HEED PRECURSORS - NINETEEN INSTANCES OF THRUST.

[194] WPPSS 2 DOCKET 50-397 LER 90-013
ENGINEERED SAFETY FEATURE ISOLATIONS AND ACTUATIONS DUE TO BREAKER TRIP DUE TO UNKNOWN CAUSES.
EVENT DATE: 061090 REPORT DATE: 070690 NSSS: GE TYPE: BWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 218801) ON JUNE 10, 1990 AT 1309 HOURS THE BREAKER (E-CB-73/7A) BETWEEN THE LOW VOLTAGE CRITICAL SWITCHGEAR (E-SL-73) AND MOTOR CONTROL CENTER 7A (E-MC-7A) TRIPPED CAUSING A LOSS OF POWER TO SEVERAL LOADS INCLUDING REACTOR PROTECTION SYSTEM (RPS) BUS "A". LOSS OF POWER TO RPS BUS "A" CAUSED A HALF-SCRAM IN RPS DIVISION "A" AND MULTIPLE PRIMARY CONTAINMENT ISOLATIONS WHICH ARE ENGINEERED SAFETY FEATURE (ESF) ACTUATIONS. AT THE TIME OF THE EVENT THE PLANT WAS IN OUTAGE STATUS IN THE COLD SHUTDOWN MODE. THE LOSS OF RPS "A" POWER CAUSES NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) CONTAINMENT OUTBOARD ISOLATIONS FOR ISOLATION GROUPS (1,2,5,6 AND 7). PLANT OPERATORS RESPONDED BY RESTORING ALL SYSTEMS, INCLUDING RESIDUAL HEAT REMOVAL (RHR-LOOP "A") SHUTDOWN COOLING, TO PRE-EVENT LINEUP STATUS BY 1400 HOURS. THE CAUSE OF THIS EVENT IS INDETERMINENT. THERE IS NO SAFETY SIGNIFICANCE ASSOCIATED WITH THIS EVENT. NO ACTUAL PLANT CONDITIONS REQUIRING THE ENGINEERED SAFETY FEATURE ISOLATIONS AND ACTUATIONS EXISTED, AND ALL ISOLATIONS AND ACTUATIONS OCCURRED AS DESIGNED.

[195] ZION 1 DOCKET 50-295 LER 89-012 REV 01
UPDATE ON INADEQUATE AFW FLOW SETTINGS WITH INOPERABLE AFW PUMP DUE TO ADMINISTRATIVE ERRORS AND PROCEDURAL INADEQUACIES.
EVENT DATE: 072389 REPORT DATE: 062790 NSSS: WE TYPE: PWR

(NSIC 218749) AT THE TIME OF THIS EVENT, THE UNIT WAS OPERATING AT 99% POWER, AND THE REACTOR COOLANT SYSTEM WAS AT NORMAL OPERATING TEMPERATURE AND PRESSURE. AT 1840 HOURS ON 7/23/89 THE 1A AUXILIARY FEEDWATER (AFW) PUMP (BA) WAS DECLARED INOPERABLE DUE TO FAILING A TRIP TEST. IN ACCORDANCE WITH TECH SPEC 3.7.2.C THE MOTOR DRIVEN AFW PUMPS WERE REALIGNED TO PROVIDE TWO OPERABLE FLOW PATHS TO THE STEAM GENERATORS AT 1940 HOURS. THE FLOW RATES PROVIDED TO THE STEAM GENERATORS WERE NOT VERIFIED WITHIN 8 HOURS TO BE 105 GPM TO EACH STEAM GENERATOR (S/G) USING THE NEW PUMP LINEUP, AS REQUIRED BY TECH SPEC 3.7.2.D AND TECH SPEC 4.7.2.A.1.B. ALL FLOWS WERE PROPERLY RESET TO 105 GPM TO EACH S/G AT 1148 HOURS ON 7/24/89. THE CAUSE OF THE EVENT WAS AN ADMINISTRATIVE ERROR IN THAT THE STATION FAILED TO RECOGNIZE THAT THE SYSTEM HEAD CURVE FOR THE TURBINE DRIVEN SUPPLY HEADER DID NOT INTERSECT THE "B" MOTOR DRIVEN AUXILIARY FEEDWATER PUMP CURVE AT THE SAME HEAD AND FLOW AS THE TURBINE DRIVEN PUMP CURVE. A CONTRIBUTING FACTOR TO THE EVENT WAS THE REVISION OF SYSTEM OPERATING INSTRUCTION (SOI-10), AUXILIARY FEEDWATER, IN USE WHICH DID NOT ADDRESS THE RESETTING OF AFW DISCHARGE THROTTLE VALVE POSITIONS. THE SHORTFALL IN AFW FLOW THAT RESULTED FROM THE FLOW SET ERROR HAS BEEN SHOWN TO BE WITHIN THE ERRORS ASSUMED IN A NEW ANALYSIS.

[196] ZION 1 DOCKET 50-295 LER 90-012
TURBINE BUILDING AIR SAMPLER FOUND INOPERABLE.
EVENT DATE: 051390 REPORT DATE: 062290 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 2 (PWR)

(NSIC 218758) ON 5/13/90, A RADIATION TECHNICIAN NOTICIED THE TURBINE BUILDING PARTICULATE/IODINE CONTINUOUS AIR SAMPLER TURNED OFF; CONSEQUENTLY, THE CONTINUOUS RELEASE POINT SAMPLE REQUIRED BY TECH SPECS TABLE 4.12-1, C.6, FOR PARTICULATE AND IODINE WAS NOT OBTAINED BETWEEN 5/8 - 5/13. THIS WAS RECOGNIZED AS REPORTABLE ON 5/23/90. THE EVENT WAS CAUSED BY PERSONNEL ERROR, IN THAT AN UNIDENTIFIED, UNAUTHORIZED INDIVIDUAL TURNED OFF THE SAMPLER. CORRECTIVE ACTION IS TO PLACE THE SAMPLER IN A LOCKED CAGE TO PREVENT TAMPERING. THERE WAS NO SAFETY SIGNIFICANCE, AS THE TURBINE BUILDING IS NOT A RELEASE PATH UNDER NORMAL OPERATIONS AND SAMPLES TAKEN IMMEDIATELY FOLLOWING THE EVENT SHOWED NO ACTIVITY.

COMPONENT INDEX

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

COMPONENT INDEX

SYSTEM INDEX

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

SYSTEM INDEX

SYSTEM INDEX

KEYWORD INDEX

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

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

VENDOR CODE INDEX

NRC FORM 335 U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER: NUREG/CR-2000, Vol.9, No. 8 ORNL/NSIC-200

2. TITLE AND SUBTITLE

Licensee Event Report (LER) Compilation

For month of August 1990

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR
September	1990

5. AUTHOR(S)

6. DATE REPORT ISSUED

MONTH	YEAR
September	1990

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

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

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER

FIN A9135

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

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

11a. TYPE OF REPORT

Monthly Report

b. PERIOD COVERED (Inclusive dates)

August 1990

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

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

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS

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

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT

Unlimited

16. SECURITY CLASSIFICATION

(This page)
Unclassified

(This report)
Unclassified

17. NUMBER OF PAGES

18. PRICE

THIS DOCUMENT WAS PRINTED USING RECYCLED PAPER.

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

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

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

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