NUREG/CR-2000
ORNL/NSIC-200
Vol. 10, No. 12

Licensee Event Report (LER) Compilation

For month of December 1991

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. 10, No. 12

Licensee Event Report (LER) Compilation

For month of December 1991

Manuscript Completed: January 1992
Date Published: January 1992

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, its contents, or LER searches should be directed to

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

Page

[1] ARKANSAS NUCLEAR 1 DOCI 50-313 LER 91-009
SAFETY RELATED 480 VAC LOAD CENTERS NOT INSTALLED .. A SEISMICALLY QUALIFIED CONFIGURATION DURING INITIAL PLANT CONSTRUCTION DUE TO AN INSUFFICIENT DESIGN DOCUMENTATION REVIEW.
EVENT DATE: 100391 REPORT DATE: 110191 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 2 (PWR)
VENDOR: ITE IMPERIAL CORPORATION

(NSIC 223326) ON OCTOBER 3, 1991 DURING AN ELECTRICAL DISTRIBUTION SYSTEM FUNCTIONAL INSPECTION ON ANO-2, IL WAS IDENTIFIED THAT 400 VAC ENGINEERED SAFETY FEATURES LOAD CENTERS 2B5 AND 2B6 WERE NOT IN A SEISMICALLY QUALIFIED CONFIGURATION. FURTHER INVESTIGATION REVEALED THAT B5 AND B6 IN ANO-1 WERE IN THE SAME CONFIGURATION AS 2B5 AND 2B6 IN ANO-2. LIFTING TROLLEYS USED TO ASSIST IN THE INSTALLATION AND REMOVAL OF CIRCUIT BREAKERS WERE MOUNTED ON THEIR GUIDE AND SUPPORT RAILS ON THE LOAD CENTERS AND WERE NOT SECURED IN PLACE. IT WAS DETERMINED THAT THE SEISMIC QUALIFICATION FOR THE LOAD CENTERS, WHICH WAS PERFORMED BY THE VENDOR, WAS NOT DONE WITH THE TROLLEYS OR THEIR GUIDE RAILS IN PLACE. THE LIFTING TROLLEYS AND RAILS WERE REMOVED AND STORED IN A REMOTE AREA AFTER DETERMINING THAT THEY WERE NOT SEISMICALLY QUALIFIED, THEREBY RESTORING THE LOAD CENTERS TO A QUALIFIED CONFIGURATION. INTERIM CORRECTIVE ACTIONS INCLUDED THE ESTABLISHMENT OF ADMINISTRATIVE CONTROLS TO PREVENT LIFTING TROLLEYS FROM BEING LEFT ATTACHED TO THE SAFETY RELATED LOAD CENTERS AFTER THE PERFORMANCE OF MAINTENANCE. A MODIFICATION TO REINSTALL THE LIFTING TROLLEYS AND RAILS IN A SEISMICALLY QUALIFIED CONFIGURATION WILL BE COMPLETED DURING THE NEXT REFUELING OUTAGE FOR EACH UNIT.

[2] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 91-017
WEEKLY STATION BATTERY SURVEILLANCE TEST NOT PERFORMED WITHIN THE REQUIRED INTERVAL DUE TO PERSONNEL ERROR.
EVENT DATE: 101191 REPORT DATE: 110891 NSSS: CE TYPE: PWR

(NSIC 223375) AT 0810 ON OCTOBER 11, 1991, IT WAS DISCOVERED THAT THE TECHNICAL SPECIFICATION SURVEILLANCE TEST FOR THE ARKANSAS NUCLEAR ONE UNIT 2 STATION BATTERIES HAD NOT BEEN PERFORMED WITHIN THE MAXIMUM ALLOWABLE TIME INTERVAL. BOTH BATTERIES WERE DECLARED INOPERABLE AND TECHNICAL SPECIFICATION 3.0.3 WAS ENTERED. THE TESTS FOR BOTH BATTERIES WERE COMPLETED SATISFACTORILY BY 0838, THREE HOURS AND EIGHT MINUTES AFTER THE END OF THE ALLOWABLE INTERVAL. THE ROOT CAUSE OF THIS EVENT WAS ATTRIBUTED TO PERSONNEL ERRORS BY ELECTRICAL MAINTENANCE PERSONNEL THAT RESULTED IN A LACK OF OVERSIGHT OF THE SCHEDULED SURVEILLANCE ACTIVITY. A CONTRIBUTING FACTOR WAS A FAILURE OF THE SURVEILLANCE COORDINATOR TO COMMUNICATE TO APPROPRIATE PERSONNEL THE LATE COMPLETION STATUS OF THE TEST. ACTIONS HAVE BEEN TAKEN TO HIGHLIGHT SURVEILLANCE TESTS ON THE SHOP SCHEDULE, STANDARDIZE ASSIGNMENT OF THE BATTERY SURVEILLANCE TO ONE CREW, AND IMPROVE COMMUNICATION BETWEEN THE SURVEILLANCE COORDINATOR AND GROUPS PERFORMING SURVEILLANCES. A "LESSONS LEARNED" TRAINING IS BEING CONDUCTED WITH THE MAINTENANCE STAFF OF BOTH UNITS. THE SURVEILLANCE PROGRAM WAS REVIEWED AND DETERMINED TO BE ADEQUATE. THIS EVENT IS REPORTABLE AS AN OPERATION PROHIBITED BY TECHNICAL SPECIFICATIONS BECAUSE OF THE ENTRY INTO SPECIFICATION 3.0.3.

[3] ARNOLD DOCKET 50-331 LER 91-011
INADEQUATE INCORPORATION OF TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS DURING DESIGN CHANGE REVIEW CAUSES MISSED VERIFICATION OF FLOWPATH TO FIRE HOSE STATIONS.
EVENT DATE: 101791 REPORT DATE: 111491 NSSS: GE TYPE: BWR

(NSIC 223435) ON OCTOBER 17, 1991, WHILE OPERATING AT 100% REACTOR POWER, IT WAS FOUND THAT SIX VALVES IN THE FLOWPATH TO THE SAFETY RELATED HOSE STATIONS WERE NOT VERIFIED OPEN UNDER SURVEILLANCE PROCEDURES AS REQUIRED BY TECHNICAL SPECIFICATIONS. SYSTEM WALKDOWN AND DRAWING REVIEWS WERE COMPLETED TO ENSURE ALL OMITTED VALVES WERE IDENTIFIED. IN EACH CASE, THE VALVE WAS FOUND TO BE IN THE PROPER POSITION. THE MONTHLY SURVEILLANCE TEST HAS BEEN MODIFIED TO INCLUDE THE SIX OMITTED VALVES. THIS EVENT HAD NO AFFECT ON THE SAFE OPERATION OF THE PLANT.

[4] BEAVER VALLEY 1 DOCKET 50-334 LER 91-027
INSUFFICIENT DIESEL GENERATOR FUEL OIL SUPPLY.
EVENT DATE: 092691 REPORT DATE: 102891 NSSS: WE TYPE: PWR

(NSIC 223320) THE UNIT WAS OPERATING AT 100 PERCENT POWER WHEN THE ELECTRICAL DISTRIBUTION SYSTEM FUNCTIONAL EVALUATION TEAM (EDSFT) IDENTIFIED A CONCERN WITH THE FUEL OIL SUPPLY TO THE EMERGENCY DIESEL GENERATORS. THERE ARE TWO UNDERGROUND FUEL OIL STORAGE TANKS, EACH WITH A CAPACITY OF 20,000 GALLONS. THE CAPACITY OF BOTH TANKS TOGETHER IS ADEQUATE FOR MORE THAN SEVEN DAYS OF FULL LOAD OPERATION OF ONE DIESEL GENERATOR AS REQUIRED BY THE UFSAR. HOWEVER, IN THE EVENT OF A LOSS OF ONE EMERGENCY BUS, THE AFFECTED TRAIN OF TRANSFER PUMPS WOULD NOT BE AVAILABLE TO SUPPLY FUEL OIL TO THE OPPOSITE TRAIN DIESEL GENERATOR. THE TRAIN A PUMPS (EE-P-1A, 1B) TAKE SUCTION FROM THE A STORAGE TANK (EE-TK-1A) AND NORMALLY SUPPLY THE TRAIN A DIESEL GENERATOR. THE TRAIN B PUMPS (EE-P-1C, 1D) TAKE SUCTION FROM THE B STORAGE TANK (EE-TK-1B) AND NORMALLY SUPPLY THE TRAIN B DIESEL GENERATOR. THUS, IN THE EVENT OF A LOSS OF ONE EMERGENCY BUS, ONLY ONE HALF OF THE REQUIRED SUPPLY OF FUEL OIL WOULD BE AVAILABLE TO THE OPERABLE DIESEL GENERATOR (APPROXIMATELY THREE AND ONE HALF DAYS' WORTH).

[5] BEAVER VALLEY 1 DOCKET 50-334 LER 91-028
INOPERABLE OVERPRESSURE PROTECTION DUE TO INADEQUATE SEISMIC QUALIFICATION.
EVENT DATE: 100291 REPORT DATE: 110191 NSSS: WE TYPE: PWR

(NSIC 223319) ON 10/2/91, WITH THE UNIT AT 100 PERCENT POWER, THE OVERPRESSURE PROTECTION SYSTEM (OPPS) WAS DECLARED INOPERABLE DUE TO INADEQUATE SEISMIC QUALIFICATION. UPON REVIEWING A VENDOR SUPPLIED 10CFR21 REPORT, THE STATION DETERMINED THAT RELAYS ASSOCIATED WITH THE OPPS WERE INSTALLED IN AUXILIARY RELAY RACKS (ARRS) WHICH ARE NOT SEISMICALLY QUALIFIED. SINCE THE OPPS, UTILIZING THE POWER OPERATED RELIEF VALVES (PORV), IS A SEISMIC, CATEGORY I SYSTEM, IT WAS DECLARED INOPERABLE. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR50.73.A.2.I.B, AS A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS. THE OPPS HAD BEEN ASSUMED OPERABLE, DURING ALL APPLICABLE MODES, SINCE IT'S ORIGINAL INSTALLATION. THIS RESULTED IN SEVERAL INSTANCES WHEN THE OPPS FUNCTION SUPPLIED BY THE PORVS WAS POTENTIALLY INOPERABLE. THE OPPS IS REQUIRED TO BE IN SERVICE WHENEVER THE REACTOR COOLANT SYSTEM (RCS) TEMPERATURE IS AT OR BELOW 292 DEGREES FAHRENHEIT. ON 10/23/91 THE UNIT WAS SHUTDOWN TO COLD SHUTDOWN FOR UNRELATED REASONS. WHILE IN COLD SHUTDOWN, A MODIFICATION WHICH UTILIZED SPARE RELAYS IN ADJACENT, SEISMICALLY QUALIFIED, AUXILIARY SAFEGUARDS CABINETS WAS COMPLETED FOR THE SUSCEPTIBLE RELAYS. ON 10/26/91, FOLLOWING COMPLETION OF THE DESIGN CHANGE, THE OPPS WAS DECLARED OPERABLE.

[6] BEAVER VALLEY 2 DOCKET 50-412 LER 91-004
POTENTIAL AUXILIARY FEEDWATER PUMP LUBE OIL COOLER OPERATION ABOVE DESIGN PRESSURE.
EVENT DATE: 101891 REPORT DATE: 111891 NSSS: WE TYPE: PWR

(NSIC 223405) UNIT 2 WAS OPERATING AT 100 PERCENT POWER WHEN THE TURBINE DRIVEN AUXILIARY FEED WATER PUMP (TDAFP) WAS DECLARED INOPERABLE AT 1600 HOURS ON 10/18/91. AN ENGINEERING EVALUATION DETERMINED THAT THE PRESSURE DEVELOPED IN THE LUBE OIL COOLER WOULD EXCEED THE COOLER DESIGN PRESSURE IN SOME SYSTEM CONFIGURATIONS. CALCULATIONS INDICATED THAT ALL PORTIONS OF THE LUBE OIL COOLERS, EXCEPT THE END BELL FLANGE BOLTS ON THE TDAFP COOLER, WOULD ACCOMMODATE THE PRESSURES EXPERIENCED. REPLACING THE FLANGE BOLTS FOR THE COOLER WITH STUD BOLTS OF A HIGHER TENSILE STRENGTH QUALIFIED THE COOLER FOR THE HIGHER MAXIMUM CALCULATED PRESSURE. THE PUMP WAS DECLARED OPERABLE ON 10/19/91 AT 0400 HOURS. AT NO TIME WAS THE MARGIN OF SAFETY TO THE GENERAL PUBLIC REDUCED. THE POTENTIAL OVERPRESSURIZATION OF THE TDAFP LUBE OIL COOLER BOLTS WAS CALCULATED BASED ON VERY CONSERVATIVE ASSUMPTIONS. ADDITIONALLY, AT NO TIME DURING THE OPERATION OF BEAVER VALLEY UNIT 2 WERE THE AUXILIARY FEEDWATER LUBE OIL COOLERS IN THE SYSTEM CONFIGURATION WHICH COULD HAVE RESULTED IN POTENTIAL OVERPRESSURIZATION. WHEN THE PUMP WAS PURCHASED DURING PLANT CONSTRUCTION, THE PUMP VENDOR HAD NOT OBSERVED THE OPERATING PRESSURE REQUIREMENTS OF ALTERNATE LINEUPS. THEREFORE, THE COOLERS WERE NOT RATED FOR A HIGHER PRESSURE.

[7] BROWNS FERRY 1 DOCKET 50-259 LER 85-016 REV 02
UPDATE ON AUTOMATIC REACTOR SCRAM DUE TO LOSS OF FEEDWATER.
EVENT DATE: 011685 REPORT DATE: 110891 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.
TECHNOLOGY FOR ENERGY CORP.
TERRY STEAM TURBINE COMPANY
TEXAS INSTRUMENTS INC.

(NSIC 223441) UNIT 1 SCRAMMED DURING NORMAL OPERATION DUE TO LOW REACTOR WATER LEVEL WHILE AT 100 PERCENT POWER. THE HIGH PRESSURE STEAM TO THE REACTOR FEED PUMP TURBINES ISOLATED DUE TO PROBLEMS WITH THE MASTER LEVEL CONTROLLER WHICH RESULTED IN A LOSS OF FEEDWATER TO THE REACTOR VESSEL. THE VESSEL LEVEL CONTINUED DECREASING, DUE TO THE FEEDWATER FLOW LOSS, PAST THE ISOLATION SETPOINT. THE PRIMARY CONTAINMENT ISOLATION SYSTEM FUNCTIONED AS DESIGNED. THE HIGH PRESSURE COOLANT INJECTION SYSTEM INITIATED AND RETURNED THE WATER LEVEL TO THE HIGH TRIP SETPOINT IN APPROXIMATELY SIX MINUTES. THE REACTOR CORE INJECTION COOLANT SYSTEM INITIATED BUT TRIPPED IMMEDIATELY ON MECHANICAL AND ELECTRICAL OVERSPEED. THE ROOT CAUSE FOR THE UNIT SCRAM WAS BELIEVED TO BE A COLD SOLDER JOINT IN THE MASTER LEVEL CONTROLLER WHICH RESULTED IN LOSS OF MOTIVE STEAM TO THE REACTOR FEED PUMP TURBINES AND SUBSEQUENT LOSS OF FEEDWATER TO THE VESSEL. THE COLD SOLDER JOINT WAS RESOLDERED AND NO FURTHER PROBLEMS HAVE OCCURRED.

[8] BROWNS FERRY 2 DOCKET 50-260 LER 91-017
FAILED SOLDERED CONNECTION ON AIR SUPPLY LINE TO STEAM PACKING EXHAUSTER BYPASS FLOW CONTROL VALVE RESULTED IN ENGINEERED SAFETY FEATURE ACTUATION.
EVENT DATE: 091491 REPORT DATE: 101591 NSSS: GE TYPE: BWR

(NSIC 22327[illegible]) ON SEPTEMBER 14, 1991 AT 1832 HOURS, WITH UNIT 2 OPERATING AT 100 PERCENT POWER, AN AUTOMATIC REACTOR SCRAM OCCURRED ON LOW REACTOR WATER LEVEL. THE LOW WATER LEVEL WAS CAUSED BY A MOMENTARY LOSS OF REACTOR FEEDWATER. THE LOSS OF REACTOR FEEDWATER WAS A RESULT OF A FAILED SOLDERED CONNECTION ON THE AIR SUPPLY LINE TO THE STEAM PACKING EXHAUSTER BYPASS FLOW CONTROL VALVE (FCV) ALLOWING THE FCV TO CLOSE. THE LOSS OF FLOW CONDITION SUBSEQUENTLY CAUSED 'A' AND 'B' CONDENSATE BOOSTER PUMPS AND 'A' AND 'B' REACTOR FEED PUMPS TO TRIP ON LOW SUCTION PRESSURE. THIS EVENT IS REPORTABLE IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(IV) AS A CONDITION WHICH RESULTED IN MANUAL OR AUTOMATIC ACTUATION OF ENGINEERED SAFETY FEATURES INCLUDING THE REACTOR PROTECTION SYSTEM. THE ROOT CAUSE OF THIS EVENT WAS POOR WORKMANSHIP ON THE FAILED AIR SUPPLY LINE SOLDERED CONNECTION. TVA'S ANALYSIS DETERMINED THAT THE BONDED AREA OF THE FAILED SOLDER POINT WAS LESS THAN 15 PERCENT. THE SOLDERED CONNECTION WAS REPAIRED. THOSE REQUIRED SOLDERED CONNECTIONS IN AIR SUPPLY LINES WHOSE FAILURE COULD CAUSE A REACTOR SCRAM WILL BE INSPECTED BY VISUAL EXAMINATION USING LIQUID LEAK DETECTION SOLUTION.

[9] BRUNSWICK DOCKET 50-325 LER 91-016 REV 01
UPDATE ON FAILURE OF THE REACTOR CORE ISOLATION COOLING STEAM LINE ISOLATION VALVES TO SEAT.
EVENT DATE: 061991 REPORT DATE: 111591 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 223425) ON JUNE 19, 1991, WITH UNIT 1 AT 100% POWER, THE REACTOR CORE ISOLATION COOLING (RCIC) STEAM LINE WAS ISOLATED TO PERFORM SURVEILLANCE TESTS. DURING SYSTEM RESTORATION, AFTER COMPLETION OF THE SURVEILLANCE TESTING, IT WAS DISCOVERED THAT WITH BOTH THE PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) VALVES (1-E51-F007/F008) CLOSED, THE DOWNSTREAM 3" DIAMETER STEAM LINE WAS STILL PRESSURIZED TO 1000 PSIG. THIS INDICATED THAT BOTH THE 1-E51-F007 AND F008 VALVES WERE LEAKING BY THEIR SEATS, AND A PCIS LCO 12-HOUR ACTION STATEMENT WAS ENTERED. WHILE QUANTIFYING THE VALVE LEAKAGE, AN ISOLATION VALVE SEATED WHEN A DIFFERENTIAL PRESSURE (DP) WAS CREATED ACROSS THE CLOSED ISOLATION VALVES BY OPENING THE RCIC TURBINE STEAM SUPPLY VALVE (1-E51-F045). THE 1-E51-F045 WAS THEN CLOSED AND THE 1-E51-F008 OPENED TO APPLY A DP ACROSS THE 1-E51-F007. ADDITIONAL STROKING OF THE 1-E51-F007 AND F008 WITH NO INITIAL DIFFERENTIAL PRESSURE SHOWED THAT EACH VALVE DID IN FACT ISOLATE THE RCIC STEAM LINE. THE VALVES SEATED AS INDICATED BY RCIC STEAM LINE PRESSURE RESPONSE AND LATER BY THE DECREASE IN STEAM

LINE TEMPERATURE. EVALUATION DETERMINED THAT THE RCIC STEAM LINE WAS NOW ISOLATED AND THE 12-HOUR ACTION STATEMENT WAS EXITED. AN ON-LINE LOCAL LEAK RATE TEST (LLRT) WAS PERFORMED THAT ESTABLISHED THE LEAKAGE RATES ACROSS THE 1-E51-F007/F008 AS WITHIN NORMAL ACCEPTABLE LEAKAGE CRITERIA.

[10] BRUNSWICK 1 DOCKET 50-325 LER 91-025
HIGH PRESSURE COOLANT INJECTION DIAPHRAGM CONTROL VALVE DIAPHRAGM FAILED CAUSING OIL LEAK.
EVENT DATE: 100391 REPORT DATE: 110491 NSSS: GE TYPE: BWR
VENDOR: ROBERTSHAW CONTROLS COMPANY

(NSIC 223369) ON 10/3/91, AT APPROX. 0230, UNIT 1 WAS OPERATING AT 100% POWER. THE CORE SPRAY (CS) SYSTEM, LOW PRESSURE COOLANT INJECTION (LPCI) SYSTEM, AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) AND DIESEL GENERATORS 1, 2 AND 4 WERE OPERABLE AND IN STANDBY. DIESEL GENERATOR 3 WAS INOPERABLE FOR SCHEDULED MAINTENANCE. THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM WAS INOPERABLE FOR SCHEDULED MAINTENANCE SURVEILLANCE TESTING. PRIOR TO THIS TIME, ON 8/23/91, DURING WEEKLY TESTING OF THE HPCI SYSTEM, AN AUXILIRY OPERATOR FOUND THAT THE DIAPHRAGM CONTROL VALVE FOR THE LOW OIL PRESSURE TRIP LEAKED DURING STROKING. ON 8/23/91, THE TECHNICAL SUPPORT HPCI SYSTEM ENGINEER EVALUATED THE LEAK. DUE TO THE FACTORS CONSIDERED IN THE HPCI SYSTEM ENGINEER'S EVALUATION, IT WAS DETERMINED THAT THE OIL LEAK DID NOT PREVENT HPCI FROM PERFORMING AS REQUIRED AND WAS NOT AN OPERABILITY CONCERN AT THAT TIME. A WORK ORDER WAS INITIATED TO REPLACE THE FAILED DIAPHRAGM, BUT REPLACEMENT PARTS COULD NOT BE LOCATED IN THE MATERIALS WAREHOUSE ON SITE. ON 10/2/91, AT 1110, A 14 DAY LIMITING CONDITION FOR OPERATION WAS INITIATED TO REMOVE THE HPCI SYSTEM FROM SERVICE FOR SCHEDULED MAINTENANCE SURVEILLANCE TESTING. ON 10/3/91, AT APPROX. 0230, OPERATIONS PERSONNEL BEGAN PERIODIC OPERABILITY TESTING OF THE HPCI SYSTEM. WHILE THE HPCI TURBINE WAS BEING STARTED, AN AUXILIARY OPERATOR IN THE FIELD NOTED AN OIL LEAK.

[11] BRUNSWICK 2 DOCKET 50-324 LER 91-014
UNPLANNED REACTOR PROTECTION SYSTEM ACTUATION RESULTED FROM SENIOR CONTROL OPERATOR RE-ALIGNING THE SCRAM DISCHARGE VOLUME HI-HI LEVEL TRIP BYPASS SWITCH WITHOUT INFORMING THE DUTY REACTOR OPERATORS.
EVENT DATE: 092791 REPORT DATE: 102891 NSSS: GE TYPE: BWR

(NSIC 223321) ON 9/27/91 AT 0443 UNIT 2 WAS IN A REFUELING OUTAGE WITH THE REACTOR CORE PARTIALLY OFF LOADED TO THE SPENT FUEL POOL. ALL CONTROL RODS WERE FULLY INSERTED AND THE CONTROL ROD DRIVE (CRD) SYSTEM WAS SECURED EXCEPT FOR A TEMPORARY MECHANICAL JUMPER TO SUPPLY COOLING WATER WHILE CRD SYSTEM MAINTENANCE WAS BEING PERFORMED. SEVERAL HOURS EARLIER, AS THE CRD SYSTEM WAS BEING PREPARED TO RETURN IT TO SERVICE, THE REACTOR PROTECTION SYSTEM (RPS) TRIP HAD BEEN RESET. THE UNIT'S SENIOR REACTOR OPERATOR (SRO) WHILE TOURING THE REACTOR CONTROL PANELS FOUND THE SCRAM DISCHARGE VOLUME (SDV) HI-HI LEVEL TRIP BYPASS SWITCH WAS STILL IN THE BYPASS POSITION WITH BOTH SDV NOT DRAINED ANNUNCIATORS CLEARED. AS THE REACTOR OPERATORS (RO) WERE BUSY, THE SRO RETURNED THE SWITCH TO THE NORMAL POSITION HIMSELF, WITHOUT INFORMING THE ROS. THE SDV HI-HI LEVEL RPS TRIP, WHOSE ANNUNCIATOR HE HAD NOT VERIFIED, RESULTED IN THE FULL RPS TRIP. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT, AS THE RPS RESPONDED AS EXPECTED AND NO CONTROL ROD MOTION OCCURRED SINCE ALL CONTROL RODS WERE PREVIOUSLY FULLY INSERTED. THE LONGER RESET TIMES OF THE SDV ELECTRONIC LEVEL SWITCHES DOES NOT AFFECT THEIR OPERABILITY. PREVIOUS RELATED LERS ARE 1-88-031 AND 1-87-010.

[12] BRUNSWICK 2 DOCKET 50-324 LER 91-015
UNPLANNED ENGINEERED SAFETY FEATURE ACTUATION WHILE REMOVING REACTOR LOW LEVEL INSTRUMENTATION FROM SERVICE FOR LOWERING OF REACTOR VESSEL LEVEL FOR INVESSEL INSPECTIONS AND DECONTAMINATION.
EVENT DATE: 100291 REPORT DATE: 110191 NSSS: GE TYPE: BWR

(NSIC 223322) ON 10/2/91, THE UNIT 2 REACTOR WAS DEFUELED. A SPECIAL PROCEDURE (SP) TO COORDINATE THE LOWERING OF THE VESSEL LEVEL FOR INVESSEL INSPECTIONS AND CHEMICAL DECONTAMINATION WAS IN PROGRESS. PORTIONS OF THE HPCI SYSTEM, ADS, RHR/LPCI SYSTEM AND CS SYSTEM ACTUATION LOGICS WERE DISABLED TO PREVENT AUTOMATIC

INJECTION. DG #3 WAS REMOVED FROM SERVICE FOR SCHEDULED MAINTENANCE. PREREQUISITE STEPS WHICH DISABLE THE ECCS/PCIS AND RPS AUTOMATIC INITIATIONS ON LOW REACTOR WATER LEVEL AND PLACE A CLEARANCE ON THE INVOLVED COMPONENTS WERE IN PROGRESS. AT 0216, A SPURIOUS LOW LEVEL (LL) TRIP SIGNAL WAS GENERATED THAT RESULTED IN CLOSURE OF THE DIVISIONS I AND II NONINTERRUPTIBLE AIR ISOLATION VALVES TO THE DRYWELL, AUTOMATIC STARTING OF DG'S 1, 2 AND 4, AND ACTUATION OF PORTIONS OF THE HPCI, ADS, LPCI AND CS LOGICS. THIS EVENT WAS CAUSED BY LACK OF A CAUTION IN THE SP STATING THAT THE ACTIONS ARE TO BE PERFORMED IN THE ORDER LISTED IN THE SP TO PREVENT AN ACTUATION. A FACTOR CONTRIBUTING TO THIS EVENT WAS THE EXPECTATION BY PERSONNEL WRITING THE CLEARANCE THAT THE PRE-REQUISITE STEPS 5.1 AND 5.2 WOULD BE PERFORMED PRIOR TO 5.3. TYPICALLY, PRE-REQUISITES DO NOT REQUIRE ACTIONS AND CAN BE VERIFIED IN ANY ORDER; AS OPPOSED TO PROCEDURE STEPS WHICH MUST BE PERFORMED SEQUENTIALLY.

[13] BRUNSWICK 2 DOCKET 50-324 LER 91-016
SEVERE WEATHER INDUCES A VOLTAGE DROP ON AN OFF-SITE DISTRIBUTION LINE THAT RESULTS IN PRIMARY CONTAINMENT ISOLATIONS.
EVENT DATE: 100591 REPORT DATE: 110491 NSSS: GE TYPE: BWR

(NSIC 223368) ON OCTOBER 5, 1991, AT 0136, UNIT 2 WAS IN A REFUELING OUTAGE WITH ALL FUEL REMOVED FROM THE REACTOR VESSEL. THE EMERGENCY CORE COOLING SYSTEMS (ECCS) INITIATIONS HAD BEEN DEFEATED TO SUPPORT HAVING THE REACTOR VESSEL DRAINED DOWN FOR IN-VESSEL MAINTENANCE. THE #3 EMERGENCY DIESEL GENERATOR (EDG) WAS OUT OF SERVICE FOR SCHEDULED MAINTENANCE. TO SATISFY THE TECHNICAL SPECIFICATION SURVEILLANCES REQUIRED DUE TO #3 EDG BEING INOPERABLE, #4 EDG OPERABILITY VERIFICATION TESTING WAS IN PROGRESS. #4 EDG WAS RUNNING BUT HAD NOT BEEN ELECTRICALLY TIED TO THE 4160 VOLT EMERGENCY BUS E4. PRIOR TO THE AUXILIARY OPERATORS COMPLETING THE ELECTRICAL TIE, AN ELECTRICAL STORM CAUSED A 230 KV OFF-SITE DISTRIBUTION LINE TO TRIP ON A PHASE TO GROUND FAULT, AND THE MASTER/SLAVE BREAKER FROM THE BALANCE OF PLANT (BOP) BUS 2C TO THE EMERGENCY BUS E4 OPENED. THIS RESULTED IN A MOMENTARY LOSS OF VOLTAGE ON EMERGENCY BUS E4 DURING THE AUTOMATIC TRANSFER TO THE #4 EDG. WHILE THE AUTOMATIC TRANSFER TO THE #4 EDG LASTED ONLY APPROX. 4 SECONDS, AS DESIGNED, THE REACTOR PROTECTION BUS B (RPS BUS B) WAS DEENERGIZED. THE LOSS OF RPS BUS B RESULTED IN THE CLOSURE OF SOME PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) DIVISION II VALVES AND, AN ISOLATION SIGNAL FOR SECONDARY CONTAINMENT WITH A STANDBY GAS TREATMENT SYSTEM INITIATION SIGNAL.

[14] CALLAWAY 1 DOCKET 50-483 LER 91-005
MISSED CONTAINMENT AIR LOCK DOOR INTERLOCK SURVEILLANCES DUE TO HUMAN PERFORMANCES.
EVENT DATE: 101191 REPORT DATE: 111291 NSSS: WE TYPE: PWR

(NSIC 223452) ON 10/11/91 WHILE IN MODE 1 AT 100%, A UTILITY QUALITY ASSURANCE ENGINEER DISCOVERED THAT THE SIX MONTH TECHNICAL SPECIFICATION (T/S) NORMAL AND EMERGENCY PERSONNEL CONTAINMENT AIR LOCK DOOR INTERLOCK TEST HAD NOT BEEN PERFORMED SINCE 11/6/90. THE AIR LOCKS WERE DECLARED INOPERABLE AT 1515 CDT AND T/S 3.0.3 WAS ENTERED. THE SURVEILLANCES WERE COMPLETED SATISFACTORILY AT 1547 CDT ON 10/11/91. THE ROOT CAUSE OF THIS EVENT IS HUMAN PERFORMANCE. THE SURVEILLANCE TASK SHEET (STS) GENERATION WAS NOT PROPERLY COORDINATED BY THE ASSIGNED UTILITY PERSONNEL DURING A SURVEILLANCE TEST PROCEDURE REVISION. THE FOLLOWING STEPS HAVE BEEN IMPLEMENTED TO PREVENT RECURRENCE. A STEP WAS ADDED TO THE PLANT'S PROCEDURE REQUEST FORM THAT REQUIRES A REVIEW OF ALL STS ASSOCIATED WITH THE SURVEILLANCE PROCEDURE BEFORE THE PROCEDURE CAN BE ISSUED. THE PLANT'S COMPUTERIZED REFERENCE TRACKING SYSTEM AND COMMITMENT TRACKING SYSTEM HAVE BEEN ENHANCED TO AID PROCEDURE WRITERS IN IDENTIFYING AN STS THAT MAY NEED REVISION. PROCEDURES WILL BE CHECKED TO ENSURE THE T/S REFERENCED INCLUDES A CURRENT STS TO PROPERLY IMPLEMENT THE SURVEILLANCE. THE OPERABILITY OF THESE HATCHES WAS DOCUMENTED BY OTHER SURVEILLANCE TESTS DURING REFUELS BETWEEN 2/88 ANT 10/11/91. A REVIEW OF SURVEILLANCE TESTS FOR THESE CONTAINMENT ACCESS HATCHES INDICATED NO SIGNIFICANT INCREASE IN LEAKAGE DURING THE PERIODS OF MISSED SURVEILLANCES.

[15] CALVERT CLIFFS 1 DOCKET 50-317 LER 91-003
REACTOR PROTECTION SYSTEM ACTUATION AND PLANT TRIP DUE TO LOW STEAM GENERATOR WATER LEVELS CAUSED BY LOOSE ELECTRICAL FUSE.
EVENT DATE: 100191 REPORT DATE: 103191 NSSS: CE TYPE: PWR
VENDOR: MARATHON ELEC MFG

(NSIC 223324) ON OCTOBER 1, 1991 CALVERT CLIFFS UNIT 1 TRIPPED FROM 93 PERCENT POWER DUE TO LOW STEAM GENERATOR (SG) WATER LEVELS. THE LOW SG WATER LEVELS RESULTED FROM A LOOSE FUSE IN THE POWER SUPPLY TO 12 FEEDWATER REGULATING VALVE (FRV) DIFFERENTIAL PRESSURE (DP) CONTROLLER AND 12 STEAM GENERATOR FEED PUMP (SGFP) TURBINE SPEED CONTROLLER. A BENT FUSE CLIP CAUSED AN ELECTRICAL DISCONTINUITY. THE FUSE HOLDER HAS BEEN REPLACED AND PERSONNEL WHOSE DUTIES INVOLVE FUSE MANIPULATION WILL BE TRAINED ON PROPER FUSE INSTALLATION AND REMOVAL.

[16] CATAWBA 1 DOCKET 50-413 LER 91-009 REV 01
UPDATE ON TECHNICAL SPECIFICATION VIOLATIONS RESULTING FROM IMPROPER OVERTEMPERATURE DELTA TEMPERATURE CIRCUIT SCALING.
EVENT DATE: 071791 REPORT DATE: 101591 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 223241) ON 7/17/91, WITH UNITS 1 AND 2 IN MODE 1, POWER OPERATION, AT 100% POWER, MAINTENANCE ENGINEERING SERVICES (MES) PERSONNEL LEARNED THAT THE OVERTEMPERATURE DELTA-TEMPERATURE (OTDT) CIRCUIT WOULD NOT WORK PROPERLY OVER THE ENTIRE RANGE. THE GAIN ASSOCIATED WITH ONE OF THE OTDT CALCULATION CONSTANTS, K2, WAS APPLIED TO ONLY ONE CIRCUIT CARD IN EACH CHANNEL WHEN THE OTDT CHANNELS WERE CALIBRATED AT BOTH MCGUIRE NUCLEAR STATION AND CATAWBA. THIS WOULD RESULT IN OVERRANGING THE CARD, UNDER CERTAIN CONDITIONS, FOR THE CURRENT VALUES OF K2. THIS WAS DISCOVERED BY MES PERSONNEL AT MCGUIRE DURING THE WEEK OF 7/8/91, DURING A SCALING CALCULATION FOR A NEW K2 VALUE. IT WAS NOT DETERMINED WHETHER OR NOT THE OTDT CHANNELS WERE INOPERZBLE PER TECH SPECS ON 7/17. IT HAS BEEN CONCLUDED THAT CHANNELS WERE INOPERABLE IN THE PAST AND THAT THE PLANT WOULD HAVE BEEN OUTSIDE ITS DESIGN BASIS UNDER SOME POSTULATED CONDITIONS. THIS INCIDENT IS ATTRIBUTED TO DEFECTIVE VENDOR DOCUMENTATION AND INAPPROPRIATE ACTION (DUE TO INADEQUATE PROCEDURE REVIEWS). THE GAIN VALUE ASSOCIATED WITH THE K2 CONSTANT WAS APPLIED PER VENDOR DOCUMENTATION, TO THE LEAD/LAG CIRCUIT IN THE OTDT CALIBRATION PROCEDURES, RESULTING IN POTENTIAL SATURATION OF THE CIRCUITRY. CORRECTIVE ACTIONS INCLUDED PROCEDURE AND VENDOR DOCUMENTATION REVISIONS, OTDT CIRCUIT MODIFICATIONS, AND INTERIM CONTROLS.

[17] CATAWBA 1 DOCKET 50-413 LER 91-025
PAST INOPERABILITY OF THE CA SUMP SYSTEM DUE TO A DESIGN DEFICIENCY.
EVENT DATE: 073091 REPORT DATE: 111491 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 223451) ON JULY 30, 1991, UNIT 1 WAS OPERATING IN MODE 1, POWER OPERATION, AT 100% POWER. THE UNIT 1 HIGH-HIGH AUXILIARY FEEDWATER TURBINE DRIVEN PUMP (CATDP) SUMP LEVEL ANNUNCIATOR WAS CONTINUOUSLY ALARMING IN THE CONTROL ROOM. UPON INVESTIGATION, IT WAS DISCOVERED THAT BOTH THE 1A & 1B CATDP SUMP PUMPS WERE OPERATING (AS EXPECTED ON RECEIPT OF A HI-HI ALARM) ALTHOUGH THE SUMP LEVEL WAS NOT DECREASING. IT WAS ALSO OBSERVED THAT THE LIQUID RADWASTE SYSTEM (WL) FLOOR DRAIN 1D1 & 1D2 SUMP PUMPS WERE RUNNING AT THIS TIME. ALL FOUR SUMP PUMPS DISCHARGE TO THE TURBINE BUILDING SUMP THROUGH RADIATION MONITOR 1EMF52 BY WAY OF A COMMON HEADER. IT WAS CONCLUDED THAT, WITH VALVE 1WL846 THROTTLED, THE WL FLOOR DRAIN D SUMP PUMPS DISCHARGE HEAD WAS OVERCOMING THE DISCHARGE HEAD OF THE CATDP SUMP PUMPS THEREBY CREATING A SITUATION IN WHICH THERE WAS NO FLOW FROM THE CATDP SUMPS. A PAST OPERABILITY EVALUATION CONCLUDED THAT THE AUXILIARY FEEDWATER (CA) SYSTEM WOULD NOT HAVE BEEN ABLE TO MEET ITS SAFETY RELATED DESIGN BASIS WITH 1WL846 THROTTLED. THIS INCIDENT IS ATTRIBUTED TO A DESIGN DEFICIENCY DUE TO THE UNANTICIPATED INTERACTION BETWEEN SYSTEMS. CORRECTIVE ACTION INCLUDED OPENING THE THROTTLED VALVE IN THE COMMON DISCHARGE HEADER TO RESTORE THE CA SUMP SYSTEM TO OPERABILITY AND INITIATION OF ALTERNATE SAMPLING OF THE SUMP DISCHARGE.

[18] CATAWBA 1 DOCKET 50-413 LER 91-018
ESSENTIAL BUS BLACKOUT DUE TO INAPPROPRIATE ACTION.
EVENT DATE: 090691 REPORT DATE: 100391 NSSS: WE TYPE: PWR

(NSIC 223193) ON SEPTEMBER 6, 1991, AT 1903 HOURS, UNIT 1 WAS OPERATING IN MODE 1, POWER OPERATIONS. A NUCLEAR OPERATIONS TECHNICIAN (NOT B) WAS SHUTTING DOWN DIESEL GENERATOR (D/G) 1B PER PT/1/A/4350/02B, DIESEL GENERATOR OPERATION, FOLLOWING CORRECTIVE MAINTENANCE. NOT B ASSUMED THE SHUTDOWN FOLLOWING TURNOVER FROM THE DAY SHIFT. AFTER REDUCING THE LOAD ON D/G 1B, NOT B INADVERTENTLY OPENED THE NORMAL B TRAIN ESSENTIAL BUS (1ETB) FEEDER BREAKER, 1ETB-3, INSTEAD OF THE D/G BREAKER, 1ETB- 18 . D/G 1B IMMEDIATELY BEGAN SUPPLYING THE LOADS ON 1ETB . NOT B CONTINUED THE SHUT DOWN AND PRESSED THE STOP PUSHBUTTON. THE BLACKOUT CONDITION INITIATED THE D/G LOAD SEQUENCER, RESTARTED D/G 1B, AND LOADED THE BLACKOUT LOADS. NOT B REALIZED THAT THE D/G HAD RESTARTED AND PRESSED THE STOP BUTTON AGAIN AND THE D/G RESTARTED . NOT B FINALLY STOPPED D/G 1B BY USING THE RUN/STOP PLUNGER. 1ETB-3 WAS CLOSED AND THE AFFECTED PLANT SYSTEMS WERE RESET. UNIT 1 CONTINUED TO OPERATE AT 100% POWER THROUGHOUT THIS EVENT. THE D/G RUN WAS REPEATED SUCCESSFULLY. THIS EVENT WAS ATTRIBUTED TO AN INAPPROPRIATE ACTION. NOT B FAILED TO SELF-VERIFY ACTIONS TAKEN TO OPEN 1ETB-18 . CONTRIBUTING CAUSES INCLUDE HARDWARE AND PROCEDURE DEFICIENCIES. CORRECTIVE ACTIONS INCLUDE AN EMPHASIS ON THE USE OF SELF VERIFICATION ALONG WITH PROCEDURE AND EQUIPMENT ENHANCEMENTS.

[19] CATAWBA 1 DOCKET 50-413 LER 91-021
TURBINE/REACTOR TRIP DUE TO INSTALLATION DEFICIENCY.
EVENT DATE: 100291 REPORT DATE: 103091 NSSS: WE TYPE: PWR

(NSIC 223378) ON OCTOBER 2, 1991, AT 1200 HOURS, WITH UNIT 1 AT 100% POWER, A TURBINE TRIP RESULTING IN A REACTOR TRIP OCCURRED. A VENDOR, CLEANING THE AREA ON THE 619 ELEVATION OF THE TURBINE BUILDING WITH HIGH PRESSURE WATER, INADVERTENTLY SPRAYED WATER INTO ELECTRICAL BOX 1TBOX0047 LOCATED ON THE 594 ELEVATION. THE WATER INFILTRATION CAUSED ERRONEOUS 2 OF 3 HIGH WATER LEVEL SIGNALS IN MOISTURE SEPARATOR REHEATER "D" TO INITIATE A TURBINE TRIP, AND, SUBSEQUENTLY, AT 12:00:04 THE REACTOR TRIPPED. AN EVALUATION OF PLANT DATA HAS SHOWN THAT ALL SYSTEMS PERFORMED AS DESIGNED FOLLOWING THE TRIP. PROCEDURES EP/1/A/5000/001A, REACTOR TRIP RESPONSE, AND AP/1/A/5500/002, TURBINE GENERATOR TRIP, WERE ENTERED AT 1205 HOURS, WITH ALL SYSTEMS RESPONDING NORMALLY. THE EP AND AP PROCEDURES WERE COMPLETED APPROXIMATELY 1235 HOURS, AND OP/1/A/6100/005, UNIT FAST RECOVERY, WAS INITIATED. THIS EVENT HAS BEEN ATTRIBUTED TO AN INSTALLATION DEFICIENCY IN FAILING TO PROPERLY SEAL THE CONDUIT CONNECTORS WHICH PENETRATE THE NEMA 4 BOX. CORRECTIVE ACTIONS INCLUDE AN INSPECTION OF NON-SAFETY RELATED NEMA 4 BOXES LOCATED IN THE UNIT 1 & 2 TURBINE BUILDINGS AND SEALING OF THE CONNECTORS AS REQUIRED. THE VENDOR WILL DEVELOP A PROCEDURE FOR PROPER USE OF THE PRESSURE WASHER.

[20] CATAWBA 1 DOCKET 50-413 LER 91-022
TECHNICAL SPECIFICATION VIOLATION AS A RESULT OF A MISSED GRAB SAMPLE ON RADIATION MONITOR EMF-36 DUE TO INAPPROPRIATE ACTION AND MANAGEMENT DEFICIENCY.
EVENT DATE: 100791 REPORT DATE: 103191 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 223379) ON OCTOBER 7, 1991, AT 0005 HOURS, WITH UNIT 1 IN MODE 1, POWER OPERATION, AT 100% POWER, A TECHNICAL SPECIFICATION (T/S) VIOLATION OCCURRED DUE TO A MISSED GRAB SAMPLE FOR THE INOPERABLE UNIT VENT RADIATION MONITOR 1EMF36. THE RADIATION PROTECTION (RP) SPECIALIST A, WHO FAILED TO OBTAIN THE INOPERABLE 1EMF36 GRAB SAMPLE AT THE APPROPRIATE TIME, INFORMED HIS RELIEF, SPECIALIST C, DURING TURNOVER THAT HE OBTAINED THE INOPERABLE 1EMF36 GRAB SAMPLE WITH THE UNIT VENT WEEKLY GRAB SAMPLE AT 0103 HOURS. SPECIALIST C DISCUSSED THIS INCIDENT WITH RP MANAGEMENT, AND IT WAS CONCLUDED THAT A TECHNICAL SPECIFICATION VIOLATION HAD OCCURRED BECAUSE THE INOPERABLE 1EMF36 GRAB SAMPLE WAS NOT OBTAINED AT 0005 HOURS TO COMPLY WITH THE SAMPLING REQUIREMENT OF ONCE PER 12 HOURS WHEN 1EMF36 IS INOPERABLE. THIS INCIDENT IS ATTRIBUTED TO INAPPROPRIATE ACTION DUE TO SPECIALIST A IMPROPERLY FOLLOWING THE CORRECT PROCEDURE AND A MANAGEMENT DEFICIENCY BECAUSE LESS THAN ADEQUATE TRAINING HAD BEEN GIVEN. CORRECTIVE ACTIONS INCLUDED

COMMUNICATING THIS INCIDENT TO ALL APPROPRIATE RP PERSONNEL. AN ENHANCEMENT TO THE RP PROCEDURE HP/0/B/1009/11, EMF LOSS, IS PLANNED TO FURTHER CLARIFY SAMPLING REQUIREMENTS, AND TRAINING WILL BE PROVIDED ON THIS PROCEDURE.

[21] CATAWBA 1 DOCKET 50-413 LER 91-024
TECHNICAL SPECIFICATION 3.0.3 ENTRY AS A RESULT OF BOTH TRAINS OF CONTROL ROOM AREA VENTILATION BEING INOPERABLE DUE TO EQUIPMENT FAILURE.
EVENT DATE: 100991 REPORT DATE: 110791 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 223406) ON OCTOBER 9, 1991, AT APPROXIMATELY 0912 HOURS, WITH UNITS 1 AND 2 IN MODE 1, POWER OPERATION, AT 100% POWER, TECHNICAL SPECIFICATION (T/S) 3.0.3 WAS ENTERED DUE TO BOTH TRAINS OF THE CONTROL ROOM AREA VENTILATION (VC) AND CHILLED WATER (YC) SYSTEMS BEING DECLARED INOPERABLE. TRAIN 'A' WAS DECLARED INOPERABLE AFTER ITS CHILLER TRIPPED DUE TO SPURIOUS ACTUATION OF THE LOW REFRIGERANT TEMPERATURE CUTOUT SWITCH. AN INVESTIGATION REVEALED THAT THE ROOT CAUSE FOR THE TRAIN 'A' YC CHILLER TRIPS WAS DUE TO CORRODED SWITCH CONTACTS. THE TRAIN 'B' YC CHILLER INOPERABILITY WAS DUE TO THE CHILLER FAILING TO START. A CHILLER MOTOR OVERLOAD OCCURRED AS A RESULT OF A PROBLEM WITH THE GUIDE VANE ACTUATOR. YC TRAIN 'A' WAS DECLARED OPERABLE AND RETURNED TO SERVICE AFTER REPLACING THE LOW REFRIGERANT TEMPERATURE CUTOUT SWITCH. TRAIN 'B' YC CHILLER WAS EVALUATED AND DECLARED OPERABLE AFTER CORRECTIVE ACTION WAS PERFORMED ON THE GUIDE VANE ACTUATOR. T/S 3.0.3 WAS EXITED AT 1515 HOURS. THIS INCIDENT IS ATTRIBUTED TO EQUIPMENT FAILURES. SUBSEQUENT CORRECTIVE ACTIONS RESTORED BOTH CHILLERS TO OPERABLE STATUS. PLANNED CORRECTIVE ACTIONS INCLUDE REPLACING THE GUIDE VANE ACTUATOR ON 'B' CHILLER. ADDITIONAL CORRECTIVE ACTIONS MAY BE IMPLEMENTED UPON COMPLETION OF AN INVESTIGATION OF BOTH INCIDENTS.

[22] CATAWBA 1 DOCKET 50-413 LER 91-026
SOLID STATE PROTECTION SYSTEM TECHNICAL SPECIFICATION VIOLATION DUE TO A POSSIBLE INAPPROPRIATE ACTION.
EVENT DATE: 101791 REPORT DATE: 111891 NSSS: WE TYPE: PWR

(NSIC 223407) ON OCTOBER 17, 1991, AT 1330 HOURS, UNIT 1 WAS OPERATING IN MODE 1, POWER OPERATION. DURING PERFORMANCE OF A SOLID STATE PROTECTION SYSTEM (SSPS) BIMONTHLY SURVEILLANCE, A JUMPER WAS DISCOVERED ACROSS TERMINALS INSIDE THE UNIT 1 TRAIN B SSPS CABINET BY INSTRUMENT AND ELECTRICAL (IAE) SPECIALISTS. SSPS TRAIN B WAS NOT REMOVED FROM SERVICE. IT WAS DETERMINED THE JUMPER WOULD PREVENT THE TRAIN B REACTOR TRIP BREAKERS (RTBS) FROM OPENING ON COINCIDENT TRAIN A AND B SSPS GENERAL WARNINGS (GW) SINCE THE TRAIN A GW INPUT TO TRAIN B WAS BLOCKED. HOWEVER, THE TRAIN A RTBS WOULD HAVE TRIPPED UPON RECEIPT OF A COINCIDENT SSPS GW. ALL GENERAL WARNING ALARMS AND ANNUNCIATORS WERE FUNCTIONAL AS WERE ALL OTHER REACTOR TRIP FUNCTIONS. IT COULD NOT BE DETERMINED BY THE INVESTIGATION, BY REVIEW OF THE SURVEILLANCE TESTS, OR ANY RELATED SSPS WORK HISTORY, AS TO WHEN THIS JUMPER WAS PLACED OR HOW IT WAS LEFT IN THE CABINET. THIS INCIDENT IS ATTRIBUTED TO A POSSIBLE INAPPROPRIATE ACTION WHICH MAY HAVE OCCURRED DURING ACTIVITIES WHICH REQUIRE BLOCKING SSPS GW SIGNALS. THE JUMPER WAS REMOVED BY 1430 HOURS. IAE INSPECTED UNIT 1 AND 2 SSPS CABINETS TO ENSURE THAT SIMILAR JUMPERS WERE NOT PRESENT. PLANNED CORRECTIVE ACTIONS INCLUDE REVISIONS TO THE IAE PROCEDURES, CABINET INSPECTIONS FOLLOWING THE UNIT 2 OUTAGE, AND AN EVALUATION OF THE CURRENT CONTROLS OVER THE USE OF JUMPERS.

[23] CLINTON 1 DOCKET 50-461 LER 91-005
INOPERABLE LEAK DETECTION FISSION PRODUCT PARTICULATE SAMPLE PANEL DUE TO AN OUT-OF-ADJUSTMENT TENSIONER PREVENTING FILTER PAPER ADVANCEMENT.
EVENT DATE: 101591 REPORT DATE: 112091 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223410) ON OCTOBER 15, 1991, WITH THE PLANT IN POWER OPERATION, A TECHNICIAN PERFORMING SCHEDULED PREVENTIVE MAINTENANCE DISCOVERED THE MOVING FILTER PAPER IN THE LEAK DETECTION SYSTEM FISSION PRODUCT PARTICULATE SAMPLE PANEL 1E31-P002 HAD STOPPED ADVANCING. THE FILTER PAPER HAD NOT ADVANCED SINCE THE LAST PREVENTIVE MAINTENANCE TASK HAD BEEN COMPLETED ON OCTOBER 3, 1991. THE

OPERATIONS SHIFT SUPERVISOR DETERMINED THAT THE FAILURE OF THE FILTER PAPER TO ADVANCE CONSTITUTED AN INOPERABLE DRYWELL ATMOSPHERE PARTICULATE RADIOACTIVITY MONITORING SYSTEM AND DIRECTED THE DRYWELL ATMOSPHERE TO BE MANUALLY SAMPLED AND ANALYZED AT LEAST ONCE PER 24 HOURS AS REQUIRED BY THE TECHNICAL SPECIFICATIONS. OPERATING THE PLANT WITH AN INOPERABLE DRYWELL ATMOSPHERE PARTICULATE RADIOACTIVITY MONITORING SYSTEM WITHOUT ENTERING AND COMPLYING WITH THE REQUIRED ACTION STATEMENT IS A CONDITION WHICH IS PROHIBITED BY TECHNICAL SPECIFICATION. TROUBLESHOOTING OF 1E31-P002 IDENTIFIED THAT THE CAPSTAN TENSIONER WAS OUT OF ADJUSTMENT. THIS CAUSED THE FILTER PAPER TO STOP ADVANCING. CORRECTIVE ACTIONS INCLUDE OBTAINING AND ANALYZING DRYWELL ATMOSPHERE GRAB SAMPLES AT LEAST ONCE EVERY 24 HOURS UNTIL THE SAMPLE PANEL IS MODIFIED TO ENABLE EXTERNAL VERIFICATION OF FILTER PAPER MOVEMENT OR UNTIL RELIABILITY OF THE SAMPLE PANEL HAS BEEN DEMONSTRATED.

[24]   COMANCHE 1   DOCKET 50-445   LER 91-023
REACTOR TRIP RESULTING FROM ERRATIC OPERATION OF THE MAIN TURBINE ELECTROHYDRAULIC CONTROLLER.
EVENT DATE: 100391   REPORT DATE: 110491   NSSS: WE   TYPE: PWR

(NSIC 223305) ON OCTOBER 3, 1991, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 WAS IN MODE 1, POWER OPERATION, WITH THE REACTOR AT APPROXIMATELY 30 PERCENT OF RATED THERMAL POWER. IN ORDER TO COLLECT DATA FOR EVALUATION OF A PROBLEM WITH THE MAIN TURBINE ELECTROHYDRAULIC CONTROL (EHC) SYSTEM, THE IDLE HYDRAULIC FLUID PUMP WAS STARTED AND ONE OF THE TWO OPERATING PUMPS WAS SECURED. FOLLOWING THE PUMP SWITCHOVER, ERRATIC OPERATION OF THE MAIN TURBINE STEAM CONTROL VALVES CREATED A PRESSURE PULSE WHICH PROPOGATED UP THE MAIN STEAM LINE TO THE STEAM GENERATOR, CAUSING A SPIKE OF SUFFICIENT MAGNITUDE ON THE STEAM GENERATOR NARROW RANGE LEVEL TRANSMITTER TO EXCEED THE HI-HI LEVEL SETPOINT. THE MAIN FEEDWATER PUMPS TRIPPED AUTOMATICALLY, AND THE REACTOR WAS MANUALLY TRIPPED DUE TO DECREASING STEAM GENERATOR LEVELS. CAUSES INCLUDE ERRATIC OPERATION OF THE EHC SYSTEM AND SENSITIVITY OF THE STEAM GENERATOR NARROW RANGE LEVEL INSTRUMENTATION. CORRECTIVE ACTIONS FOR EHC SYSTEM PERFORMANCE ANOMALIES WILL BE DETERMINED BASED ON THE RESULTS OF EVALUATION OF TEST DATA.

[25]   COMANCHE 1   DOCKET 50-445   LER 91-024
INOPERABLE MAIN STEAM SAFETY VALVES DUE TO LESS THAN ADEQUATE LIFT SETPOINT VERIFICATION PROCEDURE.
EVENT DATE: 100491   REPORT DATE: 110491   NSSS: WE   TYPE: PWR

(NSIC 223304) ON MARCH 20, 1990, THE MAIN STEAM SAFETY VALVES (MSSV) WERE TESTED AND ADJUSTED IN ACCORDANCE WITH THE APPROVED PROCEDURE WHICH CALCULATES THE LIFT SETPOINT FROM MEASURED PARAMETERS. THESE PARAMETERS ARE DOCUMENTED WHEN THE MSSV IS HEARD TO LIFT OFF OF ITS SEAT. DURING THIS TEST THE ATMOSPHERIC RELIEF VALVES (ARV) WERE OPEN, RESULTING IN EXTREMELY HIGH AMBIENT NOISE. AS A RESULT, IT WAS DIFFICULT TO DETERMINE THE EXACT LIFT POINT, AND THE MEASURED PARAMETERS WERE IN ERROR (TOO HIGH). BASED ON THESE MEASURED PARAMETERS THE CALCULATED LIFT SETPOINTS APPARENTLY SATISFIED THE ACCEPTANCE CRITERIA. ON OCTOBER 4, 1991, THE MSSVS WERE TESTED IN ACCORDANCE WITH THE APPROVED PROCEDURE. THE AMBIENT NOISE LEVEL AT THE TIME OF THE TEST WAS MINIMAL; THE ARVS WERE CLOSED. THIRTEEN OF THE FOURTEEN MSSVS TESTED WERE FOUND TO HAVE LIFT SETPOINTS LESS THAN THE SETPOINT TOLERANCE ALLOWED BY TECHNICAL SPECIFICATIONS. THE SETPOINTS WERE SUBSEQUENTLY RESET. THE ROOT CAUSE WAS DETERMINED TO BE A LESS THAN ADEQUATE PROCEDURE. CORRECTIVE ACTIONS INCLUDED A REVISION TO THE PROCEDURE.

[26]   CONNECTICUT YANKEE   DOCKET 50-213   LER 91-018
SPENT FUEL POOL COOLING SYSTEM ISOLATED FOR HEAT EXCHANGER CLEANING.
EVENT DATE: 092091   REPORT DATE: 101891   NSSS: WE   TYPE: PWR

(NSIC 223287) ON SEPTEMBER 20, 1991, AT APPROXIMATELY 0900 HOURS WITH THE PLANT IN MODE 1 AT 100 PERCENT POWER, OPERATIONS DEPARTMENT PERSONNEL ISOLATED SERVICE WATER TO THE SPENT FUEL POOL (SFP) HEAT EXCHANGERS TO SUPPORT PREPLANNED MAINTENANCE ON THE SERVICE WATER SYSTEM. THE ROOT CAUSE OF THIS EVENT WAS THE NEED TO REMOVE THE SPENT FUEL COOLING SYSTEM FROM SERVICE IN ORDER TO SWAP THE

TEMPORARY HOSES SUPPLYING THE 'A' SFP HEAT EXCHANGER TO THE 'B' SFP HEAT EXCHANGER WHICH WILL ALLOW CLEANING OF THE 'A' HEAT EXCHANGER. CORRECTIVE ACTION CONSISTED OF RETURNING THE 'B' SPENT FUEL POOL HEAT EXCHANGER TO SERVICE AT 1500 HOURS FOLLOWING THE RE-IN STALLATION OF THE TEMPORARY HOSES. THE CLEANING IS EXPECTED TO LAST APPROXIMATELY 2 WEEKS. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(V)(B) AS A CONDITION THAT ALONE COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF A SYSTEM NEEDED TO REMOVE RESIDUAL HEAT.

[27]   CONNECTICUT YANKEE   DOCKET 50-213   LER 91-019
FIRE PROTECTION SPRINKLERS DECLARED INOPERABLE DUE TO ERECTION OF STAGING.
EVENT DATE: 100491   REPORT DATE: 110191   NSSS: WE   TYPE: PWR

(NSIC 223344) AT 0230 HOURS ON OCTOBER 4, 1991, WITH THE PLANT IN MODE 1 AT 100 PERCENT POWER A PLANT OPERATOR REPORTED THAT STAGING ERECTED UNDER THE HIGH PRESSURE TURBINE WAS PARTIALLY OBSTRUCTING THE FLOW PATTERN OF 3 FIRE PROTECTION PIPING SPRINKLER HEADS. THE AREA INVOLVED HAS A TOTAL OF 6 SPRINKLER HEADS FOR FIRE PROTECTION. THE STAGING WAS INSTALLED AT APPROXIMATELY 1500 HOURS ON OCTOBER 3, 1991. OPERATORS DETERMINED THAT THE SPRINKLER HEADS WERE UNABLE TO PERFORM THEIR DESIGN FUNCTION AND DECLARED THEM INOPERABLE. THIS CONDITION IS A VIOLATION OF TECHNICAL SPECIFICATION 3.7.6.2A.4. THE REQUIRED ACTION WAS TAKEN BY POSTING AN HOURLY FIRE WATCH PATROL. THE ROOT CAUSE OF THIS EVENT WAS THE UNFAMILIARITY OF THE REQUIREMENT FOR A FIRE SPRINKLE HEAD TO BE CAPABLE OF ACHIEVING 100 PERCENT OF ITS FLOW PATTERN. CORRECTIVE ACTIONS TAKEN INCLUDED THE REMOVAL OF STAGING THAT BLOCKED THE FLOW PATTERN OF THE SPRINKLER HEADS. IN ADDITION, A MEETING WITH JOB SUPERVISORS AND WORKERS WAS HELD IN WHICH THE LESSONS LEARNED FROM THIS EVENT WERE DISCUSSED. THIS EVENT IS REPORTABLE PER 10FR5C.73(A)(2)(I)(B) SINCE IT RESULTED IN A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[28]   CONNECTICUT YANKEE   DOCKET 50-213   LER 91-020
INAPPROPRIATE REMOVAL OF A REACTOR PROTECTION SYSTEM TRIP SIGNAL.
EVENT DATE: 101891   REPORT DATE: 111591   NSSS: WE   TYPE: PWR

(NSIC 223415) ON OCTOBER 17, 1991, AT 1525 HOURS, WITH THE PLANT IN MODE 1 (POWER OPERATION) AT 35% POWER, PLANT OPERATORS DECLARED THE CHANNEL 4 STEAM FLOW-FEED FLOW MISMATCH TRIP TO BE OUT OF SERVICE. IN ACCORDANCE WITH TECHNICAL SPECIFICATION TABLE 3.3-1, ACTION STATEMENT 5, A TRIP SIGNAL WAS INSERTED FOR THE INOPERABLE CHANNEL WITHIN ONE HOUR AT 1601 HOURS. AT 2034 HOURS, WITH THE PLANT IN MODE 2 (STARTUP) AT LESS THAN 1% POWER, THE TRIP WAS REMOVED PRIOR TO RETURNING THE STEAM FLOW-FEED FLOW SIGNAL TO OPERABLE STATUS. ON OCTOBER 18, 1991, AT 1006 HOURS, IT WAS RECOGNIZED THAT IT WAS INAPPROPRIATE TO HAVE REMOVED THE TRIP AND WAS REINSERTED. THE ROOT CAUSE OF THE EVENT IS FAILURE OF THE CONTROL ROOM OPERATORS TO CONSULT THE TECHNICAL SPECIFICATIONS PRIOR TO HAVING AN INSERTED TRIP SIGNAL REMOVED. CORRECTIVE ACTION INCLUDES COUNSELING THE OPERATORS INVOLVED, REVIEWING THE EVENT WITH ALL OPERATIONS DEPARTMENT PERSONNEL, AND THE INCORPORATION OF LESSONS LEARNED INTO THE LICENSED OPERATOR TRAINING PROGRAM. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B) AS A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[29]   CONNECTICUT YANKEE   DOCKET 50-213   LER 91-021
REDUCTION IN RESIDUAL HEAT REMOVAL CAPABILITY DUE TO EQUIPMENT FAILURE.
EVENT DATE: 101991   REPORT DATE: 111891   NSSS: WE   TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 223416) ON OCTOBER 19, 1991, WITH THE PLANT IN MODE 4 (HOT SHUTDOWN) AND FORCED REACTOR COOLANT SYSTEM (RCS) FLOW BEING MAINTAINED ONLY IN LOOP 3, A FAILURE OF THE RCS LOOP 3 HOT LEG LOOP ISOLATION VALVE CAUSED A MAJOR REDUCTION IN FORCED REACTOR COOLANT FLOW. THE VALVE STEM BROKE ALLOWING THE DISC ASSEMBLY TO FALL. THE CAUSE OF THE STEM FAILURE HAS NOT YET BEEN DETERMINED. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(V)(B) SINCE THE VALVE FAILURE IMPEDED THE LOOP'S FUNCTION OF REMOVING RESIDUAL HEAT. FORCED RCS FLOW WAS REESTABLISHED IN 17 MINUTES BY STARTING THE LOOP 4 REACTOR COOLANT PUMP (RCP). DURING THE STARTING OF THE LOOP 4 RCP, THE PROVISION OF THE IDLED LOOP STARTUP TECHNICAL

SPECIFICATION (3.4.1.11) WERE NOT FULFILLED IN ORDER TO EXPEDITE THE REESTABLISHMENT OF FORCED FLOW IN THE RCS AND LIMIT RCS HEATUP. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B) SINCE IT IS A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[30] COOK 1 DOCKET 50-315 LER 91-005 REV 01
UPDATE ON DESIGN DOCUMENTS COULD NOT BE LOCATED THAT WOULD DEMONSTRATE THE CAPABILITY OF THE DIESEL GENERATORS VENTILATION AND EXHAUST STRUCTURES TO WITHSTAND THE EFFECTS OF A TORNADO.
EVENT DATE: 071891 REPORT DATE: 103191 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)
VENDOR: AMERICAN ELECTRIC POWER SERVICE CORP.

(NSIC 223325) THIS REVISION IS BEING SUBMITTED TO PROVIDE PREVENTIVE ACTIONS. DURING AN ENGINEERING REVIEW, DESIGN DOCUMENTS COULD NOT BE LOCATED THAT WOULD DEMONSTRATE THE CAPABILITY OF THE EMERGENCY DIESEL GENERATOR (EDG) COMBUSTION AIR ENGINE EXHAUST, AND ROOM VENTILATION SYSTEMS TO WITHSTAND THE EFFECTS OF A TORNADO. COMPONENTS FOUND TO BE POTENTIALLY VULNERABLE INCLUDED THE COMBUSTION AIR INTAKE FILTER AND SILENCER, THE EDG EXHAUST STACK AND SILENCERS, AND THE DUCT WORK ASSOCIATED WITH THE ROOM VENTILATION SUPPLY OF THE PLANT'S FOUR EDGS. UPON RECOGNITION OF THIS CONDITION, ACTIONS WERE TAKEN TO PROVIDE A PRESSURE EQUALIZATION PATH TO PROTECT THE VENTILATION DUCT WORK AND THE INTAKE AIR SILENCER FROM THE POSTULATED DIFFERENTIAL PRESSURE LOAD. THE NRC GRANTED A 30 DAY EXEMPTION OF THE TORNADO DESIGN CRITERION TO ALLOW MODIFICATION OF THE SPECIFIC COMPONENTS MENTIONED. MODIFICATIONS TO THE OUTSIDE STRUCTURES WERE COMPLETED ON AUGUST 16, 1991. AN ADDITIONAL DAMPER WILL BE INSTALLED ON THE EDG ROOM VENTILATION SUPPLY TO ISOLATE THE ENTERNAL DUCT WORK FROM TORNADO EFFECTS. THIS DAMPER IS EXPECTED TO BE INSTALLED BY DECEMBER 31, 1991.

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

(NSIC 223424) THIS REVISION PROVIDES INFORMATION ON THE RESULTS OF THE ROOT CAUSE ANALYSIS AND CORRECTS THE FAILED COMPONENT IDENTIFICATION INFORMATION. ON AUGUST 19, 1991, DURING ROUTINE SURVEILLANCE TESTING OF THE FULL LENGTH RODS, CONTROL BANK A WAS ORDERED INTO THE CORE BY THE ROD CONTROL SYSTEM BUT APPEARED NOT TO MOVE INTO THE CORE BY OBSERVATION OF THE ANALOG ROD POSITION INDICATION. FURTHER POSITIONING OF CONTROL BANK A RESULTED IN AN URGENT ALARM BEING RECEIVED. INITIAL TROUBLESHOOTING OF THE ROD CONTROL SYSTEM FOUND NO REASON FOR THE URGENT ALARM AND IT WAS RESET. THE RODS WERE MOVED PER THE TEST REQUIREMENTS AND THE SURVEILLANCE COMPLETED. AT APPROXIMATELY 1200, A ROD MISALIGNMENT IN SHUTDOWN BANK A, GROUP 2 WAS SUSPECTED DUE TO ADDITIONAL SMALL AMOUNT OF DILUTION REQUIRED TO KEEP THERMAL POWER AT THE PRE-SURVEILLANCE LEVEL AND A DECREASE IN THE ANALOG ROD POSITION INDICATION FOR SHUTDOWN BANK A, GROUP 2 IN RELATION TO PRE-SURVEILLANCE READINGS. AT 1305 HOURS THE FLUX MAPPING SYSTEM WAS USED TO DETERMINE THAT THE 4 RODS OF SHUTDOWN BANK A, GROUP 2 WERE INSERTED APPROXIMATELY 6 STEPS INTO THE CORE. FURTHER TROUBLESHOOTING OF THE ROD CONTROL SYSTEM DISCOVERED A FAILURE OF THE 2AC POWER CABINET MULTIPLEXING RELAY, MXR-1. IT WAS REPLACED AND PROPER OPERATION OF THE ROD CONTROL SYSTEM WAS VERIFIED.

[32] COOPER DOCKET 50-298 LER 91-011
REACTOR PROTECTION SYSTEM (RPS) AND ENGINEERED SAFETY FEATURE TRIPS DUE TO SPURIOUS TRIP OF TWO RPS REACTOR VESSEL WATER LEVEL INSTRUMENTS CAUSED BY AN AIR BUBBLE IN THE INSTRUMENT REFERENCE LEG TAP.
EVENT DATE: 100791 REPORT DATE: 110691 NSSS: GE TYPE: BWR

(NSIC 223358) ON 10/7/91, AT 01:43, WITH THE PLANT SHUTDOWN AND THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM OPERATING IN THE SHUTDOWN COOLING MODE OF OPERATION, AN RPS TRIP AND ACTUATION OF GROUPS 2 AND 3 ISOLATIONS (RCS AND RWCS) OCCURRED DUE TO A SPURIOUS LOW LEVEL ACTUATION OF TWO REACTOR VESSEL WATER LEVEL SENSORS. AT THE

TIME, WATER LEVEL WAS 10-15 INCHES ABOVE THE LEVEL MAINTAINED DURING NORMAL OPERATION. THE CAUSE OF THE EVENT IS POSTULATED TO BE AS A RESULT OF EITHER INTRODUCTION OF AIR THROUGH REFERENCE LEG INJECTION SOLENOID VALVE NBI-SOV-SPV739 INTO THE CONDENSING CHAMBER 3B REFERENCE LEG, WHICH IS COMMON TO ALL OF THE AFFECTED INSTRUMENTS, OR WATER LEAKAGE THROUGH THE SOLENOID VALVE INTO THE REFERENCE LEG, DISTURBING AN EXISTING AIR BUBBLE. THE POSTULATED REFERENCE LEG DISTURBANCE OCCURRED, EVEN THOUGH THE SOLENOID VALVE WAS CLOSED, UPON RESTORATION OF THE "B" CORE SPRAY (CS) SYSTEM FOLLOWING PERFORMANCE OF LOCAL LEAK RATE TESTING (LLRT). THE AIR BUBBLE IN THE REFERENCE LEG EVENTUALLY MIGRATED TO THE CONDENSING CHAMBER REFERENCE COLUMN. THE HYDRAULIC EFFECTS OF THE AIR BUBBLE IN THE COLUMN RESULTED IN TRIPPING THE RPS LEVEL SENSORS. UPON RESTORATION OF NORMAL WATER LEVEL FOLLOWING REFUELING ACTIVITIES, INSTRUMENT LINE VENTING OR BACKFLUSHING WILL BE CONDUCTED.

[33] COOPER DOCKET 50-298 LER 91-012
UNPLANNED ENGINEERED SAFETY FEATURE ACTUATIONS DURING DESIGN CHANGE ACTIVITIES RESULTING FROM PERSONNEL ERROR, HUMAN FACTORS, AND INADEQUATE PRECAUTIONS FOR WORK IN SENSITIVE AREAS.
EVENT DATE: 101591 REPORT DATE: 111491 NSSS: GE TYPE: BWR

(NSIC 223423) ON OCTOBER 15, 1991, AT 2:03 A.M. AND 3:31 A.M., GROUP 3 ISOLATIONS REACTOR WATER CLEANUP (RWCU) SYSTEM) OCCURRED DUE TO A BLOWN FUSE AND DE-ENERGIZING A RELAY, RESULTING IN CLOSURE OF THE OUTBOARD ISOLATION VALVE. AT 9:41 P.M., A LOSS OF NORMAL OFFSITE POWER OCCURRED WHEN A WIRE BEING MOVED INADVERTENTLY COMPLETED A RELAY TRIP CIRCUIT. THE ESSENTIAL BUSSES TRANSFERRED TO THE EMERGENCY TRANSFORMER AND THE #2 DIESEL GENERATOR STARTED. DURING THE TRANSFER TO EMERGENCY POWER, THE "A" REACTOR PROTECTION SYSTEM LOST POWER, CAUSING A HALF SCRAM AND VARIOUS GROUP ISOLATIONS. THESE ACTUATIONS WERE A RESULT OF DESIGN CHANGE ACTIVITIES ASSOCIATED WITH THE CNS ANNUNCIATOR UPGRADE PROJECT. DURING THESE EVENTS, THE PLANT WAS IN A REFUELING OUTAGE, AND FUEL WAS BEING OFFLOADED TO THE SPENT FUEL POOL. THESE ACTUATIONS RESULTED FROM AN IMPROPERLY INSTALLED RELAY IDENTIFICATION TAG AND THE FAILURE TO IMPLEMENT THE PRECAUTIONS NECESSARY TO PREVENT INADVERTENT ACTUATIONS WHEN WORKING IN SENSITIVE AREAS. CONTRIBUTING CAUSES WERE A DESIGN ERROR, FAILURE OF THE DESIGN CHANGE TO IDENTIFY THE SENSITIVE WORK AREAS, AND INCORRECTLY MOVING A WIRE. AS A RESULT OF THESE EVENTS, DESIGN CHANGE WORK IN THE CONTROL ROOM WAS STOPPED UNTIL THE DESIGN CHANGE PACKAGES WERE INDEPENDENTLY REVIEWED TO IDENTIFY SENSITIVE AREAS.

[34] CRYSTAL RIVER 3 DOCKET 50-302 LER 91-009
HIGH LOCALIZED TEMPERATURE CAUSES DIESEL GENERATOR COMPONENTS TO EXCEED TEMPERATURE RATINGS.
EVENT DATE: 100791 REPORT DATE: 110691 NSSS: BW TYPE: PWR
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)
CLARK CONTROL INC.

(NSIC 223361) ON OCTOBER 8, 1991 AT 1700, CRYSTAL RIVER UNIT 3 (CR-3) DETERMINED THAT SEVERAL RELAYS AND FUSES LOCATED IN THE CONTROL RELAY CABINETS IN THE A AND B EMERGENCY DIESEL GENERATOR (EDG) ROOMS MIGHT EXPERIENCE TEMPERATURES ABOVE THEIR DESIGN AMBIENT TEMPERATURE RATING DURING DIESEL GENERATOR OPERATION. CR-3 DISCOVERED THIS CONDITION WHILE EVALUATING NRC INFORMATION NOTICE 89-30, SUPPLEMENT 1 "HIGH TEMPERATURE ENVIRONMENTS AT NUCLEAR POWER PLANTS". THIS DESIGN CONCERN DOES NOT AFFECT THE EDG OPERABILITY. THE HIGHER TEMPERATURE IN THE CONTROL CABINETS IS CAUSED BY THE PROXIMITY OF THE CABINETS TO THE EDG. THIS PROBLEM WAS CAUSED BY FAILURE OF THE DIESEL GENERATOR VENDOR TO MEET THE DESIGN SPECIFICATION. TO RESOLVE THE CONCERNS WITH THE RELAYS AND FUSES FOR FUTURE OPERATION, ACTIONS WILL BE TAKEN TO REDUCE THE TEMPERATURE IN THE CABINETS. TO ADDRESS THE CURRENTLY INSTALLED EQUIPMENT, THE FUSES WILL BE REPLACED AND AN EVALUATION OF THE RELAYS WILL BE PERFORMED. TWO RELAYS FROM ONE DIESEL GENERATOR WILL BE REPLACED. THESE RELAYS WILL BE TESTED FOR TEMPERATURE EFFECTS ON AGING TO DETERMINE IF ADDITIONAL RELAYS MUST BE REPLACED.

[35] CRYSTAL RIVER 3 DOCKET 50-302 LER 91-010
WIRING PROBLEM CAUSES TRANSFORMER BREAKERS TO OPEN ACTUATING THE EMERGENCY DIESEL GENERATOR.
EVENT DATE: 102091 REPORT DATE: 111891 NSSS: BW TYPE: PWR

(NSIC 223427) ON OCTOBER 20, 1991, CRYSTAL RIVER UNIT 3 (CR-3) WAS IN MODE 5 (COLD SHUTDOWN) FOR A SCHEDULED MAINTENANCE OUTAGE. AT 1443 THE BREAKERS FOR THE OFFSITE POWER TRANSFORMER OPENED, DISCONNECTING THE ENGINEERED SAFEGUARDS (ES) BUSSES FROM THE OFFSITE POWER SUPPLY. DECAY HEAT COOLING WAS INTERRUPTED FOR LESS THAN A MINUTE WHILE THE EMERGENCY DIESEL GENERATOR (EDG) LOADED THE ES BUS AND OPERATORS RESTARTED THE DECAY HEAT PUMP (DHP). UPON STARTING THE DHP, A PURIFICATION RELIEF VALVE LIFTED CAUSING A DROP IN PRESSURIZER LEVEL. OPERATORS QUICKLY IDENTIFIED THE SOURCE AND ISOLATED THE PURIFICATION SYSTEM. AT 1447, THE OPERATORS MANUALLY ENERGIZED THE REMAINING ES BUS VIA THE CR-3 STARTUP TRANSFORMER. THIS EVENT WAS CAUSED BY A PREEXISTING WIRE INSTALLATION WHICH INADVERTENTLY APPLIED 115V AC TO THE CR-3 BATTERY BUS. THE ERRONEOUS WIRE HAS BEEN REMOVED. THE ASSOCIATED BREAKER RELAYS HAVE BEEN REPLACED WITH LESS SENSITIVE RELAYS TO MINIMIZE THE POSSIBILITY OF SIMILAR EVENTS IN THE FUTURE. INFORMATION WILL BE PROVIDED TO ALL OPERATORS CONCERNING THIS EVENT. THE ASSOCIATED EMERGENCY PROCEDURE (EP) WILL BE REVISED TO ADDRESS RESTARTING THE DHP.

[36] DAVIS-BESSE 1 DOCKET 50-346 LER 91-005
INADVERTENT SAFETY FEATURES ACTUATION DUE TO SPURIOUS SPIKE ON CONTAINMENT RADIATION MONITOR RE2007.
EVENT DATE: 101191 REPORT DATE: 111191 NSSS: BW TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 223448) ON OCTOBER 11, 1991 AT 1417 HOURS WITH THE PLANT IN MODE 5 FOR THE SEVENTH REFUELING OUTAGE, THE STATION EXPERIENCED AN INADVERTENT SAFETY FEATURES ACTUATION SYSTEM (SFAS) LEVEL 1 ACTUATION WHEN SFAS CHANNEL 4 CONTAINMENT RADIATION MONITOR (RE2007) SPURIOUSLY SPIKED. THE SPIKE OCCURRED AFTER RE2007 WAS CALIBRATED FOLLOWING MOVEMENT FROM CONTAINMENT INTO THE SHIELD BUILDING ANNULUS IN PREPARATION FOR ENTRY INTO MODE 4. CHANNEL 1 OF SFAS HAD BEEN PREVIOUSLY DE-ENERGIZED FOR THE INSTALLATION OF SFAS SHUTDOWN BYPASS MODIFICATION 90-0006. THIS CONDITION ALLOWED THE SPURIOUS TRIP OF A SINGLE CHANNEL TO CAUSE AN SFAS ACTUATION. THIS EVENT OCCURRED BECAUSE OF THE SPURIOUS SPIKE ON RE2007 AND OF THE DESIGN OF SFAS WHICH DOES NOT ALLOW SFAS TO BE DE-ENERGIZED WITHOUT FAILING THE OUTPUT DEVICES TO THEIR SAFETY POSITIONS/STATUS. SFAS SHUTDOWN BYPASS MODIFICATION 90-0006 IS BEING INSTALLED TO PROTECT AGAINST UNNECESSARY ENGINEERED SAFETY FEATURE ACTUATIONS.

[37] DAVIS-BESSE 1 DOCKET 50-346 LER 91-004
DEFICIENT REACTOR PROTECTION SYSTEM RESPONSE TIME SURVEILLANCE TESTING.
EVENT DATE: 101291 REPORT DATE: 111191 NSSS: BW TYPE: PWR

(NSIC 223447) ON OCTOBER 12, 1991, WITH THE PLANT IN MODE 5 FOR THE SEVENTH REFUELING OUTAGE (7RFO), SYSTEMS ENGINEERING IDENTIFIED THAT REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME TESTING DID NOT ADEQUATELY SATISFY THE REQUIREMENTS OF TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT 4.3.1.1.3. THE DELAY TIME OF AN ITE RELAY IN THE SHUNT TRIP DEVICE (STD) OF THE REACTOR TRIP BREAKERS (RTBS) WAS NOT MEASURED OR ACCOUNTED FOR IN OVERALL RPS RESPONSE TIME TESTING. AUTOMATIC ACTUATION CAPABILITY WAS PROVIDED TO THE STD UNDER FACILITY CHANGE REQUEST (FCR) 84-026 BASED ON GENERIC LETTER 83-28 GUIDANCE. TESTING SHOWED THE DELAY WAS BOUNDED BY THE 14 MSEC SPECIFIED BY THE RELAY VENDOR. A REVIEW OF PAST DATA REVEALED THAT WITH THE 14 MSEC DELAY CONSIDERED, THE HIGH FLUX/NUMBER OF REACTOR COOLANT PUMPS ON TRIP FUNCTION WOULD HAVE EXCEEDED ITS ALLOWABLE RESPONSE TIME LIMIT FROM DECEMBER 1986 TO MARCH 1988 BY 7.5 MSEC. THIS TEST DEFICIENCY WAS A RESULT OF AN INADEQUATE TECHNICAL REVIEW OF THE FCR PACKAGE AND ITS SUPPLEMENT. ON OCTOBER 22, 1991, RPS OVERALL RESPONSE TIME CALCULATION PROCEDURES WERE REVISED TO ACCOUNT FOR THE 14 MSEC DELAY AND WERE PERFORMED. ALL FUNCTIONAL UNITS WERE WITHIN THEIR RESPECTIVE ALLOWABLE RESPONSE TIME LIMITS. CURRENT PROCEDURES INVOLVING PLANT MODIFICATIONS REQUIRE MORE EXTENSIVE REVIEW AND APPROVAL WHICH SHOULD ENSURE THAT CHANGES SUCH AS THIS ARE THOROUGHLY EVALUATED AND DOCUMENTED.

[38] DIABLO CANYON 1 DOCKET 50-275 LER 91-015
VIOLATION OF TECHNICAL SPECIFICATION 3.7.10 WHEN AN HOURLY FIRE WATCH PATROL WAS NOT PERFORMED DUE TO INADEQUATE INSTRUCTIONS.
EVENT DATE: 091791 REPORT DATE: 101691 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: DIABLO CANYON 2 (PWR)

(NSIC 223275) ON SEPTEMBER 17, 1991, AT 0300 PDT, WITH UNIT 1 IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER, AND UNIT 2 DEFUELED, THE ACTION STATEMENT FOR TECHNICAL SPECIFICATION 3.7.10 WAS NOT MET FOR UNITS 1 AND 2 WHEN THE REQUIRED HOURLY FIRE WATCH PATROL WAS NOT PERFORMED IN THE REQUIRED SAFETY-RELATED EQUIPMENT ROOMS. THE HOURLY FIRE WATCH PATROL WAS NOT PERFORMED BECAUSE THE HOURLY ROVING FIRE WATCH WAS UNABLE TO EXIT FROM THE RADIOLOGICALLY CONTROLLED AREA (RCA) OF THE PLANT AND EXCHANGE DUTIES WITH ANOTHER FIRE WATCH IN THE TURBINE BUILDING. THE ROOT CAUSE OF THE EVENT WAS DETERMINED TO BE THAT NO WRITTEN INSTRUCTIONS EXISTED TO ASSURE THAT TS RELATED FIRE IMPAIRMENTS WOULD BE INSPECTED EACH HOUR DURING UNEXPECTED CONDITIONS WHICH COULD DELAY FIRE WATCH PERSONNEL. THE CORRECTIVE ACTIONS TO PREVENT RECURRENCE INCLUDE (1) PROVIDING WRITTEN INSTRUCTIONS TO FIRE WATCH PERSONNEL ON ACTIONS TO TAKE IF DELAYED DURING ROUNDS; AND (2) PREPARING AN INCIDENT SUMMARY OF THIS EVENT, REVIEWING IT WITH ALL FIRE WATCH PERSONNEL, AND INCLUDING THE INCIDENT SUMMARY IN INITIAL ROVING FIRE WATCH TRAINING.

[39] DIABLO CANYON 1 DOCKET 50-275 LER 91-016
MISSED SURVEILLANCE OF AIRLOCK DOOR SEALS DUE TO PERSONNEL ERROR CAUSED BY INADEQUATE KNOWLEDGE OF LEAK-RATE MONITOR OPERATION.
EVENT DATE: 092791 REPORT DATE: 102891 NSSS: WE TYPE: PWR
VENDOR: VOLUMETRICS

(NSIC 223332) ON JUNE 11, 1991, AT 0947 PDT, THE CONDITIONAL SURVEILLANCE REQUIRED BY TECHNICAL SPECIFICATION (TS) 4.6.1.3.A WAS MISSED WHEN A LEAK RATE TEST OF THE CONTAINMENT PERSONNEL AIRLOCK DOOR WAS NOT SATIFACTORILY PERFORMED FOLLOWING OPENING OF THE INNER AND OUTER DOORS. ON SEPTEMBER 26, 1991, THE OPERATOR PERFORMING DAILY SURVEILLANCE CHECKS NOTED THAT THE PERSONNEL AIRLOCK LEAK-RATE MONITOR WAS OUT-OF-SERVICE BECAUSE OF A CLEARANCE FOR CALIBRATION. WHEN THE SURVEILLANCE TEST WAS REVIEWED BY PLANT ENGINEERING ON SEPTEMBER 27, 1991, THE OUT-OF-SERVICE STATUS OF THE LEAK-RATE MONITOR WAS NOTED AND REPORTED TO THE SYSTEM ENGINEER, WHO INQUIRED AS TO THE OPERABILITY STATUS OF THE LEAK-RATE MONITOR. THE SHIFT SUPERVISOR AT THAT TIME DETERMINED THAT THE LEAK-RATE MONITOR WAS INOPERABLE, AFTER WHICH THE DOOR SEALS WERE SUCCESSFULLY LEAK-RATE TESTED. A REVIEW DETERMINED THAT AN ACCEPTABLE LEAK RATE TEST WAS NOT PERFORMED FOLLOWING 17 CONTAINMENT ENTRIES DURING THE PERIOD FROM JUNE 11, 1991, TO SEPTEMBER 27, 1991. THE IMMEDIATE CAUSE OF THE MISSED SURVEILLANCES WAS A FAULTY SOLENOID VALVE THAT RENDERED THE LEAK-RATE MONITOR INCAPABLE OF PERFORMING AUTOMATICALLY. THE ROOT CAUSE IS DUE TO PERSONNEL ERROR CAUSED BY INADEQUATE KNOWLEDGE OF THE LEAK-RATE MONITOR OPERATION. TO PREVENT RECURRENCE, AN OPERATIONS INCIDENT SUMMARY HAS BEEN PREPARED.

[40] DIABLO CANYON 2 DOCKET 50-323 LER 91-009
10 CFR 100 DOSE LIMITS POTENTIALLY EXCEEDED IN THE EVENT OF A DESIGN BASIS LOSS OF COOLANT ACCIDENT RECOVERY AS A RESULT OF VALVE LEAKAGE.
EVENT DATE: 092691 REPORT DATE: 110191 NSSS: WE TYPE: PWR

(NSIC 223367) ON OCTOBER 4, 1991, AT 1645 PDT, WITH UNIT 2 DEFUELED, AN EVALUATION DETERMINED THAT THE CONTROL ROOM AND EXCLUSION AREA BOUNDARY 10 CFR 100 DOSE LIMITS COULD BE POTENTIALLY EXCEEDED DURING THE DESIGN BASIS RECIRCULATION PHASE OF LOSS OF COOLANT ACCIDENT (LOCA) RECOVERY. THE POTENTIAL PROBLEM WAS IDENTIFIED DURING THE PERFORMANCE OF A HYDROSTATIC TEST ON SEPTEMBER 26, 1991. ON OCTOBER 4, 1991, AT 1800 PDT, A FOUR-HOUR, NON-EMERGENCY REPORT WAS MADE TO THE NRC IN ACCORDANCE WITH 10 CFR 50.72(B)(2)(I). ON SEPTEMBER 26, 1991, DURING THE PERFORMANCE OF A HYDROSTATIC TEST, TOTAL LEAKAGE OF APPROXIMATELY 1.3 GALLONS PER MINUTE (GPM) WAS IDENTIFIED FROM DIAPHRAGM VALVES CVCS-2-8741 AND CVCS-2-548 IN THE CHARGING PUMP SUCTION LINE. AN EVALUATION DETERMINED THAT THIS LEAKAGE COULD HAVE RESULTED IN THE CONTROL ROOM AND EXCLUSION AREA BOUNDARY 10 CFR 100 DOSE LIMITS BEING EXCEEDED DURING THE RECIRCULATION PHASE OF A DESIGN

BASIS LOCA RECOVERY. INVESTIGATIONS ARE BEING PERFORMED TO DETERMINE THE ROOT CAUSE OF THE VALVE LEAKAGE AND THE POTENTIAL AFFECT ON THE HEALTH AND SAFETY OF THE PUBLIC. THE ROOT CAUSE, SAFETY ANALYSIS, AND CORRECTIVE ACTIONS WILL BE SUBMITTED IN A SUPPLEMENTAL LER.

[41] DIABLO CANYON 2 DOCKET 50-323 LER 91-006
MOMENTARY LOSS OF POWER TO RADIATION MONITOR RM-26 CAUSES CONTROL ROOM VENTILATION SYSTEM MODE SHIFT DUE TO PERSONNEL ERROR.
EVENT DATE: 100391 REPORT DATE: 110191 NSSS: WE TYPE: PWR

(NSIC 223365) ON OCTOBER 3, 1991, AT 1256 PDT, WITH UNIT 2 IN MODE 5 (COLD SHUTDOWN) AT 0 PERCENT POWER, THE CONTROL ROOM VENTILATION SYSTEM (CRVS) SHIFTED FROM NORMAL MODE TO PRESSURIZATION MODE. THIS MODE CHANGE CONSTITUTES AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION. WHILE REMOVING INSTRUMENT AC INVERTER IY-24 FROM SERVICE IN ACCORDANCE WITH PROCEDURE OP J-10:III, "INSTRUMENT AC SYSTEM-SHUTDOWN AND CLEARING," A SENIOR LICENSED OPERATOR AND A NON-LICENSED OPERATOR INADVERTENTLY OPENED THE OUTPUT BREAKER FOR INVERTER IY-23 INSTEAD OF THE INTENDED BREAKER FOR INVERTER IY-24. THIS RESULTED IN A MOMENTARY LOSS OF POWER TO THE CONTROL ROOM VENTILATION INTAKE RADIATION MONITOR RM-26, WHICH CAUSED THE CRVS TO SHIFT FROM NORMAL MODE TO PRESSURIZATION MODE. ON OCTOBER 3, 1991, AT 1427 PDT, A FOUR-HOUR, NON-EMERGENCY REPORT WAS MADE TO THE NRC IN ACCORDANCE WITH 10 CFR 50.72(B)(2)(II). THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR (INATTENTION TO DETAIL). THE OPERATORS INVOLVED FAILED TO PERFORM AN ADEQUATE PRE-JOB REVIEW OF THE OPERATING PROCEDURE. THEY DID NOT NOTICE THAT SPECIFIC ATTACHMENTS TO THE PROCEDURE HAD BEEN PROVIDED. TO PREVENT RECURRENCE, (1) THE OPERATORS INVOLVED WERE COUNSELED, (2) AN OPERATIONS INCIDENT SUMMARY HAS BEEN ISSUED, (3) OP J-10:III WILL BE REVISED, AND (4) OPERATIONS POLICY A-6 WAS REVISED.

[42] DIABLO CANYON 2 DOCKET 50-323 LER 91-007
INADVERTENT SAFETY INJECTION WHILE IN MODE 5 DUE TO PERSONNEL ERROR.
EVENT DATE: 100691 REPORT DATE: 110191 NSSS: WE TYPE: PWR

(NSIC 223366) ON OCTOBER 6, 1991, AT 0008 PDT, WITH UNIT 2 IN MODE 5 (COLD SHUTDOWN) DURING THE FOURTH REFUELING OUTAGE, TWO SAFETY INJECTION (SI) SIGNALS OCCURRED DUE TO TECHNICIANS OPERATING SOLID STATE PROTECTION SYSTEM (SSPS) CONTROL SWITCHES OUT OF SEQUENCE WITH THE SURVEILLANCE TEST PROCEDURE (STP) THEY WERE PERFORMING. A FOUR HOUR, NON-EMERGENCY REPORT REQUIRED BY 10 CFR 50.72(B)(2)(II) WAS MADE ON OCTOBER 6, 1991, AT 0218 PDT. THE CAUSE OF THE SIS WAS PERSONNEL ERROR. THE TECHNICIANS FAILED TO UTILIZE STP I-16D4 "RECONFIGURING AN SSPS TRAIN IN MODES 5 OR 6," DURING THE SWITCH OPERATING SEQUENCE. STP I-16D4 REQUIRES THE SSPS INPUT ERROR INHIBIT SWITCH TO BE REPOSITIONED FROM "NORMAL TO "INHIBIT" AND THE MODE SELECTOR SWITCH FROM "TEST" TO "OPERATE," IN THAT ORDER. THE TECHNICIANS REPOSITIONED THE SWITCHES IN REVERSE ORDER ON SSPS TRAINS A AND B. EACH TRAIN PRODUCED AN SI SIGNAL. ALL EQUIPMENT PERFORMED AS EXPECTED IN MODE 5 DURING THE SI. UNIT 2 WAS RETURNED TO NORMAL MODE 5 ALIGNMENT ON OCTOBER 6, 1991, AT 0015 PDT. SINCE EMERGENCY CORE COOLING PUMPS WERE SECURED FOR MAINTENANCE, THERE WAS NO WATER INJECTION INTO THE REACTOR COOLANT SYSTEM. THE TECHNICIANS INVOLVED IN THIS EVENT WERE COUNSELED IN ACCORDANCE WITH PG&E'S POSITIVE DISCIPLINE PROGRAM A MEMORANDUM HAS BEEN ISSUED FROM THE VICE PRESIDENT, DIABLO CANYO OPERATIONS AND PLANT MANAGER, EMPHASIZING THE NEED FOR PROCEDURAL COMPLIANCE AND VERIFICATION.

[43] DRESDEN 2 DOCKET 50-237 LER 91-034
PRIMARY CONTAINMENT ISOLATION VALVE CLOSURE DUE TO REACTOR WATER CLEANUP SYSTEM ISOLATION.
EVENT DATE: 101291 REPORT DATE: 110691 NSSS: GE TYPE: BWR

(NSIC 223347) ON OCTOBER 12, 1991, AT 1643 HOURS, WITH UNIT 2 AT 95% POWER, A REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATION OCCURRED RESULTING IN PRIMARY CONTAINMENT ISOLATION MOTOR OPERATED VALVES (MOVS) 2-1201-1 AND 2-1201-2 FULLY CLOSING. DUE TO PRESSURE OSCILLATIONS IN THE RWCU SYSTEM, OPERATIONS PERSONNEL WERE TRANSFERRING THE PRESSURE CONTROL STATION FROM "AUTO" TO "MANUAL" WHEN THE

RWCU SYSTEM ISOLATED ON A HIGH PRESSURE SIGNAL. ON INVESTIGATION, IT WAS FOUND THAT THE NITROGEN GAS BLANKET ON THE RWCU SURGE TANK HAD BEEN LOST. CORRECTIVE ACTIONS WERE TO DRAIN THE LOW PRESSURE SENSING LINE OF THE SURGE TANK LEVEL TRANSMITTER, RE-PRESSURE THE RWCU SURGE TANK WITH NITROGEN, INSPECT RWCU PUMPS AND PIPING, AND RESET/RESTART THE RWCU SYSTEM. THIS EVENT HAD NO ADVERSE SAFETY SIGNIFICANCE BECAUSE THE RWCU SYSTEM PROPERLY ISOLATED ON THE HIGH PRESSURE SIGNAL AND ISOLATION OF THE RWCU SYSTEM FOR SHORT PERIODS HAS NO EFFECT ON POWER OPERATION OR COOLANT CHEMISTRY LIMITS. TWO RECENT PREVIOUS EVENTS INVOLVING CLOSURE OF PRIMARY CONTAINMENT ISOLATION VALVES WHILE ADJUSTING PRESSURE/FLOW ON THE RWCU SYSTEM HAVE BEEN REPORTED BY LERS 91-014 AND 91-019.

[44] DRESDEN 2 DOCKET 50-237 LER 91-033
PRIMARY CONTAINMENT ISOLATION VALVE CLOSURE DUE TO REACTOR WATER CLEANUP SYSTEM ISOLATION.
EVENT DATE: 101491 REPORT DATE: 110691 NSSS: GE TYPE: BWR

(NSIC 223346) ON 10/14/91, AT 1101 HOURS, WITH UNIT 2 AT 51% POWER, A REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATION OCCURRED RESULTING IN PRIMARY CONTAINMENT ISOLATION MOTOR OPERATED VALVES (MOVS) 2-1201-1 AND 2-1201-2 FULLY CLOSING. OPERATIONS PERSONNEL WERE RETURNING THE RWCU SYSTEM TO NORMAL OPERATION AFTER IT WAS ISOLATED TO SUPPORT 250 VDC BATTERY SWITCHING. WHILE RETURNING THE RWCU SYSTEM TO NORMAL OPERATION, IT ISOLATED ON A HIGH PRESSURE SIGNAL. THE AUTOMATIC ISOLATION OCCURRED, AS EXPECTED, TO PROTECT THE RWCU PIPING. THIS EVENT HAD NO SAFETY SIGNIFICANCE BECAUSE ISOLATION OF THE RWCU SYSTEM FOR SHORT PERIODS HAS NO AFFECT ON POWER OPERATION OR COOLANT CHEMISTRY LIMITS. WHEN RETURNING THE RWCU SYSTEM TO OPERATION WITH THE REACTOR AT OPERATING PRESSURE, RWCU SYSTEM PRESSURE OSCILLATIONS ARE FREQUENTLY GENERATED DUE TO CONTROL DIFFICULTY WITHIN THE RWCU FLOW CONTROLLER. ALTHOUGH THIS EVENT WAS NOT INITIATED BY PRIMARY CONTAINMENT ISOLATION LOGIC, IT DID RESULT IN UNPLANNED CLOSURE OF PRIMARY CONTAINMENT ISOLATION VALVES. IMMEDIATE CORRECTIVE ACTIONS WERE TO RESET/RESTART THE RWCU SYSTEM. ADDITIONAL CORRECTIVE ACTIONS INCLUDE REVIEWING THE PRESSURE AND FLOW CONTROL VALVE TRIM AND PNEUMATIC ACTUATOR CONFIGURATIONS FOR POSSIBLE IMPROVEMENT. THREE PREVIOUS EVENTS INVOLVING RWCU SYSTEM ISOLATIONS OR UNPLANNED ISOLATION VALVE CLOSURES HAVE BEEN REPORTED BY LERS 91-014, 91-019, AND 91-034.

[45] DRESDEN 3 DOCKET 50-249 LER 91-013
250 VOLT DC BATTERY DISCHARGE VOLTAGE DECREASED BELOW DESIGN BASIS LIMIT DUE TO UNKNOWN CAUSE.
EVENT DATE: 101491 REPORT DATE: 111291 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 2 (BWR)

(NSIC 223443) ON OCTOBER 14, 1991, WITH UNIT 3 IN A REFUEL OUTAGE, THE UNIT 3 250 VOLT BATTERY WAS REMOVED FROM SERVICE IN PREPARATION FOR A DISCHARGE TEST AS REQUIRED BY TECHNICAL SPECIFICATION 4.9.A.3. UNIT 2 WAS OPERATING AT POWER AT THIS TIME. THIS PLACED UNIT 2 IN A 7 DAY LIMITING CONDITION FOR OPERATION, IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.9.B.4.A. DURING THE FIRST MINUTE OF THE SERVICE TEST, THE BATTERY VOLTAGE DROPPED BELOW THE ACCEPTANCE CRITERION OF 210 VOLTS. THE VOLTAGE STABILIZED ABOVE 210 VOLTS FOR THE REMAINDER OF THE FOUR-HOUR TEST. A SUBSEQUENT PERFORMANCE TEST INDICATED A BATTERY CAPACITY OF 96% OF RATED VALUE. A COMPREHENSIVE REVIEW OF THE TESTED LOAD PROFILE WAS BEGUN. SINCE RESOLUTION OF THE ISSUE WAS STILL PENDING, AND BATTERY OPERABILITY HAD NOT BEEN RE-INSTATED, AN ORDERLY SHUTDOWN OF UNIT 2 WAS INITIATED ON OCTOBER 21, 1991. A REVIEW OF THE UNIT 3 250 VOLT BATTERY INTER-TIER CABLES RESISTANCES INDICATED THAT THEIR RESISTANCES HAD INCREASED SINCE THE REPLACEMENT BATTERY WAS ORIGINALLY INSTALLED IN MAY, 1988. THE INTER-TIER CABLES WERE REPLACED AND THEIR RESISTANCES WERE VERIFIED SATISFACTORY. AN ABBREVIATED ONE MINUTE SERVICE TEST WAS THEN PERFORMED ON OCTOBER 27, 1991; THE BATTERY VOLTAGE ACCEPTANCE CRITERION WAS AGAIN NOT MET. ON NOVEMBER 1, 1991, A SERVICE TEST WAS SUCCESSFULLY PERFORMED UTILIZING A REVISED LOAD PROFILE.

[46] DRESDEN 3 DOCKET 50-249 LER 91-014
3-1105A AND 3-1105B STANDBY LIQUID CONTROL RELIEF VALVES LIFTED UNDER THE ACCEPTANCE CRITERIA DUE TO SETPOINT DRIFT.
EVENT DATE: 102291 REPORT DATE: 110791 NSSS: GE TYPE: BWR
VENDOR: CROSBY VALVE & GAGE CO.

(NSIC 223417) ON OCTOBER 22, 1991 AT 0900 HOURS WITH UNIT 3 IN THE REFUEL MODE WITH ALL FUEL REMOVED, STANDBY LIQUID CONTROL (SBLC) RELIEF VALVES 3-1105A AND 3-1105B OPENED AT 1410 PSIG WHEN TESTED DURING THE PERFORMANCE OF DRESDEN MAINTENANCE PROCEDURE (DMP) 1100-2, STANDBY LIQUID CONTROL RELIEF VALVE SETTING. THIS EXCEEDED THE TECHNICAL SPECIFICATION LIMIT OF 1455 TO 1545 PSIG. NEW RELIEF VALVES WERE INSTALLED BY THE MECHANICAL MAINTENANCE DEPARTMENT, AND DURING SUBSEQUENT TESTING THE VALVES LIFTED WITHIN THE TECHNICAL SPECIFICATION LIMIT. THE CAUSE OF THIS EVENT IS RELIEF VALVE SETPOINT DRIFT. WHILE THE SBLC RELIEF VALVES WERE BEING TESTED PER DMP 1100-2, THE SBLC SYSTEM WAS NOT REQUIRED TO BE OPERABLE. THE AS-FOUND RELIEF VALVE SETTINGS WOULD NOT HAVE DIVERTED SBLC FLOW FROM THE REACTOR UNDER DESIGN BASIS CONDITIONS WITH BOTH PUMPS OPERATING. IN ADDITION, THE CONTROL ROD DRIVE (CRD) HYDRAULIC SYSTEM WAS OPERABLE WHEN REQUIRED DURING THE OPERATING CYCLE PRIOR TO THIS EVENT, AND THEREFORE WAS AVAILABLE FOR SHUTTING DOWN THE REACTOR. A PREVIOUS EVENT INVOLVING THE UNIT 3 SBLC RELIEF VALVES WAS REPORTED BY LER 88-07/050249.

[47] FARLEY 1 DOCKET 50-348 LER 91-010
MANUAL REACTOR TRIP DUE TO UNEXPECTED GOVERNOR VALVE CLOSURE.
EVENT DATE: 100391 REPORT DATE: 110191 NSSS: WE TYPE: PWR

(NSIC 223314) AT 0606 ON 10-03-91, WHILE OPERATING AT 100 PERCENT POWER, THE OPERATORS MANUALLY TRIPPED THE REACTOR IN ACCORDANCE WITH FNP-1-AOP-13, "LOSS OF MAIN FEEDWATER." THE SEQUENCE OF EVENTS RESULTING IN THE MANUAL REACTOR TRIP BEGAN WITH THE MAIN TURBINE CONTROL SYSTEM'S UNEXPECTED AUTOMATIC CLOSURE OF THE MAIN TURBINE GOVERNOR VALVES. THE OPERATORS RESPONDED BY PLACING THE MAIN TURBINE DIGITAL ELECTRO-HYDRAULIC CONTROL SYSTEM (DEHC) IN "TURBINE MANUAL" WHICH STOPPED THE GOVERNOR VALVES' CLOSURE AT 15 PERCENT OPEN. THE ENSUING PLANT TRANSIENT RESULTED IN BOTH STEAM GENERATOR FEED PUMPS (SGFPS) TRIPPING ON LOW SUCTION PRESSURE. THIS EVENT WAS CAUSED BY A DEGRADED DC VOLTAGE OUTPUT FROM THE PRIMARY OPERATOR AUTO CONTROLLER (OAC) POWER SUPPLY IN THE MAIN TURBINE DEHC SYSTEM AND A FAILURE OF THE CIRCUITRY WHICH SHOULD HAVE TRANSFERRED THE POWER SUPPLY. THIS DEGRADED VOLTAGE RESULTED IN A CHANGE OF SEVERAL CONTROL PARAMETER VALUES WITHIN THE PRIMARY OAC PRIOR TO ITS COMPLETE FAILURE. THESE DEGRADED CONTROL PARAMETERS WERE PASSED TO THE BACK-UP OAC WHICH RESULTED IN CONTINUED GOVERNOR VALVE CLOSURE UNTIL THE SYSTEM WAS PLACED IN MANUAL BY THE OPERATORS. THE DEGRADED POWER SUPPLY WAS REPLACED AND THE PRIMARY OAC PROGRAMMING WAS RELOADED. A DEH TASK FORCE IS CONTINUING THE INVESTIGATION OF THIS FAILURE AND WILL EVALUATE CORRECTIVE ACTION TO PREVENT RECURRENCE.

[48] FERMI 2 DOCKET 50-341 LER 91-018
AVERAGE POWER RANGE MONITOR WEEKLY CALIBRATION REQUIREMENT NOT PERFORMED PER TECHNICAL SPECIFICATION.
EVENT DATE: 100291 REPORT DATE: 110191 NSSS: GE TYPE: BWR

(NSIC 223316) ON OCTOBER 2, 1991, DETROIT EDISON DISCOVERED THAT AN ERROR HAD BEEN MADE DURING IMPLEMENTATION OF TECHNICAL SPECIFICATION AMENDMENT 75. THIS ERROR RESULTED IN THE FAILURE TO COMPLETE ONE OF TWO WEEKLY CHANNEL CALIBRATION REQUIREMENTS FOR THE AVERAGE POWER RANGE MONITORING (APRM) SYSTEM CHANNELS. PROCEDURES WHICH IMPLEMENTED CHANNEL FUNCTIONAL TEST REQUIREMENTS FOR APRM'S, WHOSE SURVEILLANCE INTERVAL HAD BEEN MODIFIED FROM WEEKLY TO QUARTERLY BY AMENDMENT 75, ALSO IMPLEMENTED A WEEKLY CHANNEL CALIBRATION REQUIREMENT. THESE PROCEDURES WERE REVISED WITHOUT ACCOUNTING FOR THIS WEEKLY CHANNEL CALIBRATION REQUIREMENT. UPON RECOGNITION OF THE MISSED SURVEILLANCE REQUIREMENT, ALL APRM'S WERE DECLARED INOPERABLE. SURVEILLANCES WERE PERFORMED SUCCESSFULLY AND THE APRM'S WERE RESTORED TO OPERABLE STATUS. THE PRIMARY ROOT CAUSE OF THIS EVENT CAN BE ATTRIBUTED TO A DEFICIENCY IN THE PROCESS UTILIZED DURING IMPLEMENTATION OF TECHNICAL SPECIFICATION AMENDMENTS. CHANGES TO THIS PROCESS ARE BEING MADE. PERSONNEL INVOLVED IN IMPLEMENTING AMENDMENT 75 SHOULD HAVE PREVENTED THE ERROR.

A REVIEW OF LESSONS LEARNED FROM THIS EVENT IS BEING DONE BY APPROPRIATE PERSONNEL. ADDITIONALLY, A HUMAN PERFORMANCE ENHANCEMENT SYSTEM (HPES) EVALUATION WILL BE PERFORMED TO CONFIRM THIS ASSESSMENT OF ROOT CAUSE, CONTRIBUTING FACTORS AND CORRECTIVE ACTIONS.

[49] FT. CALHOUN 1 DOCKET 50-285 LER 91-021
INADVERTENT CONTAINMENT ISOLATION ACTUATION SIGNAL.
EVENT DATE: 100491 REPORT DATE: 110491 NSSS: CE TYPE: PWR

(NSIC 223353) ON OCTOBER 4, 1991, AT 1248 HOURS, AN INADVERTENT PARTIAL ACTUATION OF CONTAINMENT ISOLATION ACTUATION SIGNAL (CIAS) OCCURRED AT FORT CALHOUN STATION WHILE THE PLANT WAS HEATING UP FROM MODE 4 (COLD SHUTDOWN) TO MODE 3 (HOT SHUTDOWN). DURING THE PERFORMANCE OF ENGINEERED SAFETY FEATURE (ESF) SURVEILLANCE TEST (ST) OP-ST-ESF-0009, THE COMPONENTS ASSOCIATED WITH THE "A" CONTAINMENT ISOLATION PANEL AI-43A WERE INADVERTENTLY ACTUATED AND WENT TO THE REQUIRED POSITIONS FOR A CIAS. IMMEDIATE INVESTIGATION OF THE SWITCH ALIGNMENT REVEALED THAT THE CIAS OVERRIDE SWITCH DID NOT HAVE COMPLETE CONTACT STACKUP WHEN A TRAINEE TURNED THE SWITCH TO THE "TEST" POSITION. THE ST WAS SUCCESSFULLY PERFORMED THE SECOND TIME WITH THE SWITCH PROPERLY POSITIONED. THE CONTAINMENT ISOLATION COMPONENTS THAT WERE ACTIVATED PERFORMED THEIR DESIGN FUNCTION AND THERE WAS NO SAFETY CONCERN. A FOUR-HOUR REPORT WAS SUBSEQUENTLY MADE TO THE NRC AT 1430 HOURS PURSUANT TO 10 CFR 50.72(B)(2)(II). CORRECTIVE ACTIONS INCLUDE PLACING A CAUTION TAG ON THE SWITCH TO ALERT THE OPERATORS OF THE POTENTIAL PROBLEM WITH OPERATING THE SWITCH, PLACING A CAUTION STATEMENT IN THE PROCEDURE REGARDING THE PROPER OPERATION OF THE SWITCH, AND INSPECTING THE SWITCH DURING THE NEXT REFUELING OUTAGE.

[50] FT. CALHOUN 1 DOCKET 50-285 LER 91-022
NUCLEAR INSTRUMENTATION CHANNELS B & D OUTSIDE DESIGN BASIS.
EVENT DATE: 100991 REPORT DATE: 110891 NSSS: CE TYPE: PWR

(NSIC 223354) OMAHA PUBLIC POWER DISTRICT (OPPD) INITIATED AN EVALUATION OF THE POTENTIAL FAILURE TO MEET THE REGULATORY GUIDE 1.97, REVISION 2 (RG 1.97) DESIGN BASIS REQUIREMENTS FOR NEUTRON FLUX INDICATION. THIS REVIEW WAS IN RESPONSE TO NEUTRON FLUX CHANNEL CONCERNS THAT OCCURRED DURING SHUTDOWN FOR THE FORT CALHOUN STATION (FCS) SEPTEMBER 12, 1991 BATTERY OUTAGE. OPPD CONCLUDED THAT THE NEUTRON FLUX MONITORING CHANNELS SELECTED TO MEET RG 1.97 (NE-002 AND NE-004) DO NOT MEET THE RG 1.97 SINGLE FAILURE CRITERIA. FAILURE OF DC BUS NO. 2 IN A DESIGN BASIS ACCIDENT (DBA)/POST DBA (I.E., LOSS OF COOLANT ACCIDENT OR MAIN STEAM LINE BREAK IN CONTAINMENT) COULD RESULT IN THE LOSS OF BOTH NE-002 AND NE-004. THIS CONDITION IS OUTSIDE THE FCS DESIGN BASIS AND WAS REPORTED TO THE NRC PURSUANT TO 10 CFR 50.72(B)(1)(II)(B) AT 1651 HOURS ON OCTOBER 9, 1991. THIS EVENT IS NOT SAFETY SIGNIFICANT AS REACTOR SAFE SHUTDOWN COULD BE ACHIEVED WITHOUT THIS WIDE RANGE NEUTRON FLUX INDICATION. THE ROOT CAUSE IS THE FAILURE TO FOLLOW PROCEDURE PED-QP-5. CONTRIBUTING CAUSES ARE LACK OF GUIDANCE AND FAILURE TO UNDERSTAND THE PROCEDURAL REQUIREMENTS WITH REGARD TO UPDATING ANALYSES/STUDIES. A NONCONFORMANCE REPORT WAS ISSUED TO EVALUATE AND CORRECT THE ENVIRONMENTAL QUALIFICATION PROBLEM WITH THE NEUTRON FLUX CHANNELS, AND TO TRACK FINAL RESOLUTION OF THIS CONDITION WHICH IS OUTSIDE THE DESIGN BASIS OF FCS.

[51] FT. CALHOUN 1 DOCKET 50-285 LER 91-023
FAILURE TO COMPLY WITH TECHNICAL SPECIFICATION 2.10.4(1)(B)(III).
EVENT DATE: 102291 REPORT DATE: 112191 NSSS: CE TYPE: PWR

(NSIC 223421) AT 1308 HOURS, ON OCTOBER 22, 1991, WHILE FORT CALHOUN STATION WAS OPERATING AT 99.3 PERCENT POWER (MODE 1) WITH ALL-RODS-OUT (ARO), THE EMERGENCY RESPONSE FACILITIES (ERF) COMPUTER SYSTEM WAS TAKEN OFFLINE FOR MAINTENANCE ACTIVITIES. HOWEVER, THIS WAS DONE WITHOUT MEETING THE REQUIREMENTS IN TECHNICAL SPECIFICATION (TS) 2.10.4(1)(B)(III) WHICH STATES THAT IF THE ERF IS BEING USED TO MONITOR LINEAR HEAT RATE AND BECOMES INOPERABLE, THEN POWER OPERATION MAY CONTINUE PROVIDED "POWER IS NOT INCREASED NOR HAS IT BEEN INCREASED SINCE THE TIME OF THE LAST INCORE POWER DISTRIBUTION." THE LAST INCORE POWER DISTRIBUTION HAD BEEN OBTAINED AT 0728 HOURS ON OCTOBER 22, 1991, AT 98.2 PERCENT POWER. POWER

THEN STEADILY INCREASED FROM 98 . PERCENT AT 0933 HOURS TO 99.3 PERCENT AT 1300 HOURS WHERE IT WAS STABILIZED IN THE ARO CONFIGURATION. PRIOR TO TAKING THE ERF OFFLINE, ANOTHER INCORE POWER DISTRIBUTION SHOULD HAVE BEEN OBTAINED IN ACCORDANCE WITH TS 2.10.4(1)(B)(III), SINCE POWER HAD INCREASED. THE ERF WAS RETURNED TO SERVICE AND THE POWER DISTRIBUTION WAS OBTAINED AT 1339 HOURS, INDICATING THAT ALL MONITORED PARAMETERS WERE WITHIN THEIR TS LIMITS. THIS EVENT IS BEING REPORTED PURSUANT TO 10 CFR 50.73(A)(2)(I). A HUMAN PERFORMANCE ENHANCEMENT SYSTEM (HPES) EVALUATION WAS PERFORMED AND THE CAUSE OF THIS EVENT WAS HUMAN ERROR DUE TO FAILURES IN COMMUNICATIONS AND WORK PRACTICES.

[52] GRAND GULF 1 DOCKET 50-416 LER 91-011
NONCOMPLIANCE WITH EOC/RPT SURVEILLANCE REQUIREMENT.
EVENT DATE: 100391 REPORT DATE: 110491 NSSS: GE TYPE: BWR

(NSIC 223380) WHILE SCHEDULING SURVEILLANCE ACTIVITIES FOR THE NEXT REFUELING OUTAGE, REVIEW OF PREVIOUSLY CONDUCTED SURVEILLANCES DISCOVERED THAT AN IMPROPER GROUP OF COMPONENTS HAD BEEN SURVEILLED IN 1987. THIS WAS CONTRARY TO THE SCHEDULE SPECIFIED BY TECHNICAL SPECIFICATION 4.3.4.2.3. TECHNICAL SPECIFICATION 4.3.4.2.3 PRESCRIBES THAT TESTING OF THESE REDUNDANT TRIP SYSTEMS BE PERFORMED IN A STAGGERED SCHEDULE SUCH THAT ALL CHANNELS ARE TESTED AT LEAST ONCE PER 36 MONTHS. THE SCHEDULE DEVELOPED IN 1986 DESIGNED THE IMPROPER GROUP OF CHANNELS TO BE SURVEILLED. THIS SCHEDULING DID NOT CONSIDER THE STAGGERED SCHEDULE OF THE SURVEILLANCE REQUIREMENT. CLEARER GUIDANCE WILL BE DEVELOPED TO ENSURE PROPER SCHEDULING OF SURVEILLANCES. PERFORMANCE OF THE SURVEILLANCE WAS CONDUCTED SUCCESSFULLY ON OCTOBER 5, 1991 ON THE AFFECTED TRIP SYSTEM CHANNELS TO COMPLY WITH TECHNICAL SPECIFICATION 4.3.4.2.3 REQUIREMENTS. THE ERRONEOUS SURVEILLANCE TESTING SITUATION DID NOT INCREASE THE RISK OF SAFETY FUNCTION FAILURE. THE SAFETY AND HEALTH OF THE GENERAL PUBLIC WAS NOT COMPROMISED BY THIS EVENT.

[53] HATCH 1 DOCKET 50-321 LER 91-020
AREA RADIATION MONITORS TRIP RESULTING IN ENGINEERED SAFETY FEATURE AUTOMATIC ACTUATION.
EVENT DATE: 100491 REPORT DATE: 110191 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 2 (BWR)

(NSIC 223323) ON 10/4/91 AT 2342 CDT, UNIT 1 WAS IN A REFUELING OUTAGE WITH THE MODE SWITCH IN REFUEL AND NO FUEL IN THE REACTOR VESSEL. UNIT 2 WAS IN THE RUN MODE AT 2436 CMWT (100 PERCENT RATED THERMAL POWER). AT THAT TIME VARIOUS AREA HIGH RADIATION ANNUNCIATORS ALARMED IN THE MAIN CONTROL ROOM AND THE UNIT-COMMON MAIN CONTROL ROOM ENVIRONMENTAL CONTROL SYSTEM (MCRECS) AUTOMATICALLY TRANSFERRED TO THE PRESSURIZATION MODE. ALL 38 AREA RADIATION MONITOR (ARM) TRIP UNITS, LOCATED ON MAIN CONTROL ROOM PANEL 1D21-P600, HAD TRIPPED; HOWEVER, THE TRIP UNIT INDICATORS WERE READING WITHIN THEIR NORMAL RANGES. TWO OF THE ARMS LOCATED ON THE REFUELING FLOOR PROVIDE INITIATION SIGNALS FOR AUTOMATIC ACTUATION OF THE MCRECS PRESSURIZATION MODE WHEN THEIR DESIGN SETPOINTS ARE EXCEEDED. SINCE THE ARM TRIP UNIT INDICATORS SHOWED THAT RADIATION LEVELS WERE NORMAL, THE TRIP UNITS WERE RESET. MCRECS WAS THEN RETURNED TO ITS NORMAL MODE OF OPERATION AT APPROXIMATELY 2355 CDT. THE MOST PROBABLE CAUSE OF THIS EVENT WAS CONCLUDED TO BE A VOLTAGE PERTURBATION IN THE COMMON POWER SUPPLY OF THE ARMS. HOWEVER, A REVIEW OF OUTAGE ACTIVITIES IN PROGRESS AT THE TIME OF THE EVENT DID NOT REVEAL ANY ACTIVITIES THAT WOULD HAVE RESULTED IN SUCH A PERTURBATION.

[54] HATCH 1 DOCKET 50-321 LER 91-022
SAFETY RELIEF VALVE SETPOINTS REMAIN WITHIN IN-SERVICE TESTING TOLERANCE REQUIREMENTS.
EVENT DATE: 100991 REPORT DATE: 110691 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 223364) ON 10/9/91, AT 1430 CDT, UNIT 1 WAS IN ITS THIRTEENTH REFUELING OUTAGE WITH THE MODE SWITCH IN REFUEL AND FUEL REMOVED FROM THE REACTOR VESSEL. ALL ELEVEN MAIN STEAM SYSTEM (EIIS CODE SB) SAFETY RELIEF VALVES (SRVS) HAD BEEN SENT OFFSITE FOR TESTING OF THEIR MECHANICAL LIFT SETPOINTS AS REQUIRED BY AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) BOILER AND PRESSURE VESSEL CODE,

SECTION XI, IWV-3512. AT THAT TIME, ENGINEERING PERSONNEL WERE NOTIFIED BY THE OFFSITE TESTING AGENCY THAT TESTS SHOWED THE MECHANICAL LIFT SETPOINTS FOR ALL ELEVEN SRVS WERE WITH IN THE SECTION XI TOLERANCE REQUIREMENT OF +/- 3 PERCENT. THIS REPORT IS BEING SUBMITTED VOLUNTARILY DUE TO THE POTENTIAL INDUSTRY INTEREST IN VIEW OF THE ONGOING BOILING WATER REACTOR OWNERS GROUP (BWROG) ACTIVITIES TO ADDRESS SRV MECHANICAL LIFT SETPOINT DRIFT. THE CAUSE OF THE SETPOINT DRIFT IS CORROSION-INDUCED BONDING OF THE SRV PILOT VALVE DISC AND SEAT. IN THIS EVENT, THE SRV'S ALL LIFTED WITHIN THEIR SETPOINT TOLERANCE OF +/- 3 PERCENT; HOWEVER, THIS REPORT IS BEING SUBMITTED TO APPRISE THE INDUSTRY OF THE RESULTS.

[55] HATCH 1 DOCKET 50-321 LER 91-023
BLOWN FUSE RESULTS IN ENGINEERED SAFETY FEATURE ACTUATION.
EVENT DATE: 101491 REPORT DATE: 111291 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 2 (BWR)

(NSIC 223446) ON 10/14/91, AT 1135 CDT, UNIT 1 WAS IN THE REFUEL MODE WITH THE REACTOR VESSEL DISASSEMBLED. NO FUEL WAS IN THE VESSEL, AND THE REACTOR CAVITY WAS FLOODED. UNIT 2 WAS IN THE RUN MODE AT A POWER LEVEL OF 2436 CMWT (100% RATED THERMAL POWER). AT THAT TIME, CONTROL ROOM OPERATORS OBSERVED THAT THE "B" TRAIN OF EACH UNIT'S STANDBY GAS TREATMENT SYSTEM (SGTS, EIIS CODE BH) HAD STARTED. LICENSED OPERATIONS PERSONNEL THEN VERIFIED PROPER OPERATION OF THE SGTS AND THAT AN ACTUAL CONDITION REQUIRING SGTS OPERATION DID NOT EXIST. SINCE CONDITIONS THAT WOULD HAVE INDICATED THE START WAS VALID DID NOT OCCUR, OPERATORS ATTEMPTED TO RESET THE SYSTEM AND RESTORE NORMAL VENTILATION. HOWEVER, THE SGTSS COULD NOT BE TRIPPED. ELECTRICIANS TROUBLESHOOTING THE SYSTEMS DISCOVERED FUSE 1C61-F20 HAD BLOWN. THE BLOWN FUSE WAS REPLACED, AND THE SGTSS WERE RETURNED TO SERVICE BY 1432 CDT. THE CAUSE OF THIS EVENT WAS A BLOWN FUSE. THE FUSE APPLICATION WAS REVIEWED AND DETERMINED TO BE CORRECT. THE BLOWN FUSE OCCURRENCE IS BELIEVED TO BE RANDOM. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDE: 1) REPLACING THE BLOWN FUSE, 2) SECURING SGTS, 3) RETURNING NORMAL VENTILATION SYSTEMS TO SERVICE, REVIEWING THE FUSE APPLICATION, AND 4) TESTING THE EFFICIENCY OF THE FILTER TRAIN CARBON ADSORBERS.

[56] HATCH 1 DOCKET 50-321 LER 91-024
TECHNICAL SPECIFICATIONS SURVEILLANCE NOT PERFORMED ON REQUIRED PERIODICITY.
EVENT DATE: 101691 REPORT DATE: 111291 NSSS: GE TYPE: BWR

(NSIC 223434) ON 10/16/91 AT 1130 CDT, UNIT 1 WAS IN A REFUELING OUTAGE WITH THE CORE COMPLETELY UNLOADED. AT THAT TIME, IT WAS DETERMINED THAT THE REQUIREMENTS OF UNIT 1 TECHNICAL SPECIFICATIONS SECTION 4.7.A.5 HAD NOT BEEN MET. SPECIFICALLY, PROCEDURE 34SV-SW-019-1S, "SURVEILLANCE CHECKS," REQUIRED THE UNIT 1 TORUS OXYGEN CONCENTRATION TO BE VERIFIED TO BE LESS THAN 4% ONCE PER WEEK INSTEAD OF ONCE PER DAY AS REQUIRED BY THE TECHNICAL SPECIFICATIONS. THE PROCEDURE DID REQUIRE THE UNIT 1 DRYWELL OXYGEN CONCENTRATION TO BE VERIFIED TO BE LESS THAN 4% ONCE PER DAY AS REQUIRED. WITH THE UNIT IN A REFUELING OUTAGE AND THE CORE UNLOADED, UNIT 1 TECHNICAL SPECIFICATIONS SECTION 4.7.A.5 WAS NOT APPLICABLE BECAUSE THE OXYGEN CHECK REQUIREMENT APPLIES ONLY WHEN THE UNIT IS IN POWER OPERATION. THEREFORE, NO LIMITING CONDITIONS FOR OPERATION EXISTED AT THE TIME OF DISCOVERY OF THIS MISSED SURVEILLANCE. THE CAUSE OF THIS EVENT WAS A MISINTERPRETATION OF THE REQUIREMENTS OF SECTION 4.7.A.5. THIS SPECIFICATION REQUIRES THAT "THE PRIMARY CONTAINMENT OXYGEN CONCENTRATION" BE MEASURED DAILY. IT APPEARS THAT PRIMARY CONTAINMENT HAS BEEN INTERPRETED TO MEAN THE DRYWELL. IN FACT, THE TORUS IS ALSO PART OF THE PRIMARY CONTAINMENT AND, BECAUSE THE TORUS ATMOSPHERE IS NOT IN CONSTANT COMMUNICATION WITH THE DRYWELL ATMOSPHERE, ITS OXYGEN CONCENTRATION MUST BE CHECKED AS WELL. CORRECTIVE ACTION FOR THIS EVENT INCLUDED REVISING THE PROCEDURE.

[57] HOPE CREEK 1 DOCKET 50-354 LER 90-022 REV 01
UPDATE ON ENGINEERED SAFETY FEATURES ACTUATION: AUTO START OF "A" CORE SPRAY PUMP DURING SURVEILLANCE TEST DUE TO PERSONNEL ERROR.
EVENT DATE: 102290 REPORT DATE: 110491 NSSS: GE TYPE: BWR

(NSIC 223342) ON 10/22/90 AT 1405, DURING THE COURSE OF A MAINTENANCE DEPARTMENT

CONTROLS SURVEILLANCE TEST, AN INADVERTENT AUTO-START OF THE "A" CORE SPRAY PUMP OCCURRED. AFTER VERIFYING THE INITIATING CAUSE OF THE AUTO-START, CONTROL ROOM PERSONNEL STOPPED THE PUMP AND RETURNED THE CORE SPRAY SYSTEM TO A NORMAL (STANDBY) CONFIGURATION. THE CORE SPRAY SYSTEM DID NOT INJECT TO THE REACTOR VESSEL. INVESTIGATION SUBSEQUENT TO THE EVENT DETERMINED THAT THE ROOT CAUSE OF THIS OCCURRENCE TO BE PERSONNEL ERROR, COMPOUNDED BY HUMAN FACTORS CONCERNS IN THE RELAY CABINET IN WHICH THE SUBJECT SURVEILLANCE TOOK PLACE. THE PROCEDURALLY REQUIRED SEQUENCE OF INDEPENDENT VERIFICATION ALSO CONTRIBUTED TO THE ERROR. DURING PREPARATION FOR THE SURVEILLANCE, A CONTROLS TECHNICIAN CONNECTED A TEST SWITCH LEAD TO THE WRONG TERMINAL ON A RELAY BEING TESTED. THIS RESULTED IN THE RELAY BECOMING ENERGIZED WHEN THE TEST SWITCH WAS CLOSED LATER IN THE TESTING PROCESS. CORRECTIVE ACTIONS INCLUDE COUNSELLING FOR THE TECHNICIAN, RE-PRIORITIZING A PREVIOUSLY IDENTIFIED DESIGN CHANGE TO THE SUBJECT RELAY CABINET, ENHANCING THE INDEPENDENT VERIFICATION PROCESS, AND REVIEWING THIS INCIDENT DURING THE COURSE OF CONTROLS TECHNICIAN CONTINUING TRAINING.

[58] INDIAN POINT 2 DOCKET 50-247 LER 91-018
CONTAINMENT RADIOGAS MONITOR SPURIOUS ELECTRICAL SPIKE CAUSES ENGINEERED SAFETY FEATURE ACTUATION.
EVENT DATE: 083191 REPORT DATE: 093091 NSSS: WE TYPE: PWR

(NSIC 223081) DURING THE PERFORMANCE OF A PRESSURE RELIEF OF CONTAINMENT ON AUGUST 31, 1991, WITH THE PLANT AT 100% POWER, THE CONTAINMENT RADIOGAS MONITOR (R-42) EXPERIENCED A SPURIOUS ELECTRICAL SPIKE, WHICH IN TURN INITIATED CONTAINMENT VENTILATION ISOLATION AND PARTIALLY ACTUATED THE WELD CHANNEL AND CONTAINMENT PENETRATION PRESSURIZATION (WCCPP) SYSTEM. THE WCCPP SYSTEM IS CLASSIFIED AS AN ENGINEERED SAFETY FEATURE AND IT FUNCTIONED AS DESIGNED. AFTER DETERMINING THERE HAD BEEN NO ACTUAL INCREASE IN GASEOUS ACTIVITY, RADIATION MONITOR R-42 WAS RESET AND PRESSURE RELIEF WAS REINSTITUTED. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED BY THIS EVENT.

[59] INDIAN POINT 2 DOCKET 50-247 LER 91-019
CONTAINMENT SPRAY PUMP AND NAOH SPRAY EDUCATOR INOPERABLE.
EVENT DATE: 091191 REPORT DATE: 101191 NSSS: WE TYPE: PWR
VENDOR: ITT GRINNELL VALVE CO INC

(NSIC 223284) ON SEPTEMBER 11, 1991, AND AGAIN ON SEPTEMBER 12, DURING THE CONDUCT OF SURVEILLANCE TESTING OF A CONTAINMENT SPRAY PUMP, THE PLANT ENTERED TECHNICAL SPECIFICATION 3.0.1 WHICH REQUIRES THE UNIT TO BE SHUTDOWN WITHIN 7 HOURS UNLESS CORRECTIVE MEASURES ARE COMPLETED THAT RESTORE COMPLIANCE TO THE LIMITING CONDITION FOR OPERATION (LCO). IN EACH CASE ONE CONTAINMENT SPRAY PUMP AND ITS ASSOCIATED SPRAY ADDITIVE EDUCATOR FLOW PATH WERE SIMULTANEOUSLY DECLARED INOPERABLE AND IN EACH CASE THE PROVISION OF TECHNICAL SPECIFICATION 3.0.1 WAS EXITED WITHIN 20 MINUTES.

[60] INDIAN POINT 2 DOCKET 50-247 LER 91-020
DETERMINATION OF 480V SWITCHGEAR ROOM TEMPERATURE.
EVENT DATE: 092791 REPORT DATE: 102891 NSSS: WE TYPE: PWR

(NSIC 223338) BASED ON AN ENGINEERING EVALUATION, IT WAS DETERMINED ON SEPTEMBER 27, 1991 THAT THE 480 VOLT SWITCHGEAR ROOM TEMPERATURE, UNDER CERTAIN CONDITIONS, COULD EXCEED THE MAXIMUM TEMPERATURE RECOMMENDED BY THE SAFEGUARDS EQUIPMENT SWITCHGEAR MANUFACTURERS. SPECIFICALLY, DURING A LOSS OF COOLANT ACCIDENT WITH OFFSITE POWER AVAILABLE, A POSTULATED SINGLE FAILURE COULD RENDER THE ROOM'S VENTILATION SYSTEM UNAVAILABLE, RESULTING IN A CALCULATED MAXIMUM AMBIENT TEMPERATURE OF 135F AFTER SEVERAL HOURS FOR THE WORST SUMMER DESIGN OUTSIDE TEMPERATURE OF 95F. IMMEDIATE CORRECTIVE ACTIONS CONSISTED OF PROCEDURAL CHANGES TO VERIFY OPERATION OF THE VENTILATION SYSTEM DURING PERIODIC OPERATOR ROUNDS AND, IN THE EVENT OF AN ACCIDENT, TO OPEN DOORS TO ESTABLISH NATURAL CIRCULATION (STACK EFFECT).

[61] INDIAN POINT 2 DOCKET 50-247 LER 91-021
INADVERTENT ACTUATION OF HYDROGEN CYANIDE TOXIC GAS MONITOR.
EVENT DATE: 100491 REPORT DATE: 110491 NSSS: WE TYPE: PWR
VENDOR: WISCONSIN BRIDGE & IRON

(NSIC 223348) ON OCTOBER 4, 1991, AT APPROXIMATELY 2340 HOURS, WITH REACTOR POWER AT 99.5%, THE HYDROGEN CYANIDE (HCN) TOXIC GAS MONITOR CHANNEL 2 ALARMED INADVERTENTLY, RESULTING IN THE TRANSFER OF THE CENTRAL CONTROL ROOM (CCR) VENTILATION SYSTEM FROM THE NORMAL MODE TO THE INCIDENT MODE. SUBSEQUENT TOXIC GAS MONITOR CHANNEL ACTUATIONS OCCURRED ON OCTOBER 11, 12 AND 17. AS DESIGNED, THE DETECTION OF THE RESPECTIVE GAS BY EITHER CHANNELS 1 OR 2 OF THE TOXIC GAS MONITORS WILL GENERATE AN ALARM IN THE CCR AND ISOLATE THE CCR VENTILATION SYSTEM. THE TOXIC GAS MONITORING SYSTEM IS CLASSIFIED AS AN ENGINEERED SAFETY FEATURE (ESF). NO TECHNICAL SPECIFICATION OR NRC LIMITS WERE EXCEEDED.

[62] KEWAUNEE DOCKET 50-305 LER 89-012 REV 01
UPDATE ON FAILURE TO SATISFY SAFEGUARD CABLE SEPARATION REQUIREMENTS RESULT IN THE PLANT BEING IN A CONDITION OUTSIDE OF THE DESIGN BASIS.
EVENT DATE: 062289 REPORT DATE: 110591 NSSS: WE TYPE: PWR

(NSIC 223341) ON 6/22/89, AT 1600, WHILE THE PLANT WAS OPERATING AT 10% POWER, A DESIGN DEFICIENCY WAS IDENTIFIED THAT CAUSED BOTH MOTOR DRIVEN AUXILIARY FEEDWATER (MDAFW) PUMPS, ALTHOUGH REMAINING FUNCTIONAL, TO BE DECLARED INOPERABLE. THE LO-LO STEAM GENERATOR SIGNAL FOR BOTH MDAFW PUMPS START CIRCUITS WERE FOUND TO USE CONTROL CABLES THAT WERE ROUTED IN DIFFERENT SECTIONS OF THE SAME CABLE TRAY RACEWAY, THEREFORE NOT MEETING SEPARATION CRITERIA. THEY ALSO WERE DESIGNATED AS NORMAL CABLE AND ROUTED THROUGH NORMAL CABLE TRAYS. THIS CONDITION WAS NOT CONSISTENT WITH THE REQUIREMENTS SPECIFIED IN THE PLANT'S UPDATED SAFETY ANALYSIS REPORT (USAR). PREVIOUSLY, ON JUNE 14, 1989, AT 1400, WHILE THE PLANT WAS AT 100% POWER, A DESIGN DEFICIENCY WAS IDENTIFIED IN THE TURBINE DRIVEN AUXILIARY FEEDWATER (TDAFW) PUMP START CIRCUITRY. THE DC MOTOR OPERATED VALVE WHICH ADMITS STEAM TO THE TDAFW PUMP WAS IDENTIFIED AS CONTAINING A NORMAL CABLE IN THE SAFEGUARD CONTROL CIRCUITRY. ON 6/17/89, THE PLANT WAS SHUTDOWN TO REPAIR A REACTOR COOLANT PUMP SEAL. AT THIS TIME, THE CIRCUIT WAS MODIFIED TO USE A SAFEGUARD CABLE AS SCHEDULED. THE ROOT CAUSE OF THE EVENTS HAS BEEN DETERMINED TO BE A DESIGN DEFICIENCY AS THE ORIGINAL CABLES WERE DESIGNATED NORMAL. IF THE CABLES HAD BEEN PROPERLY DESIGNATED AS SAFEGUARD, THEY WOULD HAVE BEEN INSTALLED IN SAFEGUARD TRAYS.

[63] KEWAUNEE DOCKET 50-305 LER 91-010
UNIT TRIP DUE TO INADEQUATE NON-SAFEGUARD BREAKER COORDINATION.
EVENT DATE: 101291 REPORT DATE: 111191 NSSS: WE TYPE: PWR

(NSIC 223428) AT 1400 CDT ON 10/12/91, WITH THE PLANT AT FULL POWER, A REACTOR/TURBINE TRIP OCCURRED WHILE RETURNING EQUIPMENT TO SERVICE DURING THE IMPLEMENTATION OF PREVENTIVE MAINTENANCE PROCEDURE (PMP) 38-10, "DC SUPPLY AND DISTRIBUTION (EDC) 20 KVA INVERTER MAINTENANCE." DURING THIS PROCESS 125 VDC DISTRIBUTION CABINET BRD-103 WAS DE-ENERGIZED AS A RESULT OF INADEQUATE NON-SAFEGUARD BREAKER COORDINATION. THIS DE-ENERGIZATION CAUSED THE FEEDWATER REGULATING VALVES (FW-7A AND B) TO CLOSE, RESULTING IN A TRIP SIGNAL FROM THE SUBSEQUENT STEAM FLOW-FEEDWATER FLOW MISMATCH COINCIDENT WITH LOW WATER LEVEL IN STEAM GENERATOR A. AS EXPECTED, THE AUXILIARY FEEDWATER PUMPS STARTED, WHICH WAS AN ESF ACTUATION, IN RESPONSE TO PLANT CONDITIONS. A MAIN CONTRIBUTOR TO THIS EVENT WAS INADEQUATE COORDINATION BETWEEN THE 300 AMP BREAKER WHICH SUPPLIES 125 VDC DISTRIBUTION CABINET BRD-103 AND THE 200 AMP BREAKER FROM BRD-103 TO A NON-SAFEGUARD INVERTER. TO PREVENT RECURRENCE THE MAGNETIC TRIP SETTINGS OF THE RESPECTIVE BREAKERS WERE ADJUSTED ACCORDINGLY. IN ADDITION, ANY MODIFICATIONS NECESSARY TO FURTHER ENSURE PROPER COORDINATION OF NONSAFEGUARD DC BREAKERS WILL BE SCHEDULED FOR THE 1992 REFUELING OUTAGE. THE PLANT SAFETY SYSTEMS RESPONDED AS DESIGNED AND THE OPERATORS FOLLOWED APPROPRIATE RECOVERY PROCEDURES FOR PLANT STABILIZATION. A TECH SPEC REQUIRED INSTRUMENT CHANNEL CHECK WAS NOT PERFORMED DURING RECOVERY ACTIVITIES.

[64] LA SALLE 1 DOCKET 50-373 LER 91-011
MISSED TECHNICAL SPECIFICATION SURVEILLANCE DURING PRIMARY CONTAINMENT PURGE DUE TO MISSED PREREQUISITE IN PROCEDURE.
EVENT DATE: 100291 REPORT DATE: 110191 NSSS: GE TYPE: BWR

(NSIC 223310) ON OCTOBER 2, 1991 AT 0820 HOURS WITH UNIT 1 IN OPERATIONAL CONDITION 1 (RUN) AT 100% POWER, A PURGE OF THE UNIT 1 DRYWELL WAS INITIATED IN ACCORDANCE WITH LASALLE OPERATING PROCEDURE LOP-VQ-11, "NITROGEN INERTING OF PRIMARY CONTAINMENT WITH THE PRIMARY CONTAINMENT VENT PURGE SYSTEM", WITHOUT THE SAMPLES FOR PRINCIPAL GAMMA EMITTERS AND TRITIUM AS REQUIRED BY THE PROCEDURE AND TECHNICAL SPECIFICATION TABLE 4.11.2-1 BEING TAKEN. THERE WERE NO SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT. AT THE TIME OF THE EVENT THE PRIMARY CONTAINMENT CONTINUOUS AIR MONITOR SYSTEM SHOWED NORMAL LEVELS OF NOBLE GAS AND PARTICULATE ACTIVITY, THE DRYWELL FLOOR DRAIN AND EQUIPMENT DRAIN SUMP FLOWRATES WERE NORMAL WITH NO INCREASING TRENDS. A UNIT ONE PRIMARY CONTAINMENT SAMPLE FOR NOBLE GASES WAS ALREADY IN PROGRESS AT THE TIME OF THE EVENT. NO GASEOUS EFFLUENT RELEASE LIMITS WERE EXCEEDED DURING THE EVENT. THIS EVENT IS BEING REPORTED PURSUANT TO 10CFR50.73(A)(2)(I)(B) ANY OPERATION OR CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATION.

[65] LA SALLE 2 DOCKET 50-374 LER 91-013
LOSS OF AUXILIARY ELECTRIC EQUIPMENT ROOM VENTILATION SUPPLY FAN DUE TO OVERHEATING OF STARTING COIL FOR THE BREAKER.
EVENT DATE: 100791 REPORT DATE: 110191 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 1 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223309) ON OCTOBER 7, 1991, AT 0400 HOURS WITH UNIT 1 AND 2 IN OPERATIONAL CONDITION 1 (RUN) AT 95% AND 85% POWER RESPECTIVELY, A SEVEN DAY TIMECLOCK WAS ENTERED PER TECHNICAL SPECIFICATION 3.7.2 DUE TO THE VE(VI) SUPPLY BREAKER TRIPPING. AT THIS TIME THE "A" VC/VE(VI) TRAIN WAS OUT OF SERVICE FOR SCHEDULED MAINTENANCE AND A ONE HOUR TIMECLOCK WAS ENTERED AT THIS TIME PER TECHNICAL SPECIFICATION 3.0.3, SINCE BOTH VC/VE(VI) EMERGENCY MAKE-UP TRAINS WERE INOPERABLE AT THE SAME TIME. THE "A" TRAIN WAS RETURNED TO SERVICE AT 0450 HOURS ON OCTOBER 7, 1991, AND THE ONE HOUR TIMECLOCK WAS EXITED. NUCLEAR REGULATORY COMMISSION (NRC) NOTIFICATION WAS MADE AT 0755 HOURS ON OCTOBER 7, 1991, ON THE BASIS OF A LOSS OF A SAFETY SYSTEM FUNCTION, SINCE BOTH AUXILIARY ELECTRIC ROOM VENTILATION SYSTEMS WERE UNAVAILABLE FOR SERVICE. WORK REQUEST L10621 WAS INITIATED AT THIS TIME FOR THE ELECTRICAL MAINTENANCE DEPARTMENT TO INVESTIGATE AND CORRECT THE PROBLEM. THE 52X RELAY, THE STARTING COIL, WAS FOUND TO BE THE CAUSE OF THE PROBLEM. THIS RELAY WAS REPLACED AND ON OCTOBER 7, 1991, AT 2030 HOURS THE VE(VI) SUPPLY FAN, OVE01CB, WAS SUCCESSFULLY RUN. ON OCTOBER 7, 1991, AT 2120 HOURS THE TIMECLOCK WAS EXITED. THIS EVENT IS REPORTABLE PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(V)(C) DUE TO THE LOSS OF A SAFETY SYSTEM FUNCTION.

[66] LIMERICK 1 DOCKET 50-352 LER 91-023
ENGINEERED SAFETY FEATURE ACTUATIONS DUE TO A LOSS OF POWER TO AN RPS/UPS POWER DISTRIBUTION PANEL CAUSED BY THE INADVERTENT ACTUATION OF AN UNDERFREQUENCY RELAY.
EVENT DATE: 100791 REPORT DATE: 110491 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 223371) ON OCTOBER 7, 1991, AN INADVERTENT ACTUATION OF AN UNDERFREQUENCY RELAY RESULTED IN A TRIP OF THE '1A' REACTOR PROTECTION SYSTEM (RPS) INVERTER OUTPUT CIRCUIT BREAKER WHICH CAUSED A LOSS OF POWER TO THE '1A' RPS/UNINTERRUPTIBLE POWER SUPPLY POWER DISTRIBUTION PANEL. THIS LOSS OF POWER RESULTED IN AUTOMATIC ACTUATIONS OF THE PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM, AND OTHER VARIOUS ENGINEERED SAFETY FEATURES. OPERATIONS PERSONNEL RESET THE UNDERFREQUENCY RELAY AND RECLOSED THE BREAKER THEREBY RESTORING POWER TO PANEL 1AY160. ALL REMAINING ISOLATIONS WERE RESET 36 MINUTES AFTER THE TRIP. THE ISOLATIONS WERE BYPASSED OR RESET IN ACCORDANCE WITH PLANT PROCEDURES AND THE SYSTEMS WERE RESTORED EXPEDITIOUSLY BY OPERATORS, PREVENTING ANY ADVERSE IMPACTS ON PLANT SYSTEMS. THE CAUSE OF THIS EVENT WAS THE INADVERTENT ACTUATION OF AN UNDERFREQUENCY RELAY DURING PERFORMANCE OF A

SURVEILLANCE TEST (ST) PROCEDURE. A NEW TEST METHODOLOGY IS BEING DEVELOPED FOR THE PERFORMANCE OF THE ST PROCEDURE, TO MINIMIZE THE POSSIBILITY OF ACTUATION OF THE UNDERFREQUENCY RELAY. ADDITIONALLY, AN EVALUATION IS BEING PERFORMED BY THE OFF-SITE ENGINEERING ORGANIZATION TO INVESTIGATE POSSIBLE REVISIONS TO THE BREAKER TRIP LOGIC TO INCREASE THE RELIALILITY OF THIS POWER SUPPLY.

[67] LIMERICK 2 DOCKET 50-353 LER 91-016
HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION CAUSED BY AN INSTRUMENTATION AND CONTROL TECHNICAN SELECTING AND TESTING THE INCORRECT TRIP UNIT.
EVENT DATE: 100791 REPORT DATE: 110491 NSSS: GE TYPE: BWR

(NSIC 223372) ON OCTOBER 7, 1991, DURING PERFORMANCE OF A SURVEILLANCE TEST (ST) PROCEDURE, AN INSTRUMENTATION AND CONTROLS (I&C) TECHNICIAN WAS SWITCHING ELECTRONICALLY BETWEEN TRIP UNITS AND INADVERTENTLY PLACED THE SELECTOR SWITCH TO THE NEXT NUMBERED TRIP UNIT. THE TECHNICIAN THEN DEPRESSED THE SWITCH TO BEGIN THE FUNCTIONAL TEST WHICH RESULTED IN A PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM ACTUATION, AN ENGINEERED SAFETY FEATURE ACTUATION, WHICH CLOSED THE UNIT 2 HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM STEAM SUPPLY LINE OUTBOARD ISOLATION VALVE. THIS WAS AN EVENT WHICH ALONE COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF A SYSTEM. THE HPCI SYSTEM WAS RESTORED TO NORMAL SYSTEM ALIGNMENT WITHIN 21 MINUTES. THE ACTUAL CONSEQUENCES OF THIS EVENT WERE MINIMAL IN THAT AN ACCIDENT CONDITION DID NOT OCCUR WHILE THE HPCI SYSTEM WAS ISOLATED, AND THE SYSTEM WAS NOT CALLED UPON TO PERFORM ITS INTENDED SAFETY FUNCTION. THE CAUSE OF THIS EVENT WAS A PERSONNEL ERROR BY THE I&C TECHNICIAN PERFORMING THE ST PROCEDURE IN THAT INSUFFICIENT ATTENTION TO DETAIL WAS APPLIED TO THE PERFORMANCE OF THE ST PROCEDURE. THE TECHNICIAN INVOLVED WAS ADMONISHED BY THE MAINTENANCE AND I&C SUPERINTENDENT FOR FAILURE TO APPLY THE PRINCIPLES OF SELF-CHECK AND ATTENTION TO DETAIL WHILE PERFORMING THE ST PROCEDURE.

[68] MAINE YANKEE DOCKET 50-309 LER 91-007 REV 01
UPDATE ON COMPONENT COOLING VALVES NOT LOCKED OPEN.
EVENT DATE: 062591 REPORT DATE: 110491 NSSS: CE TYPE: PWR

(NSIC 223362) DURING NORMAL OPERATION AT 100% POWER, ON JUNE 25, 1991, IT WAS DETERMINED THAT THE OPEN OUTLET VALVES FROM THE COMPONENT COOLING SURGE TANKS WERE NOT LOCKED AND THEREFORE NOT IN ACCORDANCE WITH PLANT TECHNICAL SPECIFICATION 3.6. THIS TECHNICAL SPECIFICATION REQUIRES EMERGENCY CORE COOLING SYSTEM MANUAL VALVES BE ALIGNED AND LOCKED IN THE POSITION REQUIRED FOR SAFEGUARDS OPERATION. THE VALVES WERE SUBSEQUENTLY LOCKED, CONTROLLING PROCEDURES WERE REVISED, AND A REVIEW OF THE COMPONENT COOLING SYSTEM REVEALED THAT NO ADDITIONAL UNLOCKED MANUAL VALVES WERE REQUIRED TO BE LOCKED. A DETAILED ROOT CAUSE EVALUATION WAS COMPLETED AND NO CHANGE TO THIS LER IS REQUIRED.

[69] MAINE YANKEE DOCKET 50-309 LER 91-009
INOPERABLE MAIN FEEDWATER REGULATING VALVE FEEDWATER TRAIN TRIP SOLENOID.
EVENT DATE: 092791 REPORT DATE: 102891 NSSS: CE TYPE: PWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)

(NSIC 223327) ON SEPTEMBER 27, 1991 AT 1030, A PLANT SHUTDOWN WAS INITIATED AS REQUIRED BY TECHNICAL SPECIFICATIONS WHEN BOTH TRAINS OF THE FEEDWATER TRAIN TRIP (FTT) SYSTEM WAS DECLARED INOPERABLE. MAINE YANKEE DETERMINED THAT THE MAIN FEEDWATER REGULATING VALVE (MFRV) FIT SOLENIOD VALVES WERE RECEIVING INSUFFICIENT AIR TO ACTUATE UNDER ALL PLANT CONDITIONS. THE POWER REDUCTION WAS STABILIZED AT 15.5% POWER WHICH ALLOWED ISOLATION OF THE MFRVS. THE SOLENOIDS WERE RETURNED TO OPERABLE STATUS BY INCREASING VALVE POSITIONING AIR PRESSURE. THE MFRV FTT SOLENOIDS WERE DECLARED OPERABLE ON SEPTEMBER 30, 1991 AT 0542.

[70] MAINE YANKEE DOCKET 50-309 LER 91-010
PLANT TRIP ON CONDENSATE PUMP MOTOR FAILURE.
EVENT DATE: 100591 REPORT DATE: 110491 NSSS: CE TYPE: PWR
VENDOR: ALLIS CHALMERS
RAYCHEM CORP.

(NSIC 223363) AT 0237 ON OCTOBER 5, 1991, AN AUTOMATIC REACTOR TRIP FROM 100% POWER OCCURRED DUE TO ACTUATION OF THE REACTOR PROTECTIVE SYSTEM FROM A LOSS OF LOAD TRIP WHEN THE MAIN TURBINE TRIPPED. THE TURBINE TRIP OCCURRED WHEN THE STEAM DRIVEN FEEDWATER PUMP (P-2C) TRIPPED AFTER A CONDENSATE PUMP (P-27C) BREAKER TRIPPED DUE TO A PHASE TO PHASE SHORT AT THE MOTOR LEADS CONNECTION. THE GENERATOR PRIMARY RELAY (86P) ACTUATION WAS DELAYED DUE TO THE SLOW CLOSURE OF A MAIN TURBINE STOP VALVE; THE BACKUP RELAY (86BU) ACTUATED AND TRANSFERRED STATION SERVICE TO OFFSITE POWER AS DESIGNED. ALL SAFETY SYSTEMS PERFORMED AS EXPECTED.

[71] MONTICELLO DOCKET 50-263 LER 91-021
INOPERABLE FIRE PENETRATION DUE TO PIPE MOVEMENT RESULTING FROM POSSIBLE WATER HAMMER.
EVENT DATE: 093091 REPORT DATE: 103091 NSSS: GE TYPE: BWR

(NSIC 223337) WHILE CONDUCTING A RADIATION PROTECTION SURVEY, HEALTH PHYSICS PERSONNEL DISCOVERED A PIPE PENETRATION WHICH DID NOT APPEAR TO BE PROPERLY SEALED. THE FIRE PROTECTION ENGINEER WAS NOTIFIED. THE ENGINEER DETERMINED THAT THE PENETRATION WAS IN A FIRE BARRIER AND WAS NOT PROPERLY SEALED. THE PENETRATION WAS DECLARED INOPERABLE AND A ONE HOUR ROVING FIRE WATCH WAS ESTABLISHED. THE PENETRATION WAS INSPECTED, SEALED, AND RETURNED TO OPERABLE STATUS. THE POSTULATED CAUSE OF THE EVENT IS PIPE MOVEMENT DUE TO POSSIBLE WATER HAMMER. THIS OCCURRED AS A RESULT OF A MOMENTARY LOSS OF POWER TO THE CONDENSATE SERVICE PUMPS ON AUGUST 25, 1991. IT IS BELIEVED THAT THE AUGUST EVENT CAUSED THE FAILURE OF THE PENETRATION SINCE THE PENETRATION WAS INSPECTED IN MAY 1991 AND DETERMINED TO BE SEALED. SYSTEM IMPROVEMENTS TO ENSURE LONG TERM FIRE BARRIER INTEGRITY WILL BE INVESTIGATED. INTERIM ANY INSPECTIONS OF THIS PENETRATION SEAL WILL BE PERFORMED FOLLOWING AN IDENTIFIED WATER HAMMER EVENTS IN THE CONDENSATE SERVICE SYSTEM.

[72] NINE MILE POINT 1 DOCKET 50-220 LER 91-009 REV 01
UPDATE ON PLANT OPERATED OUTSIDE OF DESIGN BASIS AND IN VIOLATION OF TECHNICAL SPECIFICATION DUE TO PERSONNEL ERROR CAUSED BY POOR WORK PRACTICES.
EVENT DATE: 081291 REPORT DATE: 110491 NSSS: GE TYPE: BWR

(NSIC 223442) ON AUGUST 13, 1991, AT 0130 HOURS, IT WAS DETERMINED THAT NINE MILE POINT UNIT 1 (NMP1) WAS IN A CONDITION OUTSIDE THE DESIGN BASIS OF THE PLANT. SPECIFICALLY, THE BATTERY CHARGER OUTPUT BREAKER TO BATTERY BOARD #12 TRIP OPEN SETPOINT WAS AT A VALUE WHEREBY THE BATTERY CHARGER WAS PREVENTED FROM PERFORMING ITS REQUIRED SAFETY FUNCTION. AT THE TIME OF THIS DETERMINATION, THE MODE SWITCH WAS IN THE "RUN" POSITION WITH THE REACTOR AT 97 PERCENT POWER. THE ROOT CAUSE OF THIS EVENT WAS A PERSONNEL ERROR CAUSED BY POOR WORK PRACTICES IN THAT THE DESIGN ANALYSIS AND DESIGN REVIEW FAILED TO IDENTIFY THE DEFICIENCY. CORRECTIVE ACTIONS CONSISTED OF THE COMPLETION OF A PLANT CHANGE REQUEST TO INCREASE TRIP OPEN SETPOINT TO AN APPROPRIATE VALUE, PERFORMING A BATTERY CHARGER BREAKER COORDINATION CALCULATION, VERIFYING BATTERY #11 OPERABLE AND CONDUCTING AN ACCOUNTABILITY MEETING. TO PREVENT RECURRENCE, PROCEDURES WILL BE REVISED AND A LESSONS LEARNED TRANSMITTAL WILL BE ISSUED. TO ASSESS POTENTIAL IMPACT TO OTHER ELECTRICAL EQUIPMENT/SYSTEMS, A REVIEW OF SETPOINT CHANGES MADE VIA I&C SETPOINT PROGRAM WILL BE CONDUCTED.

[73] NINE MILE POINT 1 DOCKET 50-220 LER 91-010
ENGINEERING SAFETY FEATURE ACTUATION DUE TO POOR DESIGN, POOR ENVIRONMENTAL CONDITIONS DUE TO MACHINE INTERFACE.
EVENT DATE: 090991 REPORT DATE: 100291 NSSS: GE TYPE: BWR

(NSIC 223078) ON SEPTEMBER 9, 1991, AT 1716 HOURS, NINE MILE POINT UNIT 1 (NMP1) EXPERIENCED AN ACTUATION OF AN ENGINEERING SAFETY FEATURE (ESF). SPECIFICALLY, AN ISOLATION OF #12 EMERGENCY COOLING LOOP. AT THE TIME OF THE EVENT, THE PLANT WAS OPERATING AT APPROXIMATELY 97 PERCENT POWER WITH THE MODE SWITCH IN THE "RUN" POSITION. THE PRIMARY CAUSE OF THE EVENT WAS THE COMBINATION OF POOR HUMAN FACTORS DESIGN OF A CALIBRATION UNIT, POOR ENVIRONMENTAL CONDITIONS, DEGRADED HARDWARE AND THE FAILURE OF THE TECHNICIANS TO APPRECIATE THE CONSEQUENCES OF THESE LESS THAN OPTIMUM CONDITIONS. THE IMMEDIATE CORRECTIVE ACTIONS WERE TO

INFORM OPERATIONS PERSONNEL AS TO THE CAUSE OF THE INADVERTENT ISOLATION, RETURNING THE INSTRUMENT LOOP TO NORMAL AND OPENING THE ISOLATION VALVES. OTHER CORRECTIVE ACTIONS INCLUDE: ASSESSED PLANT IMPACT BY VERIFYING SELECTOR SWITCHES ON OTHER APPLICATIONS OF THIS EQUIPMENT WERE SECURELY FASTENED, ISSUING A LESSONS LEARNED TRANSMITTAL, AND A PLANT CHANGE REQUEST TO MODIFY THE CALIBRATION UNIT SELECTOR KNOBS HAS BEEN WRITTEN.

[74] NINE MILE POINT 2 DOCKET 50-410 LER 91-020
HIGH PRESSURE CORE SPRAY SYSTEM INOPERABLE DUE TO FAILED INSTRUMENT.
EVENT DATE: 092991 REPORT DATE: 102991 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 223307) ON SEPTEMBER 29, 1991, AT 0341 HOURS, THE HIGH PRESSURE CORE SPRAY (HPCS) SYSTEM WAS DECLARED INOPERABLE FOLLOWING A FAILED CHANNEL CHECK OF A HPCS INITIATION LOGIC TRIP UNIT. FURTHER INVESTIGATION IDENTIFIED A DEFECTIVE WIDE RANGE REACTOR VESSEL WATER LEVEL TRANSMITTER. AT THE TIME THIS CONDITION WAS DISCOVERED, NINE MILE POINT UNIT 2 (NMP2) WAS OPERATING AT 55 PERCENT RATED THERMAL POWER WITH THE REACTOR MODE SWITCH IN THE "RUN" POSITION (OPERATIONAL CONDITION 1). THE ROOT CAUSE FOR THIS EVENT WAS DETERMINED TO BE EQUIPMENT FAILURE. CORRECTIVE ACTIONS INCLUDED: REPLACING THE DEFECTIVE TRANSMITTER; COMPLETING CALIBRATION ON THE NEW TRANSMITTER; DECLARING HPCS SYSTEM OPERABLE; AND MAKING PREPARATIONS TO RETURN THE FAILED INSTRUMENT TO THE MANUFACTURER FOR FAILURE ANALYSIS.

[75] NORTH ANNA 1 DOCKET 50-338 LER 91-019
INADEQUATE PROCEDURE CAUSES AN IMPROPER LOW HEAD SAFETY INJECTION RELIEF VALVE BLOWDOWN RING SETTING RESULTING IN OPERABILITY ISSUES.
EVENT DATE: 090491 REPORT DATE: 110691 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)

(NSIC 223370) ON SEPTEMBER 3, 1991 WITH UNIT 1 IN MODE 1 (100 PERCENT POWER), IT WAS DISCOVERED DURING TESTING OF THE "B" LOW HEAD SAFETY INJECTION (LHSI) PUMP, 1-SI-P-1B, THAT DISCHARGE RELIEF VALVE, 1-SI-RV-1845C, LIFTED AND DID NOT RESEAT UNTIL AFTER THE PUMP WAS SECURED. FOLLOWING AN ENGINEERING EVALUATION, IT WAS DETERMINED THAT, PREVIOUSLY DOCUMENTED AND MORE SEVERE RELIEF VALVE MISADJUSTMENTS COULD HAVE RESULTED IN THE FAILURE OF A RELIEF VALVE TO RESEAT. THIS RELIEF VALVE LEAKAGE IS ASSUMED TO BE 20 GPM FOR 30 DAYS WHICH IS GREATER THAN THE UFSAR ASSUMED LEAKAGE OF 50 GPM FOR 10 MINUTES DUE TO A LHSI PUMP SEAL FAILURE. THIS IS REPORTABLE PURSUANT TO 10CFR50.3(A)(2)(II)(B) FOR A CONDITION THAT WAS OUTSIDE THE DESIGN BASIS OF THE PLANT. THE CAUSE OF THE EVENT WAS AN INADEQUATE PROCEDURE. THE MAINTENANCE PROCEDURE DID NOT PROVIDE ADEQUATE INSTRUCTIONS FOR ADJUSTING THE RELIEF VALVE BLOWDOWN RING SETTING ON THE LHSI DISCHARGE LINE RELIEF VALVES. THE PROCEDURE HAS BEEN SUBSEQUENTLY REVISED, THE BLOWDOWN RING ON ALL RELIEF VALVES HAVE BEEN PROPERLY SET, AND ALL RELIEF VALVES HAVE BEEN SUCCESSFULLY TESTED. THIS INCIDENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE PROCEDURAL ACTION COULD HAVE BEEN TAKEN TO RESEAT THE RELIEF VALVE THEREBY ELIMINATING THE POSTULATED RADIOACTIVE RELEASE.

[76] NORTH ANNA 2 DOCKET 50-339 LER 91-010
CONDENSER AIR EJECTOR ISOLATION AND SUBSEQUENT BYPASSING OF FLOW TO THE RADIATION MONITOR.
EVENT DATE: 100691 REPORT DATE: 103191 NSSS: WE TYPE: PWR

(NSIC 223318) ON OCTOBER 6, 1991, WITH UNIT 2 OPERATING AT 100 PERCENT POWER (MODE 1) INSPECTIONS WERE IN PROGRESS FOR POSSIBLE CONDENSER AIR INLEAKAGE SOURCES. AT 2000 HOURS BOTH CONDENSER AIR EJECTORS WERE OBSERVED TO BE ALIGNED WITH FLOW TO THE TURBINE BUILDING ATMOSPHERE, BYPASSING THE INSTALLED RADIATION MONITOR. WITH THE RADIATION MONITOR BYPASSED, THE CAPABILITY FOR AUTOMATIC DIVERSION OF AIR EJECTOR FLOW TO CONTAINMENT UPON A HIGH-HIGH RADIATION MONITOR ALARM WAS DEFEATED. THE EVENT WAS DISCOVERED WHILE INVESTIGATING LOW BACKGROUND RADIATION LEVELS ON THE AIR EJECTOR RADIATION MONITOR. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73 (A)(2)(V)(C), AND A FOUR HOUR REPORT WAS MADE PURSUANT TO 10CFR50.72 (B)(2)(III)(C). THE CAUSE OF THE EVENT WAS FAILURE TO PROVIDE

ADEQUATE SUPERVISORY CONTROLS OVER A NON-ROUTINE EVOLUTION. THE LACK OF DETAILED PROCEDURAL GUIDANCE WAS A CONTRIBUTING FACTOR. NO SIGNIFICANT SAFETY CONSEQUENCES RESULTED FROM THIS EVENT BECAUSE THE CALCULATED TOTAL GASEOUS RELEASE ACTIVITY WAS A SMALL FRACTION OF THE ALLOWABLE EFFLUENT LIMIT. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED AT ANY TIME DURING THIS EVENT.

[77] OCONEE 1 DOCKET 50-269 LER 91-006 REV 01
UPDATE ON TRIP FOR UNKNOWN REASON, POSSIBLE INAPPROPRIATE ACTION.
EVENT DATE: 051691 REPORT DATE: 102991 NSSS: BW TYPE: PWR

(NSIC 223335) ON 5/16/91, AT 1507 HOURS, UNIT 1 TRIPPED FROM 100% FULL POWER, DURING A THUNDERSTORM. THE UNIT EVENTS RECORDER FAILED IMMEDIATELY PRIOR TO THE TRIP, SO IT IS UNKNOWN WHICH TRIP SIGNAL WAS RECEIVED FIRST. COMPUTER AND CHART RECORDER OUTPUTS INDICATED THAT AN INSTRUMENT AND ELECTRICAL (I&E) TECHNICIAN MAY HAVE ERRONEOUSLY OPENED THE HIGH PRESSURE IMPULSE LINE ISOLATION VALVE INSTEAD OF THE LOW PRESSURE VALVE ON A NON-SAFETY RELATED REACTOR COOLANT SYSTEM FLOW TRANSMITTER FOLLOWING CALIBRATION. THIS MAY HAVE CAUSED A MOMENTARY DIP IN THE PRESSURE SIGNAL TO FOUR CHANNELS OF REACTOR PROTECTIVE SYSTEM FLOW INSTRUMENTS WHICH SHARE THE IMPULSE LINES FROM THE FLOW ELEMENT. THE PRESSURE DIP WOULD HAVE CAUSED A TRIP ON FLUX/FLOW/IMBALANCE. ALTERNATIVELY, THE TRIP COULD HAVE OCCURRED DUE TO A SWITCHYARD TRANSIENT. OPERATORS TOOK IMMEDIATE ACTION TO STABILIZE THE UNIT FOLLOWING THE TRIP. THE ROOT CAUSE IS UNKNOWN, POSSIBLE INAPPROPRIATE ACTION. ADDITIONAL INVESTIGATION PERFORMED DURING A SUBSEQUENT UNIT 1 REFUELING OUTAGE FAILED TO CLARIFY THE CAUSE.

[78] OCONEE 1 DOCKET 50-269 LER 91-010
HIGH PRESSURE INJECTION SYSTEM TECHNICALLY INOPERABLE FOR SOME SINGLE FAILURE LOCA SCENARIOS DUE TO DESIGN DEFICIENCY.
EVENT DATE: 091891 REPORT DATE: 102191 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 223260) ON 9/19/91, AT 1225 HOURS, WHILE REVIEWING AN ALARM SETPOINT CHANGE, DESIGN ENGINEERING DETERMINED THAT THE OPERATING LIMIT CURVE FOR LETDOWN STORAGE TANK (LDST) PRESSURE VERSUS LEVEL WAS INADEQUATE. USE OF THIS CURVE COULD PERMIT OPERATION OUTSIDE THE DESIGN BASIS FOR THE EMERGENCY INJECTION FUNCTION OF THE HIGH PRESSURE INJECTION (HPI) SYSTEM. IT WAS DETERMINED THAT, UNDER CERTAIN SMALL BREAK LOCA SCENARIOS, A SINGLE FAILURE COULD RESULT IN HYDROGEN GAS FROM THE LDST EXPANDING INTO THE SUCTION PIPING OF THE HPI PUMPS, CAUSING THE PUMPS TO BE DAMAGED. THIS DEFICIENCY HAS EXISTED ON ALL THREE OCONEE UNITS SINCE INITIAL STARTUP. AT THE TIME OF DISCOVERY, UNIT 1 WAS SHUTDOWN FOR REFUELING AND UNITS 2 AND 3 WERE BOTH AT 100% FULL POWER. CORRECTIVE ACTIONS WERE TO REVISE THE OPERATING LIMIT CURVE AND TO PROVIDE ADDITIONAL INSTRUCTIONS FOR OPERATOR ACTION. THE ROOT CAUSE WAS DESIGN DEFICIENCY.

[79] OCONEE 1 DOCKET 50-269 LER 91-011
REACTOR TRIP RESULTS FROM ELECTRICAL GENERATOR LOCKOUT AFTER EQUIPMENT FAILURE IN A GENERATOR PROTECTIVE RELAY CIRCUIT.
EVENT DATE: 100291 REPORT DATE: 110[illegible]1 NSSS: BW TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 2237[illegible]) ON OCTOBER 2, 1991, UNIT 1 REACTOR TRIPPED FROM 73 PERCENT FULL POWER ON A REACTOR PROTECTIVE SYSTEM TURBINE ANTICIPATORY TRIP SIGNAL DUE TO A GENERATOR LOCKOUT. THE LOCKOUT OCCURRED WHEN A PROTECTIVE RELAY CIRCUIT SPURIOUSLY ACTUATED. INVESTIGATION FOUND THAT THE RELAY CIRCUITRY WAS INTERRUPTED WHEN EITHER A TERMINAL WIRE OR A CONNECTOR BECAME LOOSE. THE ROOT CAUSE WAS ASSIGNED EQUIPMENT FAILURE IN THE 41MXA CIRCUITRY. POSTTRIP OPERATOR RESPONSE STABILIZED THE PLANT. AFTER THE TRIP, THE EMERGENCY FEEDWATER SYSTEM WAS ACTUATED DUE TO LOW FEEDWATER PUMP DISCHARGE PRESSURE DURING A CONDENSATE-FEEDWATER PRESSURE OSCILLATION. ONE MAIN FEEDWATER PUMP WAS MANUALLY SECURED. THE SECOND MAIN FEEDWATER PUMP TRIPPED DUE TO HIGH PRESSURE AT THE PUMP DISCHARGE. CORRECTIVE ACTIONS INCLUDE TIGHTENING CONNECTIONS, EVALUATING THE NEED FOR A

PREVENTIVE MAINTENANCE PROGRAM ON ELECTRICAL CONNECTORS NEAR VIBRATING EQUIPMENT, AND INVESTIGATION INTO LOW DEMAND FEEDWATER OSCILLATIONS.

[80] OYSTER CREEK DOCKET 50-219 LER 91-006
DEGRADATION OF INSTRUMENT RESPONSE DUE TO INADEQUATE DESIGN CONTROL OF INSTRUMENT SNUBBER USE.
EVENT DATE: 100191 REPORT DATE: 103191 NSSS: GE TYPE: BWR

(NSIC 223345) ON SEPTEMBER 24, 1991, A ROUTINE CALIBRATION OF THE PIPE BREAK SENSORS WAS PERFORMED ON ISOLATION CONDENSER SYSTEMS 1 AND 2. DURING THE SURVEILLANCE FOR EACH INSTRUMENT, A CHECK REVEALED A DEGRADATION IN THE FLOW DAMPENING SNUBBER PERFORMANCE FOR SEVEN OF THE EIGHT SENSORS WHEN COMPARED TO THE PROCEDURE CRITERIA. THE FLOW DAMPENING SNUBBERS WERE REMOVED AND TESTED. BASED UPON LIMITED TESTING OF THE REMOVED SNUBBERS AND AN EXTRAPOLATION OF THE TEST RESULTS, THE PIPE BREAK SENSORS WOULD HAVE EXPERIENCED A DELAY IN RESPONSE TIME SUCH THAT THE DESIGN BASIS ISOLATION VALVE CLOSURE TIME OF SIXTY SECONDS WOULD NOT HAVE BEEN MET FOR ALL POSTULATED PIPE BREAK EVENTS. THE CAUSE OF THIS OCCURRENCE HAS BEEN ATTRIBUTED TO INADEQUATE DESIGN CONTROL. DURING THE RECENT 13R REFUELING OUTAGE, THE EXISTING SNUBBERS WERE REPLACED WITH A DIFFERENT MANUFACTURER AND TYPE SNUBBER WITHOUT PERFORMING AN EVALUATION TO ASSESS THE IMPACT ON THE SENSOR RESPONSE TIME. CORRECTIVE ACTION CONSISTED OF REMOVING THE SNUBBERS FROM THE EIGHT PIPE BREAK SENSORS. ADDITIONAL CORRECTIVE ACTION WILL CONSIST OF PERFORMING A REVIEW OF OTHER APPLICATIONS WHERE SNUBBER FITTINGS ARE USED ON CRITICAL PLANT INSTRUMENTATION TO ENSURE APPROPRIATE USE AND PROPER DESIGN DOCUMENTATION.

[81] PALISADES DOCKET 50-255 LER 91-019
STACK GAS RADIATION MONITOR CARTRIDGE/FILTER ASSEMBLY O-RING FALURE.
EVENT DATE: 100391 REPORT DATE: 103191 NSSS: CE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 223351) ON OCTOBER 3, 1991, AT APPROX. 0830 HOURS, IT WAS DETERMINED THAT THE RADIOACTIVE GASEOUS EFFLUENT MONITOR (RGEM) ON THE PLANT STACK WAS NOT FUNCTIONING PROPERLY. AT THE TIME OF DISCOVERY THE PLANT WAS OPERATING AT 100% POWER. THE CONDITION WAS DISCOVERED DURING THE PERFORMANCE OF TECHNICAL SPECIFICATION SURVEILLANCE TEST DWR-10, "STACK EFFLUENT SAMPLING, CALCULATIONS AND RECORDS". THE CAUSE OF THIS EVENT WAS THE FAILURE OF THE O-RING ON THE RGEM SYSTEM THAT SEALS THE INLET SIDE OF THE SAMPLE HOLDER FROM THE OUTLET SIDE. BECAUSE OF THE O-RING FAILURE, CONTINUOUS SAMPLING WAS NOT PROVIDED. THE IMMEDIATE CORRECTIVE ACTION INCLUDED NOTIFYING THE SHIFT ENGINEER AND THE RMC ADMINISTRATOR OF THE CONDITION, REPLACING THE WORN O-RING AND REESTABLISHING SAMPLE FLOW AND CONSERVATIVELY ESTIMATING RADIOACTIVE IODINE AND PARTICULATE ACTIVITY FOR THE PERIOD OF SEPTEMBER 26, 1991 THROUGH OCTOBER 3, 1991. LONG TERM CORRECTIVE ACTION INCLUDES REVISING THE TECHNICAL SPECIFICATION SURVEILLANCE PROCEDURE TO INCLUDE RECORDING OF THE DIFFERENTIAL PRESSURE READING ACROSS THE FILTER, TRAINING ALL RADIATION SAFETY DEPARTMENT PERSONNEL ON RGEM SYSTEM OPERATION AND SURVEILLANCE PROCEDURES AND VERIFYING THE CONSERVATISM OF THE IODINE AND PARTICULATE ACTIVITIES ESTIMATE.

[82] PALO VERDE 1 DOCKET 50-528 LER 91-008 REV 01
UPDATE ON REACTOR COOLANT SYSTEM POSSIBLY EXCEEDING MAKEUP DUE TO A POSTULATED FIRE.
EVENT DATE: 081691 REPORT DATE: 102391 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 2 (PWR)
PALO VERDE 3 (PWR)

(NSIC 223300) ON 8/16/91, PALO VERDE UNITS 1 AND 3 WERE IN MODE 1 AT APPROXIMATELY 100 PERCENT POWER AND UNIT 2 WAS IN MODE 1 AT APPROXIMATELY 64 PERCENT POWER (UNTIL A REACTOR TRIP OCCURRED AT APPROXIMATELY 0839 MST: REFERENCE LER 529/91-004) WHEN APS ENGINEERING PERSONNEL DETERMINED THAT A DESIGN BASIS APPENDIX R FIRE IN THE CONTROL ROOM COULD RESULT IN LOSS OF REACTOR COOLANT PUMP SEAL COOLING. THE LOSS OF RCP SEAL COOLING COULD RESULT IN RCP SEAL DAMAGE WHICH MAY RESULT IN REACTOR COOLANT SYSTEM LEAKAGE IN EXCESS OF AVAILABLE CHARGING.

UPON DISCOVERY OF THIS POTENTIAL EVENT, APPROPRIATE COMPENSATORY MEASURES WERE ESTABLISHED IN ACCORDANCE WITH THE PVNGS FIRE PROTECTION PROGRAM. SUBSEQUENTLY, APS ENGINEERING DETERMINED THAT A FIRE OUTSIDE THE CONTROL ROOM COULD ALSO RESULT IN THE LOSS OF RCP SEAL INTEGRITY. THE CAUSE OF THIS POSTULATED EVENT WAS A FAILURE OF THE ORIGINAL APPENDIX R EVALUATIONS TO RECOGNIZE THE POTENTIAL FOR RCP SEAL LEAKAGE GREATER THAN THE MAKEUP CAPACITY OF ONE CHARGING PUMP (44 GALLONS PER MINUTE) DUE TO A CONTROL ROOM FIRE. NO PREVIOUS SIMILAR EVENTS HAVE BEEN REPORTED PURSUANT TO TECHNICAL SPECIFICATION 6.9.3 AND 10CFR50.73.

[83] PALO VERDE 2 DOCKET 50-529 LER 91-005
MAIN STEAM SAFETY VALVE SETPOINTS OUT OF TOLERANCE.
EVENT DATE: 100991 REPORT DATE: 111291 NSSS: CE TYPE: PWR
VENDOR: DRESSER INDUSTRIES, INC.

(NSIC 223455) ON OCTOBER 9, 1991 WHILE UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 97.5 PERCENT POWER, AN ENGINEERING EVALUATION OF ASME SURVEILLANCE TESTING RESULTS DETERMINED THAT ELEVEN (11) OF THE TWENTY (20) MAIN STEAM SAFETY VALVE (MSSV) AS-FOUND RELIEF SETTINGS WERE OUT OF THE TOLERANCE LIMITS SPECIFIED IN TECHNICAL SPECIFICATION (TS) 3.7.1.1 AND IN THE TESTING REQUIREMENTS ESTABLISHED BY APS. THE TESTING AND ADJUSTMENTS WERE PERFORMED DURING THE PERIOD OF OCTOBER 8 THROUGH OCTOBER 9, 1991, WHILE UNIT 2 WAS IN MODE 1, TO VERIFY THE RELIEF SETTINGS OF THE MSSVS. THE CAUSE OF THE EVENT IS SETPOINT DRIFT. AS IMMEDIATE CORRECTIVE ACTION THE MSSVS HAVE BEEN ADJUSTED AND TESTED SATISFACTORILY. PREVIOUS SIMILAR EVENTS WERE REPORTED IN MSSV LERS 528/88-014-01, 528/89-01000, 529/89-002-00, 529/89-007-00 AND 530/91-001-00.

[84] PALO VERDE 3 DOCKET 50-530 LER 91-007
REACTOR SHUTDOWN REQUIRED BY TECHNICAL SPECIFICATION DUE TO INVERTER FAILURE.
EVENT DATE: 083191 REPORT DATE: 092791 NSSS: CE TYPE: PWR
VENDOR: COMBUSTION ENGINEERING, INC.
ELGAR, CORP.

(NSIC 223108) AT APPROXIMATELY 0119 MST ON AUGUST 31, 1991, PALO VERDE UNIT 3 WAS IN MODE 3 (HOT STANDBY) AT NORMAL TEMPERATURE AND PRESSURE WHEN A PLANT SHUTDOWN WAS COMPLETED IN ACCORDANCE WITH ACTION B.(2) OF TECHNICAL SPECIFICATION (TS) 3.8.3.1 DUE TO A CLASS 1E INVERTER THAT COULD NOT BE RETURNED TO OPERABLE STATUS WITHIN THE ALLOWED ACTION TIME. THE PLANT WAS STABLE AT NORMAL TEMPERATURE AND PRESSURE. NO SAFETY SYSTEM RESPONSES OCCURRED AND NONE WERE REQUIRED. THE EVENT WAS DETERMINED TO BE AN UNCOMPLICATED NORMAL PLANT SHUTDOWN. AT APPROXIMATELY 1902 MST ON AUGUST 31, 1991, TS 3.8.3.1 ACTION B.(2) WAS EXITED WHEN THE CLASS 1E INVERTER WAS RETURNED TO SERVICE AND THE 120-VOLT VITAL A.C. ELECTRICAL BUS WAS REENERGIZED FROM ITS ASSOCIATED INVERTER. THE CAUSE OF THIS EVENT WAS A COGNITIVE PERSONNEL ERROR BY AN OPERATIONS COMPUTER SYSTEMS MAINTENANCE TECHNICIAN WHO IMPROPERLY ATTACHED A VARIABLE POWER SOURCE TO THE ENERGIZED CONTROL ELEMENT ASSEMBLY CALCULATOR CIRCUIT WHILE IT WAS BEING FED FROM THE INVERTER. THIS RESULTED IN THE INVERTER FAILURE. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED PURSUANT TO 10CFR50.73.

[85] PEACH BOTTOM 2 DOCKET 50-277 LER 91-033
HIGH PRESSURE COOLANT INJECTION SYSTEM BEING INOPERABLE AS A RESULT OF A PACKING LEAK ON THE STEAM SUPPLY INBOARD ISOLATION VALVE.
EVENT DATE: 101691 REPORT DATE: 111591 NSSS: GE TYPE: BWR

(NSIC 223419) ON 10/15/91 AT 2112 HOURS, DURING THE PERFORMANCE OF A SURVEILLANCE TEST, A DRYWELL PRESSURE AND TEMPERATURE INCREASE WAS IDENTIFIED WHEN THE HIGH PRESSURE COOLANT INJECTION (HPCI) STEAM SUPPLY INBOARD ISOLATION VALVE WAS STROKED. THE VALVE WAS BACKSEATED TO ISOLATE THE LEAK AND PLANT SHUTDOWN WAS COMPLETED ON 10/17/91. THE CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE EXCESSIVE VALVE PACKING LEAKAGE DUE TO A GOUGED STEM. THIS IS THE RESULT OF A REVERSED PACKING CONFIGURATION AND THE VALVE OPERATOR BEING HORIZONTALLY MOUNTED. THIS COMBINATION CHANGED THE LOAD SUPPORTING CHARACTERISTICS OF THE VALVE WHICH RESULTED IN THE VALVE BONNET AREA GOUGING THE STEM. THE PACKING HAD BEEN PREVIOUSLY REVERSED IN ACCORDANCE WITH A NON CONFORMANCE REPORT (NCR) BECAUSE OF

AN UNQUALIFIED STEM LEAK-OFF PLUG WELD. THE EVALUATION PERFORMED AS PART OF THE NCR DID NOT ADEQUATELY EVALUATE THE EFFECT OF PACKING CONFIGURATION REVERSAL ON HORIZONTALLY MOUNTED MOTOR OPERATORS. THE WELD WAS REWORKED AND THE VALVE PACKING WAS REINSTALLED TO ITS NORMAL CONFIGURATION AND THE VALVE STEM WAS REPLACED. THE PLANT WAS RETURNED TO POWER OPERATION ON 10/21/91. A REVIEW HAS BEEN CONDUCTED TO ENSURE THAT SIMILAR PACKING CONFIGURATIONS DO NOT EXIST ON HORIZONTALLY MOUNTED VALVES. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THESE EVENTS. THERE WERE NO PREVIOUS SIMILAR LERS.

[86] PEACH BOTTOM 2 DOCKET 50-277 LER 91-034
REACTOR CORE ISOLATION SYSTEM BEING INOPERABLE DUE TO A DIRTY TRANSFER SWITCH CONTACT.
EVENT DATE: 102291 REPORT DATE: 111991 NSSS: GE TYPE: BWR

(NSIC 223445) ON 10/21/91 A TECHNICAL SPECIFICATION VIOLATION OCCURRED WHEN THE MODE SWITCH WAS PLACED IN THE STARTUP POSITION WITH THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM INOPERABLE. ON 10/22/91 AT 0545 HOURS, THE SHIFT TECHNICAL ADVISOR IDENTIFIED THE RCIC FLOW CONTROLLER OUTPUT SIGNAL TO BE 0% INSTEAD OF THE EXPECTED 100%. THE CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE A DIRTY CONTACT ON THE RCIC FLOW CONTROLLER OUTPUT SIGNAL TRANSFER SWITCH. IT IS PRESUMED THAT THE TRANSFER SWITCH CONTACTS FAILED TO MAKEUP DURING THE PERFORMANCE OF THE SURVEILLANCE TEST (ST) ON 10/19/91 EVEN THOUGH THE SWITCH WAS IN THE PROPER POSITION. THE ST USED TO TEST THIS TRANSFER SWITCH PROVIDED NO MEANS TO ENSURE THAT THE RCIC FLOW CONTROLLER CONTACTS WERE PROPERLY MADE-UP AFTER SWITCH OPERATION. THIS TYPE OF TRANSFER SWITCH IS NOT CHECKED ON A PERIODIC BASIS TO ENSURE THAT THE SWITCH CONTACTS ARE CLEAR OF DIRT OR OXIDATION. THE RCIC FLOW CONTROLLER TRANSFER SWITCH WAS EXERCISED AND THE OUTPUT SIGNAL FROM THE FLOW CONTROLLER WAS VERIFIED TO BE FUNCTIONING PROPERLY. AN EVALUATION WILL BE PERFORMED TO IDENTIFY IF ADDITIONAL TESTING OR MAINTENANCE MAY BE NECESSARY. THE ST WILL BE REVISED TO ENHANCE THE SYSTEM RESTORATION REQUIREMENTS. OTHER SIMILAR TESTS WHICH INVOLVE TRANSFER SWITCH OPERATION WILL BE EVALUATED TO DETERMINE ADDITIONAL APPROPRIATE CORRECTIVE ACTIONS.

[87] PERRY 1 DOCKET 50-440 LER 91-017
HIGH PRESSURE CORE SPRAY PUMP INOPERABLE DUE TO MOTOR BEARING OIL RESERVOIR DRAIN PLUG FAILURE.
EVENT DATE: 100291 REPORT DATE: 110191 NSSS: GE TYPE: BWR

(NSIC 223306) ON 10/2/91, DURING ROUTINE OPERATOR ROUNDS, OIL WAS OBSERVED LEAKING FROM THE LOWER BEARING OIL RESERVOIR FOR THE HIGH PRESSURE CORE SPRAY (HPCS) PUMP. DURING THE SUBSEQUENT INVESTIGATION, THE RESPONSIBLE SYSTEM ENGINEER (RSE) DETERMINED THE SOURCE OF THE LEAK TO BE THE BEARING OIL RESERVOIR DRAIN PLUG. UPON FURTHER INVESTIGATION THE RSE DETERMINED THAT THE DRAIN PLUG COULD NOT BE TIGHTENED SUFFICIENTLY TO ALLOW OPERATION OF THE HPCS MOTOR. THE HPCS SYSTEM WAS DECLARED INOPERABLE AND A FOUR-HOUR NOTIFICATION MADE PURSUANT TO THE REQUIREMENTS OF 10CFR50.72(B)(2)(III)(D). THE DRAIN PLUG WAS SUBSEQUENTLY REPLACED AND THE HPCS SYSTEM WAS RETURNED TO OPERABLE STATUS. BEARING OIL DRAIN PLUGS OF SIMILAR DESIGN WERE VISUALLY INSPECTED FOR SIGNS OF LEAKAGE. NO ADDITIONAL PROBLEMS WERE IDENTIFIED. AS A RESULT OF SIMILAR PREVIOUS OCCURRENCES, THE ROOT CAUSE HAS BEEN DETERMINED TO BE A DESIGN DEFICIENCY DUE TO THE USE OF THE DRAIN PLUG AS AN OIL SAMPLE COLLECTION POINT. EXISTING INTERIM CORRECTIVE ACTION MEASURES WILL BE UTILIZED UNTIL DESIGN CHANGE PACKAGE (DCP) 88-0256 IS IMPLEMENTED. THE REVISED DESIGN WILL REPLACE THE DRAIN PLUGS WITH A BALL VALVE ARRANGEMENT TO FACILITATE LUBE OIL SAMPLING ACTIVITIES. DRAIN PLUG REPLACEMENT WILL OCCUR PRIOR TO THE END OF THE NEXT REFUELING OUTAGE ON A SYSTEM AVAILABILITY BASIS.

[88] PERRY 1 DOCKET 50-440 LER 91-018
CONTROL ROD SCRAM INSERTION TIME TESTING RESULTS IN DISCOVERY OF COMMON MODE FAILURE AND TECHNICAL SPECIFICATION REQUIRED SHUTDOWN.
EVENT DATE: 100691 REPORT DATE: 110191 NSSS: GE TYPE: BWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)

(NSIC 223317) ON 10/6/91, AT 1037, CONTROL ROD 46-23 EXCEEDED ITS MAXIMUM SCRAM TIME TO POSITION 43. OPERATORS THEN TOOK ACTION TO DEMONSTRATE THAT THE EIGHT ADJACENT CONTROL RODS SURROUNDING CONTROL ROD 46-23 COULD SATISFY THE MAXIMUM SCRAM INSERTION TIME LIMITS. CONTROL ROD 42-19 ALSO EXCEEDED ITS MAXIMUM SCRAM INSERTION TIME TO POSITION 43. BECAUSE CONTROL RODS 46-23 AND 42-19 ARE ADJACENT RODS, THE PLANT WAS REQUIRED BY TECH SPEC TO BE IN AT LEAST HOT SHUTDOWN WITHIN 12 HOURS. ON 10/6/91, AT 1436, OPERATORS COMMENCED A SHUTDOWN OF THE PLANT. THE CAUSE OF THE "SLOW" CONTROL RODS IS ATTRIBUTED TO COMPONENT FAILURE. THE AUTOMATIC SWITCH COMPANY (ASCO) MODEL NUMBER HVA176-816-1 SCRAM SOLENOID PILOT VALVE (SSPV) FOR EACH OF THE AFFECTED CONTROL RODS WAS DETERMINED TO BE THE CAUSE OF THE "SLOW" CONTROL RODS. ALL OF THE SUSPECT SSPVS WERE FROM THE SAME LOT REMANUFACTURED BY ASCO IN 11/90. AFTER SHUTDOWN OF THE PLANT, ALL 49 OF THE SSPVS FROM LOT NUMBER 184010001 WERE REMOVED FROM THEIR ASSOCIATED HCUS AND REPLACED. PRELIMINARY EVALUATION OF THE REMOVED SSPVS HAS NOT YIELDED ANY CONCLUSIONS AS TO THE REASON BEHIND THE FAILURES. TWO OF THE SUSPECT VALVES HAVE BEEN SENT TO ASCO FOR DISASSEMBLY AND ANALYSIS. INVESTIGATION OF THE SSPV PROBLEM AND MONITORING OF CONTROL ROD SCRAM TIMES BY ENGINEERING PERSONNEL IS PRESENTLY IN PROGRESS.

[89] PERRY 1 DOCKET 50-440 LER 91-019
MISSED ROD PATTERN CONTROLLER SURVEILLANCE RESULTS IN TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 100691 REPORT DATE: 110591 NSSS: GE TYPE: BWR

(NSIC 223381) ON OCTOBER 6, 1991, AT 1839, DURING A PLANT SHUTDOWN, A TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT FOR THE ROD PATTERN CONTROLLER WAS PERFORMED LATE. AT 1827 THE FIRST ROD/GANG (GANG 46) HAD REACHED THE FIRST BANKED POSITION DURING THE SHUTDOWN; HOWEVER, TWELVE ADDITIONAL ROD/GANG MOVEMENTS WERE PERFORMED BEFORE PROPERLY TESTING THE ROD PATTERN CONTROLLER BEFORE OPERATORS REALIZED THE ERROR. THE CAUSE OF THE VIOLATION OF THE ROD PATTERN CONTROLLER TECHNICAL SPECIFICATION IS PERSONNEL ERROR, INATTENTION TO DETAIL/FAILURE TO RECOGNIZE. THE SENIOR REACTOR OPERATOR IN CONTROL OF THE SHUTDOWN ACTIVITIES KNEW OF THE REQUIREMENT TO TEST THE ROD PATTERN CONTROLLER, BUT WAS UNAWARE OF ACTUAL ROD POSITION WITH RESPECT TO THE BANK LIMITS. A CONTRIBUTING FACTOR TO THIS EVENT IS INADEQUATE TRAINING, IN THAT THE INFREQUENCY AND LEVEL OF DETAIL OF TRAINING TO THIS SURVEILLANCE LEFT THE OPERATING CREW UNFAMILIAR WITH THIS ACTIVITY. THE SENIOR REACTOR OPERATOR RESPONSIBLE FOR THE MISSED SURVEILLANCE REQUIREMENT HAS BEEN COUNSELED CONCERNING IMPORTANCE OF ATTENTION TO DETAIL. PROCEDURES ARE BEING REVISED TO CLARIFY THE ROD PATTERN CONTROLLER TESTING REQUIREMENT. ADDITIONALLY, THE FREQUENCY AND DETAIL OF SIMULATOR TRAINING, ON PERFORMANCE OF THIS SURVEILLANCE, IS BEING INCREASED.

[90] PERRY 1 DOCKET 50-440 LER 91-020
CABLE TRAY RACEWAYS FOUND TO BE IMPAIRED AS A FIRE BARRIER, ADVERSELY AFFECTING SAFE SHUTDOWN REQUIREMENTS.
EVENT DATE: 100791 REPORT DATE: 110491 NSSS: GE TYPE: BWR

(NSIC 223456) ON 10/7/91, DISCREPANCIES IN THE INSTALLATION OF THE FIRE WRAP ON APPENDIX R RACEWAYS WAS DETERMINED TO BE A FIRE BARRIER IMPAIRMENT WHICH COULD ADVERSELY AFFECT SAFE SHUTDOWN REQUIREMENTS. TECH SPECS REQUIRE THIS CONDITION TO BE REPORTED TO THE NRC THROUGH THE LER PROCESS THIS CONDITION IS ALSO CONSIDERED REPORTABLE UNDER 10CFR21. THE CAUSE OF THIS EVENT WAS INADEQUATE DESIGN. THE INFORMATION, PROVIDED BY GILBERT COMMONWEALTH, RESULTED IN THE INSTALLATION STANDARD SPECIFICATION, THE DESIGN DRAWING, AND THE PERIODIC TEST INSTRUCTION (PTI-P54-P0075) "APPENDIX R FIRE WRAP INSPECTION" NOT REFLECTING THE THERMAL SCIENCE, INC. (TSI) INSTALLATION CRITERIA IN THE VENDOR MANUAL CONCERNING MAXIMUM SPACING OF THE MECHANICAL FASTENERS. THE IMMEDIATE CORRECTIVE ACTION WAS TO INITIATE HOURLY FIRE WATCHES FOR THE FIRE IMPAIRMENT SITUATION, WHICH WILL CONTINUE UNTIL THE APPENDIX R RACEWAYS ARE UPGRADED TO ENSURE THE BANDING SPACING IS NO GREATER THAN TWELVE INCHES. THE INSTALLATION STANDARD SPECIFICATION, DESIGN DRAWING, AND PTI-P54-P0075 WILL BE REVISED TO REFLECT THE INSTALLATION CRITERIA SPECIFIED BY THE TSI VENDOR MANUAL.

[91] PERRY 1 DOCKET 50-440 LER 91-021
INOPERABLE PREFERRED SOURCE BREAKER RESULTS IN EXCEEDING THE ALLOWABLE OUTAGE TIME LIMIT PROVIDED BY TS 3.8.1.1.
EVENT DATE: 101191 REPORT DATE: 110891 NSSS: GE TYPE: BWR
VENDOR: BROWN BOVERI

(NSIC 223388) ON OCTOBER 11, 1991 AT 1138, PREFERRED SOURCE BREAKER EH1114 WOULD NOT CLOSE ON DEMAND FROM ITS SWITCH ON CONTROL ROOM PANEL 1H13-P877-1. INVESTIGATION DETERMINED THAT THIS BREAKER HAD BEEN INOPERABLE SINCE SEPTEMBER 4, 1991. AS A RESULT, THE ALLOWABLE OUTAGE TIME LIMIT PROVIDED BY TECHNICAL SPECIFICATION 3.8.1.1 WAS EXCEEDED . THE PRIMARY CAUSE OF THIS EVENT WAS EQUIPMENT MALFUNCTION, OTHER. OVER THE YEARS, MODIFICATIONS WERE PERFORMED ON THE RACKING TOOL UTILIZED TO ENGAGE AND TURN T HE RACKING LEAD SCREW. THESE MODIFICATIONS ALLOWED INTERFERENCE WITH THE END OF THE RACKING RELEASE LEVER AS IT ATTEMPTS TO SETTLE IN THE FINAL DETENTE (RACKED-IN) POSITION; THEREBY ALLOWING THE RACKING LEAD SCREW TO BE OVERTURNED, WHICH PLACES THE BREAKER INTO A TRIP FREE CONDITION. TO PREVENT RECURRENCE, OPERATIONS PERSONNEL VISUALLY INSPECTED SAFETY AND NONSAFETY RELATED BUS BREAKERS TO ENSURE THAT THEIR MECHANICAL INTERLOCKS WERE DISENGAGED. ALL AUXILIARY OPERATORS WERE RETRAINED TO PERFORM VISUAL CHECKS OF THE MECHANICAL INTERLOCK DURING FUTURE RACKING EVOLUTIONS. ADDITIONALLY, SPRING PINS ON THE APPLICABLE RACKING TOOLS WILL BE SHORTENED TO ONE INCH IN LENGTH. LICENSED PERSONNEL AND AUXILIARY OPERATORS WILL BE TRAINED TO THIS EVENT DURING CURRENT EVENTS TRAINING.

[92] PERRY 1 DOCKET 50-440 LER 91-022
PREVIOUSLY UNRECOGNIZED RADIATION MONITOR SPIKING RESULTS IN BACKUP HYDROGEN PURGE SYSTEM ISOLATION.
EVENT DATE: 101191 REPORT DATE: 110891 NSSS: GE TYPE: BWR

(NSIC 223382) ON OCTOBER 11, 1991, AT 1845, THE SENIOR REACTOR OPERATOR MANIPULATED THE FRONT PANEL SWITCHES ON THE DRYWELL ATMOSPHERIC GASEOUS RADIATION MONITOR TO CHECK THE "ALERT ALARM SETPOINT". DURING THE SWITCH MANIPULATIONS, THE RADIATION MONITOR INDICATION SPIKED UPSCALE CAUSING A BACKUP HYDROGEN PURGE SYSTEM ISOLATION. THE CAUSE OF THE EVENT IS A PREVIOUSLY UNRECOGNIZED EQUIPMENT CHARACTERISTIC. DURING TROUBLESHOOTING IT WAS NOTED THAT MANIPULATING THE FRONT PANEL SWITCHES OF THE RADIATION MONITOR TO PERFORM SETPOINT CHECKS WOULD RESULT IN THE SAME TRIP SIGNAL THAT WAS RECEIVED DURING THIS EVENT. THIS IS DUE TO A COMBINATION OF EQUIPMENT DESIGN AND THE VALUE OF THE TRIP SETPOINT, AND THE SPIKING IS CONSIDERED NORMAL FOR THIS EVOLUTION. TO PREVENT RECURRENCE, AN INFORMATION TAG HAS BEEN ATTACHED TO THE FRONT OF THE MONITOR TO WARN OF THE POTENTIAL OF AN INADVERTENT BACKUP HYDROGEN PURGE SYSTEM ISOLATION DUE TO SPIKING DURING SWITCH MANIPULATIONS. THE APPROPRIATE OPERATING INSTRUCTION IS BEING REVISED TO PROVIDE THE REQUIRED GUIDANCE TO AVOID POTENTIAL PROBLEMS WITH THE EQUIPMENT CHARACTERISTICS FOR THIS AND OTHER APPLICABLE RADIATION MONITORS. LICENSED OPERATOR TRAINING IS BEING UPDATED TO REFLECT THE ADDITIONAL PROCEDURAL GUIDANCE FOR THE APPLICABLE RADIATION MONITORS. DURING REQUALIFICATION TRAINING, OPERATO RS WILL BE INSTRUCTED ON THE LESSONS LEARNED FROM THIS EVENT.

[93] PILGRIM 1 DOCKET 50-293 LER 91-021
REACTOR CORE ISOLATION COOLING SYSTEM DECLARED INOPERABLE DUE TO INSUFFICIENT BATTERY CHARGER TEST.
EVENT DATE: 100991 REPORT DATE: 110891 NSSS: GE TYPE: BWR

(NSIC 223422) ON OCTOBER 9, 1991 AT 1802 HOURS, THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM WAS DECLARED INOPERABLE AND A SEVEN DAY LIMITING CONDITION FOR OPERATION (LCO) BEGAN AT THAT TIME. THE SYSTEM WAS DECLARED INOPERABLE BECAUSE SUFFICIENT TEST DATA WAS NOT AVAILABLE TO DEMONSTRATE THAT SUFFICIENT MARGIN TO THE RCIC INVERTER TRIP SETPOINT EXISTED IF A 125 VDC BUS 'A' VOLTAGE TRANSIENT WERE TO OCCUR. THE DC VOLTAGE TRANSIENT IS POSSIBLE IF AN AC VOLTAGE TRANSIENT OF SUFFICIENT MAGNITUDE OCCURS AT THE INPUT OF THE 125 VDC BATTERY CHARGER. THE 125 VDC BATTERY 'A' AND BACKUP BATTERY CHARGER WERE IN SERVICE SUPPLYING POWER TO THE RCIC INVERTER VIA 125 VDC BUS 'A' WHEN THIS CONDITION WAS IDENTIFIED. THE RCIC SYSTEM WAS MAINTAINED IN THE NORMAL STANDBY MODE AND WAS NOT REMOVED FROM SERVICE AS A RESULT OF DECLARING THE SYSTEM INOPERABLE. A

WRITTEN REQUEST FOR RELIEF FROM THE REQUIREMENT TO SHUT DOWN ON OCTOBER 16, 1991 BECAUSE OF THIS CONDITION WAS GRANTED BY THE NRC ON OCTOBER 16, 1991. COMPENSATORY MEASURES WERE IMPLEMENTED AS A RESULT OF THIS CONDITION. THE PLANT WAS SHUT DOWN ON OCTOBER 30, 1991 FOR REASONS UNRELATED TO THE RCIC SYSTEM LCO. CORRECTIVE ACTION PLANNED CONSISTS OF TESTING AND/OR THE IMPLEMENTATION OF MODIFICATIONS TO PRECLUDE UNACCEPTABLE VOLTAGE TRANSIENTS FROM OCCURRING ON THE 125 VDC BUS 'A'.

[94]   POINT BEACH 1   DOCKET 50-266   LER 91-012
NUCLEAR INSTRUMENTATION TURBINE RUNBACK.
EVENT DATE: 092491   REPORT DATE: 102291   NSSS: WE   TYPE: PWR
VENDOR: ELGAR, CORP.

(NSIC 223277) AT 1027 ON 9/24/91 UNIT 1 EXPERIENCED A TURBINE RUNBACK FROM 100% TO APPROXIMATELY 80% POWER DURING MAINTENANCE ON INVERTER DYOD. THE RUNBACK WAS CAUSED BY A VOLTAGE LOSS ON INSTRUMENT BUSES 1Y04 AND 1Y104 WHEN THE SUPPLY BREAKER FROM DC BUS D04 TO INVERTER DYOD WAS CLOSED. PRIOR TO THIS EVENT, SWING INVERTER DYOD (ELGAR MODEL 253-1-103) HAD BEEN TAKEN OUT OF SERVICE FOR MAINTENANCE. THE TURBINE RUNBACK OCCURRED WHEN THE D04 TO DYOD SUPPLY BREAKER WAS CLOSED WHILE BRINGING THE INVERTER BACK INTO SERVICE. THE INPUT CIRCUIT BREAKER FILTER CAPACITORS TO INVERTER DYOD HAD NOT BEEN CHARGED PRIOR TO ATTEMPTED CLOSURE. THE UNCHARGED CAPACITORS CAUSED A VOLTAGE SPIKE TO OCCUR ON THE DC SUPPLY BUS TO INVERTER 1DY04 (BUS D04). THIS VOLTAGE SPIKE SCRAMBLED THE INVERTER'S LOGIC, BLOWING A FUSE AND SHUTTING DOWN INVERTER 1DY04. ALSO, THE UNCHARGED CAPACITOR CREATED AN EXCESSIVE CURRENT DRAW ON INVERTER DYOD, CAUSING THE DC SUPPLY BREAKER TO TRIP ON OVERCURRENT. POWER TO YELLOW INSTRUMENT BUSES 1Y04 AND 1Y104, WHICH WAS BEING SUPPLIED BY INVERTER 1DY04, WAS SUBSEQUENTLY LOST. BECAUSE POWER RANGE NUCLEAR INSTRUMENT CHANNEL N44 IS POWERED BY THE YELLOW INSTRUMENT BUS, THIS LOSS OF POWER PRODUCED A NEGATIVE VOLTAGE SPIKE ON N44, ACTIVATING THE DROPPED ROD TURBINE RUNBACK CIRCUITRY. ALL INSTRUMENTATION AND CONTROL SYSTEMS OPERATED AS DESIGNED. UNIT 1 WAS RETURNED TO FULL POWER AT 1257. THIS EVENT IS AN ACTUATION OF A REACTOR PROTECTION SYSTEM.

[95]   POINT BEACH 1   DOCKET 50-266   LER 91-013
NUCLEAR INSTRUMENTATION TURBINE RUNBACK CAUSED BY A VOLTAGE SPIKE ON POWER RANGE CHANNEL N44.
EVENT DATE: 100391   REPORT DATE: 110191   NSSS: WE   TYPE: PWR

(NSIC 223336) AT 1917 ON OCTOBER 3, 1991, WHILE UNIT 1 WAS OPERATING AT 100% POWER, A 20% RUNBACK OF THE TURBINE GENERATOR OCCURRED. THIS RUNBACK WAS CAUSED BY AN OUTPUT TRANSIENT ON POWER RANGE NUCLEAR INSTRUMENTATION CHANNEL N44. THE REACTOR PROTECTION SYSTEM SENSED A DROPPED CONTROL ROD AND INITIATED THE TURBINE RUNBACK. FOLLOWING THE RUNBACK, THE DROPPED ROD CIRCUIT WAS RESET AT 1922. THE LOAD INCREASE WAS COMMENCED AT 1935. FULL POWER WAS ATTAINED AT 2010.

[96]   POINT BEACH 2   DOCKET 50-301   LER 91-001
FAILURE OF MAIN STEAM ISOLATION VALVES TO CLOSE.
EVENT DATE: 092991   REPORT DATE: 102891   NSSS: WE   TYPE: PWR
OTHER UNITS INVOLVED: POINT BEACH 1 (PWR)
VENDOR: ATWOOD & MORRILL CO., INC.

(NSIC 223329) AT 0930 ON SEPTEMBER 29, 1991, AN ATTEMPT TO SHUT THE POINT BEACH NUCLEAR PLANT UNIT 2 MAIN STEAM ISOLATION VALVES (MSIVS) WAS MADE FROM THE CONTROL ROOM. BOTH MSIVS FAILED TO LEAVE THE FULLY OPEN POSITION. AN OPERATOR WAS DISPATCHED TO THE VALVES AND SHUT THEM BY APPLYING MECHANICAL FORCE TO THE VALVE OPERATORS. AFTER THE VALVE OPERATORS WERE FREED BY MECHANICAL FORCE, THE VALVES SHUT UNASSISTED. UNIT 2 HAD BEEN SHUT DOWN AND COOLED DOWN TO APPROXIMATELY 325 DEGREES F FOR THE BEGINNING OF ITS ANNUAL MAINTENANCE AND REFUELING OUTAGE WHEN THIS EVENT OCCURRED. AN EXTENSIVE INVESTIGATION INTO THE CAUSE OF THE FAILURE TO CLOSE IS BEING PERFORMED. THE CAUSE FOR THE VALVES FAILURE TO CLOSE HAS INITIALLY BEEN ATTRIBUTED TO DEGRADATION OF THE VALVE OPERATORS DUE TO CORROSION. STEPS ARE BEING TAKEN TO PREVENT RECURRENCE OF THE CORROSION AND RETURN THE OPERATORS TO SERVICE. MODIFICATIONS TO THE VALVES AND ASSOCIATED OPERATORS AND CHANGES TO

THE VALVE MAINTENANCE PROGRAM ARE BEING CONSIDERED. AN INDEPENDENT NUCLEAR POWER DEPARTMENT TEAM INVESTIGATION WAS ALSO PERFORMED TO ASSESS MSIV PERFORMANCE AND PERCEPTIONS OF VALVE PERFORMANCE.

[97] POINT BEACH 2 DOCKET 50-301 LER 91-003
CONTAINMENT HATCH TEMPORARY THIRD DOOR BLOCKED OPEN DURING FUEL MOVEMENT.
EVENT DATE: 101091 REPORT DATE: 110891 NSSS: WE TYPE: PWR

(NSIC 223360) AT APPROXIMATELY 1250 ON OCTOBER 10, 1991, WITH UNIT 2 IN REFUELING SHUTDOWN WITH FUEL MOVEMENT IN PROGRESS, CONTRACTOR PERSONNEL BLOCKED OPEN THE TEMPORARY THIRD DOOR TO THE PERSONNEL HATCH ON THE CONTAINMENT 66 FOOT ELEVATION. THIS IS IN VIOLATION OF TECHNICAL SPECIFICATION SECTION 15.3.8.1, WHICH STATES, "A TEMPORARY THIRD DOOR ON THE OUTSIDE THE PERSONNEL LOCK SHALL BE IN PLACE WHENEVER BOTH DOORS IN A PERSONNEL LOCK ARE OPEN." THE DOOR REMAINED OPEN UNTIL APPROXIMATELY 1325 WHEN IT WAS SHUT BY A PLANT EMPLOYEE WHO RECOGNIZED THAT THE DOOR WAS OPEN IN VIOLATION OF TECHNICAL SPECIFICATIONS.

[98] POINT BEACH 2 DOCKET 50-301 LER 91-002
STEAM GENERATOR TUBE DEGRADATION.
EVENT DATE: 101491 REPORT DATE: 110891 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 223359) UNIT 2 WAS SHUT DOWN FOR REFUELING 17 (U2R17) ON SEPTEMBER 27, 1991. LEAK TESTING AND EDDY CURRENT EXAMINATION OF THE STEAM GENERATOR TUBES BEGAN ON OCTOBER 4, 1991 AND WERE COMPLETED ON OCTOBER 12, 1991. INSPECTION OF THE "A" STEAM GENERATOR HOT LEG REVEALED 5 TUBES DEGRADED GREATER THAN OR EQUAL TO 40% OF THE WALL THICKNESS, 30 TUBES WITH AXIAL INDICATIONS IN THE TUBE-END AREA, 12 TUBES WITH SQUIRREL INDICATIONS, AND 2 TUBES WITH DISTORTED ROLL INDICATIONS. ALL OF THESE TUBES (49 TOTAL) WERE PLUGGED. IN THE "B" STEAM GENERATOR, EDDY CURRENT TESTING REVEALED 5 TUBES (4 HOT LEG, 1 COLD LEG) DEGRADED GREATER THAN OR EQUAL TO 40% OF THE WALL THICKNESS, 20 TUBES WITH AXIAL INDICATIONS IN THE TUBE-END AREA, AND 4 TUBES WITH SQUIRREL INDICATIONS. ALL OF THESE TUBES (29 TOTAL) WERE PLUGGED. THE RESULTS OF THE 800 PSID LEAK TEST WERE SATISFACTORY. THE "A" STEAM GENERATOR REVEALED 5 SLEEVES, 6 PLUGS, AND ONE OPEN TUBE LEAKING LESS THAN 3 DROPS/MIN. THE "B" STEAM GENERATOR REVEALED 12 PLUGS AND 8 OPEN WERE LEAKING LESS THAN OR EQUAL TO 3 DROP/MIN. NO CORRECTIVE ACTIONS WERE REQUIRED AS A RESULT OF THIS TEST.

[99] QUAD CITIES 1 DOCKET 50-254 LER 90 026 REV 01
UPDATE ON CONTROL ROOM ISOLATION ON HIGH TOXIC GAS CONCENTRATION DUE TO THE ERASABLE/PROGRAMMABLE READ-ONLY MEMORY NOT BEING COMPATIBLE WITH THE SOFTWARE.
EVENT DATE: 122090 REPORT DATE: 101591 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 223259) ON 12/20/90, AT 2306 HOURS, UNIT ONE WAS IN THE SHUTDOWN MODE AT 0% POWER AND UNIT TWO WAS IN THE RUN MODE AT 100% OF RATED CORE THERMAL POWER. ALARM, CONTROL ROOM STANDBY HVAC SYSTEM MAJOR TROUBLE, ANNUNCIATED AT THIS TIME. THE CONTROL ROOM VENTILATION SYSTEM (HVAC) ISOLATED ON HIGH CHLORINE GAS CONCENTRATION. THIS RESULTED IN A CONTROL ROOM HVAC ENGINEERED SAFETY FEATURE (ESF) ACTUATION. THE INSTRUMENT MAINTENANCE (IM) DEPARTMENT REFILLED THE CHLORINE PROBE WITH ELECTROLYTIC SOLUTION WHEN IT WAS DISCOVERED THAT THE PROBE HAD DRIED OUT. AN EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE NOTIFICATION WAS COMPLETED AT 0114 HOURS ON 12/21/90, AS REQUIRED BY 10CFR50.72(B)(2)(II). ON 12/22/90 AT 1800 HOURS THE CONTROL ROOM VENT TOXIC GAS MONITOR WAS DELCARED OPERABLE AGAIN. THE CAUSE OF THIS EVENT WAS THE ERASABLE/PROGRAMMABLE READ ONLY MEMORY (EPROM) WAS NOT COMPATIBLE WITH CURRENT SOFTWARE. INITIAL CORRECTIVE ACTION WAS TO REDUCE SYSTEM AIR FLOW AS RECOMMENDED BY THE MANUFACTURER. THE MANUFACTURER COMPLETED AN INSPECTION OF THE SYSTEM, AND THE STATION HAS UPDATED THE EPROM VIA MINOR DESIGN CHANGE P04-0-91-01B PER THE MANUFACTURERS RECOMMENDATION. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV).

[100] QUAD CITIES 1 DOCKET 50-254 LER 91-013
THREE OF TEN CONTROL ROOM SMOKE DETECTORS OUT OF SERVICE FOR GREATER THAN 14 DAYS.
EVENT DATE: 051491 REPORT DATE: 061491 NSSS: GE TYPE: BWR
VENDOR: ALLEN-BRADLEY CO.

(NSIC 223146) AT 0630, ON 5/14/91, QUAD CITIES UNITS ONE AND TWO WERE BOTH IN THE RUN MODE AT 100% CORE THERMAL POWER. AT THIS TIME THE CONTROL ROOM (CR) HEATING, VENTILATION, AND AIR CONDITIONING (HVAC) FIRE DETECTION SYSTEM EXCEEDED THE 14 DAY REPORTING CRITERIA DUE TO BEING OUT OF SERVICE FOR CORRECTIVE MAINTENANCE. PREVIOUSLY, ON 4/30/91, THE CR HVAC FIRE DETECTION SYSTEM WAS TAKEN OUT OF SERVICE TO REPAIR A RELAY, PER WORK REQUEST 092384, THAT PLACES THE CR HVAC IN RECIRCULATION MODE. THIS FAULT WAS IDENTIFIED DURING THE SURVEILLANCE OF SMOKE DETECTOR 1/2-4160-003, USING QUAD CITIES INSTRUMENT SURVEILLANCE (QIS) 57-1 CR HVAC FIRE PROTECTION SURVEILLANCE PROCEDU[illegible]E 14 DAY CRITERIA WAS EXCEEDED DUE TO THE FAULTY RELAY, AND DELAYS IN CO[illegible]NANCE. CORRECTIVE ACTION FOR THE EVENT WAS TO REPLACE THE FAULTY R[illegible]IN[illegible]TE AN INTERNAL INVESTIGATION TO DETERINE THE ROOT CAUSE [illegible]D[illegible] DELAYED CORRECTIVE MAINTENANCE PER ERROR FREE INVESTIGATION 9[illegible]RT IS SUBMITTED TO COMPLY WITH TECHNICAL SPECIFICATIONS 3.12.A.[illegible] 3.A.[illegible].

[101] QUAD CITIES 1 DOCKET 50-254 LER 91-020
MISSED OFF GAS RECOMBINER TECHNICAL SPECIFICATION SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 091991 REPORT DATE: 110691 NSSS: GE TYPE: BWR

(NSIC 223350) ON SEPTEMBER 19, 1991, UNIT ONE WAS IN THE RUN MODE AT 100% OF RATED CORE THERMAL POWER. THE SHIFT THREE NUCLEAR STATION OPERATOR (NSO) NOTICED THAT THE RECOMBINER TEMPERATURES HAD NOT BEEN VERIFIED TO BE IN THE ALLOWABLE BAND AS REQUIRED BY TECHNICAL SPECIFICATION 4.8.A.5 SINCE SHIFT THREE THE PREVIOUS DAY. THE RECOMBINER TEMPERATURE RECORDER WAS NOT WORKING PROPERLY DURING SHIFTS ONE AND TWO ON SEPTEMBER 19, 1991. THE SHIFT ONE NSO TRIED TO INTERPOLATE THE TEMPERATURE VALUES, AND NOTIFIED THE SHIFT ENGINEER OF THE MALFUNCTIONING RECORDER. THE SHIFT TWO NSO RECOGNIZED THAT THE RECORDER WAS NOT WORKING, BUT TOOK NO CORRECTIVE ACTIONS. THE TEMPERATURE RECORDER APPEARS TO HAVE BEEN IN STANDBY MODE DURING THIS EVENT. THE RECORDER WAS REPAIRED AND DISCIPLINARY ACTION WAS GIVEN TO THE SHIFT TWO NSO FOR FAILING TO NOTIFY THE SHIFT ENGINEER OF THE PROBLEM. THIS REPORT IS BEING WRITTEN PER 10CFR50.73(A)(2)(I)(B).

[102] QUAD CITIES 1 DOCKET 50-254 LER 91-019
"A" TRAIN CR HVAC EMERGENCY FILTRATION UNIT UNABLE TO ATTAIN PROPER DT ACROSS HEATER AND PROPER DP ACROSS FILTER TRAIN DUE TO IMPROPER EQUIPMENT CONFIGURATION AND PARTIALLY FOULED HEPA FILTERS.
EVENT DATE: 100591 REPORT DATE: 102891 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 223349) ON OCTOBER 5, 1991 AT 1255 HOURS, UNIT ONE WAS IN THE RUN MODE AT 80 PERCENT OF RATED CORE THERMAL POWER. UNIT TWO WAS IN THE SHUTDOWN MODE. AT THIS TIME THE CONTROL ROOM (VI) (CR) "B" TRAIN AIR FILTRATION UNIT (AFU) WAS DECLARED INOPERABLE. DURING SURVEILLANCE TESTING, THE HEATER (EHTR) FAILED TO ATTAIN A 15 DEGREE FAHRENHEIT DIFFERENTIAL TEMPERATURE (DT). ALSO, THE DIFFERENTIAL PRESSURE (DP) ACROSS THE FILTER TRAIN EXCEEDED 6 INCHES OF WATER. AN EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE CALL WAS COMPLETED AT 1358 HOURS PER 10CFR50.72(B)(2)(III)(D). THE APPARENT CAUSES OF THIS EVENT WERE A DESIGN DEFICIENCY INVOLVING THE THERMOWELLS OF THE RESISTANCE TEMPERATURE DEVICE (RTD) USED TO MEASURE DT ACROSS THE HEATER, AND HEPA FILTERS (FLT) THAT WERE PARTIALLY FOULED DUE TO NORMAL USAGE. CORRECTIVE ACTIONS INCLUDED FILLING THE THERMOWELLS WITH THERMALLY CONDUCTIVE OIL AND REPLACING THE HEPA FILTERS. THIS REPORT IS BEING SUBMITTED TO COMPLY WITH 10CFR50.73(A)(2)(V).

[103] QUAD CITIES 2 DOCKET 50-265 LER 91-007 REV 01
UPDATE ON REACTOR WATER LOW LEVEL SCRAM DUE TO 2B FEEDWATER REGULATING VALVE FAILING TO FULLY CLOSE.
EVENT DATE: 071391 REPORT DATE: 091191 NSSS: GE TYPE: BWR

VENDOR: BABCOCK & WILCOX COMPANY

(NSIC 222971) ON JULY 13, 1991 UNIT TWO WAS BEING SHUTDOWN FOR A SCHEDULED WEEKEND OUTAGE WHEN A MANUAL SCRAM WAS INSERTED PER THE SHUTDOWN PROCEDURE. WHILE PERFORMING A NORMAL SCRAM RECOVERY, REACTOR (RCT) WATER LEVEL CHANGES OCCURRED WHICH WERE QUICKER THAN EXPECTED. THE REACTOR HIGH WATER LEVEL TRIP SETPOINT WAS REACHED AND THE FEEDWATER (JB) PUMP (P) TRIPPED. REACTOR WATER CLEANUP (CE) BLOWDOWN WAS INITIATED AND A TURBINE (TRB) BYPASS VALVE (V) WAS OPENED TO DECREASE WATER LEVEL AND ESTABLISH A COOLDOWN. WHEN THE BYPASS VALVE WAS CLOSED TO AVOID AN EXCESSIVE COOLDOWN RATE, REACTOR WATER LEVEL DROPPED RAPIDLY AS THE VOIDS COLLAPSED. A LOW LEVEL REACTOR SCRAM OCCURRED. THE REACTOR FEEDWATER PUMP WAS RESTARTED AND SUBSEQUENTLY TRIPPED ON HIGH LEVEL AGAIN. THE OPERATORS RECOGNIZED A PROBLEM WITH THE 2B FEEDWATER REGULATING VALVE LEAKING THROUGH, ISOLATED THE VALVE, AND CONTINUED WITH A NORMAL SHUTDOWN. THE ROOT CAUSE OF THIS INCIDENT WAS THE LEAKING FEEDWATER REGULATING VALVE WHICH CREATED THE HIGH LEVEL CONDITIONS. THE MAINTENANCE PROCEDURE USED FOR THE REPACKING OF VALVES IS TO BE REVISED TO INCLUDE ADDITIONAL DETAIL AND TESTING REQUIREMENTS.

[104] QUAD CITIES 2 DOCKET 50-265 LER 91-013
ENTERING ECONOMIC GENERATION CONTROL WITHOUT PERFORMING REQUIRED SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 081991 REPORT DATE: 111491 NSSS: GE TYPE: BWR

(NSIC 223418) ON 8/18/91, QUAD CITIES UNIT TWO WAS IN THE RUN MODE AT 95% OF RATED CORE THERMAL POWER. AT 0730 HOURS, THE NUCLEAR STATION OPERATOR (NSO) PLACED THE UNIT INTO ECONOMIC GENERATION CONTROL (EGC) WITHOUT PERFORMING THE REQUIRED SURVEILLANCES TO COMPLY WITH TECHNICAL SPECIFICATION 4.3.F. AN INVESTIGATION REVEALED THAT THE CAUSE FOR THIS EVENT WAS PERSONNEL ERROR. THE NSO DID NOT FOLLOW QUAD CITIES STATION OPERATING PROCEDURES FOR EGC OPERATION. UPON IDENTIFICATION OF THE ERROR, THE SHIFT CONTROL ROOM ENGINEER (SCRE) INSTRUCTED THE NSO TO TRIP THE UNIT OUT OF EGC. SUBSEQUENT ACTIONS WILL INCLUDE TRAINING ON EGC OPERATION FOR ALL THE NSOS, AS WELL AS ADDITIONAL TRAINING FOR THE NSO WHO WAS INVOLVED IN THE EVENT. THIS REPORT IS PROVIDED TO SATISFY THE REQUIREMENTS OF 10CFR50.73(A)(2)(I)(B).

[105] QUAD CITIES 2 DOCKET 50-265 LER 91-012
INADVERTENT OPENING OF ELECTROMATIC RELIEF VALVE 2-203-3C DUE TO BINDING IN THE PILOT VALVE ASSEMBLY.
EVENT DATE: 100791 REPORT DATE: 110591 NSSS: GE TYPE: BWR
VENDOR: DRESSER INDUSTRIES, INC.

(NSIC 223352) AT 0223 HOURS ON OCTOBER 7TH, 1991, UNIT TWO WAS IN THE STARTUP MODE AT 1 PERCENT RATED REACTOR CORE THERMAL POWER. WHEN THE REACTOR (RCT) (RPV) REACHED APPROXIMATELY 63 PSIG, A SPURIOUS ACTUATION OF ELECTROMATIC RELIEF VALVE (ERV) (RV) 2-203-3C OCCURRED. AT 0253 HOURS, UNIT TWO WAS MANUALLY SCRAMMED AND COOLDOWN WAS INITIATED. LATER, WHEN A DRYWELL ENTRY WAS MADE, THE PILOT VALVE (V) FOR THE 2-203-3C WAS FOUND STUCK OPEN. THE CAUSE OF THIS EVENT WAS BINDING OF THE PILOT VALVE FOR THE 2-203-3C ERV. THE MECHANICAL MAINTENANCE DEPARTMENT REPLACED THE PILOT VALVE INTERNALS FOR EACH ERV AS A PRECAUTIONARY MEASURE. ON OCTOBER 7TH 1991, AT 1912 HOURS, THE ERV'S WERE STROKED MANUALLY AND TESTED SUCCESSFULLY. UNIT TWO STARTUP WAS INITIATED AT 2054 HOURS ON OCTOBER 7TH, 1991. QCOS 203-3, MAIN STEAM VALVES OPERABILITY TEST, WAS PERFORMED ON OCTOBER 8TH, 1991 AT 1020 HOURS, AND THE ERV DECLARED OPERABLE. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV).

[106] RIVERBEND 1 DOCKET 50-458 LER 91-003 REV 02
UPDATE ON DAMPER ISOLATIONS AND AUTOMATIC SWAP OF DIVISIONAL CONTROL BUILDING VENTILATION/CHILLER TRAINS DUE TO INADEQUATE WORK PLAN.
EVENT DATE: 032291 REPORT DATE: 102991 NSSS: GE TYPE: BWR

(NSIC 223384) AT 1055 ON 3/22/91, DURING MAINTENANCE ON THE DIVISION II CONTROL BUILDING LOCAL AIR INTAKE RADIATION MONITOR 1RMS*RE13B, THE DIVISION II CONTROL POWER CIRCUIT WAS DE-ENERGIZED. THIS RESULTED IN THE DE-ENERGIZATION OF THE

DIVISION II CHARCOAL FILTER TRAIN SUCTION DAMPERS 1HVC*AOD19D AND 1HVC*AOD19F, AND ISOLATION OF THE AIR OPERATED DAMPERS (AODS) TO THE DIVISION II AIR HANDLING UNITS, 1HVC*AOD6B AND 1HVC*AOD8B. NOTE THAT DAMPERS 19D AND 19F WERE CLOSED AT THE TIME OF THE EVENT. THE ISOLATIONS RESULTED IN A TRIP OF THE DIVISION II CONTROL BUILDING VENTILATION SYSTEM/CHILLER AND AUTOMATIC SWAP TO THE DIVISION I VENTILATION SYSTEM/CHILLER. THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73 TO DOCUMENT THE ENGINEERED SAFETY FEATURE (ESF) ACTUATIONS DESCRIBED ABOVE. THE EVENT OCCURRED DURING THE IMPLEMENTATION OF MODIFICATION REQUEST (MR) 90-0071. THIS MR SPECIFIED THAT THE RM-80 MOTHER BOARD WAS TO BE REMOVED FROM 1RMS*RE13B. THE ROOT CAUSE OF THIS EVENT IS THAT THE MAINTENANCE PLANNER OVERLOOKED THE 115 VAC CONTROL POWER TO THE RM-80 MOTHER BOARD AND THUS, THE POTENTIAL FOR THE ESF ACTUATIONS. THIS EVENT CONCERNED THE ENGINEERING/MAINTENANCE PLANNING INTERFACE AND RESPONSIBILITY. AS PREVIOUSLY REPORTED IN LER 90-033, REVISION 3, A TASK FORCE EVALUATION OF THIS ISSUE WAS PERFORMED AND THE TASK FORCE RECOMMENDATIONS HAVE BEEN APPROVED BY MANAGEMENT.

[107] RIVERBEND 1 DOCKET 50-458 LER 91-014 REV 01
UPDATE ON ENGINEERED SAFETY FEATURE ACTUATIONS DUE TO RELAY MALFUNCTIONS HAVING THE SAME FAILURE MODE.
EVENT DATE: 071991 REPORT DATE: 111391 NSSS: GE TYPE: BWR
VENDOR: POTTER & BRUMFIELD

(NSIC 223404) ON 07/19/91 AT 2028 AND ON 07/23/91 AT 2110, WITH THE PLANT AT 100 PERCENT POWER (OPERATIONAL CONDITION 1) IN EACH CASE, ENGINEERED SAFETY FEATURE (ESF) ACTUATIONS OCCURRED DUE TO RELAY MALFUNCTIONS HAVING THE SAME FAILURE MODE. ON 07/19/91, THE FAILURE OF RELAY 1C71*K45D RESULTED IN THE ISOLATION OF NUMEROUS BALANCE-OF-PLANT (BOP) CONTAINMENT ISOLATION VALVES, AND ACTUATION OF THE CONTROL ROOM FILTER TRAINS, THE STANDBY GAS TREATMENT SYSTEM, AND THE FUEL BUILDING FILTER TRAINS. ON 07/23/91, THE FAILURE OF RELAY 1B21H*K204C RESULTED IN THE ISOLATION OF VALVE 1B33*MOVF019, REACTOR WATER UPSTREAM SAMPLE VALVE. THIS VALVE IS A CONTAINMENT ISOLATION VALVE. THIS EVENT IS REPORTED PURSUANT TO 10CFR50.73(A)(2)(IV) TO DOCUMENT THE ESF ACTUATIONS. THE IMMEDIATE CORRECTIVE ACTION WAS TO RESTORE THE ESF SYSTEMS AND CLEAR THE ISOLATIONS. SUBSEQUENT ACTION WAS TO REPLACE BOTH SUSPECT RELAYS, 1C71A*K45D AND 1B21H*K204C. GSU HAS COMPLETED ITS EVALUATION AND IS SUBMITTING THIS SUPPLEMENTAL REPORT TO DOCUMENT THE RESULTS AND CORRECTIVE ACTIONS. GSU'S SAFETY ASSESSMENT INDICATES THAT THE FAILURE RATES OF THESE RELAYS RESULT IN AN INCREASED PROBABILITY OF RPS FAILURES. HOWEVER, THE IMPACT ON CORE DAMAGE FREQUENCY DUE TO ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) EVENTS IS INSIGNIFICANT.

[108] RIVERBEND 1 DOCKET 50-458 LER 91-018
ENGINEERED SAFETY FEATURE ACTUATION WHEN REACTOR WATER CLEANUP ISOLATION VALVE STROKED CLOSED, CAUSE UNKNOWN BUT MAY HAVE BEEN OPERATOR ERROR.
EVENT DATE: 092991 REPORT DATE: 102891 NSSS: GE TYPE: BWR

(NSIC 2[illegible]303) AT APPROXIMATELY 1654 ON 09/29/91 WITH THE UNIT IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) FOR THE PLANNED MIDCYCLE MAINTENANCE OUTAGE, AN UNANTICIPATED ENGINEERED SAFETY FEATURE (ESF) ACTUATION OCCURRED WHEN VALVE G33*MOVF004, OUTBOARD REACTOR WATER CLEANUP SYSTEM (RWCU) ISOLATION VALVE STROKED CLOSED. PER DESIGN, THE LOW FLOW CONDITION CAUSED BY THE CLOSURE OF THIS VALVE CAUSED BOTH OF THE RWCU PUMPS TO TRIP. APPROXIMATELY FOUR MINUTES LATER, A SECOND ESF ACTUATION OCCURRED WHILE OPERATORS WERE ATTEMPTING TO RESTART THE REACTOR WATER CLEANUP SYSTEM PUMPS. DUE TO A PROBLEM WITH THE PROGRAMMABLE CONTROLLER WHICH CONTROLS THE FILTER/DEMINERALIZER PORTION OF THE SYSTEM, FLOW OSCILLATIONS OCCURRED UPON STARTING THE FIRST RWCU PUMP. AFTER 45 SECONDS, THE SENSED FLOW DIFFERENTIAL RESULTED IN ALL OF THE RWCU ISOLATION VALVES CLOSING. THE ROOT CAUSE OF THIS EVENT IS INDETERMINATE. IT IS POSSIBLE, HOWEVER, THAT THE OPERATOR DID NOT DEPRESS THE RESET PUSHBUTTONS HARD ENOUGH OR LONG ENOUGH TO ENSURE ALL OF THE ISOLATION RELAYS HAD TIME TO RESET. INDICATING LIGHTS WILL BE INSTALLED ON THE OPERATOR PANEL ADJACENT TO THE RESET PUSHBUTTONS THAT WILL INDICATE IF AN ISOLATION SIGNAL IS PRESENT TO A PORTION OF THE NSSSS LOGIC. THESE TWO RELATED EVENTS ARE BEING REPORTED PER 10CFR50.73(A)(2)(IV). THIS EVENT DID NOT AFFECT THE HEALTH OR SAFETY OF THE PUBLIC .

[109] SALEM 1 DOCKET 50-272 LER 91-022
TWO ENGINEERED SAFETY FEATURE ACTUATIONS, 1B AND 1C 4KV VITAL BUS UNDERVOLTAGE.
EVENT DATE: 060691 REPORT DATE: 070891 NSSS: WE TYPE: PWR

(NSIC 223199) THIS LER ADDRESSES 2 ENGINEERED SAFETY FEATURE (ESF) SIGNAL ACTUATION EVENTS. BOTH INVOLVE SAFEGUARD EQUIPMENT CONTROL (SEC) SYSTEM OPERATION "MODE OP" ON A SINGLE 4 KV VITAL BUS DUE TO UNDERVOLTAGE. THE FIRST EVENT OCCURRED ON 6/6/91 AND INVOLVED 1C 4KV VITAL BUS. THE SECOND EVENT OCCURRED ON 6/13/91 AND INVOLVED 1B 4KV VITAL BUS. AT THE TIME OF BOTH EVENTS, AN UNDERVOLTAGE FUNCTIONAL SURVEILLANCE WAS BEING CONDUCTED. THE ROOT CAUSE OF THE 1C 4KV EVENT IS INADEQUATE HUMAN FACTORS DESIGN. THE INITIATING CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THE TECHNICIAN DID NOT MAINTAIN ADEQUATE ATTENTION TO DETAIL WHILE POSITIONING HIMSELF FOR PLACEMENT OF A TEMPORARY ELECTRICAL JUMPER. THE ROOT CAUSE OF THE 1B 4KV VITAL BUS EVENT IS PERSONNEL ERROR. THE TECHNICIAN PERFORMING UNDERVOLTAGE RELAY FUNCTIONAL TESTING DID NOT FULLY COMPLY WITH THE PROCEDURE RESULTING IN THE BLOWING OF A PT FUSE AND THE SUBSEQUENT ESF ACTUATION. THE 1C 4KV VITAL BUS AND THE 1B 4KV VITAL BUS EVENTS WILL BE REVIEWED WITH APPLICABLE SUPERVISORS AND TECHNICIANS TO MAKE THEM AWARE OF THE DETAILS SURROUNDING THESE EVENTS.

[110] SALEM 1 DOCKET 50-272 LER 91-030
BOTH PRESSURIZER PRESSURE OPERATED RELIEF VALVES FAILED AN OPERABILITY CHECK.
EVENT DATE: 092091 REPORT DATE: 101891 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)
VENDOR: COPES-VULCAN, INC.

(NSIC 223276) ON 9/20/91, A PLANT SHUTDOWN WAS IN PROGRESS. THE PRESSURIZER PRESSURE OPERATED RELIEF VALVES (PORVS) ARE USED TO PROVIDE OVERPRESSURE PROTECTION AT LOW REACTOR COOLANT SYSTEM TEMPERATURES. IN ACCORDANCE WITH SURVEILLANCE 4.4.9.3.1.1 THE PORVS WERE FUNCTIONALLY CHECKED. BOTH VALVES FAILED TO OPEN. THE 1PR1 VALVE DID NOT LOSE ITS CLOSED LIMIT INDICATION AND THOUGH THE 1PR2 VALVE INDICATED MOVEMENT, IT APPARENTLY DID NOT REACH ITS FULL OPEN LIMIT. INVESTIGATION SHOWED THAT BOTH THE 1PR1 AND 1PR2 VALVE ACTUATORS LEAKED. THE VALVE ACTUATOR DIAPHRAGM BOLTS WERE OBSERVED TO BE LOOSE ALLOWING AIR LEAKAGE FROM THE FLANGE BOLTING AREA. THE ROOT CAUSE OF THE VALVES FAILING TO OPEN IS CONTINUING. THE 1PR1 AND 1PR2 VALVES ARE COPES-VULCAN REVERSE ACTING AIR ACTUATED GLOBE VALVES. INVESTIGATION INDICATED THAT THE DIAPHRAGMS APPEARED TO BE IN A FUNCTIONAL CONDITION. FURTHER ASSESSMENT HAD SHOWN THAT THE DIAPHRAGM MATERIAL (BUNA-N RUBBER) THIS PHENOMENA CAN BE EXACERBATED BY UNEVEN TORQUING. WHEN THE DIAPHRAGMS WERE INSTALLED (APRIL 1991), THE SPECIFIC INSTALLATION PROCEDURE WAS NOT USED. THE ADMINISTRATIVE CONTROLS FOR IMPLEMENTING NEW MAINTENANCE PROCEDURES WILL BE MODIFIED. A REVIEW OF NEW PROCEDURES WILL BE CONDUCTED TO ENSURE THEY HAVE BEEN INCORPORATED INTO EXISTING PLANNED WORK. THE DIAPHRAGMS WERE REPLACED IN ALL COPES-VULCAN ACTUATORS LOCATED I NSIDE THE UNIT 1 PRESSURIZER.

[111] SALEM 1 DOCKET 50-272 LER 91-031
MAIN STEAM LINE ISOLATION SIGNAL DUE TO DESIGN CONCERN.
EVENT DATE: 092391 REPORT DATE: 101891 NSSS: WE TYPE: PWR

(NSIC 223272) ON 9/23/91 AT 1414 HOURS, A MAIN STEAM ISOLATION (MSI) ACTUATION OCCURRED. AT THE TIME, THE UNIT WAS IN MODE 4 AND HEATING UP IN PREPARATION FOR STARTUP. THE MSI SIGNAL OCCURRED UPON RECEIPT OF A HIGH STEAMLINE FLOW SIGNAL COINCIDENT WITH A LOW STEAMLINE PRESSURE SIGNAL. IN MODE 4, THE BISTABLES FOR LOW STEAMLINE PRESSURE ARE TRIPPED PROVIDING HALF THE LOGIC SIGNAL REQUIRED FOR MSI. AT THE TIME OF THE EVENT, THE NO. 14 STEAM GENERATOR (S/G) STEAMLINE FLOW CHANNEL NO. I BISTABLE WAS IN THE TRIPPED POSITION TO SUPPORT CHANNEL FUNCTIONAL TESTING. WHEN NO. 11 S/G STEAMLINE FLOW CHANNELS I AND II BISTABLES TRIPPED THE MSI SIGNAL LOGIC WAS SATISFIED. MSI IS AN ENGINEERED SAFETY FEATURE. SIMILAR MSI EVENTS HAVE OCCURRED (REFERENCE LERS 272/90-019-00 AND 272/90-027-00). THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO DESIGN INADEQUACY ASSOCIATED WITH THE MAIN STEAMLINE FLOW TRANSMITTER SENSING LINES DURING PLANT HEATUP. ASSESSMENT OF THIS EVENT, BY MAINTENANCE PERSONNEL, WAS THAT THE FALSE HIGH STEAM FLOW SIGNALS WERE NOT CAUSED BY FAILED COMPONENTS. THE FALSE SIGNALS CLEARED AFTER A FEW HOURS. AT THE TIME OF THE EVENT, MAINTENANCE WAS PERFORMING CHANNEL MONTHLY FUNCTIONAL TESTING AS PER

TECH. SPEC. SURVEILLANCES 4.3.1.1.1 AND 4.3.2.1.1. THE CHANNEL I AND CHANNEL II FLOW TRANSMITTERS FUNCTIONAL TESTING, FOR ALL 4 STEAMLINES, WAS SUCCESSFULLY COMPLETED ON 9/23/91 AND 9/28/91, RESPECTIVELY.

[112] SALEM 2 DOCKET 50-311 LER 91-014
CONTAINMENT VENTILATION ISOLATION DUE TO 2R12B RADIATION MONITORING SYSTEM CHANNEL FAILURE.
EVENT DATE: 101091 REPORT DATE: 110791 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 223429) ON OCTOBER 10, 1991, AT 0138 HOURS, THE 2R12B (CONTAINMENT RADIOACTIVE IODINE MONITOR) RADIATION MONITORING SYSTEM (RMS) 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. AT THE TIME OF THE EVENT THE CP/P-VR SYSTEM VALVES WERE CLOSED. THE ROOT CAUSE OF THIS EVENT IS EQUIPMENT DESIGN. THE 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, AND 311/91-007-00). THE 2R12B DETECTOR USES A VICTOREEN MODEL 843-34 SCINTILLATOR. THE VICTOREEN RMS CHANNELS ARE SUBJECT TO VOLTAGE TRANSIENTS. AN ENGINEERING ASSESSMENT WAS DONE TO IDENTIFY POSSIBLE CAUSES OF THE MOMENTARY CHANNEL FAILURE. A FUNCTIONAL TEST WAS CONDUCTED WHICH CHECKED THESE POSSIBLE CAUSES. THE FUNCTIONAL TEST WAS SUCCESSFULLY COMPLETED WITHOUT IDENTIFYING A COMPONENT FAILURE. THE CHANNEL WAS OBSERVED FOR TWO (2) WEEKS, WITH THE OUTPUT FUNCTIONS BLOCKED, WITHOUT RECURRENCE OF A CHANNEL FAILURE. SUBSEQUENTLY, THE CHANNEL WAS RETURNED TO SERVICE. AS INDICATED IN LER 311/90-040-00, ENGINEERING HAS INVESTIGATED THE CONCERNS IDENTIFIED WITH THE VICTOREEN RMS CHANNELS.

[113] SALEM 2 DOCKET 50-311 LER 91-015
POWER RANGE NUCLEAR INSTRUMENTATION SYSTEM CHANNELS INOPERABLE.
EVENT DATE: 102191 REPORT DATE: 111991 NSSS: WE TYPE: PWR

(NSIC 223430) ON 10/21/91, UNIT LOAD WAS BEING INCREASED. AS REQUIRED BY TECH. SPEC. 3.3.1.1 TABLE 4.3-1, THE REQUIRED DAILY HEAT BALANCE CHECK WAS PERFORMED. THE POWER INCREASE WAS STOPPED AND STABILIZED AT A CALCULATED RATED THERMAL POWER (RTP) OF 29.8%. THE 4 NUCLEAR INSTRUMENTATION SYSTEM (NIS) POWER RANGE CHANNELS INDICATED LESS THAN 29.8% RTP. CHANNELS 2N41 AND 2N44 SATISFIED THE OPERABILITY CRITERIA (WITHIN 1.74% OF RTP). CHANNELS 2N43 AND 2N44 DID NOT MEET THE OPERABILITY CRITERIA. THEY WERE DECLARED INOPERABLE AT 0400 HOURS ON 10/21/91. SINCE TECHNICAL SPECIFICATION TABLE 3.3-1 ACTION STATEMENTS FOR INOPERABLE POWER RANGE CHANNELS DOES NOT ADDRESS INOPERABILITY OF TWO (2) OR MORE CHANNELS, TECHNICAL SPECIFICATION ACTION 3.0.3 WAS ENTERED. THIS EVENT WAS CAUSED BY THE APPLICATION OF THE CURRENT METHODOLOGY FOR DEFINING WHEN SPECIFIC NIS OPERABILITY CRITERIA MUST BE CONSIDERED. THIS RESULTED IN OVERLY CONSERVATIVE REQUIREMENTS FOR NIS OPERABILITY CRITERIA DURING INFREQUENT OPERATION AT LOW POWER LEVELS AND DEEP CONTROL ROD INSERTIONS. THE FINE GAIN POTENTIOMETERS OF ALL FOUR (4) POWER RANGE NIS CHANNELS WERE ADJUSTED TO 30 RTP AT 0425 HOURS ON 10/21/91 AT WHICH TIME TECHNICAL SPECIFICATION ACTION 3.0.3 WAS EXITED. REACTOR ENGINEERING PERSONNEL VALIDATED THE CALORIMETRIC CALCULATIONS AS TO THE DETERMINATION OF POWER LEVEL AND VERIFIED THE APPROPRIATE APPLICATION OF THE PROCEDURES.

[114] SAN ONOFRE 1 DOCKET 50-206 LER 91-018
DELINQUENT TECHNICAL SPECIFICATION SURVEILLANCES DUE TO INADEQUATE ADMINISTRATIVE CONTROLS.
EVENT DATE: 102191 REPORT DATE: 112091 NSSS: WE TYPE: PWR

(NSIC 223414) ON 10/21/91 WITH UNIT 1 IN MODE 3 FOLLOWING A REACTOR TRIP ON 10/17/91, IT WAS DETERMINED THAT BOTH POWER OPERATED RELIEF VALVE (PORV) BLOCK VALVES HAD EXCEEDED THEIR QUARTERLY INSERVICE TESTING (ITS) INTERVAL PLUS THE ALLOWABLE EXTENSION PERMITTED BY TECHNICAL SPECIFICATIONS (TS) ON 10/19/91 AND 10/20/91, RESPECTIVELY. THE VALVES WERE DECLARED INOPERABLE AND THE TS 3.1.5, "PRESSURIZER RELIEF VALVES," ACTION STATEMENT WAS ENTERED. UPON SATISFACTORY

COMPLETION OF THE IST ON 10/21/91, BOTH VALVES WERE DECLARED OPERABLE AND THE TS 3.1.5 ACTION STATEMENT WAS EXITED. THE CAUSE OF THIS EVENT WAS A WEAKNESS IN THE ADMINISTRATIVE CONTROLS FOR THE OPERATIONS IST SURVEILLANCE PROGRAM WHICH RESULTED IN THE FAILURE TO PROVIDE THE CONTROL ROOM WITH A LIST OF VALVES REQUIRING ISTS ON 10/18/91. THE ADMINISTRATIVE CONTROLS WERE NOT SUFFICIENT TO ENSURE THAT THE IST VALVE LIST WOULD BE DELIVERED TO THE CONTROL ROOM IN A TIMELY FASHION. ADDITIONALLY, THESE ADMINISTRATIVE CONTROLS DID NOT INCLUDE PROVISIONS TO ALERT CONTROL ROOM OPERATORS WHEN IST SURVEILLANCE INTERVAL EXTENSIONS ARE USED, THEREFORE REQUIRING MORE ATTENTION THAN THAT GIVEN TO NORMALLY SCHEDULED ISTS. SINCE SURVEILLANCE INTERVAL EXTENSIONS WERE USED IN THIS CASE, THERE WAS LITTLE MARGIN IN THE TEST SCHEDULE TO ACCOUNT FOR UNFORSEEN EVENTS SUCH AS THE 10/17/91 REACTOR TRIP.

[115] SAN ONOFRE 2 DOCKET 50-361 LER 91-015
CONTAINMENT PURGE ISOLATION SYSTEM TRAIN "A" ACTUATION WHILE PERFORMING RADIOGRAPHY.
EVENT DATE: 100191 REPORT DATE: 103091 NSSS: CE TYPE: PWR

(NSIC 223313) ON 10/1/91, AT 0323, WITH UNIT 2 IN MODE 6 AND CONTAINMENT PURGE IN PROGRESS, A CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) (JM) TRAIN "A" ACTUATION OCCURRED DURING RADIOGRAPHY IN THE VICNITY OF CONTAINMENT AREA RADIATION MONITOR 2RT-7856 (MON). CONTROL ROOM OPERATORS (UTILITY, LICENSED) WERE ALERTED TO THE TRAIN "A" CPIS ACTUATION BY CONTROL ROOM ANNUNCIATORS (ANN). OPERATORS RESPONDED PROPERLY TO THE ACTUATION BY 1) VERIFYING THAT ALL CPIS TRAIN "A" COMPONENTS OPERATED IN ACCORDANCE WITH DESIGN AND 2) VERIFYING THAT THE ACTUATION WAS SPURIOUS BEFORE RESETTING CPIS AND RESTORING CONTAINMENT PURGE AT 0357. AT APPROXIMATELY 0240, PRIOR TO THE PERFORMANCE OF RADIOGRAPHY AND IN ACCORDANCE WITH PROCEDURE, HEALTH PHYSICS (HP) TECHNICIANS NOTIFIED THE CONTROL OPERATOR (CO) THAT RADIOGRAPHY WAS TO BE PERFORMED IN CONTAINMENT AND THAT RADIATION MONITORS MAY BE AFFECTED. THE DISCUSSION THAT TOOK PLACE BETWEEN THE HP TECHNICIANS AND THE CO WAS NOT SUFFICIENTLY CLEAR FOR THE CO TO RECOGNIZE THAT THE PROPOSED RADIOGRAPHY COULD CAUSE A CPIS ACTUATION. THE CONTROL OPERATOR DID NOT ANTICIPATE THAT THE RADIOGRAPHY WOULD CAUSE A CPIS ACTUATION; THE ACTUATION IS THEREFORE NOT CONSIDERED PREPLANNED.

[116] SAN ONOFRE 2 DOCKET 50-361 LER 91-016
CONTROL ROOM ISOLATION SYSTEM TRAIN "B" SPURIOUS ACTUATION DUE TO INSTRUMENT FAILURE.
EVENT DATE: 101091 REPORT DATE: 110891 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 223373) AT 1238 ON OCTOBER 10, 1991, WITH UNIT 2 IN A REFUELING OUTAGE AND UNIT 3 AT 100% POWER, A CONTROL ROOM ISOLATION SYSTEM (CRIS) TRAIN B ACTUATION OCCURRED. THE ACTUATION WAS VERIFIED TO BE SPURIOUS, AND ALL REQUIRED COMPONENTS WERE VERIFIED TO HAVE ACTUATED AS DESIGNED. THE CRIS TRAIN B RADIATION MONITOR WAS PLACED IN BYPASS, AND AT 1340, THE CONTROL ROOM VENTILATION LINEUP WAS RETURNED TO NORMAL. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE RADIATION LEVELS REMAINED NORMAL AND ALL CRIS TRAIN B COMPONENTS ACTUATED AS REQUIRED BY THE DESIGN. THE CAUSE OF THE ACTUATION WAS DETERMINED TO BE A MOMENTARY (APPROXIMATELY 48 MSEC) INSTRUMENT FAILURE OF THE CRIS TRAIN B PARTICULATE/IODINE CHANNEL. THE INSTRUMENT FAILURE WAS CAUSED BY A SPURIOUS NOISE SIGNAL EITHER ON THE POWER INPUT TO THE CRIS PARTICULATE/IODINE CHANNEL MODULE OR ON THE SIGNAL PATH FROM THE DETECTOR. A DESIGN CHANGE TO REDUCE THE CRIS RADIATION MONITOR SUSCEPTIBILITY TO NOISE IS BEING PREPARED. AS AN INTERIM CORRECTIVE ACTION TO REDUCE THE SUSCEPTIBILITY OF THE MODULE TO NOISE, A TEMPORARY MODIFICATION HAS BEEN INSTALLED WHICH (1) ADDS A FILTERING CIRCUIT TO THE (AC) INPUT TO THE RADIATION MONITOR MODULE, AND (2) ENHANCES THE FILTERING OF THE INTERNAL (DC) POWER SUPPLY REGULATORS. SIMILAR LER IS 361/91-011.

[117] SAN ONOFRE 2 DOCKET 50-361 LER 91-018
SAFETY RELATED INSTRUMENTATION NOT INSTALLED IN A SEISMICALLY QUALIFIED CONFIGURATION.
EVENT DATE: 101591 REPORT DATE: 111491 NSSS: CE TYPE: PWR

VENDOR: FOXBORO CO., THE

(NSIC 223436) ON OCTOBER 16, 1991, WHILE UNIT 2 WAS IN MODE 5 DURING A REFUELING OUTAGE, AN IN INSPECTION OF FOXBORO SPECIFICATION 200 ANALOG INSTRUMENTATION CABINETS (CAB) DETERMINED THAT SEVERAL INSTRUMENT MODULES WERE MISSING THE UPPER AND/OR LOWER GUIDE RAILS AND/OR VIBRATION DAMPENING MATERIAL. WITHOUT THE GUIDE RAILS AND/OR VIBRATION DAMPENING MATERIAL INSTALLED, THIS EQUIPMENT WOULD NOT CONFORM TO THE VENDOR'S GENERIC SEISMIC DESIGN REQUIREMENTS AND WAS THEREFORE CONSIDERED INOPERABLE, CAUSING THE APPLICABLE TECHNICAL SPECIFICATION LIMITING CONDITION FOR OPERATION FOR THE FOLLOWING AFFECTED INSTRUMENT LOOPS TO HAVE BEEN EXCEEDED DURING PREVIOUS PERIODS OF OPERATION IN MODES 1, 2, 3, AND 4: 1) 2LT-5903-1, DIESEL FUEL STORAGE TANK T035 LEVEL (LT); 2) 2PT-1013-1, STEAM GENERATOR E089 PRESSURE (PT); 3) 2PT-1023-1, STEAM GENERATOR E088 PRESSURE (PT); AND 4) 2LT-9386, 2LT-9387, 2LT-9388, AND 2LT-9389, CONTAINMENT EMERGENCY SUMP LEVEL (LT). ALTHOUGH INSPECTIONS OF THE FOXBORO SPECIFICATION 200 INSTRUMENTATION CABINETS IN UNITS 1 AND 3 IDENTIFIED SIMILAR DEFICIENCIES, NO REPORTABLE CONDITIONS EXISTED. SUBSEQUENT SEISMIC TESTING DEMONSTRATED THAT WITHOUT VIBRATION DAMPENING MATERIAL INSTALLED, THE FOXBORO INSTRUMENT MODULES WOULD HAVE FUNCTIONED PROPERLY DURING AND AFTER THE SONGS DESIGN BASIS SEISMIC EVENTS.

[118] SAN ONOFRE 3 DOCKET 50-362 LER 91-004
DELINQUENT TECHNICAL SPECIFICATION SURVEILLANCES DUE TO PERSONNEL ERROR.
EVENT DATE: 093091 REPORT DATE: 102991 NSSS: CE TYPE: PWR

(NSIC 223312) AT APPROX. 0815 ON 9/30/91, WITH UNIT 3 IN MODE 1 AT 100% POWER, IT WAS DISCOVERED THAT TECH SPEC (TS) SURVEILLANCES REQUIRED TO BE PERFORMED AT LEAST ONCE PER 24-HOURS HAD EXCEEDED THEIR SURVEILLANCE INTERVAL BY 40 MINUTES. AT 2225 ON 9/28/91, SURVEILLANCE OPERATING INSTRUCTION S023-3-3.26 WAS PERFORMED. THE SURVEILLANCE WAS SUBSEQUENTLY PERFORMED AT 0505 ON 9/30/91. THE INTERVAL BETWEEN COMPLETED SURVEILLANCES WAS 40 MINUTES GREATER THAN THE TS SURVEILLANCE INTERVAL PLUS THE ALLOWABLE EXTENSION PERMITTED BY TS 4.0.2. THE CAUSE OF THIS EVENT WAS A COGNITIVE ERROR BY CONTROL ROOM PERSONNEL. AT THE BEGINNING OF THEIR SHIFT, THE CONTROL ROOM OPERATOR AND THE CONTROL ROOM SUPERVISOR REVIEWED THE LIST OF SURVEILLANCES REQUIRED TO BE PERFORMED ON THEIR SHIFT. ALTHOUGH S023-3-3.26 WAS SPECIFIED TO BE PERFORMED PRIOR TO 2400 HOURS, DUE TO A LACK OF ATTENTION TO DETAIL, THIS TIME REQUIREMENT WAS NOT IDENTIFIED BY EITHER THE CONTROL ROOM OPERATOR OR THE CONTROL ROOM SUPERVISOR. AS CORRECTIVE ACTION, APPROPRIATE DISCIPLINARY ACTION HAS BEEN ADMINISTERED TO THE PERSONNEL INVOLVED IN THIS EVENT. IN ADDITION, THIS EVENT HAS BEEN REVIEWED BY APPROPRIATE OPERATIONS PERSONNEL. THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE THE DELINQUENT PERFORMANCE OF S023-3-3.26 DEMONSTRATED THAT APPLICABLE TS COMPONENTS MET THEIR ASSOCIATED TS REQUIREMENTS AND THEREFORE REMAINED OPERABLE.

[119] SEABROOK 1 DOCKET 50-443 LER 91-013
REACTOR PROTECTION SYSTEM ACTUATION DURING ROD POSITION INDICATION 18-MONTH SURVEILLANCE.
EVENT DATE: 100791 REPORT DATE: 110691 NSSS: WE TYPE: PWR

(NSIC 223383) ON OCTOBER 7, 1991, AT 6:45 P.M., EDT, WHILE IN MODE 3, MANUAL REACTOR PROTECTION SYSTEM (RPS) ACTUATION WAS PERFORMED PER TECHNICAL SPECIFICATION 3.1.3.3 WHEN THE DIGITAL ROD POSITION INDICATOR (DRPI) FOR SHUTDOWN BANK C, ROD N5, WAS GREATER THAN PLUS OR MINUS 12 STEPS OF THE POSITION INDICATED ON THE DEMAND COUNTER. A SECOND PLANNED MANUAL ACTUATION OF THE REACTOR PROTECTION SYSTEM WAS PERFORMED AT 9:36 P.M., EDT, DURING THE CONDUCT OF TROUBLESHOOTING ACTIVITIES. THE ROOT CAUSE OF THIS EVENT HAS NOT YET BEEN DETERMINED. NHY BELIEVES THAT THE DRPI SYSTEM OPERATED PROPERLY AND PROVIDED AN ACCURATE INDICATION OF THE POSITION OF ROD N5, WHICH HAD DROPPED DUE TO A MALFUNCTION OF THE ROD CONTROL SYSTEM. NHY WILL IMPLEMENT CORRECTIVE ACTIONS TO ASSIST WITH THE DETERMINATION OF THE ROOT CAUSE OF THE MALFUNCTION OF THE ROD CONTROL SYSTEM. SPECIFICALLY, NHY WILL CONDUCT ADDITIONAL TROUBLESHOOTING OF THE SHUTDOWN BANK C DURING THE MONTHLY DRPI OPERABILITY TEST (OX1410.02) FOR AT LEAST THE NEXT THREE MONTHS UNLESS THE ROOT CAUSE IS DETERMINED PRIOR TO THE END OF THIS PERIOD. ADDITIONAL CORRECTIVE ACTIONS WILL BE FORMULATED IF NECESSARILY. FOLLOWING COMPLETION OF THE ABOVE STATED CORRECTIVE ACTIONS, NHY WILL SUBMIT A

FOLLOWUP REPORT REGARDING THE ROOT CAUSE OF THIS EVENT. IT IS EXPECTED THAT THIS REPORT WILL BE SUBMITTED TO THE NRC BY MARCH 2, 1992.

[120] SOUTH TEXAS 1 DOCKET 50-498 LER 91-012 REV 01
UPDATE ON REACTOR TRIP DUE TO MOTOR GENERATOR MALFUNCTION.
EVENT DATE: 041291 REPORT DATE: 103191 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 223301) ON 4/12/91, AT 0418, THE UNIT 1 REACTOR TRIPPED FROM 40% POWER. A TURBINE TRIP, FEEDWATER ISOLATION AND AUXILIARY FEEDWATER ACTUATION OCCURRED AS A RESULT OF THE REACTOR TRIP. SYSTEMS OPERATED AS DESIGNED IN RESPONSE TO THE REACTOR TRIP. IT WAS DETERMINED THAT ROD DRIVE MOTOR GENERATOR (RDMG) SET #11 TRIPPED DUE TO A TRANSIENT INDUCED BY RDMG #12 WHICH WAS FOUND RUNNING WITH ITS MOTOR AND GENERATOR BREAKERS CLOSED WITH NO OUTPUT VOLTAGE TO THE REACTOR TRIP SWITCHGEAR. IT IS BELIEVED THAT INTERMITTENT PICK-UP AND DROP-OUT OF THE 2R RELAY, WHICH ACTUATES CONTACTS TO SUPPLY POWER TO THE RDMG SET #12'S GENERATOR VOLTAGE REGULATOR, CAUSED INSTABILITY IN THE VOLTAGE REGULATOR OPERATION. THE 2R RELAY MALFUNCTION WAS DUE TO A DEFECTIVE OUTPUT SWITCH. THE INSTABILITY OF THE VOLTAGE REGULATION RESULTED IN TRANSIENTS THAT CAUSED A REVERSE CURRENT TO THE RDMG SET #11 AND A SUBSEQUENT TRIP OF THE GENERATOR OUTPUT BREAKER. IT IS ALSO BELIEVED THAT THE 2R RELAY CONTACTS SUPPLYING POWER TO THE VOLTAGE REGULATOR EVENTUALLY REMAINED OPEN LONG ENOUGH TO ALLOW A LOSS OF THE GENERATOR FIELD IN THE RDMG SET #12. A LOSS OF THE GENERATOR FIELD RESULTS IN ZERO OUTPUT VOLTAGE FROM THE GENERATOR. THE LOSS OF BOTH OF THE POWER SOURCES TO THE REACTOR TRIP SWITCHGEAR RESULTED IN A REACTOR TRIP. THE 2R RELAY'S TIMER AND CONTROL RELAY WERE REPLACED AND A PROCEDURAL CHANGE HAS BEEN MADE TO ENHANCE DETECTION OF MALFUNCTION.

[121] SOUTH TEXAS 1 DOCKET 50-498 LER 91-022
REACTOR TRIP DURING PERFORMANCE OF SOLID STATE PROTECTION SYSTEM LOGIC FUNCTIONAL TEST.
EVENT DATE: 101491 REPORT DATE: 111391 NSSS: WE TYPE: PWR

(NSIC 223453) ON OCTOBER 14, 1991, AT 2304 HOURS, UNIT 1 WAS IN MODE 1 AT 100 POWER. SOLID STATE PROTECTION SYSTEM (SSPS) LOGIC TRAIN R FUNCTIONAL TEST WAS IN PROGRESS WHEN THE LICENSED OPERATOR PERFORMING THE SURVEILLANCE MISUNDERSTOOD THE INTENT OF A NOTE IN THE PROCEDURE AND FAILED TO BLOCK THE TURBINE TRIP SIGNAL BEFORE PROCEEDING TO THE NEXT STEP. THE "MEMORIES" TEST SWITCH WAS PLACED IN POSITION 16 AND AN AUTOMATIC TRAIN R TRIP SIGNAL WAS GENERATED. TRAIN R TRIP SIGNAL GENERATED A "TURBINE TRIP UPON REACTOR TRIP" SIGNAL WHICH HAD NOT BEEN BLOCKED AND THE "MEMORIES" TEST SWITCH ALSO MALFUNCTIONED, WHICH IF IT HAD FUNCTIONED PROPERLY SHOULD HAVE ALSO BLOCKED THE TRIP SIGNAL. SUBSEQUENTLY, THE MAIN TURBINE TRIPPED AND, COINCIDENT WITH A "REACTOR POWER ABOVE 50" SIGNAL, A VALID TRAINS REACTOR TRIP SIGNAL WAS GENERATED TRIPPING THE REACTOR. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR BY THE LICENSED OPERATOR WHO EXERCISED POOR JUDGEMENT WHILE PERFORMING THE TEST. CONTRIBUTING FACTORS WERE A LESS THAN IDEAL PROCEDURE AND THE MALFUNCTION OF THE "MEMORIES" TEST SWITCH. CORRECTIVE ACTIONS INCLUDE SITE-WIDE TRAINING SESSIONS FOR APPROPRIATE PLANT PERSONNEL STRESSING THE APPLICATION OF SELF VERIFICATION DURING WORK PERFORMANCE, COUNSELING OF THE LICENSED OPERATOR INVOLVED IN THE EVENT, REVISION OF ALL SP SERIES SURVEILLANCES PERFORMED AT POWER THAT HAVE THE POTENTIAL TO TRIP THE UNIT/MAIN TURBINE.

[122] SOUTH TEXAS 1 DOCKET 50-498 LER 91-024
A SAFETY ANALYSIS DEFICIENCY CONCERNING THE PRESSURIZER SAFETY RELIEF VALVE (SRV) LOOP SEAL DELAY TIME.
EVENT DATE: 103191 REPORT DATE: 112291 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SOUTH TEXAS 2 (PWR)

(NSIC 223450) DURING A REVIEW OF NRC INFORMATION NOTICE 89-90 SUPPLEMENT 1, DATED SEPTEMBER 5, 1991 AND WCAP-12910, WITH UNIT 1 IN MODE 1 AT 100 PERCENT POWER AND UNIT 2 IN A MODE 6 REFUELING OUTAGE, IT WAS DISCOVERED THAT THE UFSAR CHAPTER 15 SAFETY ANALYSIS DID NOT CONSIDER THE TIME REQUIRED TO PURGE THE LOOP SEAL FOR THE PRESSURIZER SAFETY RELIEF VALVES (SRVS). IMMEDIATE ACTIONS TAKEN TO INVESTIGATE

THE PROBLEM CONFIRMED THAT THE CALCULATED PEAK RCS PRESSURE FOR THE LOCKED ROTOR EVENT WITH THE PRESSURIZER SRV LOOP SEAL DELAY TIME WOULD EXCEED THE NRC SAFETY LIMIT OF 110% DESIGN PRESSURE. ON OCTOBER 30, 1991, A STATION PROBLEM REPORT WAS ISSUED IDENTIFYING THE DEFICIENCY IN THE SAFETY ANALYSIS. ON NOVEMBER 5, 1991, A JUSTIFICATION FOR CONTINUED OPERATION (JCO) WAS ISSUED. THE JCO CONCLUDED THAT THE CONDITION DOES NOT RESULT IN EITHER STPEGS UNITS 1 OR 2 BEING IN AN UNSAFE CONDITION. THE CAUSE OF THE EVENT WAS THAT THE NSSS VENDOR DID NOT CONSIDER THE DELAY TIME ASSOCIATED WITH PURGING THE PRESSURIZER SRV LOOP SEALS IN THE SAFETY ANALYSIS. SINCE THE JCO FOR THIS ISSUE IDENTIFIED THAT NO UNSAFE CONDITION EXISTS, NO IMMEDIATE CORRECTIVE ACTIONS ARE PLANNED. AFTER THE NRC APPROVAL OF WCAP-12910 AND WESTINGHOUSE OWNERS GROUP (WOG) RESOLUTION OF THIS ISSUE, ADDITIONAL ACTIONS WILL BE DEVELOPED AS NECESSARY.

[123] SOUTH TEXAS 2 DOCKET 50-499 LER 91-005 REV 01
UPDATE ON ENGINEERED SAFETY FEATURES ACTUATION OF CONTROL ROOM ENVELOPE HVAC TRAINS B AND C DUE TO UNKNOWN CAUSE.
EVENT DATE: 041191 REPORT DATE: 111191 NSSS: WE TYPE: PWR

(NSIC 223454) ON APRIL 11, 1991, UNIT 2 WAS IN MODE 1 AT 100 PERCENT POWER. AT 1130, AN AUTOMATIC ENGINEERED SAFETY FEATURES (ESF) ACTUATION OF CRE HVAC TRAINS B AND C TO EMERGENCY MODE OCCURRED. CONTROL ROOM ENVELOPE (CRE) HVAC TRAIN A HAD BEEN MANUALLY ACTUATED TO THE EMERGENCY MODE IN SUPPORT OF A SURVEILLANCE PROCEDURE. NO INDICATION OF A HIGH RADIATION OR SAFETY INJECTION SIGNAL WAS FOUND. THERE HAS BEEN NO CAUSE ESTABLISHED FOR THIS ACTUATION.

[124] SUMMER 1 DOCKET 50-395 LER 91-009
VENDOR CONCERNS LEAD TO DISCOVERY ECCS FLOW DISCREPANCIES.
EVENT DATE: 101591 REPORT DATE: 111491 NSSS: WE TYPE: PWR

(NSIC 223449) ON SEPTEMBER 17, 1991, SOUTH CAROLINA ELECTRIC & GAS COMPANY (SCE&G) WAS NOTIFIED BY WESTINGHOUSE OF A DEVELOPING ISSUE BASED ON A VENDOR RECOMMENDATION TO LIMIT ITS PREVIOUSLY APPROVED PUMP RUNOUT LIMIT OF 680 GPM TO A NEW LIMIT OF 675 GPM. THIS NEW LIMIT IS RELATED TO SEVERAL ISSUES THAT COULD AFFECT THE RUNOUT MARGIN AVAILABLE FOR THE CHG/SI PUMPS. BASED ON THESE ISSUES, IT WAS DETERMINED THAT SURVEILLANCE TEST PROCEDURE (STP) 230.006, VALVE/CHARGING PUMP OPERABILITY TESTING, SHOULD BE REVISED TO ADDRESS THESE ISSUES AND PERFORMED. THE FOLLOWING DEFICIENCY WAS FOUND TO CONSTITUTE OPERATION OUTSIDE THE PROVISIONS OF TECHNICAL SPECIFICATION 3.5.2 AND THUS IS REPORTABLE PER 10CFR50.73(A)(2)(I): FOR THE HOT LEG RECIRCULATION PATH, ALL THREE PUMPS WOULD EXCEED THE SURVEILLANCE REQUIREMENT RUNOUT LIMIT OF 680 GPM. IT WAS ALSO NOTED THAT THE B PUMP PERFORMANCE WAS BELOW THE MINIMUM PUMP CURVE ASSUMED IN THE FSAR ACCIDENT ANALYSIS FOR THE HIGH FLOW REGION OF THE CURVE. THE CAUSE OF THE CHG/SI PUMPS EXCEEDING THEIR RUNOUT LIMIT IN THE RECIRCULATION ALIGNMENTS WAS DIRECTLY RELATED TO THE NEW INFORMATION SUPPLIED BY THE VENDOR REGARDING THE EFFECTS OF INSTRUMENT UNCERTAINTY AND SUCTION BOOST. THIS CONDITION WAS EVALUATED AND FOUND NOT TO BE EXPECTED TO IMPACT PLANT SAFETY.

[125] SURRY 1 DOCKET 50-280 LER 91-017 REV 01
UPDATE ON EMERGENCY DIESEL GENERATOR RENDERED INOPERABLE DUE TO PERSONNEL ERROR IN THAT SPECIFIED TESTING WAS NOT PERFORMED.
EVENT DATE: 050991 REPORT DATE: 111391 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)
VENDOR: WOODWARD GOVERNOR COMPANY

(NSIC 223412) ON AUGUST 9, 1991, WITH UNIT 1 AND UNIT 2 AT 100% POWER, IT WAS DETERMINED THAT EMERGENCY DIESEL GENERATOR (EDG) #3 HAD BEEN INOPERABLE SINCE MAY 9, 1991. THIS DETERMINATION WAS MADE WHILE PERFORMING A ROOT CAUSE EVALUATION OF THE OBSERVED PERFORMANCE OF EDG #3 DURING AN AUGUST 2, 1991, ENGINEERED SAFEGUARDS FEATURE (ESF) ACTUATION ON UNIT 2. THIS SAFETY INJECTION/REACTOR TRIP, WHICH OCCURRED AS A RESULT OF VITAL BUS POWER OSCILLATIONS ON ONE CHANNEL AND A FAILED STEAM GENERATOR PRESSURE TRANSMITTER ON ANOTHER CHANNEL, IS BEING REPORTED SEPARATELY BY LICENSEE EVENT REPORT S2-91-007-00. A ROOT CAUSE INVESTIGATION TEAM APPOINTED TO DETERMINE THE CAUSE OF THE FAILURE OF EDG #3 TO ACHIEVE RATED

SPEED FOUND THAT INADEQUATE POST MAINTENANCE TESTING (PMT) WAS PERFORMED FOLLOWING REPLACEMENT OF THE GOVERNOR ON MAY 7, 1991. THIS WAS DUE TO A COGNITIVE ERROR ON THE PART OF UTILITY PERSONNEL IN THAT AN APPROVED WORK ORDER STEP WHICH SPECIFIED A FAST START TEST OF THE EDG WAS NOT PERFORMED. DURING THE PERIOD EDG #3 WAS INOPERABLE, THE UNIT'S OTHER SOURCE OF EMERGENCY POWER, EDG #2, WAS INOPERABLE FOR APPROXIMATELY THIRTEEN HOURS ON JULY 15, 1991. THIS EVENT IS BEING REPORTED PURSUANT TO 10CFR50.73(A)(2)(I)(B) AND (II)(B).

[126] SURRY 1 DOCKET 50-280 LER 91-021
EMERGENCY DIESEL GENERATOR NO. 1 ROOM FIRE SUPPRESSION SYSTEM INOPERABLE DUE TO PERSONNEL ERROR IN ADMINISTRATIVELY CONTROLLING THE EXIT DOOR.
EVENT DATE: 102891 REPORT DATE: 111891 NSSS: WE TYPE: PWR

(NSIC 223420) ON OCTOBER 28, 1991, AT APPROXIMATELY 0900 HOURS, WITH UNIT 1 AT 100% POWER AND UNIT 2 AT INTERMEDIATE SHUTDOWN, UTILITY QA PERSONNEL FOUND THE EMERGENCY DIESEL GENERATOR (EDG) NUMBER 1 ROOM REAR EXIT DOOR TIED OPEN. THIS CONDITION, WHICH LIMITED THE ABILITY OF THE CARBON DIOXIDE FIRE SUPPRESSION SYSTEM TO PERFORM ITS INTENDED FUNCTION, HAD EXISTED FOR A PERIOD IN EXCESS OF THAT ALLOWED BY TECHNICAL SPECIFICATIONS (TS) 3.21.B.4. ON OCTOBER 28, 1991, AT APPROXIMATELY 0920 HOURS, THE AUTOMATIC DOOR RELEASE DEVICE WAS REPAIRED AND ENGAGED. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO A COGNITIVE ERROR ON THE PART OF CONTRACTOR PERSONNEL IN FAILING TO PROVIDE EXPLICIT INSTRUCTIONS ON HOW TO PROPERLY SECURE OPEN THE EDG NUMBER 1 ROOM REAR EXIT DOOR. TO PREVENT RECURRENCE, THE GENERAL EMPLOYEE TRAINING PROGRAM WILL BE REVISED TO INCLUDE A DISCUSSION OF THE SIGNIFICANCE OF FIRE SUPPRESSION SYSTEM BOUNDARY DOORS AND HOW TO USE THE AUTOMATIC DOOR RELEASE DEVICE. SIGNS ARE ALSO BEING INSTALLED ON BOTH SIDES OF DOORS EQUIPPED WITH AN AUTOMATIC RELEASE DEVICE, CAUTIONING THAT THE DOOR IS A FIRE PROTECTION SYSTEM BOUNDARY. IN ADDITION, STATION DOORS ARE BEING EVALUATED AND LABELED OR COLOR CODED, AS APPROPRIATE, TO INDICATE THEIR SAFETY SIGNIFICANCE TO STATION OPERATION. THE EVENT IS BEING REPORTED, PURSUANT TO 10CFR 50.73(A)(2)(I)(B).

[127] SURRY 2 DOCKET 50-281 LER 91-007 REV 01
UPDATE ON SAFETY INJECTION/REACTOR TRIP FROM VITAL BUS IVA ELECTRICAL FAULT AND FAILURE OF STEAM GENERATOR PRESSURE TRANSMITTER.
EVENT DATE: 080291 REPORT DATE: 111291 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE
ROSEMOUNT, INC.
WOODWARD GOVERNOR COMPANY

(NSIC 223413) AT 1657 HOURS ON AUGUST 2, 1991, WITH UNIT 1 AT 100% POWER AND UNIT 2 AT 92% POWER, AN ENGINEERED SAFEGUARDS FEATURE (ESF) ACTUATED IN UNIT 2. A SAFETY INJECTION/REACTOR TRIP OCCURRED AS A RESULT OF A HIGH STEAM FLOW SIGNAL COINCIDENT WITH A LOW STEAM LINE PRESSURE SIGNAL. THESE CONDITIONS DID NOT ACTUALLY EXIST, BUT WERE GENERATED BY A COMBINATION OF AN ERRATIC STEAM GENERATOR PRESSURE CHANNEL AND AN ELECTRICAL FAULT IN A VITAL BUS DISTRIBUTION PANEL. TROUBLESHOOTING WAS IN PROGRESS ON BOTH PROBLEMS, BUT THE ACTUATION OCCURRED BEFORE CORRECTIVE ACTION COULD BE COMPLETED. ESF FUNCTIONS PERFORMED AS EXPECTED EXCEPT FOR THE FAILURE OF EMERGENCY DIESEL GENERATOR #3 TO ACHIEVE RATED SPEED AND THE FAILURE OF THE CONTAINMENT SUMP PUMP DISCHARGE TRIP VALVE (02-DA-TV-200A) TO FULLY SHUT. A NOTIFICATION OF UNUSUAL EVENT (NOUE) WAS DECLARED AT 1710 HOURS, AND APPROPRIATE REPORTS WERE MADE. ON-SHIFT OPERATING PERSONNEL PROMPTLY PLACED THE PLANT IN A STABLE CONDITION. THE NOUE WAS TERMINATED AT 1840 HOURS. A ROOT CAUSE EVALUATION OF THE EMERGENCY DIESEL GENERATOR PROBLEM IS BEING CONDUCTED. THE RESULTS OF THIS EVALUATION WILL BE REPORTED IN LICENSEE EVENT REPORT S1-91-017. A COMPONENT FAILURE ANALYSIS OF THE CONTAINMENT SUMP DISCHARGE TRIP VALVE HAS BEEN CONDUCTED, AND THE RESULTS ARE INCLUDED IN THIS SUPPLEMENTARY REPORT.

[128] SUSQUEHANNA 1 DOCKET 50-387 LER 91-012
VALVES NOT SURVEILLANCE TESTED - OPERATION PROHIBITED BY TECHNICAL SPECIFICATIONS.
EVENT DATE: 100791 REPORT DATE: 110691 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)

(NSIC 223376) ON OCTOBER 7, 1991, WITH BOTH UNITS 1 & 2 AT 100%, IT WAS DISCOVERED THAT CERTAIN MANUAL CONTAINMENT BOUNDARY VALVES WERE NOT INCLUDED IN REQUIRED SURVEILLANCE PROCEDURES. THIS CONDITION WAS DISCOVERED DURING A PERIODIC REVIEW OF THE PROCEDURE. THE SUBJECT VALVES ARE NORMALLY LOCKED CLOSED TEST CONNECTION VALVES AND WERE IN FACT IN THIS CONFIGURATION. SEVERAL CAUSES WERE DETERMINED FOR THIS EVENT. THE FIRST WAS PERSONNEL ERROR DURING THE INITIAL PREPARATION OF THE PROCEDURE FOR THE VALVES IN THE RHR SYSTEM. CONCERNING THE VALVES IN THE CRM SYSTEM, ONE CAUSE WAS ATTRIBUTED TO NOT INCLUDING ALL APPLICABLE DISCIPLINES IN THE INSTALLATION AND/OR CLOSEOUT REVIEW PROCESS OF THE MODIFICATION. ANOTHER CAUSE WAS THE METHODS UTILIZED BY OPERATIONS DURING THE MODIFICATION CLOSEOUT REVIEW PROCESS. THE EVENT WAS DETERMINED TO BE REPORTABLE PER 10CFR50.73(A)(2)(I)(B) AS A CONDITION PROHIBITED BY TECHNICAL SPECIFICATION IN THAT SURVEILLANCES WERE NOT PERFORMED AS REQUIRED BY TECHNICAL SPECIFICATION 4.6.1.1.B. THERE WERE NO SAFETY CONSEQUENCES OR COMPROMISES AS A RESULT OF THIS CONDITION SINCE THE VALVES REMAINED IN THE REQUIRED LOCKED CLOSED POSITION. NECESSARY SURVEILLANCE PROCEDURE CHANGES HAVE BEEN COMPLETED AND THE OPERATIONS SECTION AND SYSTEMS ENGINEERING GROUP ARE REVIEWING THEIR PRESENT PROCEDURAL GUIDANCE FOR ENHANCEMENTS TO PREVENT RECURRENCE OF THIS TYPE OF EVENT.

[129] SUSQUEHANNA 2 DOCKET 50-388 LER 91-013
ENGINEERED SAFETY FEATURE ACTUATIONS DUE TO REACTOR PROTECTION SYSTEM M-G SET OUTPUT BREAKER TRIP.
EVENT DATE: 101491 REPORT DATE: 111391 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 223439) AT 1159 HOURS ON OCTOBER 14, 1991 WITH UNIT 2 OPERATING AT 100% POWER, THE PRIMARY POWER SUPPLY TO THE "A" REACTOR PROTECTION SYSTEM (RPS) POWER DISTRIBUTION PANEL WAS LOST WHEN ITS MOTOR-GENERATOR (MG) SET OUTPUT BREAKER TRIPPED. RPS AS WELL AS OTHER PLANT SYSTEMS AND COMPONENTS FUNCTIONED PROPERLY AND AS EXPECTED IN RESPONSE TO THE EVENT. NO REACTOR PARAMETERS WERE AFFECTED AND NO EMERGENCY CORE COOLING SYSTEMS WERE ACTUATED. THE 'A' RPS DISTRIBUTION PANEL WAS SWAPPED TO THE ALTERNATE POWER SUPPLY UNTIL THE PRIMARY POWER SUPPLY WAS RESTORED. ALL ISOLATION SIGNALS WERE RESET BY 1420 HOURS. THE PRIMARY RPS POWER SOURCE WAS RESET AT 0955 HOURS ON 10/16/91. FULL POWER OPERATION OF THE UNIT CONTINUED WITHOUT INTERRUPTION. THE CAUSE WAS DETERMINED TO BE ABNORMAL BEHAVIOR OF THE MG SET OUTPUT REGULATOR MECHANICAL RELAYS. THESE RELAYS WERE REWORKED AND RETURNED TO SERVICE. THE REMAINING MG SET REGULATORS OF BOTH UNITS WILL BE INSPECTED FOR RELAY ABNORMALITIES AND LOOSE CONNECTIONS DURING THE NEXT RESPECTIVE PLANNED EQUIPMENT OUTAGE. IN ADDITION, PERIODIC MAINTENANCE ACTIVITIES FOR RPS MG SETS WILL BE REVIEWED AND INCORPORATE MAINTENANCE OF RELAYS AND SECURITY OF CONNECTIONS. SINCE ALL ENGINEERED SAFETY FEATURE (ESF) SYSTEMS AND COMPONENTS FUNCTIONED PROPERLY PER DESIGN, THERE WERE NO SAFETY CONSEQUENCES OR COMPROMISES TO THE HEALTH OR SAFETY OF THE PUBLIC.

[130] THREE MILE ISLAND 1 DOCKET 50-289 LER 91-003
REACTOR TRIP DURING TURBINE VALVE TESTING DUE TO INADEQUATE PROCEDURE.
EVENT DATE: 092791 REPORT DATE: 102891 NSSS: BW TYPE: PWR

(NSIC 223331) ON SEPTEMBER 27, 1991 DURING PLANT SHUTDOWN FOR THE 9R OUTAGE REACTOR POWER WAS STABLIZED AT 13% TO SUPPORT COMPLETION OF TURBINE VALVE TIGHTNESS TESTING AND A SPECIAL TEST OF THE TURBINE OVERSPEED MECHANICAL TRIP DEVICE. THE TURBINE LOAD LIMIT CONTROL WAS TURNED TO THE MINIMUM SETTING TO CLOSE THE TURBINE CONTROL VALVES AND BEGIN THE TIGHTNESS TEST WHILE THE STOP VALVES REMAINED OPEN. THE CONTROL VALVES CLOSED AS EXPECTED AND THE TURBINE DECELERATED INDICATING THE LEAK TIGHTNESS OF THE VALVES. AFTER ABOUT SEVEN MINUTES OF DECELERATION, THE TURBINE SPEED WAS 1200 RPM. IN ACCORDANCE WITH PROCEDURE, THE CRO SELECTED THE "FAST" ACCELERATION RATE ON THE TURBINE CONTROL PANEL AND TURNED THE LOAD LIMIT CONTROL TO MAXIMUM SETTING. THE TURBINE CONTROL VALVES IMMEDIATELY STARTED OPENING RAPIDLY AND OTSG PRESSURE BEGAN DROPPING RAPIDLY. APPROXIMATELY TEN SECONDS LATER AT 18:37, THE TURBINE TRIPPED ON OVERSPEED. SUBSEQUENTLY, MAIN FEEDWATER WAS ISOLATED TO BOTH OTSGS DUE TO LOW OTSG PRESSURE AND BOTH EMERGENCY FEEDWATER TRAINS WERE AUTO INITIATED DUE TO THE LOW OTSG LEVELS. THE REACTOR TRIPPED AT 18:38 ON HIGH PRESSURE. NORMAL FEEDWATER FLOW TO THE OTSG'S WAS

RE-ESTABLISHED AND POST TRIP RESPONSE WAS CONSIDERED NORMAL. THE NRC WAS NOTIFIED IN ACCORDANCE WITH 10 CFR 50.72(B)(2)(II).

[131] THREE MILE ISLAND 1 DOCKET 50-289 LER 91-004
MOVEMENT OF IRRADIATED FUEL ASSEMBLY WITHOUT CONTAINMENT INTEGRITY DUE TO PROCEDURAL WEAKNESSES AND PERSONNEL ERROR.
EVENT DATE: 100891 REPORT DATE: 110791 NSSS: BW TYPE: PWR

(NSIC 223355) ON OCTOBER 8, 1991, TMI-1 WAS IN REFUELING SHUTDOWN. LICENSED OPERATORS WERE PERFORMING 1303-11.4, "REFUELING SYSTEMS INTERLOCKS" TEST OF THE MAIN FUEL BRIDGE HOIST. THIS TEST IS NORMALLY PERFORMED IN CONJUNCTION WITH 1505-1, "FUEL AND CONTROL COMPONENT SHUFFLES." SECTION 6.3.3.1, OF THE PROCEDURE REQUIRES FUEL MOVEMENT; HOWEVER, THIS SECTION SHOULD HAVE REQUIRED THAT NO MOVEMENT OF FUEL TAKE PLACE UNTIL THE PREREQUISITES OF 1505-1 WERE COMPLETED. IN THIS EVENT, THE BRIDGE CREW MOVED FUEL TO TEST THE HOIST INTERLOCKS WHEN CONTAINMENT INTEGRITY WAS NOT SET. MOST OF THE 1505-1 PREREQUISITES FOR CONTAINMENT INTEGRITY WERE COMPLETED EXCEPT FOR THE OPEN REACTOR BUILDING PERSONNEL AND EMERGENCY AIRLOCK DOORS (NH/AL). TECHNICAL SPECIFICATION 3.8.6 REQUIRES THAT AT LEAST ONE DOOR IN EACH AIRLOCK BE CLOSED WHEN MOVING IRRADIATED FUEL IN THE REACTOR BUILDING. THIS EVENT WAS CAUSED BY: PROCEDURAL WEAKNESSES, BECAUSE 1303-11.4 DID NOT CLEARLY CAUTION THE OPERATORS THAT CONTAINMENT INTEGRITY WAS REQUIRED PRIOR TO TEST OF THE INTERLOCKS; AND, BY PERSONNEL ERROR, DUE TO A LACK OF UNDERSTANDING BY PERSONNEL THAT THE TEST INVOLVED REFUELING OPERATIONS. THE CAUSES OF THE EVENT WERE REVIEWED WITH ALL FUEL HANDLING PERSONNEL PRIOR TO THE COMMENCEMENT OF THE FUEL SHUFFLE. 1303-11.4 WAS REVISED TO STRENGTHEN THE PROCEDURE, PROVIDE PREREQUISITES, AND ADD WARNINGS.

[132] TROJAN DOCKET 50-344 LER 91-008 REV 01
UPDATE ON LESS THAN ADEQUATE CONTROL OF ANALYSIS AND PROCEDURE CHANGES RESULT IN POTENTIAL FOR FIRE INDUCED SPURIOUS OPERATIONS TO ADVERSELY AFFECT ABILITY TO SAFELY SHUT DOWN.
EVENT DATE: 031291 REPORT DATE: 101591 NSSS: WE TYPE: PWR

(NSIC 223218) ON 3/12/91, THE TROJAN NUCLEAR PLANT WAS IN MODE 5. AT 1645 PRELIMINARY RESULTS INDICATED THAT POTENTIAL, FIRE-INDUCED SPURIOUS OPERATIONS OF THE PRESSURIZER AUXILIARY SPRAY VALVE, STEAM GENERATOR POWER OPERATED RELIEF VALVES, AND THE REACTOR COOLANT SYSTEM CHARGING FLOW CONTROL VALVE COULD HAVE ADVERSELY AFFECTED THE ABILITY TO SAFELY SHUT DOWN THE PLANT IN THE EVENT OF A FIRE. FINAL ANALYSIS RESULTS HAVE INDICATED THAT IF THE FIRE-INDUCED SPURIOUS OPERATIONS OF THESE COMPONENTS HAD OCCURRED, THEN THE ABILITY TO SAFELY SHUT DOWN THE PLANT WOULD NOT HAVE BEEN ADVERSELY AFFECTED. IT APPEARS THAT THESE POTENTIAL SPURIOUS OPERATIONS OF CONCERN WERE NOT COMPLETELY ADDRESSED IN THE "TROJAN NUCLEAR PLANT 10 CFR 50 APPENDIX R REVIEW" AND ITS IMPLEMENTING PROCEDURES. THE ROOT CAUSE OF THE EVENT WAS INADEQUATE CONTROL OF CHANGES TO THE SPURIOUS OPERATION ANALYSIS AND ITS IMPLEMENTING PROCEDURES ANALYSIS AND PROCEDURE CHANGES HAVE BEEN IMPLEMENTED TO ADDRESS THE SPECIFIC DEFICIENCIES. LONG TERM CORRECTIVE ACTIONS INCLUDED CHANGES TO THE ADMINISTRATIVE CONTROLS FOR ANALYSIS AND PROCEDURE REVISIONS AND THE ESTABLISHMENT OF A CENTRALIZED FIRE PROTECTION ENGINEERING GROUP. THE EXISTENCE OF THE DEFICIENCIES HAD MINIMAL SAFETY SIGNIFICANCE.

[133] TROJAN DOCKET 50-344 LER 91-011 REV 01
UPDATE ON POTENTIAL DEGRADATION OF ELECTRICAL PENETRATION ASSEMBLY MODULE SEALS.
EVENT DATE: 051091 REPORT DATE: 102891 NSSS: WE TYPE: PWR
VENDOR: AMPHENOL
PARKER PACKING COMPANY

(NSIC 223315) ON MAY 10, 1991, THE TROJAN NUCLEAR PLANT WAS IN MODE 6, FOLLOWING THE RELOADING OF FUEL IN THE REACTOR VESSEL, DURING THE 1991 REFUELING AND MAINTENANCE OUTAGE. NUCLEAR PLANT ENGINEERING PERFORMED AN EVALUATION OF THE POTENTIAL DEGRADATION OF ELECTRICAL PENETRATION ASSEMBLY (EPA) SEALS AND DETERMINED: (1) LUBRICANTS USED TO INSTALL THE SEALS IN ACCORDANCE WITH ORIGINAL MANUFACTURER'S RECOMMENDATIONS MAY CAUSE SEAL DEGRADATION; AND (2) THE SEAL

MANUFACTURER INDICATED THE SEALS MAY DEGRADE IF SUBJECTED TO DESIGN BASIS ACCIDENT (DBA) MOISTURE AND/OR TEMPERATURE CONDITIONS. THE PRIMARY CAUSE OF THIS CONDITION WAS THAT THE ORIGINALLY INSTALLED SEAL AND LUBRICANT MATERIALS WERE INAPPROPRIATE FOR THE APPLICATION. CORRECTIVE ACTIONS INCLUDED REPLACEMENT OF THE EPA SEALS USING A MATERIAL ENVIRONMENTALLY QUALIFIED FOR DBA CONDITIONS, AND ADDITION OF A SEAL BACKUP O-RING IN THE EPA MODULE ASSEMBLIES. THE IMPACT ON ACCIDENT CONSEQUENCES REPORTED IN CHAPTER 15 OF THE TROJAN FINAL SAFETY ANALYSIS REPORT IS INDETERMINATE WITH RESPECT TO THE AS-FOUND CONDITION. HOWEVER, THE IMPROVED AND QUALIFIED REPLACEMENT SEALS, WITH ACCEPTABLE TYPE B LEAK TEST RESULTS, SUPPORT THE CONCLUSION THAT THE TROJAN LICENSING BASIS REMAINS VALID.

[134] TURKEY POINT 3 DOCKET 50-250 LER 91-007
MODE 2 ENTERED WITH ONE OF TWO INTERMEDIATE RANGE NUCLEAR INSTRUMENTATION CHANNELS INOPERABLE.
EVENT DATE: 092591 REPORT DATE: 102291 NSSS: WE TYPE: PWR

(NSIC 223280) AT 0851 ON SEPTEMBER 25, 1991, UNIT 3 LOGGED ENTRY INTO MODE 2. AT 0950, THE UNIT 3 REACTOR CONTROL OPERATOR DECLARED EXCORE NUCLEAR INSTRUMENT INTERMEDIATE RANGE CHANNEL N-35 INOPERABLE, PRIOR TO GOING CRITICAL, WHEN IT DID NOT RESPOND TO INCREASING NEUTRON COUNTS. THE DETECTOR CABLES FOR THE N-35 CHANNEL WERE FOUND DISCONNECTED AT THE BACK OF THE DRAWER. THE ROOT CAUSE OF THE EVENT WAS PERSONNEL ERROR BY NON-LICENSED UTILITY PERSONNEL, IN THAT INADEQUATE CONTROL OF THE LIFTED LEADS BETWEEN THE N-35 DRAWER AND THE DETECTOR OCCURRED. THE CABLES FOR ALL OTHER UNIT 3 EXCORE DETECTORS WERE CHECKED TO ENSURE THAT NO OTHER CABLES WERE DISCONNECTED. THE MAINTENANCE PROCEDURE HAS BEEN REVISED TO INCLUDE LIFTED LEAD/CONNECTOR DOCUMENTATION AND INDEPENDENT VERIFICATION. OUTSTANDING PLANT WORK ORDERS INVOLVING MODE-DEFERRED TESTING HAVE BEEN REVIEWED TO ENSURE SIMILAR CONCERNS FOR OTHER SYSTEMS DO NOT EXIST. MAINTENANCE PERSONNEL HAVE BEEN TRAINED ON THE SIGNIFICANCE OF THE EVENT. A POLICY LETTER HAS BEEN ISSUED REQUIRING THE USE OF LIFTED LEAD CONTROL PROCEDURES FOR WORK INVOLVING LIFTED LEADS, WHEN THE LEADS ARE NOT SPECIFIED AND INDEPENDENTLY VERIFIED IN A PROCEDURE. TURKEY POINT'S LIFTED LEAD CONTROLS WILL BE REVIEWED AGAINST INPO AND INDUSTRY PRACTICES.

[135] TURKEY POINT 3 DOCKET 50-250 LER 91-008
MANUAL REACTOR TRIP FOLLOWING LOSS OF MAIN TURBINE GENERATOR LOAD DUE TO A MECHANICAL FAILURE OF A PIPING NIPPLE IN THE CONTROL OIL SYSTEM.
EVENT DATE: 100391 REPORT DATE: 110191 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 223339) ON OCTOBER 3, 1991, AT 1200 EDT, WITH UNIT 3 AT 50 PERCENT POWER AND UNIT 4 IN MODE 6 (REFUELING) THE UNIT 3 REACTOR WAS TRIPPED MANUALLY DUE TO A LOSS OF MAIN TURBINE GENERATOR LOAD. THE UNIT WAS STABILIZED IN MODE 3 (HOT STANDBY) WITH STEAM GENERATOR WATER LEVELS BEING MAINTAINED BY THE MAIN FEEDWATER SYSTEM AND STEAM BEING DUMPED THROUGH THE ATMOSPHERIC DUMP VALVES. AN EVENT RESPONSE TEAM WAS FORMED TO DETERMINE THE CAUSE OF THIS EVENT AND TO RECOMMEND APPLICABLE CORRECTIVE ACTIONS. AS PART OF THIS INVESTIGATION, A PIPE BREAK WAS DISCOVERED IN THE CONTROL OIL PIPING TO THE NORTH-EAST INTERCEPT VALVE. EXAMINATION OF THE BREAK AREA INDICATED THE CAUSE OF THE PIPE BREAK TO BE FATIGUE FAILURE AT THE ROOT OF THE PIPE NIPPLE AT THE CONTROL ORIFICE. THE FATIGUE WAS CAUSED BY PIPE VIBRATION. A SIGNIFICANT CONTRIBUTOR TO THE VIBRATION WAS A SPRING-CAN SUPPORT FOR THE LOW PRESSURE STEAM LINE THAT HAD ONE OF THE COMPRESSION NUTS RESTRAINING SPRING-CAN MOVEMENT. THE BROKEN PIPE WAS REPAIRED AND THE SPRING RETAINERS HAVE BEEN REMOVED FROM THE SPRING-CANS. THE NIPPLES UPSTREAM OF THE OTHER THREE UNIT 3 INTERCEPT VALVES WILL BE REPLACED DURING THE NEXT REFUELING OUTAGE. THE SUPPORTS FOR THE CONTROL OIL PIPING ON BOTH UNITS WILL BE MODIFIED TO PREVENT EXCESSIVE VIBRATION.

[136] VERMONT YANKEE DOCKET 50-271 LER 91-012 REV 01
UPDATE ON REDUCED COOLING WATER FLOW TO DIESEL GENERATOR HEAT EXCHANGERS AND STATION SERVICE AIR COMPRESSORS DUE TO HIGH SERVICE WATER SYSTEM BACKPRESSURE CAUSED BY WEAK DESIGN.
EVENT DATE: 042391 REPORT DATE: 110791 NSSS: GE TYPE: BWR

(NSIC 223411) ON APRIL 23, 1991, AT 1448 HOURS, AND AT 100% POWER, A LOSS OF NORMAL POWER (LNP) WAS EXPERIENCED. FOLLOWING THE EXPECTED START OF BOTH EMERGENCY DIESEL GENERATORS (EDG), IT WAS OBSERVED THAT THE EDG HEAT EXCHANGERS WERE OPERATING AT REDUCED FLOW AND THAT THE STATION AIR COMPRESSOR COOLERS WERE OPERATING WITH REDUCED/REVERSE FLOW. THE ROOT CAUSE OF THE EVENT WAS A WEAK DESIGN MODIFICATION RESULTING IN AN INCORRECT PROCEDURE. THE INCORRECT PROCEDURE ESTABLISHED AN ALTERNATE COOLING DISCHARGE PATH TO THE COOLING TOWERS AND PRODUCED A HIGH SERVICE WATER SYSTEM BACKPRESSURE OF APPROXIMATELY 40 PSID. STEM BACKPRESSURE WAS FURTHER INCREASED DUE TO VARIOUS SYSTEM DESIGN AND OPERATING CHARACTERISTICS PRESENT DURING THE LNP. A TASK FORCE WAS CONVENED TO ANALYZE AND TEST THE RESPONSE OF THE SERVICE WATER SYSTEM. THE SERVICE WATER SYSTEM WAS RECONFIGURED TO ELIMINATE THE PRIMARY CONTRIBUTOR OF BACKPRESSURE. A CORRECTIVE ACTION REPORT WAS GENERATED TO FURTHER IDENTIFY AND CONFIRM THE ROOT CAUSE OF THE EVENT AND PROVIDE DETAILED LONG TERM CORRECTIVE ACTIONS.

[137] WATERFORD 3 DOCKET 50-382 LER 91-013 REV 01
UPDATE ON MANUAL REACTOR TRIP IN RESPONSE TO HIGH STEAM GENERATOR WATER LEVEL DUE TO A FAILED STARTUP FEEDWATER REGULATING VALVE.
EVENT DATE: 062491 REPORT DATE: 111591 NSSS: CE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 223426) AT 1315 HOURS ON JUNE 24, 1991, THE REACTOR AT WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS MANUALLY TRIPPED DUE TO HIGH WATER LEVEL IN STEAM GENERATOR (SG) #2. FOUR SG #2 HIGH LEVEL REACTOR PRE-TRIP ALARMS WERE RECEIVED PRIOR TO MANUALLY INITIATING A REACTOR TRIP. A MAIN STEAM ISOLATION SIGNAL WAS MANUALLY ACTUATED SUBSEQUENT TO THE REACTOR TRIP AS A RESULT OF EXCESSIVE COOL DOWN RATE. THIS EVENT IS REPORTABLE AS AN UNPLANNED REACTOR PROTECTION SYSTEM ACTUATION. THE ROOT CAUSE OF THIS EVENT IS SG #2 STARTUP FEEDWATER REGULATING VALVE FAILING OPEN WHILE AT APPROXIMATELY 25 PERCENT REACTOR POWER. A PROCESS ANALOG CONTROL (PAC) CARD IN THE VALVE CONTROL CIRCUITRY FAILED, RESULTING IN A CONSTANT OPEN SIGNAL TO SG #2 STARTUP FEEDWATER REGULATING VALVE. THE PAC CARD WAS REPLACED AND PAC CARD PERFORMANCE WILL BE TREATED AS PART OF THE LONG TERM RELIABILITY PROGRAM. PLANT PROTECTIVE FEATURES FUNCTIONED AS DESIGNED; THEREFORE, THIS EVENT DID NOT THREATEN THE HEALTH OR SAFETY OF THE GENERAL PUBLIC OR PLANT PERSONNEL.

[138] WATERFORD 3 DOCKET 50-382 LER 91-021
SAFETY INJECTION TANK LEVEL BELOW TECHNICAL SPECIFICATIONS DUE TO LOW REFERENCE LEG LEVEL.
EVENT DATE: 101691 REPORT DATE: 111591 NSSS: CE TYPE: PWR

(NSIC 223438) ON OCTOBER 16, 1991, SAFETY INJECTION TANK (SIT) 1A NARROW RANGE INSTRUMENT SI ILT 0313 INDICATED A LOWER SIT LEVEL THAN THE OTHER TWO SIT LEVEL DETECTORS AND WAS PLACED OUT OF SERVICE AT 1106 HOURS. ON OCTOBER 17, 1991, AFTER A REVIEW OF SIT LEVEL TREND DATA, INSTRUMENTATION & CONTROL INFORMED OPERATIONS THAT SI ILT 0313 MAY HAVE DISPLAYED THE CORRECT SIT 1A LEVEL AND THAT THE OTHER SIT 1A LEVEL DETECTORS, SI ILT 0311 AND SI ILT 0312, MAY INDICATE ERRONEOUSLY HIGH SIT LEVELS . WHILE SI ILT 0313 WAS OUT OF SERVICE, SIT 1A LEVEL FELL BELOW TECHNICAL SPECIFICATION (TS) LIMITS. THE ROOT CAUSE OF THIS EVENT WAS LEAKAGE OF THE COMMON REFERENCE LEG FOR SI ILT 0311 AND SI ILT 0312. SINCE THE PLANT WAS OPERATED IN A CONDITION OUTSIDE THE DESIGN BASIS OF THE PLANT, AT 1319 HOURS ON OCTOBER 17, 1991 OPERATORS MADE A 1-HOUR NON-EMERGENCY EVENT NOTIFICATION. THIS EVENT IS REPORTABLE UNDER 10 CFR 50.73, OPERATION OR CONDITION PROHIBITED BY PLANT TSS AND CONDITION OUTSIDE THE DESIGN BASIS OF THE PLANT. SIT 1A LEVEL WAS INCREASED, REFERENCE LEGS FILLED AND INSTRUMENT VALVES INSPECTED. SINCE THE DIFFERENCE BETWEEN THE MINIMUM REQUIRED SIT LEVEL AND LOWEST SIT LEVEL WAS SLIGHT, PEAK CLAD TEMPERATURES WOULD BE EXPECTED TO REMAIN WITHIN THE ACCEPTABLE BOUNDS OF EXISTING ANALYSIS; THEREFORE, THIS EVENT DID NOT THREATEN THE HEALTH AND SAFETY OF THE PUBLIC OR PLANT PERSONNEL.

[139] WOLF CREEK 1 DOCKET 50-482 LER 87-061
FAILURE TO RECOGNIZE THE EMERGENCY DIESEL GENERATOR VENTILATION SYSTEM AS ESSENTIAL FOR EMERGENCY DIESEL GENERATOR OPERABILITY.
EVENT DATE: 031287 REPORT DATE: 111191 NSSS: WE TYPE: PWR
VENDOR: ITT-BARTON
WESTINGHOUSE ELECTRIC CORP.

(NSIC 223385) ON 2/6/91, IT WAS DETERMINED THAT THREE SEPARATE SITUATIONS INVOLVING THE EMERGENCY DIESEL GENERATOR (EDG) BUILDING VENTILATION SYSTEM MAY HAVE CAUSED EDGS A OR B TO BE INOPERABLE. THE FIRST SITUATION INVOLVED THE REMOVAL OF THE MOTOR TO EDG VENTILATION SUPPLY FAN CGM01B, ON 3/12/87. ON 5/15/87, CGM01B WAS RESTORED TO OPERABLE STATUS. THE SECOND EVENT INVOLVED A PROCEDURE CHANGE, INITIATED ON 8/29/88, PLACING THE HANDSWITCHES TO EDG VENTILATION SUPPLY FANS CGM01A & B IN PULL-TO-LOCK WITHOUT IMPLEMENTING PROPER COMPENSATORY OPERATOR ACTIONS. THIS PROCEDURE WAS CORRECTED IN AUGUST 1990. THE THIRD EVENT INVOLVED CLAMPING EDG ROOM "A" RECIRCULATION AIR DAMPER GMD003 IN ITS NON-SAFEGUARDS POSITION BETWEEN 7/17/88 AND 4/28/90. A SUBSEQUENT EVALUATION CONCLUDED THAT THE FIRST TWO SITUATIONS RENDERED THE EDGS INOPERABLE. THE ROOT CAUSE FOR THESE EVENTS STEMS FROM PERSONNEL FAILING TO RECOGNIZE THAT THE EDG VENTILATION SYSTEM IS ESSENTIAL TO OPERABILITY OF THE EDGS. SINCE THE TIME THESE EVENTS OCCURRED, ADMINISTRATIVE GUIDANCE HAS BEEN REVISED TO INCLUDE THE EDG VENTILATION SUPPLY SYSTEM AS ESSENTIAL FOR EDG OPERATION.

[140] WOLF CREEK 1 DOCKET 50-482 LER 91-015
PROCEDURAL INADEQUACY AND FAILURE TO FOLLOW PROCEDURES LEAD TO FAILURE TO REMOVE TEMPORARY FILTER RESULTING IN RESIDUAL HEAT REMOVAL PUMP INOPERABILITY.
EVENT DATE: 090791 REPORT DATE: 100791 NSSS: WE TYPE: PWR

(NSIC 223105) BETWEEN SEPTEMBER 7, 1991, AT APPROXIMATELY 0900 CDT, AND SEPTEMBER 9, 1991, AT APPROXIMATELY 0830 CDT, A TEMPORARY FILTER WAS IN PLACE ON THE INTAKE TO THE RESIDUAL HEAT REMOVAL (RHR) PUMP ROOM "A" COOLER SO THAT THE ROOM COOLER COULD BE PERIODICALLY OPERATED DURING PREPARATIONS FOR THE APPLICATION OF DECONTAMINATION COATINGS IN THE ROOM. SUBSEQUENT EVALUATION DETERMINED THAT WITH A DIRTY TEMPORARY FILTER IN PLACE, THE DESIGN HEAT REMOVAL FOR THE ROAM COOLERS COULD NOT BE OBTAINED. ALTHOUGH THIS EVENT DID NOT RESULT IN A TECHNICAL SPECIFICATION VIOLATION SINCE BOTH RHR PUMPS WERE NOT INOPERABLE SIMULTANEOUSLY NOR WAS ONE RHR PUMP INOPERABLE FOR MORE THAN 72 HOURS, IT WAS DETERMINED THAT A VOLUNTARY LICENSEE EVENT REPORT (LER) WOULD BE SUBMITTED. THE ROOT CAUSES OF THIS EVENT ARE PROCEDURAL INADEQUACY AND FAILURE TO FOLLOW PROCEDURES. THE WORKERS HAVE BEEN COUNSELLED ON THE IMPORTANCE OF FOLLOWING ALL THE STEPS IN THE PROCEDURES. ON SEPTEMBER 16, 1991, THE PROCEDURE WAS REVISED TO INCLUDE AN EVALUATION OF POTENTIAL IMPACTS OF THE ACTIVITIES ON SYSTEMS THAT REMAIN IN SERVICE.

[141] WOLF CREEK 1 DOCKET 50-482 LER 91-018
CONTAINMENT PURGE ISOLATION AND CONTROL ROOM VENTILATION ISOLATION CAUSED BY HIGH GASEOUS ACTIVITY DURING CONTAINMENT PURGE.
EVENT DATE: 100291 REPORT DATE: 102891 NSSS: WE TYPE: PWR

(NSIC 223302) ON OCTOBER 2, 1991, AT 0352 CDT, SHORTLY AFTER INITIATION OF A CONTAINMENT PURGE, A CONTAINMENT PURGE ISOLATION SIGNAL AND CONTROL ROOM VENTILATION ISOLATION SIGNAL WERE GENERATED BY SIGNALS FROM THE GASEOUS CHANNELS OF THE CONTAINMENT PURGE RADIATION MONITORS, GT RE-22 AND GT RE-33. ALL REQUIRED ENGINEERED SAFETY FEATURES EQUIPMENT RESPONDED PROPERLY TO THE SIGNALS. AFTER CONFIRMING THERE WERE NO UNUSUAL RADIOLOGICAL CONDITIONS IN CONTAINMENT, THE PURGE WAS RE-ESTABLISHED AT 0713 CDT WITHOUT FURTHER INCIDENT. THE ROOT CAUSE OF THIS EVENT IS FAILURE TO FOLLOW PROCEDURES BY LICENSED CONTROL ROOM OPERATORS. TO PREVENT RECURRENCE, THE CONTROL ROOM OPERATORS WHO WERE INVOLVED IN THIS EVENT WERE COUNSELED ON THE IMPORTANCE OF USING ALL AVAILABLE RESOURCES, INCLUDING PROCEDURES, WHEN RESTORING SYSTEMS TO OPERATION RATHER THAN RELYING ON MEMORY. EVALUATIONS OF ALTERNATE VENTING METHODS WILL BE COMPLETED AND ANY NECESSARY CHANGES TO THE VENTING PROCESS WILL BE IN PLACE BEFORE THE NEXT VENTING OPERATION IS SCHEDULED IN REFUELING OUTAGE VI.

[142] WOLF CREEK 1 DOCKET 50-482 LER 91-019
FRETTING BETWEEN GRID STRAP AND FUEL ROD RESULTS IN BROKEN FUEL ROD.
EVENT DATE: 101191 REPORT DATE: 110591 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 223387) ON 10/11/91, AT 0240 CDT, DURING FUEL OFF-LOAD OPERATIONS, CONTROL ROOM OPERATORS WERE NOTIFIED THAT A BROKEN FUEL ROD HAD BEEN IDENTIFIED IN FUEL ASSEMBLY F31 LOCATED IN CORE POSITION L-9. SUBSEQUENTLY, IT WAS DETERMINED THAT THE FUEL ROD WAS BROKEN IN TWO LOCATIONS RESULTING IN A SEGMENT APPROXIMATELY FOUR INCHES IN LENGTH. ADDITIONAL INSPECTION ACTIVITIES RESULTED IN THE DISCOVERY OF FUEL DEFECTS IN FUEL ASSEMBLIES F05 AND F17 AND ADDITIONAL DEFECTS IN FUEL ASSEMBLY F31. FROM THE INVESTIGATIONS AND EVALUATIONS CONDUCTED, GRID-TO-ROD FRETTING HAS BEEN IDENTIFIED AS THE PRINCIPAL FAILURE MECHANISM AND APPEARS TO HAVE OCCURRED SOLELY IN THE SECOND CYCLE OF THE FUEL ASSEMBLIES' OPERATION. EXTENSIVE INVESTIGATION HAS BEEN UNSUCCESSFUL IN IDENTIFYING THE ROOT CAUSE FOR THIS FAILURE MECHANISM.

[143] WPPSS 2 DOCKET 50-397 LER 89-001 REV 02
UPDATE ON UNANALYZED FAILURE MODES FOR CONTAINMENT NITROGEN SYSTEM CAUSED BY INADEQUATE DESIGN PROCEDURES.
EVENT DATE: 011289 REPORT DATE: 111491 NSSS: GE TYPE: BWR

(NSIC 223440) ON JANUARY 12, 1989, DURING AN ENGINEERING EVALUATION OF THE CONTAINMENT NITROGEN INERTING (CN) SYSTEM, FOUR NEW UNANALYZED FAILURE MODES WERE DISCOVERED. THESE FAILURE MODES ALL HAVE THE POTENTIAL TO IMPACT SAFETY RELATED EQUIPMENT REQUIRED TO ATTAIN SAFE SHUTDOWN OF THE REACTOR. A SUMMARY OF THE FAILURE MODES IS AS FOLLOWS: 1) FAILURE MODE 1 - LOSS OF AUXILIARY STREAM OR PRESSURE CONTROL TO THE "HIGH FLOW" NITROGEN LINE, 2) FAILURE MODE 2 - A BREAK IN THE "LOW FLOW" NITROGEN LINE OR LOSS OF ELECTRIC HEATER ON THE "LOW FLOW" LINE 3) FAILURE MODE 3 - A TORNADO MISSILE CAUSES FAILURE OF LIQUID NITROGEN STORAGE TANK AND/OR ASSOCIATED PIPING 4) FAILURE MODE 4 - NON-MECHANISTIC RUPTURE OF LIQUID NITROGEN STORAGE TANK OR LIQUID LINES BENEATH THE TANK. FAILURE MODES 1 AND 2 INVOLVE POTENTIAL DAMAGE TO SAFETY RELATED COMPONENTS DUE TO CONTACT WITH LIQUID NITROGEN AND/OR LOW TEMPERATURES. FAILURE MODES 3 AND 4 INVOLVE THE POTENTIAL FOR OXYGEN STARVATION OF ALL THREE DIVISIONS OF EMERGENCY DIESEL GENERATORS UNDER CERTAIN LOW PROBABILITY CONDITIONS. IMMEDIATE CORRECTIVE ACTION WAS TO MODIFY PROCEDURES TO REQUIRE ADDITIONAL OPERATOR COVERAGE AND PROVIDE SPECIFIC GUIDANCE TO ENSURE CORRECT RESPONSE TO FAILURE MODE CONDITIONS. THE ROOT CAUSE OF THE EVENT WAS DETERMINED TO BE INADEQUATE DESIGN PROCEDURES.

[144] WPPSS 2 DOCKET 50-397 LER 91-028
CONTAINMENT AIRLOCK SEAL TEST NOT PERFORMED IN ALLOWABLE TIME PERIOD.
EVENT DATE: 093091 REPORT DATE: 103091 NSSS: GE TYPE: BWR

(NSIC 223308) ON SEPTEMBER 30, 1991 AT 1036 HOURS THE CONTAINMENT AIRLOCK DOOR SEAL LEAKAGE TEST REQUIRED BY TECHNICAL SPECIFICATION 3/4.6.1.3 WAS NOT PERFORMED WITHIN THE ALLOWABLE SURVEILLANCE INTERVAL. THIS IS A DEVIATION FROM THE PLANT'S TECHNICAL SPECIFICATIONS AND IS REPORTABLE IN ACCORDANCE WITH THE REQUIREMENTS OF 10 CFR 50.7 3(A)(2)(I)(B). WNP-2 ENTERED MODE 2 FROM MODE 4 AT 1636 HOURS ON SEPTEMBER 26, 1991. FROM THAT TIME THE PLANT HAD 72 HOURS (PLUS 25% OR 18 HOURS) TO COMPLETE THE REQUIRED AIRLOCK SEAL TEST. THE TEST WAS COMPLETED APPROXIMATELY FOUR HOURS AFTER EXPIRATION OF THE TECHNICAL SPECIFICATION ALLOWED SURVEILLANCE INTERVAL BUT WITHIN THE 24 HOUR PERIOD PROVIDED IN ACTION STATEMENT 3.6.1.3.C. THE ROOT CAUSE OF THIS EVENT IS PERSONNEL WORK PRACTICES LESS THAN ADEQUATE IN THAT AN INTENDED OR REQUIRED VERIFICATION WAS NOT PERFORMED. A PROCEDURALLY REQUIRED VERIFICATION THAT THE AIRLOCK SEAL TEST BE PERFORMED IN THE ALLOTTED TIME PERIOD WAS NOT SATISFIED. CORRECTIVE ACTIONS INCLUDE: 1) IMMEDIATE PERFORMANCE OF THE REQUIRED TESTING WITHIN THE ACTION STATEMENT TIME REQUIREMENTS; 2) DEVIATIONS TO PLANT STARTUP PROCEDURES TO INCLUDE SPECIFIC GUIDANCE AND REQUIREMENTS TO ENSURE THE TEST IS PERFORMED PRIOR TO ENTRY INTO THE APPLICABLE OPERATIONAL CONDITION; AND 3) THIS LER WILL BE REQUIRED READING FOR OPERATIONS PERSONNEL.

[145] ZION 1 DOCKET 50-295 LER 91-012 REV 02
UPDATE ON SNUBBER MODIFICATION MISINTERPRETATION.
EVENT DATE: 071191 REPORT DATE: 110191 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 2 (PWR)
VENDOR: BINDICATOR CORP.

(NSIC 223330) IN PREPARATION FOR A HYDRAULIC SNUBBER REPLACEMENT MODIFICATION, IT WAS NOTICED THAT THE FIELD LOCATION OF SNUBBERS 1SIRS-1018 AND 1SIRS-1019 DID NOT AGREE WITH THE DRAWING. THE SNUBBERS WERE LABELED AS 1SIRS-1018 AND SIRS-1019, BUT THEIR POSITION ON THE DRAWING IDENTIFIED THEM AS 1SIRS-1017 AND 1SIRS-1020. INVESTIGATION SHOW THAT SNUBBERS 1SIRS-1017 AND 1SIRS-1020 AS IDENTIFIED ON THE DRAWING WERE SUPPOSED TO BE REMOVED BY MODIFICATION M22-1-89-14, BUT DUE TO INCORRECT LABELS, ISIRS-1018 AND ISIRS-1019 WERE REMOVED INSTEAD. ON 09/30/91 PER THE CORRECTIVE ACTIONS OF THIS ORIGINAL INVESTIGATION, A WALKDOWN OF THE INACCESSIBLE UNIT 2 HYDRAULIC SNUBBERS WAS PERFORMED. TWO DISCREPANCIES WERE FOUND DURING THIS WALKDOWN. SNUBBER 2RCRS-2082 WAS REMOVED INSTEAD OF 2RCRS-2083A, AND SNUBBER 2SIRS-2132 WAS REMOVED INSTEAD OF 2SIRS-2133. THIS EVENT WAS CAUSED BY A PERSONNEL ERROR DURING THE INSTALLATION OF MODIFICATIONS M22-1-89-14 AND M22-2-88-69. THE NUCLEAR ENGINEERING DEPARTMENT PERFORMED CALCULATIONS TO DETERMINE IF THE REVISED SNUBBER CONFIGURATION COULD ADEQUATELY SUPPORT THE DESIGN BASIS LOADS. THE CALCULATIONS CONCLUDED THAT ALL AFFECTED SUPPORTS WOULD HAVE REMAINED FUNCTIONAL THROUGHOUT A SEISMIC EVENT AND PIPING STRESSES WOULD NOT HAVE EXCEEDED DESIGN BASIS LIMITS. THE CORRECT SNUBBERS WERE REINSTALLED AND THE INCORRECT SNUBBERS WERE REMOVED.

[146] ZION 1 DOCKET 50-295 LER 91-015
CONTAINMENT INTEGRITY BROKEN ON POWER.
EVENT DATE: 100391 REPORT DATE: 110491 NSSS: WE TYPE: PWR
VENDOR: CHICAGO BRIDGE AND IRON COMPANY

(NSIC 223356) ON 10/3/91 AT 0031 HOURS, OPERATING PERSONNEL WERE EXITING UNIT 1 CONTAINMENT AFTER PERFORMING THEIR NIGHTLY ROUNDS. THE EQUIPMENT ATTENDANT INADVERTENTLY OPENED THE OUTER CONTAINMENT AIR LOCK DOOR FROM THE INSIDE OF CONTAINMENT. THE EQUIPMENT ATTENDANT IMMEDIATELY ATTEMPTED TO CLOSE THE OUTER DOOR, BUT FAILED TO CLOSE IT COMPLETELY. THE EQUIPMENT ATTENDANT CONTINUED TO MAKE HIS CONTAINMENT EXIT BY OPENING THE INNER DOOR. THIS SHOULD HAVE BEEN PREVENTED BY THE CONTAINMENT DOOR INTERLOCK MECHANISM, BUT THE INTERLOCK MECHANISM FAILED, WAS ABLE TO OPEN THE INNER DOOR. AS THE INNER DOOR WAS OPENED, THE EQUIPMENT ATTENDANT NOTICED AN OUT-RUSH OF AIR THROUGH THE DOOR. REALIZING THAT BOTH DOORS WERE OPEN, THE EQUIPMENT ATTENDANT EXITED CONTAINMENT AND IMMEDIATELY, SECURED THE INNER DOOR. HE THEN SECURED THE OUTER DOOR AND INFORMED THE SHIFT SUPERVISOR ABOUT WHAT HAD OCCURRED. SINCE THIS WAS CONSIDERED TO BE A BREECH OF CONTAINMENT INTEGRITY, THE UNIT AIR LOCK WAS DECLARED INOPERABLE PLACING THE UNIT ON A 24 HOUR CLOCK TO HOT SHUTDOWN. THE AIR LOCK WAS RETURNED TO OPERABLE STATUS 10/3/91 AT 1810 HOURS PRIOR TO THE 24 HOUR CLOCK EXPIRATION. THE CAUSE OF THIS EVENT WAS A COMPONENT FAILURE OF THE CONTAINMENT INTERLOCK MECHANISM. THIS EVENT DID NOT AFFECT THE HEALTH AND SAFETY OF THE PUBLIC.

[147] ZION 2 DOCKET 50-304 LER 91-005
MISSED POST MAINTENANCE VERIFICATION TEST OF 2FCV-PR19B DUE TO PERSONNEL ERROR.
EVENT DATE: 092291 REPORT DATE: 102291 NSSS: WE TYPE: PWR

(NSIC 223266) ON 09/22/91 AT 1600, THE LICENSED SHIFT SUPERVISOR (LSS) INCORRECTLY INSTRUCTED THE UNIT OPERATOR TO PERFORM PERIODIC TEST (PT)-300, CONTAINMENT ISOLATION VALVE STROKE TIME VERIFICATION, ON THE TRAIN A REACTOR LEAK DETECTOR AIR SPACE ISOLATION SAMPLE VALVE (IL), 2FCV-PR19A, INSTEAD OF THE TRAIN B SAMPLE VALVE, 2FCV-PR1 9B. THIS TEST WAS BEING PERFORMED FOR POST-MAINTENANCE VERIFICATION. THE LSS REVIEWED AND APPROVED THE COMPLETED PT-300, SIGNED THE "TESTS COMPLETED" AND "APPROVED COMPLETION" SECTIONS OF THE MAINTENANCE WORK REQUEST, AND COMPLETED THE PT-14, INOPERABLE EQUIPMENT SURVEILLANCE, PAPERWORK FOR 2FCV-PR19B. ON 09/24/91, THE SURVEILLANCE COORDINATOR, DURING REVIEW OF COMPLETED PT'S, IDENTIFIED THAT THE PT-300 WAS COMPLETED ON THE WRONG SAMPLE VALVE. PT-300 WAS PERFORMED ON 2FCV-PR19B AND THE VALVE STROKED WELL WITHIN THE MAXIMUM ALLOWABLE STROKE TIME. THIS EVENT WAS CAUSED BY A PERSONNEL ERROR ON THE

PART OF THE LSS WHO INSTRUCTED THE UNIT OPERATOR TO PERFORM POST MAINTENANCE VERIFICATION ON THE WRONG SAMPLE VALVE. THIS EVENT HAD MINIMAL SAFETY SIGNIFICANCE. CORRECTIVE ACTIONS INCLUDED COUNSELLING THE LSS, REVIEWING THE EVENT WITH ALL OPERATING DEPARTMENT PERSONNEL, REVIEWING THE SELF CHECK PROGRAM WITH THE OPERATING DEPARTMENT, AND PERFORMING A HUMAN PERFORMANCE EVALUATION SYSTEM (HPES) REVIEW ON THE INOPERABLE EQUIPMENT PROCESS TO IDENTIFY WHAT IMPROVEMENTS NEED TO BE MADE.

[148] ZION 2 DOCKET 50-304 LER 91-006
PRESSURIZER COOLDOWN RATE IN EXCESS OF 200F/HR.
EVENT DATE: 093091 REPORT DATE: 103091 NSSS: WE TYPE: PWR

(NSIC 223328) ON 9/30/91 AT 0328 HOURS, THE UNIT OPERATORS WERE PERFORMING GENERAL OPERATING PROCEDURE (GOP)-4, PLANT SHUTDOWN AND COOLDOWN. THE UNIT OPERATORS WERE ATTEMPTING TO COOL DOWN THE PRESSURIZER WHILE AT THE SAME TIME TAKE THE PRESSURIZER TO A WATER SOLID CONDITION USING CHARGING FLOW GREATER THAN LETDOWN FLOW TO INCREASE THE REACTOR COOLANT SYSTEM (RCS) INVENTORY. THE HIGH RATE OF CHARGING FLOW RESULTED IN DIFFICULTY IN CONTROLLING RCS PRESSURE DURING THIS EVOLUTION OPERATOR VERIFICATION SUBSEQUENT TO THIS EVENT SHOWED PRESSURIZER LIQUID SPACE TEMPERATURE DECREASED FROM 421F AT 0328 TO 214F AT 0428. THIS IS IN EXCESS OF TECHNICAL SPECIFICATION 3.3.2.D COOLDOWN LIMIT OF 200 F/HOUR. THIS EVENT WAS CAUSED BY INADEQUATE PROCEDURAL GUIDANCE. AN EVALUATION DETERMINED THAT THE PRESSURIZER STRUCTURAL INTEGRITY WAS NOT COMPROMISED BY THE HIGH COOL DOWN RATE. THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED. CORRECTIVE ACTIONS INCLUDE PROCEDURE CHANGES TO GOP-4, ENHANCED CLASSROOM AND SIMULATOR TRAINING ON PRESSURIZER COOLDOWN, AND REVIEWING THE EVENT WITH THE OPERATING DEPARTMENT.

COMPONENT INDEX

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

COMPONENT INDEX

SYSTEM INDEX

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

SYSTEM INDEX

KEYWORD INDEX

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

KEYWORD INDEX

KEYWORD INDEX

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. 10, No. 12
ORNL/NSIC-200

2. TITLE AND SUBTITLE

Licensee Event Report (LER) Compilation

For month of December 1991

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR
January	1992

5. AUTHOR(S)

Prepared by Oak Ridge National Laboratory

6. DATE REPORT ISSUED

MONTH	YEAR
January	1992

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)

December 1991

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

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

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS

Licensee Event Report
Sequence Coding and Search System
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

NUREG/CR-2000, Vol. 10, No. 12 LICENSEE EVENT REPORT (LER) COMPILATION JANUARY 1992