NUREG/CR-2000
ORNL/NSIC-200
Vol. 11, No. 3

Licensee Event Report (LER) Compilation

For month of March 1992

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, DC 20013-7082

and

National Technical Information Service
Springfield, VA 22161

NUREG/CR-2000
ORNL/NSIC-200
Vol. 11, No. 3

Licensee Event Report (LER) Compilation

For month of March 1992

Manuscript Completed: May 1992
Date Published: May 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

NOTE TO RECIPIENTS: CANCELLATION OF NUREG/CR-2000

The LICENSEE EVENT REPORT (LER) COMPILATION will no longer be issued. The last report will include data for the month of March 1992, Vol. 11, No. 3.

Licensee Event Report information is maintained for the Nuclear Regulatory Commission at the Oak Ridge National Laboratory by the Nuclear Operations Analysis Center (NOAC). This information is maintained in the Sequence Coding and Search System database which can be searched in a number of ways to obtain nuclear power plant operating experience data.

Contact W.P. Poore of NOAC to obtain information from this database:

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

Abstract

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

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

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

[1] ARNOLD DOCKET 50-331 LER 92-002
MISSED FIRE WATCH DUE TO PROCEDURAL INADEQUACIES.
EVENT DATE: 012692 REPORT DATE: 022592 NSSS: GE TYPE: BWR

(NSIC 224120) ON JANUARY 26, 1992, THE PLANT WAS OPERATING AT 83% POWER. DURING SURVEILLANCE TESTING A FIRE DOOR WAS BLOCKED OPEN WITHOUT THE REQUIRED TECHNICAL SPECIFICATION COMPENSATORY ACTION OF A FIRE WATCH. THE FIRE DOOR IS LOCATED AT THE SUMP ROOM ENTRANCE IN THE TURBINE BUILDING BASEMENT. UPON DISCOVERY OF THE LACK OF A REQUIRED FIRE WATCH, A FIRE WATCH WAS INITIATED. THE CAUSE OF THE EVENT WAS INADEQUATE PROCEDURAL INSTRUCTIONS. THE APPLICABLE SURVEILLANCE TEST PROCEDURES, SYSTEM OPERATING INSTRUCTIONS AND ANNUNCIATOR RESPONSE PROCEDURES HAVE BEEN CHANGED TO REFLECT THE REQUIREMENT FOR A FIRE WATCH WITH THE VENTILATION FANS SECURED AND THE SUMP ROOM FIRE DOOR BLOCKED OPEN. ALL THE FIRE DOORS THROUGHOUT THE PLANT ARE IN THE PROCESS OF BEING RELABELED TO HIGHLIGHT THE REQUIREMENT OF A FIRE WATCH IF THEY ARE BLOCKED OPEN. FIRE DETECTION EQUIPMENT IN THE ADJOINING AREA THAT CONTAINED SAFETY RELATED CABLES WERE OPERABLE THROUGHOUT THE EVENT. THE EVENT HAD NO EFFECT ON THE SAFE, CONTINUED OPERATION OF THE FACILITY.

[2] BIG ROCK POINT DOCKET 50-155 LER 92-004
ARCING CABLE OBSERVED IN RECIRCULATING PUMP ROOM CABLE TRAY.
EVENT DATE: 011492 REPORT DATE: 022492 NSSS: GE TYPE: BWR

(NSIC 224111) ON 1/14/92 @ 1100, ELECTRICIANS INSTALLING NEW ELECTRICAL CABLES IN THE RECIRCULATING PUMP (RCP) ROOM OBSERVED ARCING IN A CABLE TRAY ORIGINATING FROM A SINGLE CONDUCTOR CABLE. THE CABLE WAS ISOLATED AND PROTECTED BY THE ELECTRICIANS UNTIL TROUBLE-SHOOTING COULD BE DONE. THE INVESTIGATION DISCOVERED A TRIPPED BREAKER FEEDING THE RCP MOTOR HEATER, THEREFORE THE DEGRADED CABLE WAS QUICKLY IDENTIFIED AS A NON-SAFETY RELATED 120 VOLT HEATER CIRCUIT FOR THE NO. 1 RCP MOTOR. THE FACILITY HAS BEEN SHUTDOWN SINCE 11/30/91 FOR A SCHEDULED REFUELING OUTAGE. THE POLYETHYLENE CABLE HAD BECOME BRITTLE. AGE, HEAT, RADIATION AND LACK OF A PV C JACKET CONTRIBUTED TO THE BRITTLE CONDITION OF THE CABLE INSULATION. INDUSTRY EXPERIENCE WITH CABLE REPLACEMENT HAS SHOWN THAT MOVEMENT OF THE CABLE, WHETHER DIRECTLY OR AS A CONSEQUENCE OF REPLACING ADJACENT CABLES, CAN LEAD TO INSULATION DAMAGE IN THIS DEGRADED CONDITION. AN IMMEDIATE REVIEW WAS PERFORMED TO DETERMINE WHICH CABLES IN THE TRAY WERE SAFETY-RELATED, AND THE AFFECT THAT A HOT SHORT WOULD HAVE HAD ON THE EQUIPMENT THEY SERVICED. THESE CABLES WERE THEN REPLACED. SUBSEQUENTLY, A REVIEW WAS CONDUCTED TO IDENTIFY WHICH NON SAFETY-RELATED CABLES IN THE TRAY WERE STILL ENERGIZED. REPLACEMENT OF THE ENERGIZED CABLES (AS A PRECAUTION) AND DE-ENERGIZATION OF UNESSENTIAL CABLES WAS COMPLETED AS A RESULT OF THE REVIEW.

[3] BRUNSWICK 1 DOCKET 50-325 LER 92-002
AUTOMATIC ESF ACTUATION OF THE CONTROL BUILDING EMERGENCY AIR FILTRATION SYSTEM DUE TO SIMULTANEOUS FIRE ALARMS IN TWO ZONES OF THE CONTROL ROOM.
EVENT DATE: 011792 REPORT DATE: 021792 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)

(NSIC 224110) ON 1/17/92, UNITS 1 AND 2 WERE OPERATING AT 100% AND 85% POWER, RESPECTIVELY. AT 0215 SIMULTANEOUS FIRE ALARMS IN 2 ZONES OF THE CONTROL ROOM RESULTED IN AN AUTOMATIC INITIATION OF THE CONTROL BUILDING EMERGENCY AIR FILTRATION (CBEAF) SYSTEM. THE ALARMS WERE CAUSED BY COOKING IN THE CONTROL ROOM KITCHEN AND A PERSON SMOKING IN THE SECONDARY ALARM STATION FOR SECURITY. THE KITCHEN AND THE ALARM STATION ARE RESPECTIVELY LOCATED IN THE ELECTRONIC EQUIPMENT ROOM FIRE DETECTION ZONES. AT 0219, AFTER VERIFICATION THAT AN ACTUAL FIRE DID NOT EXIST, THE ALARMS WERE RESET AND THE CONTROL BUILDING HEATING, VENTILATING, AND AIR CONDITIONING AND CBEAF SYSTEMS WERE RESTORED TO NORMAL LINE UP. AN INVESTIGATION REVEALED THAT PRIOR TO THE ACTUATION, ONE OF THE INVOLVED FIRE PROTECTION ZONES HAD BEEN PLACED IN A CONDITION REQUIRING AN LCO, BUT THE RESPONSIBLE GROUP HAD NOT BEEN NOTIFIED TO INITIATE THE LCO. HOWEVER, THE FIRE PROTECTION ANNUNCIATORS ARE IN THE MAIN CONTROL ROOM AND THE AFFECTED FIRE DETECTION ZONE IS LOCATED IN A FREQUENTLY TRAVELED AREA. ADDITIONALLY, THE AREA IS IN CLOSE PROXIMITY TO AND ENVIRONMENTALLY COMMUNICATES WITH THE CONTROL ROOM PROPER. IT IS THEREFORE REASONABLE THAT A FIRE WOULD HAVE READILY BEEN DETECTED

WITHIN THE SAME FRAME AS THAT PROVIDED BY A FIRE WATCH. SIMILAR LERS 1-85-012 AND 1-86-007.

[4] BRUNSWICK 1 DOCKET 50-325 LER 92-003
PRIMARY UNINTERRUPTIBLE POWER SUPPLY INTERNAL FAILURE RESULTS IN REACTOR SCRAM.
EVENT DATE: 011792 REPORT DATE: 021892 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)
VENDOR: CYBEREX INC.

(NSIC 224079) AT 0852 ON JANUARY 17, 1992, UNIT 1 REACTOR WAS OPERATING AT 100% STEADY STATE REACTOR POWER. UNIT 2 WAS AT 85% REACTOR POWER. UNIT 1 AND 2 EMERGENCY CORE COOLING SYSTEMS (ECCS) WERE OPERABLE. UNIT 1 UNINTERRUPTABLE POWER SUPPLY (UPS) SUSTAINED MOMENTARY VOLTAGE LOSSES DUE TO FAILURE IN THE PRIMARY UPS STATIC SWITCH WHICH PREVENTED AUTOMATIC AND MANUALLY INITIATED ELECTRON TRANSFERS OF THE UPS LOADS TO THE ALTERNATE SOURCE. THE LOSS OF UNIT 1 UPS RESULTED IN THE COMMON STACK RADIATION MONITOR INITIATING A PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) GROUP 6 ISOLATION (CONTAINMENT ATMOSPHERE CONTROL), REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TRAIN (SBGT) INITIATIONS ON BOTH UNITS. ON UNIT 1, THE REACTOR VESSEL WATER LEVEL DECREASE AS A RESULT OF THE REACTOR FEED PUMP CONTROL SYSTEM RUNBACKS WITH A LOSS C UPS AND WITH SPEED CONTROL LOCKOUTS OCCURRING WHEN THE UPS VOLTAGE RETURNED. REACTOR VESSEL LEVEL PASSED THE LOW LEVEL (LL) #1 (162.5N) SETPOINT AND MOMENTARILY THE LL #2 (112.5N) SETPOINT. THESE INITIATED A REACTOR SCRAM AND THE FOLLOWING PCIS ISOLATIONS SIGNALS: GROUP (DRYWELL FLOOR AND EQUIPMENT DRAINS), GROUP 3 (REACTOR WATER CLEANUP), GROUP 6, AND A GROT 8 (RESIDUAL HEAT REMOVAL). AT LL #2, HIGH PRESSURE COOLANT INJECTION (HPCI) AND REACTOR C ISOLATION COOLING (RCIC) INITIATED AND INJECTED INTO THE REACTOR VESSEL UNTIL THEY TRIPPED ON HIGH REACTOR VESSEL LEVEL (210N).

[5] BRUNSWICK 2 DOCKET 50-324 LER 91-011 REV 01
UPDATE ON POTENTIAL LEAKAGE PAST REACTOR WATER CLEANUP SYSTEM PRIMARY CONTAINMENT INLET ISOLATION VALVES MAY HAVE RESULTED IN A PRIMARY CONTAINMENT PENETRATION NOT BEING COMPLETELY ISOLATED.
EVENT DATE: 082491 REPORT DATE: 121691 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 223638) ON 8/24/91, OPERATIONS PERSONNEL WERE ATTEMPTING TO PLACE THE REACTOR WATER CLEANUP (RWCU) SYSTEM IN SERVICE. THE LIGHT BULB ON THE MOTOR CONTROL CENTER FOR THE RWCU INLET OUTBOARD ISOLATION VALVE WAS BURNED OUT. DURING REPLACEMENT, THE BULB BROKE OFF IN THE SOCKET, CAUSING AN ELECTRICAL SHORT AND BLOWING THE CONTROL POWER FUSE. THIS RENDERED THE VALVE INOPERABLE. PER TECHNICAL SPECIFICATIONS FOR PRIMARY CONTAINMENT ISOLATION PURPOSES, THE INLET INBOARD ISOLATION VALVE WAS CLOSED FROM THE REACTOR-TURBINE GENERATOR BOARD. OPERATORS NOTED THAT THE PROCESS LINE DOWNSTREAM OF THE VALVE DID NOT DEPRESSURIZE AS EXPECTED. SEVERAL VALVES DOWNSTREAM WERE CYCLED WHICH CREATED SUFFICIENT DIFFERENTIAL PRESSURE TO CAUSE THE VALVE TO SEAT. THE CONTROL POWER FUSE FOR THE OUTBOARD ISOLATION VALVE WAS REPLACED. A SPECIAL TEST WAS DEVELOPED AND PERFORMED ON 8/25/91 TO QUANTIFY LEAKAGE AND ASSESS THE OPERABILITY OF THE INBOARD ISOLATION VALVE. THE RESULTS INDICATED THAT BOTH THE INBOARD AND OUTBOARD ISOLATION VALVES WERE LEAKING BY. WHEN A SUFFICIENT DIFFERENTIAL PRESSURE WAS PLACED ON THE VALVES, THEY BOTH APPEARED TO SEAT. PREVIOUS SIMILAR EVENTS INCLUDE: PCIS-LER 1-91-016, 2-87-001, 2-84-016, 2-79-078, 2-91-005; MCC LIGHTS 2-90-020.

[6] BRUNSWICK 2 DOCKET 50-324 LER 92-001
SCRAM DURING MAIN TURBINE CONTROL VALVE TESTING.
EVENT DATE: 020292 REPORT DATE: 030192 NSSS: GE TYPE: BWR

(NSIC 224154) ON 1/29/92, UNIT 2 WAS OPERATING AT APPROX. 100% STEADY STATE POWER. AN ANNUNCIATOR FOR LOW ELECTROHYDRAULIC (EHC) FLUID PRESSURE WAS RECEIVED. EHC PRESSURE SWINGS OCCURRED. REACTOR POWER WAS REDUCED TO ABOUT 85% POWER TO CLOSE #4 TURBINE CONTROL VALVE (TCV) WHICH REDUCED THE EHC PRESSURE SWINGS. ON 2/2/92, REACTOR POWER WAS REDUCED ABOUT TO 79% AND TCV TESTING BEGAN.

DURING #2 TCV TESTING, THE CONTROL OPERATOR (CO) DID NOT HAVE TIME TO RESET THE SCRAM LOGIC FOR THE 'B1' TRIP (RECEIVED AS EXPECTED) BEFORE THE 'A1' SWITCH TRIPPED AND A FULL SCRAM SIGNAL WAS RECEIVED CAUSING A REACTOR SCRAM. FOLLOWING THE REACTOR SCRAM, RCIC AUTOMATICALLY INITIATED AND INJECTED. HPCI INITIATED BUT LEVEL RECOVERED PRIOR TO ALLOWING INJECTION. THE CAUSE OF THIS EVENT IS PERCEIVED TO BE AIR OR NITROGEN TRAPPED IN THE TURBINE CONTROL VALVE FAST CLOSURE (TCVFC) LINE CREATING A PRESSURE PERTURBATION ON THE TCVFC PRESSURE SWITCH NO. 1 CAUSING IT TO TRIP. THE SEAL FAILURE OF THE ACCUMULATORS ASSOCIATED WITH THIS LINE ALLOWED NITROGEN TO ENTER THE EHC FLUID AND IT WAS LATER FOUND THAT THE GAS SIDE OF THE ACCUMULATOR WAS NEAR SOLID WITH EHC FLUID. THE ACCUMULATOR SEAL FAILURE IS ATTRIBUTED TO EXCESSIVE CYCLING WHICH BEGAN AFTER INSTALLATION OF THE PARTIAL ARC CONVERSION MODIFICATION DURING THE LAST REFUELING OUTAGE. SIMILAR LER: 90-017.

[7] BYRON 1 DOCKET 50-454 LER 92-001
REACTOR TRIP DUE TO TURBINE/GENERATOR TRIP CAUSED BY ANTI-MOTORING.
EVENT DATE: 012992 REPORT DATE: 022192 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 2 (PWR)

(NSIC 224099) ON 01/29/92, UNIT ONE REACTOR TRIPPED AS A RESULT OF A TURBINE TRIP FROM AN ANTI-MOTORING SIGNAL. MAINTENANCE WAS BEING PERFORMED ON A VALVE LABELLED 1MS096B, WHICH ISOLATES PRESSURE TRANSMITTER 1PT-506. THIS VALVE WAS NOT PROPERLY LABELLED AND IN FACT WAS 1MS096C WHICH ISOLATED 1PDS-T0071, THE ANTI-MOTORING RELAY. THIS ISOLATION RESULTED IN LOW DIFFERENTIAL PRESSURE SIGNAL ACROSS THE HIGH PRESSURE TURBINE AND ACTUATED THE ANTI-MOTORING TURBINE TRIP AND THUS THE REACTOR TRIP OCCURRED. THE INCORRECT VALVE TAGS WERE REMOVED AND REPLACED WITH CORRECT TAGS. THE OTHER UNIT WAS ALSO FOUND TO HAVE INCORRECT VALVE TAGS. THE INCORRECT TAGS WERE REPLACED EXCEPT FOR ONE LOCATED IN AN INACCESSIBLE AREA. THIS WILL BE REPLACED AT A LATER DATE. THIS EVENT IS REPORTABLE IN ACCORDANCE WITH 10CFR50.73(A)(Z)(IV).

[8] CALLAWAY 1 DOCKET 50-483 LER 92-001
'A' TRAIN EMERGENCY EXHAUST SYSTEM INCOMPLETE SURVEILLANCE DUE TO A HUMAN PERFORMANCE ERROR.
EVENT DATE: 091291 REPORT DATE: 021892 NSSS: WE TYPE: PWR

(NSIC 224101) ON 1/21/92, A UTILITY QUALITY ASSURANCE ENGINEER DISCOVERED FLOW HAD NOT BEEN MAINTAINED THROUGH THE 'A' TRAIN EMERGENCY EXHAUST FOR THE REQUIRED 10 HOURS DURING A TECHNICAL SPECIFICATION (T/S) 4.7.7.A MONTHLY SURVEILLANCE ON 9/12/91. ON 9/12/91 AT 0108 CDT, LICENSED OPERATORS PERFORMED THE T/S 4.7.7.A SURVEILLANCE FOR ONLY 9 HOURS AND 21 MINUTES. THE PLANT WAS IN MODE 1-POWER OPERATION, 93 PERCENT REACTOR POWER AT THE TIME OF THE EVENT. THE ROOT CAUSE OF THIS EVENT IS COGNITIVE HUMAN PERFORMANCE ERROR DURING THE COMPLETION AND REVIEW OF THE SURVEILLANCE PROCEDURE ACCEPTANCE CRITERIA DATA SHEET. THIS DATA SHEET DID NOT INCLUDE THE 10 HOUR ACCEPTANCE CRITERIA NOR REQUIRE THE TOTAL RUN TIME BE CALCULATED AND RECORDED. THE INDIVIDUALS INVOLVED HAVE BEEN INSTRUCTIVELY COACHED AND THEY HAVE REVISED THE EMERGENCY EXHAUST SURVEILLANCE PROCEDURE TO ADD THE 10 HOUR T/S ACCEPTANCE CRITERIA AND RUN TIME CALCULATION STEPS TO THE ACCEPTANCE CRITERIA DATA SHEET. OTHER SIMILAR T/S SURVEILLANCE PROCEDURES WERE REVIEWED. ONE WAS REVISED TO REQUIRE THE CALCULATED TOTAL RUN TIME TO BE RECORDED. THIS EVENT WILL BE INCLUDED IN LICENSED OPERATOR REQUALIFICATION TRAINING.

[9] CALLAWAY 1 DOCKET 50-483 LER 92-002
REACTOR TRIP ON A REACTOR COOLANT SYSTEM LOOP LOW FLOW SIGNAL DUE TO A WORKER INADVERTENTLY BUMPING OPEN AN INSTRUMENT VALVE.
EVENT DATE: 012292 REPORT DATE: 022092 NSSS: WE TYPE: PWR

(NSIC 224102) ON 1/22/92, AT 1106 CST, AN AUTOMATIC REACTOR COOLANT SYSTEM LOOP 'C' LOW FLOW REACTOR TRIP, AUXILIARY FEEDWATER ACTUATION AND FEEDWATER ISOLATION OCCURRED. THE PLANT WAS IN MODE 1 AT 100% POWER AT THE TIME OF THE TRIP. SEVEN UTILITY NON-LICENSED PERSONNEL WERE WORKING ON VALVES IN CONTAINMENT AT THE TIME OF THE EVENT. THE EVENT WAS CAUSED BY ONE OF THE WORKERS IN CONTAINMENT

[15] COMANCHE 1 DOCKET 50-445 LER 91-025 REV 01
UPDATE ON COMPONENT COOLING WATER STOP CHECK VALVES INOPERABLE DUE TO LESS THAN ADEQUATE PREVENTIVE MAINTENANCE.
EVENT DATE: 102991 REPORT DATE: 021992 NSSS: WE TYPE: PWR
VENDOR: ROCKWELL MANUFACTURING COMPANY

(NSIC 224098) AT 1215 ON OCTOBER 29, 1991, THE STOP CHECK VALVES (SCV) IN THE COMPONENT COOLING WATER LINES TO THE REACTOR COOLANT PUMP THERMAL BARRIERS WERE BEING TESTED TO SATISFY INSERVICE TESTING REQUIREMENTS. DURING THE TEST, FIVE OF THE EIGHT SCVS FAILED TO STROKE CLOSED. THE FIVE SCVS WERE SUBSEQUENTLY MANUALLY EXERCISED, AFTER WHICH THEY OPERATED AS DESIGNED. ON NOVEMBER 6, 1991, TWO OF THE SCVS WERE INSPECTED. A SMALL ACCUMULATION OF CORROSION PRODUCTS BETWEEN THE PLUG AND STEM OF THE VALVES, AND A SLIGHT SCALING ALONG THE BORE, WERE FOUND. LARGER ACCUMULATIONS OF CORROSION PRODUCTS THAT MAY HAVE BEEN PRESENT WERE FLUSHED OUT WHEN THE VALVES WERE MANUALLY EXERCISED. AT APPROXIMATELY 0215 ON NOVEMBER 23, 1991, ALL FIVE SCVS WERE IN SERVICE AND DETERMINED TO BE CAPABLE OF PERFORMING THEIR INTENDED FUNCTION. THE ROOT CAUSE OF THIS EVENT IS BELIEVED TO BE THE ACCUMULATION OF CORROSION PRODUCTS IN THE SCVS DUE TO LESS THAN ADEQUATE PREVENTIVE MAINTENANCE (PM). CORRECTIVE ACTION INCLUDES THE DEVELOPMENT OF A PM PROCEDURE TO MANUALLY EXERCISE THESE VALVES AND MONITORING OF THE EFFECTIVENESS OF THE PM.

[16] COMANCHE 1 DOCKET 50-445 LER 92-003
USE OF NONCONSERVATIVE INPUT ASSUMPTIONS LEADING TO INADEQUATE BORON DILUTION EVENT ANALYSIS.
EVENT DATE: 020392 REPORT DATE: 030492 NSSS: WE TYPE: PWR

(NSIC 224151) DURING A QUALITY ASSURANCE SURVEILLANCE OF SELECTED EVENT ANALYSES FOR COMANCHE PEAK UNIT 2, DEFICIENCIES WERE IDENTIFIED IN THE VENDORS ASSUMPTIONS AND METHODOLOGIES FOR ANALYSIS OF THE INADVERTENT BORON DILUTION EVENT. THE DEFICIENCIES WERE DETERMINED TO BE APPLICABLE TO UNIT 1, AND RENDERED INVALID THE MODE 3, 4, AND 5 BORON DILUTION EVENT ANALYSES. THE CONDITION WAS CAUSED BY THE USE OF INADEQUATE ANALYSIS INPUTS. CORRECTIVE ACTIONS INCLUDED REANALYSIS USING PLANT SPECIFIC DATA, CORRECTIONS TO LICENSING BASIS DOCUMENTS, AND RELIANCE ON COMPENSATORY MEASURES.

[17] CONNECTICUT YANKEE DOCKET 50-213 LER 91-024 REV 01
UPDATE ON INOPERABLE CONTAINMENT ISOLATION VALVE DUE TO A DESIGN DEFICIENCY.
EVENT DATE: 111491 REPORT DATE: 020792 NSSS: WE TYPE: PWR

(NSIC 224052) ON 10/15/91, AT 1600, WITH THE PLANT IN MODE 1 AT 100% POWER, A DESIGN DEFICIENCY WAS IDENTIFIED THAT COULD POSSIBLY PREVENT THE CONTAINMENT ISOLATION VALVE INSIDE OF CONTAINMENT FOR THE REACTOR COOLANT LETDOWN LINE (LD-TV-230) FROM CLOSING FOLLOWING RECEIPT OF A CONTAINMENT ISOLATION SIGNAL. IMMEDIATE CORRECTIVE ACTIONS CONSISTED OF DECLARING LD-TV-230 INOPERABLE AND ISOLATING THE LETDOWN LINE CONTAINMENT PENETRATION WITH A CLOSED MANUAL VALVE IN ACCORDANCE WITH TECH SPEC 3.6.3.C. AN ALTERNATE PATH FOR REACTOR COOLANT LETDOWN WAS ESTABLISHED VIA THE REACTOR COOLANT SYSTEM DRAIN LINE BACK TO THE CHEMICAL AND VOLUME CONTROL SYSTEM. THE NORMAL LETDOWN FLOWPATH THROUGH LD-TV-230 WAS REESTABLISHED AFTER APPROXIMATELY 14 HOURS, FOLLOWING IMPLEMENTATION OF A DESIGN CHANGE. AT THE TIME OF THE EVENT, A DETAILED OPERABILITY EVALUATION HAD BEEN INITIATED. ON 11/14/91, THE DETAILED OPERABILITY EVALUATION CONCLUDED THAT LD-TV-230 WAS INOPERABLE WITH THIS CONDITION EXISTING. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B), SINCE THE PLANT WAS IN OPERATION WITH A CONDITION PROHIBITED BY THE TECHNICAL SPECIFICATIONS.

[18] CONNECTICUT YANKEE DOCKET 50-213 LER 92-001
POTENTIAL AUXILIARY FEEDWATER PUMP INOPERABILITY DURING LI TING OPERATING CONDITIONS.
EVENT DATE: 012192 REPORT DATE: 022092 NSSS: WE TYPE: PWR
VENDOR: WORTHINGTON PUMP CORP.

(NSIC 224053) ON 1/21/92 AT 1000 HOURS WITH THE PLANT IN MODE 6 (REFUELING), AN

ENGINEERING EVALUATION DETERMINED THAT THE TURBINE DRIVEN AUXILIARY FEEDWATER (AFW) PUMPS WOULD POTENTIALLY BECOME INOPERABLE DURING MAXIMUM POSTULATED NON-HIGH ENERGY LINE BREAK (HELB) ENVIRONMENT OPERATING CONDITIONS. SPECIFICALLY, IT WAS DETERMINED THAT AFW PUMP BEARING COOLING IS POTENTIALLY INADEQUATE UNDER THESE LIMITING OPERATING CONDITIONS. CORRECTIVE ACTIONS INCLUDE REPLACING THE AFW PUMP BEARING LUBRICANT WITH A SYNTHETIC OIL WHICH HAS BETTER HIGH TEMPERATURE STABILITY, ADMINISTRATIVELY LOWERING THE DEMINERALIZED WATER STORAGE TANK (DWST) MAXIMUM TEMPERATURE FROM 120 TO 110 DEGREES (F), ADJUSTING THE DIGITAL SPEED CONTROL SYSTEM ON THE AFW SYSTEM TO LIMIT AFW PUMP DISCHARGE PRESSURE TO 1250 PSIG, AND IMPLEMENTING ADMINISTRATIVE CONTROLS TO LIMIT THE MAXIMUM TERRY TURBINE BUILDING ROOM TEMPERATURE TO 135 DEGREES (F). THESE ACTIONS WILL ASSURE TURBINE DRIVEN AFW PUMP OPERABILITY UNDER POSTULATED LIMITING, NON-HELB OPERATING CONDITIONS. A PROMPT REPORT OF THIS EVENT WAS ISSUED UNDER 10CFR50.72(b)(2)(III)(B) ON 1/21/92. THE EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(VII)(B) SINCE IT REPRESENTS A SINGLE CONDITION THAT COULD RESULT IN TWO INDEPENDENT TRAINS BEING INOPERABLE IN A SYSTEM DESIGNED TO REMOVE RESIDUAL HEAT.

[19] CONNECTICUT YANKEE DOCKET 50-213 LER 92-002
TEMPORARY LOSS OF SPENT FUEL COOLING DUE TO PREPLANNED SURVEILLANCE TESTING ACTIVITIES.
EVENT DATE: 013192 REPORT DATE: 022792 NSSS: WE TYPE: PWR

(NSIC 224112) ON JANUARY 31, 1992, AT 0926 HOURS WITH THE PLANT IN MODE 5 (COLD SHUTDOWN), THE SPENT FUEL COOLING PUMPS WERE TEMPORARILY DEENERGIZED DURING THE PERFORMANCE OF SURVEILLANCE PROCEDURE SUR 5.1-18 (TEST OF TRAIN A SAFETY INJECTION ACTUATION SIGNAL WITH A PARTIAL LOSS OF AC). THE PERFORMANCE OF THIS PROCEDURE WAS A PREPLANNED ACTIVITY IN THE PLANT REFUELING SURVEILLANCE TESTING SEQUENCE. DUE TO THE DESIGN OF THE HADDAM NECK ELECTRICAL DISTRIBUTION SYSTEM, PERFORMANCE OF SUR 5.1-18 RESULTS IN A TEMPORARY LOSS OF ELECTRICAL POWER TO THE SPENT FUEL COOLING PUMPS. POWER TO THE SPENT FUEL COOLING PUMPS WAS RESTORED AT 1100 HOURS AS PART OF THE RESTORATION ACTIVITIES ASSOCIATED WITH SUR 5.1-18. FORCED COOLING TO THE SPENT FUEL POOL WAS UNAVAILABLE FOR APPROXIMATELY 94 MINUTES. A PROMPT REPORT UNDER THE PROVISIONS OF 10CFR50.72(B)(2)(III)(B) WAS MADE ON JANUARY 31, 1992 IN ADVANCE OF THE PERFORMANCE OF SUR 5.1-18. THIS EVENT IS BEING CONSERVATIVELY REPORTED UNDER 10CFR50.73(A)(2)(V)(B) AS A CONDITION THAT ALONE COULD HAVE PREVENTED THE FULFILLMENT OF THE FUNCTION OF A SYSTEM NEEDED TO REMOVE RESIDUAL HEAT.

[20] COOK 2 DOCKET 50-316 LER 92-001
REFUELING WATER STORAGE TANK BORON CONCENTRATION GREATER THAN SPECIFICATION LIMIT DUE TO INCOMPLETE MIXING.
EVENT DATE: 012292 REPORT DATE: 022192 NSSS: WE TYPE: PWR

(NSIC 224077) ON 1-22-92, SAMPLE RESULTS OF THE UNIT 2 REFUELING WATER STORAGE TANK (RWST) IDENTIFIED THAT THE BORON CONCENTRATION WAS GREATER THAN THE TECHNICAL SPECIFICATION (TS) LIMIT. THE RWST WAS DECLARED INOPERABLE AND CONFIRMATORY SAMPLES WERE DRAWN WHICH CONFIRMED THE OUT-OF-SPECIFICATION BORON CONCENTRATION. DILUTION AND RECIRCULATION OF THE RWST WERE COMPLETED FOLLOWED BY A REACTOR SHUTDOWN AND DECLARATION OF AN UNUSUAL EVENT TO MEET THE TS ACTION STATEMENT REQUIREMENTS. DILUTIONS, RECIRCULATIONS, AND INCREASED SAMPLING CONTINUED UNTIL THE RWST BORON CONCENTRATION WAS RESTORED TO WITHIN SPECIFICATION. THE REACTOR SHUTDOWN AND UNUSUAL EVENT WERE TERMINATED ON THE RECEIPT OF A TEMPORARY WAIVER OF COMPLIANCE. AN ENGINEERING EVALUATION CONCLUDED THAT THE OUT-OF-SPECIFICATION RWST BORON CONCENTRATION WAS CAUSED BY THE COMBINATION OF INCOMPLETE MIXING IN THE RWST, LESS THAN PRECISE BLENDER CONTROL, AND INADEQUATE ADMINISTRATIVE CONTROL OF RWST BORON CONCENTRATION. THE FINDINGS OF THIS EVENT WERE REVIEWED BY THE OPERATIONS SUPERINTENDENT WITH THE OPERATIONS DEPARTMENT PERSONNEL. IN ADDITION, OPERATIONS DEPARTMENT PROCEDURES HAVE BEEN SCHEDULED TO BE REVISED TO ADDRESS THE FINDINGS OF THIS EVENT.

[21] COOPER DOCKET 50-298 LER 91-021
DIESEL GENERATOR START AND ESF GROUP ISOLATION DURING SURVEILLANCE TESTING DUE TO HUMAN PERSONNEL ERROR.
EVENT DATE: 121491 REPORT DATE: 011392 NSSS: GE TYPE: BWR

(NSIC 223788) ON DECEMBER 14, 1991, AT 9:38 AM, WITH THE PLANT IN COLD SHUTDOWN FOR A REFUELING OUTAGE, AN AUTOMATIC START OF THE #1 DIESEL GENERATOR AND ACTUATION OF THE GROUP 2 ISOLATION (SHUTDOWN COOLING), GROUP 3 ISOLATION (REACTOR WATER CLEANUP), GROUP 6 ISOLATION (REACTOR BUILDING, INCLUDING STANDBY GAS TREATMENT START), AND GROUP 7 ISOLATION (REACTOR COOLANT SAMPLING) OCCURRED. LICENSED PERSONNEL WERE IN THE PROCESS OF TIMING TWO RELAYS TO DETERMINE THE REASON FOR DISCREPANCIES IN PREVIOUSLY REPORTED TEST RESULTS. WHILE PERFORMING THE SURVEILLANCE PROCEDURE, A STEP WHICH WOULD BLOCK THE UNDERVOLTAGE TRIP OF BREAKER 1FA WAS OVERLOOKED. UPON ACTUATING THE TEST SWITCH, BREAKER 1FA TRIPPED, RESULTING IN THE ACTUATIONS. THE REACTOR WAS IN COLD SHUTDOWN, WITH A REACTOR COOLANT TEMPERATURE OF 150 DEGREES FAHRENHEIT. ALL EQUIPMENT FUNCTIONED AS DESIGNED. THE CAUSE OF THIS EVENT WAS THE INADVERTENT FAILURE OF THE LICENSED OPERATOR TO PERFORM AN ACTION SPECIFIED BY THE PROCEDURE. A CONTRIBUTING CAUSE WAS THE USE OF A NORMAL SURVEILLANCE PROCEDURE WITH A SIGNIFICANT NUMBER OF STEPS MARKED "NOT APPLICABLE (N/A)". THIS RESULTED IN THE STEPS TO BE PERFORMED BEING SEPARATED BY SEVERAL PAGES. THE PERSONNEL INVOLVED IN THIS EVENT WILL BE COUNSELLED, AND THIS EVENT WILL BE COVERED IN INDUSTRY EVENTS TRAINING.

[22] COOPER DOCKET 50-298 LER 92-002
REACTOR VESSEL WATER LEVEL SETPOINT INACCURACY RESULTING FROM REFERENCE LEG TEMPERATURE EFFECTS THAT HAD NOT BEEN CORRECTLY ADDRESSED.
EVENT DATE: 020392 REPORT DATE: 030492 NSSS: GE TYPE: BWR

(NSIC 224145) ON 2/3-4/92, TECHNICAL SPECIFICATION LIMITING CONDITIONS FOR OPERATION (LCOS) WERE ENTERED, REQUIRING PLANT LOAD REDUCTION AND SHUTDOWN DUE TO DISCOVERY OF NON-CONSERVATIVE REACTOR VESSEL WATER LEVEL 1 NOMINAL INSTRUMENT SETPOINTS. FOR REACTOR VESSEL WATER LEVEL INSTRUMENTS NBI-LIS-72A,B,C AND D, THIS DETERMINATION WAS MADE UPON REVIEWING SETPOINT CALCULATIONS PERFORMED IN 1981 IN RESPONSE TO GENERAL ELECTRIC (GE) SERVICE INFORMATION LETTER (SIL) NUMBER 299. FOR REACTOR VESSEL WATER LEVEL INSTRUMENTS NBI-LIS-57A AND B AND 58A AND B, THIS DETERMINATION WAS MADE UPON REVIEWING THE SETPOINT CHANGE REQUEST ISSUED IN 1983 IN CONJUNCTION WITH A MARK I CONTAINMENT PROGRAM DESIGN CHANGE. THE FUNCTIONS OF THE AFFECTED INSTRUMENTS ARE TO INITIATE LOW PRESSURE EMERGENCY CORE COOLING SYSTEMS, START THE DIESEL GENERATORS, SATISFY A PORTION OF THE AUTOMATIC DEPRESSURIZATION SYSTEM LOGIC AND INITIATE THE GROUP 1 AND 7 ISOLATIONS. THE REVIEWS WERE PERFORMED AS A RESULT OF ISSUANCE OF SIL 299, SUPPLEMENT 2 BY GE. WHEN THESE DISCREPANCIES WERE FOUND ON FEBRUARY 3, THE PLANT WAS AT FULL POWER; ON FEBRUARY 4, AT APPROXIMATELY 80 PERCENT POWER. THE REVISED INSTRUMENT SETPOINTS WERE IMPLEMENTED WITHIN THE RESPECTIVE LCO TIME FRAMES, ELIMINATING THE REQUIRED SHUTDOWNS AND ALLOWING A RETURN TO POWER.

[23] DRESDEN 2 DOCKET 50-237 LER 91-038
PRIMARY CONTAINMENT ISOLATION VALVE CLOSURE DUE TO REACTOR WATER CLEANUP SYSTEM ISOLATION.
EVENT DATE: 111391 REPORT DATE: 120391 NSSS: GE TYPE: BWR

(NSIC 223598) ON NOVEMBER 13, 1991, AT 1723 HOURS, WITH UNIT 2 SHUTDOWN, A REACTOR WATER CLEAN UP (RWCU) SYSTEM ISOLATION OCCURRED, RESULTING IN PRIMARY CONTAINMENT ISOLATION MOTOR OPERATED VALVES (MOVS) 2-1201-1 AND 2-1201-3 FULLY CLOSING. OPERATIONS PERSONNEL HAD JUST SECURED THE RWCU 2B RECIRCULATING PUMP AND WERE ATTEMPTING TO PLACE THE AUXILIARY PUMP IN SERVICE. THE RWCU SYSTEM ISOLATED ON A HIGH PRESSURE SIGNAL. WHEN TURNING RWCU PUMPS ON, RWCU SYSTEM PRESSURE OSCILLATIONS CAN BE 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. THIS EVENT HAD MINIMAL SAFETY SIGNIFICANCE BECAUSE THE ISOLATION OF THE RWCU SYSTEM WAS PROMPTLY RESET AND THERE WAS NO AFFECT ON REACTOR COOLANT CHEMISTRY. 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. THE MOST RECENT SIMILAR OCCURRENCE WAS REPORTED BY LER 91-31/050237.

[24] DRESDEN 2 DOCKET 50-237 LER 92-003
POSTULATED 2/3 DIESEL GENERATOR COOLING WATER PUMP RELATED FAILURE DUE TO DESIGN DEFICIENCY.
EVENT DATE: 012192 REPORT DATE: 021392 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 224055) ON JANUARY 21, 1992, WITH UNIT 2 IN COLD SHUTDOWN AND UNIT 3 IN A REFUEL OUTAGE, IT WAS DETERMINED DURING AN ENGINEERING REVIEW THAT A POSTULATED FLOOD IN THE CIRCULATING WATER INTAKE STRUCTURE RESULTING FROM A CIRCULATING WATER PUMP DISCHARGE PIPE BREAK COULD RENDER THE UNIT 2/3 DIESEL GENERATOR COOLING WATER PUMP (DGCWP) INOPERABLE IF FLOOD DAMAGE WERE TO OCCUR TO CONTROL CIRCUITRY CONTAINED IN AN UNSEALED POWER TRANSFER SWITCH JUNCTION BOX LOCATED BELOW THE DESIGN BASIS FLOOD LEVEL. THE ROOT CAUSE WAS ATTRIBUTED TO A DESIGN DEFICIENCY WITHIN A 1986 MODIFICATION INSTALLING THE TRANSFER SWITCH. THE SAFETY SIGNIFICANCE IS CONSIDERED MINIMAL BECAUSE THE SEQUENCE OF EVENTS REQUIRED FOR THIS EVENT TO OCCUR IS UNIQUE AND EXTREMELY REMOTE. CORRECTIVE ACTIONS INCLUDED PROPERLY SEALING THE JUNCTION BOX A CONDUIT INVOLVED, AND IDENTIFYING ON THE APPROPRIATE PRINTS THE LOCATIONS AND TYPE OF SEALANT INSTALLED.

[25] DRESDEN 2 DOCKET 50-237 LER 92-004
REACTOR WATER CLEANUP SYSTEM PRIMARY CONTAINMENT ISOLATION VALVE CLOSURE DUE TO PROCEDURE DEFICIENCY.
EVENT DATE: 012692 REPORT DATE: 022192 NSSS: GE TYPE: BWR

(NSIC 224056) ON 1/26/92, AT 0112 HOURS, WITH UNIT 2 SHUTDOWN, A REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATION OCCURRED, RESULTING IN PRIMARY CONTAINMENT ISOLATION (PCI) MOTOR OPERATED VALVES (MOVS) 2-1201-1 AND 2-1201-3 FULLY CLOSING. OPERATIONS PERSONNEL WERE ATTEMPTING TO START UP THE RWCU SYSTEM WITH THE RWCU AUXILIARY PUMP WHEN THE EVENT OCCURRED. THE RWCU SYSTEM ISOLATED TWICE ON SYSTEM HIGH PRESSURE SIGNALS. THE ROOT CAUSE WAS A DEFICIENCY WITHIN THE RWCU OPERATING PROCEDURE; THE SEQUENCE FOR STARTUP OF THE RWCU AUXILIARY PUMP RESULTED IN FLOW RESTRICTION. FLOW WAS RESTRICTED BY THE FLOW CONTROLLER AT THE DISCHARGE OF THE SYSTEM UPON THE STARTUP OF THE AUXILIARY PUMP. THE FLOW RESTRICTION CAUSED A PRESSURE SPIKE WHICH COULD NOT BE ADEQUATELY MITIGATED BY THE OPERATOR ADJUSTING THE PRESSURE CONTROLLER AT THE SYSTEM INLET. ALTHOUGH THIS EVENT WAS NOT INITIATED BY PCI LOGIC, IT DID RESULT IN CLOSURE OF PCI VALVES. THIS EVENT HAD MINIMAL SAFETY SIGNIFICANCE BECAUSE THE UNIT WAS SHUTDOWN AND THERE WAS NO ADVERSE EFFECT ON REACTOR WATER CHEMISTRY. IMMEDIATE CORRECTIVE ACTIONS WERE TO RESET AND RESTART THE RWCU SYSTEM. THE RWCU SYSTEM WAS RESTORED TO NORMAL OPERATION AT 0122 HOURS ON 1/26/92. A TEMPORARY PROCEDURE CHANGE WAS WRITTEN. A SIMILAR RWCU TRIP WAS REPORTED BY LER 91-38/050237.

[26] DRESDEN 2 DOCKET 50-237 LER 92-005
VIOLATION OF TECHNICAL SPECIFICATION LIMIT FOR INTERMEDIATE RANGE MONITOR OPERABILITY DUE TO PERSONNEL ERROR.
EVENT DATE: 020392 REPORT DATE: 022192 NSSS: GE TYPE: BWR

(NSIC 224136) AT 1024 HOURS ON 2/3/92, UNIT 2 WAS IN THE REFUEL MODE WITH PREPARATIONS FOR STARTUP IN PROGRESS. THE HIGH VOLTAGE CABLE WAS OBSERVED TO BE DISCONNECTED FROM THE CHASSIS CONNECTOR FOR INTERMEDIATE RANGE MONITOR (IRM 1). IRM 15 HAD BEEN BYPASSED AT 1400 HOURS ON 2/2/92, FOR MAINTENANCE WORK. BECAUSE IRM'S 15 AND 17 ARE BOTH PART OF REACTOR PROTECTION SYSTEM (RPS) CHANNEL 6, THIS LEFT TWO OPERABLE IRM'S (16 AND 18) FOR RPS CHANNEL B. THIS WAS LESS THEN THE TECHNICAL SPECIFICATION TABLE 3.1.1 REQUIREMENT OF A MINIMUM OF THREE OPERABLE IRM'S FOR EACH RPS CHANNEL WHILE IN THE REFUEL MODE. AS IMMEDIATE CORRECTIVE ACTION, THE CABLE WAS RECONNECTED. THE ROOT CAUSE OF THE EVENT WAS ATTRIBUTED TO PERSONNEL ERROR IN THAT THE CABLE WAS NOT RECONNECTED AS REQUIRED BY AN INSTRUMENT MAINTENANCE SURVEILLANCE PROCEDURE PERFORMED ON 2/1/92. DURING THE PERFORMANCE OF THE SURVEILLANCE, TWO OF THE FOUR IRM'S WERE OPERABLE FOR RPS CHANNEL 5 FOR THE LATTER PART OF THE SURVEILLANCE DUE TO THE DISCONNECTED HIGH

VOLTAGE CABLE FOR IRM 17. CONTRIBUTING CAUSES WERE PROCEDURE DEFICIENCY IN THAT THE SURVEILLANCE PROCEDURE DID NOT HAVE A VERIFICATION REQUIREMENT; AND MANAGEMENT DEFICIENCY IN THAT NO ADMINISTRATIVE CONTROLS WERE IN PLACE THAT WOULD HAVE REQUIRED SUCH A VERIFICATION REQUIREMENT IN THIS TYPE OF SURVEILLANCE PROCEDURE.

[27] DRESDEN 3 DOCKET 50-249 LER 91-015 REV 01
UPDATE ON UNPLANNED STANDBY GAS TREATMENT SYSTEM AUTOMATIC INITIATION SIGNAL DUE TO PERSONNEL ERROR.
EVENT DATE: 110591 REPORT DATE: 120591 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 2 (BWR)

(NSIC 224057) ON NOVEMBER 5, 1991, WITH UNIT 2 SHUTDOWN AND UNIT 3 IN A REFUEL OUTAGE, THE ELECTRICAL MAINTENANCE DEPARTMENT (EMD) WAS SCHEDULED TO PERFORM A PREVENTIVE MAINTENANCE SURVEILLANCE OF THE UNIT 3 REACTOR PROTECTION SYSTEM (RPS) MOTOR GENERATOR (MG) SET OUTPUT BREAKERS. WITH THE 'B' TRAIN OF THE STANDBY GAS TREATMENT SYSTEM (SBGTS) ALREADY OPERATING AND THE REACTOR BUILDING VENTILATION SYSTEM (RBVS) ISOLATED, THE EMD INSTALLED A TEMPORARY POWER SUPPLY TO THE CHANNEL 'B' REFUEL FLOOR AND REACTOR BUILDING VENTILATION RADIATION MONITORS TO PREVENT AN AUTOMATIC START OF THE SBGTS AND AN ISOLATION OF THE RBVS UPON THE REMOVAL OF THE 'A' RPS MG SET FROM SERVICE. THE OPERATIONS DEPARTMENT THEN REMOVED THE 'B' RPS MG SET FROM SERVICE, RESULTING IN THE DE-ENERGIZATION OF THE 'A' RPS BUS. THIS PRODUCED AN UNPLANNED SBGTS AUTOMATIC START SIGNAL AND AN UNPLANNED RBVS ISOLATION SIGNAL. THE CONTROL ROOM OPERATORS ANTICIPATED AN AUTOMATIC START OF THE 'A' TRAIN OF THE SBGTS UNDER THESE CONDITIONS; HOWEVER, THIS DID NOT OCCUR. THIS EVENT HAD MINIMAL SAFETY SIGNIFICANCE BECAUSE THERE WAS NO EFFECT ON PLANT STATUS. CORRECTIVE ACTIONS INCLUDE REVIEW OF THIS EVENT WITH OPERATIONS PERSONNEL, PROCEDURAL IMPROVEMENTS, AND A LABELLING EVALUATION. A PREVIOUS EVENT INVOLVING A UNPLANNED AUTOMATIC SBGTS START WAS REPORTED BY LER 91-04/050249.

[28] DRESDEN 3 DOCKET 50-249 LER 92-004
IMPROPER SETPOINT OF SECOND LEVEL UNDERVOLTAGE RELAYS DUE TO MANGEMENT DEFICIENCY.
EVENT DATE: 012292 REPORT DATE: 022192 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 2 (BWR)

(NSIC 224058) ON 7/31/91, AN EMERGENCY NOTIFICATION SYSTEM (ENS) NOTIFICATION WAS PERFORMED CONCERNING PRELIMINARY CALCULATIONS PERFORMED BY THE NUCLEAR ENGINEERING DEPARTMENT (NED) WHICH INDICATED THAT THE EXISTING SECOND LEVEL UNDERVOLTAGE SETPOINTS FOR 4KV EMERGENCY BUSES WERE TOO LOW TO PROVIDE ADEQUATE PROTECTION TO THE MOST LIMITING SAFETY RELATED COMPONENT. THE MOST LIMITING COMPONENT WAS DETERMINED TO BE THE UNIT 2 DIESEL GENERATOR COOLING WATER PUMP (DGCWP), WHICH IS FED FROM UNIT 2 DIVISION II AC DISTRIBUTION SYSTEM. DRESDEN STATION WAS NOTIFIED BY NED ON NOVEMBER 20, 1991, THAT FURTHER UNIT 2 DIVISION II CALCULATIONS HAD IDENTIFIED ADDITIONAL LOADS MORE LIMITING THAN THE DGCWP; THIS WAS REPORTED VIA EMERGENCY NOTIFICATION SYSTEM (ENS). BOTH UNIT 2 AND UNIT 3 WERE IN COLD SHUTDOWN AT THE TIME. ON JANUARY 22, 1992, NED NOTIFIED THE STATION THAT DEGRADED VOLTAGE CALCULATIONS HAD BEEN COMPLETED FOR UNIT 3 DIVISIONS I AND II. THESE CALCULATIONS, WHICH INCLUDED CREDIT FOR PLANNED MODIFICATIONS TO IMPROVE THE AVAILABLE VOLTAGE TO CRITICAL SAFETY RELATED COMPONENTS, INDICATED MINIMUM REQUIRED 4KV SAFETY BUS VOLTAGES FOR BUSES 33-1 AND 34-1 TO BE 3832 VOLTS AND 3792 VOLTS, RESPECTIVELY. THIS 4KV SAFETY BUS VOLTAGE CONCERN WAS PREVIOUSLY REPORTED BY LER 91-21/050237; A SUPPLEMENT TO LER 91-21/050237 WILL BE SUBMITTED TO PROVIDE FURTHER INFORMATION ON THIS TOPIC BY 3/31/92.

[29] FARLEY 2 DOCKET 50-364 LER 92-001
MANUAL REACTOR TRIP DUE TO A SERVICE WATER LEAK ON AN EXCITER COOLING WATER LINE.
EVENT DATE: 012292 REPORT DATE: 021892 NSSS: WE TYPE: PWR

(NSIC 224091) AT 0731 ON 1-22-92 A MANUAL REACTOR TRIP WAS INITIATED DUE TO A SERVICE WATER (SW) LEAK ON A MAIN GENERATOR EXCITER COOLING WATER LINE. BECAUSE OF THE LEAK, THE UNIT WAS BEING RAMPED DOWN AT A RATE OF 25MW/MIN AND WAS AT 65 PERCENT POWER WHEN THE TRIP WAS INITIATED. THE LEAK WAS ATTRIBUTED TO A GASKET THAT WAS DAMAGED DURING INSTALLATION WHEN THE EXCITER COOLING WATER LINES WERE

REMOTE MANUAL CLOSURE CAPABILITY OF THE PUMP SUCTION VALVES DESCRIBED IN THE FINAL SAFETY ANALYSIS REPORT IS DEFEATED. THE EVENT WAS CAUSED BY A DESIGN ERROR AND INADEQUATE REVIEW OF INDUSTRY OPERATING EXPERIENCE. THE DESIGN ERROR WILL BE C[illegible]TED PRIOR TO PLANT START-UP AND SYSTEMATIC REVIEW OF OTHER POTENTIALLY SI[illegible] CONDITIONS WILL BE COMPLETED. LER-91-026 IS A SIMILAR EVENT.

[33] FITZPATRICK DOCKET 50-333 LER 92-006
INADEQUATE PERFORMANCE OF FIRE WATCH DUTIES.
EVENT DATE: 012292 REPORT DATE: 022192 NSSS: GE TYPE: BWR

(NSIC 224109) THE PLANT WAS SHUTDOWN AND IN THE COLD CONDITION FOR MAINTENANCE AND REFUEL. ON JANUARY 22, 1992 A CONTINUOUS FIRE WATCH REQUIRED BY TECHNICAL SPECIFICATION 3.12.F IN AN AREA WITH FIRE BARRIER PENETRATION SEAL DEFICIENCIES AND VENTILATION DUCTWORK DAMPER DEFICIENCIES WAS OBSERVED TO BE LESS THAN FULLY ATTENTIVE TO THE ASSIGNED FIRE WATCH DUTIES. THE FIRE WATCH WAS RELIEVED BY ANOTHER FIRE WATCH PERSON AND DISCIPLINED. THE FIRE WATCH WAS ASSIGNED DUTIES IN AN AREA WITH VENTILATION SYSTEM FIRE DAMPER AND/OR FIRE BARRIER PENETRATION SEAL DEFICIENCIES. THE FIRE AREAS OF CONCERN CONTAIN PORTIONS OF SYSTEMS REQUIRED FOR ACCIDENT MITIGATION AND FOR SAFE SHUTDOWN IN THE EVENT OF POSTULATED FIRES. LER-92-001 DESCRIBES RELATED EVENTS CONCERNING INADEQUATE PERFORMANCE OF FIRE WATCH DUTIES.

[34] FITZPATRICK DOCKET 50-333 LER 92-007
FAILURE OF ANALOG TRANSMITTER TRIP SYSTEM TRIP RELAYS DUE TO THERMAL AGING.
EVENT DATE: 012392 REPORT DATE: 022492 NSSS: GE TYPE: BWR
VENDOR: AMERACE CORP.

(NSIC 224121) ON 1/23/91 WHILE THE PLANT WAS SHUT DOWN AND IN THE COLD CONDITION, TECHNICIANS WERE PERFORMING A ROUTINE SURVEILLANCE OF THE ANALOG TRANSMITTER TRIP SYSTEM (ATTS) WHEN AN EXCESSIVE TIME DELAY DUE TO A STICKING RELAY WAS NOTED. THE FAILED RELAY WAS REPLACED AND VISUAL EXAMINATION INDICATED SIGNS OF THERMAL STRESS ON THE RELAY INTERNALS. THE COIL SPOOL WAS DISCOLORED, CRACKED AND BRITTLE. THE DELAY IN RELAY DROP OUT TIME IS ATTRIBUTED TO COIL SPOOL DEBRIS INHIBITING RELAY PLUNGER MOVEMENT. THE NORMALLY ENERGIZED RELAY HAD BEEN IN SERVICE FOR APPROXIMATELY 4 YEARS LONGER THAN THE VENDOR RECOMMENDED SERVICE LIFE OF 3 YEARS. THESE RELAYS WERE ALSO USED IN THE NUMEROUS REACTOR PROTECTION SYSTEM, EMERGENCY CORE COOLING SYSTEM AND PRIMARY CONTAINMENT ISOLATION SYSTEM ACTUATION LOGIC (JE). A DETAILED ROOT CAUSE ANALYSIS SHALL BE PERFORMED TO VERIFY THAT FAILURE WAS DUE TO EXCESSIVE THERMAL AGING AND IDENTIFY POTENTIAL PROGRAMMATIC DEFICIENCIES. A SUPPLEMENTAL LER WILL BE SUBMITTED WHEN THIS ANALYSIS IS COMPLETE.

[35] FITZPATRICK DOCKET 50-333 LER 92-008
PRIMARY CONTAINMENT ISOLATION VALVE STEM PACKING NOT SUBJECTED TO LOCAL LEAK RATE TESTING FOLLOWING MAINTENANCE ON THE VALVES.
EVENT DATE: 012992 REPORT DATE: 022892 NSSS: GE TYPE: BWR
VENDOR: FISHER CONTROLS CO.

(NSIC 224122) THE PLANT WAS SHUTDOWN AND IN THE COLD CONDITION FOR MAINTENANCE AND REFUEL. ON 1/29/92 IT WAS DETERMINED THAT LOCAL LEAK RATE TESTING (LLRT) OF 2 PRIMARY CONTAINMENT (NH) ISOLATION VALVES FOLLOWING MAINTENANCE, WHICH MAY HAVE CHANGED THE LEAKAGE RATE OF THE VALVE STEM PACKINGS, DID NOT SUBJECT THE VALVE STEM PACKINGS TO TEST PRESSURE. DESIGN AND PHYSICAL ORIENTATION OF THE VALVES PREVENTS APPLICATION OF PRESSURE ON THE VALVE STEM PACKING DURING LLRT. THE DEFICIENCY WAS DISCOVERED AS PART OF REVIEW OF NRC INFORMATION NOTICE 86-16 AND WAS CAUSED BY A DESIGN ERROR AND INADEQUATE (NOT TIMELY) REVIEW OF OPERATING EXPERIENCE. CORRECTIVE ACTIONS HAVE NOT BEEN DETERMINED AT THIS TIME. AN UPDATE REPORT WILL BE SUBMITTED. LER-92-005 DESCRIBED AN EVENT WITH SIMILAR CAUSES.

REASSEMBLED AS PART OF THE FALL 1990 REFUELING OUTAGE. THIS EVENT WAS CAUSED BY IMPROPER GASKET INSTALLATION DUE TO LACK OF DETAILED INFORMATION ON REASSEMBLY OF VICTAULIC COUPLINGS IN THE EXCITER COOLER INSPECTION PROCEDURE. THE PROCEDURE IS IN THE PROCESS OF BEING REVISED TO INCLUDE SPECIFIC INSTRUCTIONS TO FACILITATE PROPER MATING AND ASSEMBLY OF VICTAULIC COUPLINGS. ALL ACCESSIBLE GASKETS IN THE NORTH (UPPER AND LOWER) EXCITER COOLERS WERE INSPECTED AND REPLACED AND VERIFIED TO BE LEAK FREE. ALL GASKETS IN THE NORTH AND SOUTH COOLERS WILL BE REPLACED DURING THE MARCH 1992 UNIT 2 OUTAGE. ALSO, DURING EACH 54 MONTH EXCITER INSPECTION, THE ASSOCIATED GASKETS WILL BE REPLACED.

[30]   FERMI 2   DOCKET 50-341   LER 92-001
UNPLANNED BREACH OF HPCI OIL SYSTEM.
EVENT DATE: 020592   REPORT DATE: 022892   NSSS: GE   TYPE BWR

(NSIC 224130) AT 1030 HOURS ON 2/5/92, DURING A PLANNED PREVENTIVE MAINTENANCE CALIBRATION OF HIGH PRESSURE COOLANT INJECTION (HPCI) TURBINE OIL COOLER DISCHARGE TEMPERATURE SWITCH 3R1N203, THE PRESSURE BOUNDARY OF THE HPCI TURBINE OIL SYSTEM WAS INADVERTENTLY BREACHED. THE INSTRUMENTATION AND CONTROL (I&C) TECHNICIAN PERFORMING THE WORK UNINTENTIONALLY REMOVED BOTH THE TEMPERATURE SENSOR AND THE THERMOWELL IN WHICH THE TEMPERATURE SENSOR WAS INSTALLED. THIS EXPOSED THE OIL SYSTEM TO ATMOSPHERE AND EFFECTIVELY RENDERED HPCI INOPERABLE. UPON RECOGNIZATION OF THIS CONDITION, THE I&C TECHNICIAN IMMEDIATELY RESTORED THE OIL SYSTEM PRESSURE BOUNDARY BY REPLACING THE SENSOR AND THERMOWELL AFTER APPROXIMATELY 30 SECONDS, AND INFORMED THE CONTROL ROOM OF HIS ACTIONS. OPERATIONS PERSONNEL DOCUMENTED THE HPCI SYSTEM INOPERABILITY WHILE THE OIL SYSTEM PRESSURE BOUNDARY WAS OPEN AND ENTRY WAS MADE INTO THE TECH SPEC ACTION STATEMENT FOR HPCI INOPERABILITY, INCLUDING REQUIRED VERIFICATION OF ALTERNATE SAFETY SYSTEM OPERABILITY. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE INADEQUATE DOCUMENTATION OF THE INSTALLED CONFIGURATION OF E41N203 WHICH WAS INCLUDED IN THE WORK INSTRUCTION TO THE I&C TECHNICIAN. TO PREVENT RECURRENCE OF THIS EVENT, A CORRECT DRAWING OF THE INSTALLED CONFIGURATION FOR 3R1N203 WILL BE CREATED AND REFERENCED IN THE PREVENTIVE MAINTENANCE INSTRUCTION.

[31]   FITZPATRICK   DOCKET 50-333   LER 92-004
AUTOMATIC FIRE SUPPRESSION SYSTEMS IN SAFETY-RELATED CABLE TUNNELS DECLARED INOPERABLE DUE TO INADEQUATE DESIGN AND REVIEW FOR APPENDIX R REQUIREMENTS.
EVENT DATE: 011592   REPORT DATE: 021492   NSSS: GE   TYPE: BWR

(NSIC 224082) THE PLANT WAS SHUTDOWN AND IN THE COLD CONDITION FOR MAINTENANCE AND REFUELING. ON 1/15/92 FIRE SUPPRESSION SYSTEMS (KP) FOR THE EAST AND WEST CABLE TUNNELS, WHICH CONTAIN POWER, CONTROL, AND INSTRUMENTATION CABLES REQUIRED FOR SAFE SHUTDOWN, WERE ADMINISTRATIVELY DECLARED INOPERABLE. THE FIRE SUPPRESSION SYSTEMS WERE DECLARED INOPERABLE (BUT REMAIN IN SERVICE) BECAUSE THE DESIGN OF THE ORIGINAL INSTALLATION WAS INADEQUATE. A CONTINUOUS FIRE WATCH WAS POSTED IN EACH TUNNEL AS REQUIRED BY TECHNICAL SPECIFICATION 3.12.B.1.B AND WILL REMAIN POSTED UNTIL MODIFICATION OF THE FIRE SUPPRESSION SYSTEMS ARE COMPLETED TO PROVIDE ADEQUATE PROTECTION AGAINST THE HAZARDS OF THE AREA. LER-91-010, 91-021, 91-024, AND 91-032 DESCRIBE ADDITIONAL FIRE PROTECTION INADEQUACY EVENTS. THIS REPORT IS ALSO SUBMITTED TO SATISFY THE SPECIAL REPORT REQUIREMENTS OF TECHNICAL SPECIFICATIONS 3.12.B.2 AND 6.9.B.2.

[32]   FITZPATRICK   DOCKET 50-333   LER 92-005
PRIMARY CONTAINMENT ISOLATION VALVES REMOTE MANUAL CLOSURE FUNCTION INOPERABLE DUE TO A DESIGN ERROR AND INADEQUATE OPERATING EXPERIENCE REVIEW.
EVENT DATE: 012292   REPORT DATE: 022192   NSSS: GE   TYPE: BWR

(NSIC 224083) THE PLANT WAS SHUTDOWN AND IN THE COLD CONDITION FOR MAINTENANCE AND REFUEL. ON 1/22/92 IT WAS DETERMINED THAT THE REMOTE MANUAL CLOSURE FEATURE OF PRIMARY CONTAINMENT (NH) ISOLATION VALVES FROM THE PRESSURE SUPPRESSION CHAMBER (TORUS) FOR HIGH PRESSURE COOLANT INJECTION (HPCI) (BJ) AND REACTOR CORE ISOLATION COOLING (RCIC) (BN) PUMP SUCTION LINES WAS DEFEATED UNDER CERTAIN CONDITIONS. WHEN HPCI AND/OR RCIC SUCTION IS AUTOMATICALLY TRANSFERRED TO THE TORUS DUE TO HIGH TORUS LEVEL AND/OR LOW CONDENSATE STORAGE TANK (KA) LEVEL, THE

[36] FT. CALHOUN 1 DOCKET 50-285 LER 91-028 REV 01
UPDATE ON UNMONITORED RELEASE ON LOSS OF 161KV SYSTEM.
EVENT DATE: 120191 REPORT DATE: 022892 NSSS: CE TYPE: PWR

(NSIC 224129) AT 0205 ON 12/1/91, IT WAS DISCOVERED THAT THE SAMPLE PUMP FOR THE LABORATORY AND RADIOACTIVE WASTE PROCESSING BUILDING (LRWPB) EXHAUST STACK GAS, IODINE AND PARTICULATE MONITORS WAS NOT RUNNING. AT THE TIME OF THE DISCOVERY, THE CHEMISTRY AND RADIATION PROTECTION (CARP) BUILDING EXHAUST FAN WAS RUNNING. OPERATION OF THE EXHAUST FAN WHILE THE SAMPLE PUMP WAS NOT RUNNING WAS DETERMINED TO CONSTITUTE AN UNMONITORED RELEASE IN VIOLATION OF PLANT TECHNICAL SPECIFICATION 2.9.1(2)H(I), AND IS REPORTABLE UNDER 10 CFR 50.73(A)(2)(I)(B). THE EVENT IS CONSIDERED TO HAVE BEEN CAUSED BY A POWER EXCURSION THE PREVIOUS DAY DUE TO SEVERE WINTER WEATHER. 161KV OFFSITE ELECTRICAL POWER WAS LOST AND SUBSEQUENTLY RESTORED TWICE ON 11/30/91. IT IS ASSUMED THAT WHEN THE INITIAL LOSS OF 161KV POWER OCCURRED AT 0430, THE LRWPB STACK RADIATION MONITORS SAMPLE PUMP STOPPED, AND DID NOT RESTART WHEN POWER WAS RESTORED. THE SAMPLE PUMP IS DESIGNED NOT TO RENERGIZE FOLLOWING A POWER INTERRUPTION UNTIL THE START/STOP SWITCH IS MANUALLY PUSHED. THE CARP BUILDING EXHAUST FAN RESTARTED WHEN POWER WAS RESTORED, RESULTING IN THE UNMONITORED RELEASE. DURING THE TIME THAT THE SAMPLE PUMP IS ASSUMED TO HAVE BEEN DEENERGIZED, ROUTINE RELEASES OCCURRED WHICH WERE FOUND TO BE NOT SIGNIFICANT. A TEMPORARY MODIFICATION HAS BEEN INSTALLED TO AUTOMATICALLY RESTART THE SAMPLE PUMP UPON RESTORATION.

[37] FT. CALHOUN 1 DOCKET 50-285 LER 92-001
UNMONITORED RELEASE ON LOSS OF 13.8 KV SYSTEM.
EVENT DATE: 012192 REPORT DATE: 022092 NSSS: CE TYPE: PWR

(NSIC 224071) IN 1/21/92, AT 1258 HOURS, FORT CALHOUN STATION UNIT NO. 1 WAS OPERATING AT 82% POWER, COASTING DOWN IN PREPARATION FOR A REFUELING OUTAGE, WHEN CONTROL ROOM PERSONNEL RECEIVED ALARMS THAT INDICATED A LOSS OF THE 13.8 KV ELECTRICAL SYSTEM. IMMEDIATION WAS TAKEN TO DETERMINE THE STATUS OF THE LABORATORY AND RADIOACTIVE WASTE PROCESSING BUILDING (LRWPB) EXHAUST STACK GAS, IODINE, AND PARTICULATE RADIATION MONITORS (RM41/042/043) AND THE ASSOCIATED SAMPLE PUMP. DURING THE FIVE MINUTES THAT THE SAMPLE PUMP AS DEENERGIZED DUE TO THE LOSS OF POWER, THE EXHAUST FANS WERE STILL IN OPERATION RESULTING AN UNMONITORED RELEASE. THIS IS IN VIOLATION OF TECHNICAL SPECIFICATION 2.9 AND IS BEING REPORTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 50.73(A)(2)(I)(B). THE CAUSE OF THIS EVENT IS THE INADEQUATE DESIGN OF THE RM-041/042/043 SAMPLE PUMP MOTOR CONTROL AND SUPERVISORY CIRCUITRY FOR A LOSS OF POWER. SINCE THERE WERE NO RADIOACTIVE RELEASES FROM THE 1RWPB STACK DURING THE TIME THAT THE SAMPLE SUMP WAS DEENERGIZED, THIS EVENT HAS MINIMAL NUCLEAR SAFETY SIGNIFICANCE. HOWEVER, NORMAL VENTILATION RELEASES WERE RESTARTED WHEN THE 13.8 KV ELECTRICAL SYSTEM WAS RESTORED. THE CAUSE OF THIS MOMENTARY LOSS OF THE 13.8 KV ELECTRICAL SYSTEM IS UNDETERMINED. CORRECTIVE ACTIONS INCLUDED IMMEDIATELY RESTARTING THE SAMPLE PUMP.

[38] FT. CALHOUN 1 DOCKET 50-285 LER 92-002
COMPROMISE OF CONTAINMENT INTEGRITY DUE TO PERSONNEL AIR LOCK DOOR SEAL LEAKAGE.
EVENT DATE: 012792 REPORT DATE: 022692 NSSS: CE TYPE: PWR

(NSIC 224114) ON JANUARY 27, 1992, WITH FORT CALHOUN STATION OPERATING AT 75 PERCENT POWER (MODE 1), THE INNER CONTAINMENT PERSONNEL AIR LOCK (PAL) DOOR FAILED TO MEET THE ACCEPTANCE CRITERIA OF THE PAL O-RING SEAL SURVEILLANCE TEST. THIS FAILURE TO MEET THE ACCEPTANCE CRITERIA INDICATED LEAKAGE THROUGH AT LEAST ONE OF TWO SEALS ON THE INNER DOOR. THE OUTER PAL DOOR WAS OPEN AT THE TIME TO ALLOW ACCESS TO THE INNER DOOR FOR TESTING. WITH THE OUTER DOOR OPEN AND THE INABILITY TO VERIFY CONTAINMENT INTEGRITY WAS SATISFIED BY THE INNER DOOR, THE CONDITION WAS CONSIDERED TO BE A VIOLATION OF TECHNICAL SPECIFICATION 6.(1)A. THIS REPORT IS BEING SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(I)(B). THE ROOT CAUSE OF THIS EVENT WAS CONDENSATE FROM NEARBY COMPONENT COOLING WATER PIPING DRIPPING ONTO THE INNER PAL DOOR BULKHEAD STRUCTURE AND UPPER LATCH BOLT BRACKET CAUSING SURFACE CORROSION. AN APPROXIMATELY 3/8-INCH BY 3/8-INCH BY 1/64-INCH FLAKE OF CORROSION DUCT WAS FOUND BETWEEN THE FLEXIBLE CIRCUMFERENTIAL SEAL AND ITS MATING SEAL ROD WHICH CAUSED THE LEAK. THIS EVENT DID NOT PRESENT A SIGNIFICANT HAZARD TO THE HEALTH AND SAFETY OF THE PUBLIC TO THE EXTREMELY LOW

PROBABILITY OF A DESIGN BASIS ACCIDENT OCCURRING WHEN THE OUTER PAL DOOR WAS OPEN. CORRECTIVE ACTIONS HAVE BEEN TAKEN WHICH ALLOW TESTING OF THE INNER PAL DOOR SEAL WITH OUTER PAL DOOR CLOSED.

[39] FT. CALHOUN 1 DOCKET 50-285 LER 92-003
TECHNICAL SPECIFICATION 2.19 VIOLATION DUE TO MISSED FIRE WATCH.
EVENT DATE: 012892 REPORT DATE: 022792 NSSS: CE TYPE: PWR

(NSIC 224115) AT 1035 HOURS ON JANUARY 28, 1992, WHILE IN MODE 1 AT 74 PERCENT POWER, ZONE 2 ALARM ON FIRE ALARM PANEL AI-54A ACTUATED. THIS ALARM WAS ACKNOWLEDGED AT THE AI-54A AND AI-56 (XL-3) PANELS. APPROXIMATELY FIVE HOURS LATER, THE ON-COMING OPERATING CREW REVIEWED THE COMPUTER PRINTOUT OF ACTIVITIES ON XL-3 AND NOTED THAT THIS ALARM HAD BEEN ACKNOWLEDGED AT 1035 HOURS, BUT HAD NOT BEEN RESET. DURING THIS TIME THE ZONE WAS INOPERABLE, AND COMPENSATORY MEASURES WERE NOT TAKEN AS REQUIRED BY TECHNICAL SPECIFICATION 2.19(1). THIS EVENT IS BEING REPORTED PURSUANT TO 10 CFR 50.73(A)(2)(I)(B). THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE PERSONNEL PERFORMANCE DUE TO A LACK OF ATTENTION/CONCENTRATION. A CONTRIBUTING CAUSE WAS INADEQUATE ADMINISTRATIVE CONTROL TO IDENTIFY FIRE DETECTORS POTENTIALLY IMPACTED BY WELDING ACTIVITIES, SO THAT APPROPRIATE COMPENSATORY MEASURES WOULD BE CONSIDERED. THE FAILURE TO ESTABLISH THE HOURLY FIRE WATCH FOR ROOM 13 HAD LIMITED SAFETY SIGNIFICANCE BECAUSE THE FIRE HAZARDS ANALYSIS DETERMINED THAT A LOSS OF ALL EQUIPMENT AND CABLING IN THE AREA WOULD NOT ADVERSELY AFFECT SAFE PLANT SHUTDOWN. UPON DETERMINATION THAT ZONE 2 WAS INOPERABLE, THE FIRE DETECTORS IN THE WELDING AREA OF ROOM 18 WERE DISABLED AND APPROPRIATE COMPENSATORY MEASURES WERE ESTABLISHED.

[40] GINNA DOCKET 50-244 LER 92-001 REV 01
UPDATE ON FAILURE OF CONTAINMENT RADIATION MONITOR DUE TO UNKNOWN CAUSE, CAUSES CONTAINMENT VENTILATION ISOLATION.
EVENT DATE: 010592 REPORT DATE: 021492 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INC

(NSIC 224025) ON JANUARY 5, 1992 AT APPROXIMATELY 0240 EST, WITH THE REACTOR AT APPROXIMATELY 98% FULL POWER, A CONTAINMENT VENTILATION ISOLATION OCCURRED DUE TO AN ACTUATION SIGNAL FROM THE CONTAINMENT PARTICULATE RADIATION MONITOR (R-11). ALL CONTAINMENT ISOLATION VALVES THAT WERE OPEN, CLOSED AS DESIGNED. IMMEDIATE OPERATOR ACTION WAS TO PERFORM THE APPLICABLE ALARM RESPONSE PROCEDURES ACTIONS. THIS INCLUDED VERIFYING AUTOMATIC ACTIONS, DETERMINING THE CAUSE OF THE CONTAINMENT VENTILATION ISOLATION, AND MAKING APPROPRIATE NOTIFICATIONS. THE IMMEDIATE CAUSE OF THE EVENT WAS DETERMINED TO BE THE FAILURE OF R-11. CORRECTIVE ACTION TAKEN WAS TO RETURN THE CONTAINMENT VENTILATION ISOLATION SYSTEM TO PRE-EVENT NORMAL STATUS, SEQUENTIALLY FOLLOWED BY A TROUBLESHOOTING EFFORT BY THE INSTRUMENT AND CONTROL DEPARTMENT, AND THEN CHANGEOUT OF THE R-11 DRAWER WITH QUALIFIED SPARE. FURTHER INVESTIGATION TO DETERMINE THE ROOT CAUSE IS CONTINUING.

[41] HATCH 1 DOCKET 50-321 LER 92-003
FAILURE OF SOLENOID OPERATED VALVES CAUSES LOSS OF EMERGENCY EQUIPMENT ROOM COOLERS.
EVENT DATE: 012192 REPORT DATE: 022092 NSSS: GE TYPE: BWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)

(NSIC 224078) ON 1/21/92 AT 0900 CST, UNIT 1 WAS IN THE RUN MODE AT A POWER LEVEL OF 2436 CMWT (100% RATED THERMAL POWER). AT THAT TIME, VALVES 1P41-F039A AND B, AIR OPERATED COOLING WATER SUPPLY VALVES TO EMERGENCY EQUIPMENT ROOM COOLERS 1T41-B002A AND B, FAILED TO OPEN AUTOMATICALLY AS REQUIRED DURING THE ROUTINE PERFORMANCE OF THE CORE SPRAY PUMP OPERABILITY TEST. THESE VALVES ARE DESIGNED TO OPEN AUTOMATICALLY TO PROVIDE COOLING WATER TO THE ROOM COOLERS TO MAINTAIN THE TEMPERATURE BELOW 148 DEGREES F WHEN THE CORE SPRAY AND/OR RESIDUAL HEAT REMOVAL PUMPS ARE IN OPERATION. WITH BOTH THE NORMAL AND STANDBY COOLERS FOR THIS ROOM INOPERABLE, CORE SPRAY PUMP 1E21-C001A AND RESIDUAL HEAT REMOVAL PUMPS 1E11-C002A AND C WERE DECLARED INOPERABLE. LIMITING CONDITION FOR OPERATION (LCO) 1-92-045 WAS INITIATED PER UNIT 1 TECHNICAL SPECIFICATIONS SECTIONS 3.5.A.3 AND 3.5.B.3. AT 1535 CST, A TEMPORARY MODIFICATION WAS IMPLEMENTED TO PLACE VALVES 1P41-F039A

AND B IN THE OPEN POSITION TO ASSURE A SUPPLY OF COOLING WATER TO THE EMERGENCY EQUIPMENT ROOM COOLERS. THIS RESTORED THE COOLERS TO AN OPERABLE STATUS AND LCO 1-92-043 WAS THEN TERMINATED. THE CAUSE OF THIS EVENT IS COMPONENT FAILURE. THE COOLING WATER SUPPLY VALVES FAILED TO OPEN BECAUSE THE SOLENOID OPERATED VALVES (SOVS) IN THE AIR SUPPLY LINES TO THESE VALVES FAILED TO REPOSITION WHEN GIVEN A SIGNAL TO DO SO. FAILURE OF THE SOVS TO REPOS ITION HAS NOT YET BEEN DETERMINED.

[42] HATCH 2 DOCKET 50-366 LER 92-002
PERSONNEL ERROR RESULTS IN AN UNPLANNED ESF ACTUATION.
EVENT DATE: 012792 REPORT DATE: 021992 NSSS: GE TYPE: BWR

(NSIC 224092) ON 1/27/92 AT 1017 CST, UNIT 2 WAS IN THE COLD SHUTDOWN MODE. AT THAT TIME, GROU P 5 PRIMARY CONTAINMENT ISOLATION SYSTEM VALVE 2G31-F001 CLOSED ON A REACTOR WAT ER CLEANUP (RWCU) SYSTEM HIGH DIFFERENTIAL FLOW ISOLATION SIGNAL. LICENSED OPERA TIONS PERSONNEL VERIFIED THAT THE AUTOMATIC ISOLATION SIGNAL WAS NOT VALID. NO A CTUAL SYSTEM LEAKAGE OCCURRED. THE SIGNAL OCCURRED AS INSTRUMENT & CONTROL (I&C) TECHNICIANS WERE PERFORMING A MONTHLY SCHEDULED SURVEILLANCE ON RWCU DIFFERENTI AL FLOW INSTRUMENTS 2G31-N603A & B IN ACCORDANCE WITH PROCEDURE 57SV-G31-002-2S, "RWCU SYSTEM DIFFERENTIAL FLOW INSTRUMENT FT&C." PER THE PROCEDURE, JUMPERS HAD BEEN PLACED IN THE HIGH DIFFERENTIAL FLOW TRIP LOGIC TO PRECLUDE CLOSURE OF GRO UP 5 ISOLATION VALVES 2G31-F001 AND F004 DURING THE SURVEILLANCE. WHILE CLOSING A LINK IN THE ISOLATION LOGIC FOR 2G31-F001, THE JUMPER IN THAT CIRCUIT WAS ACCI DENTALLY BUMPED, CAUSING IT TO DISENGAGE FROM THE CIRCUIT. THIS RESULTED IN A CL OSURE SIGNAL BEING GENERATED BY THE ISOLATION LOGIC AND THE VALVE CLOSED PER DES IGN. BY APPROXIMATELY 1032 CST, THE ISOLATION SIGNAL WAS RESET; VALVE 2G31-F001 WAS REOPENED; PROCEDURE 57SV-G31-002-2S WAS COMPLETED SATISFACTORILY; AND THE RW CU SYSTEM WAS RETURNED TO SERVICE. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR.

[43] HOPE CREEK 1 DOCKET 50-354 LER 92-001
TRIP OF REACTOR PROTECTION SYSTEM ELECTRICAL PROTECTION ASSEMBLY RESULTED IN HALF SCRAM AND INITIATION OF A & C CHANNELS OF NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM DUE TO PERSONNEL ERROR.
EVENT DATE: 012192 REPORT DATE: 021292 NSSS: GE TYPE: BWR

(NSIC 224090) ON 1/21/92 AT 1226, CONTROL ROOM PERSONNEL RECEIVED INDICATIONS OF A REACTOR PROTECTION SYSTEM (RPS) LOSS OF POWER, A HALF SCRAM, AND INITIATION OF CHANNEL A & C NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (N54). THE NUCLEAR CONTROL OPERATOR (NCO RO LICENSED) VERIFIED PROPER SYSTEM RESPONSE IAW SYSTEM OPERATING PROCEDURES. THE RPS BUS WAS RE-ENERGIZED VIA THE ALTERNATE POWER SUPPLY, THE HALF SCRAM AND ISOLATIONS RESET AND THE REACTOR WATER CLEAN UP SYSTEM WAS PLACED IN SERVICE. CONCURRENT TO THIS EVENT INSTRUMENT AND CONTROLS TECHNICIANS (I&C TECHS) WERE TAKING VOLTAGE READINGS ON THE OUTPUT OF THE RPS MG SET AT ONE OF THE ELECTRICAL PROTECTION ASSEMBLY (EPA) BREAKERS. WHEN THE TECHNICIAN PLACED THE TEST LEADS ON THE TERMINALS OF THE BREAKER, THE LEADS ARCED AND THE EPA BREAKER TRIPPED AS DID THE REDUNDANT EPA BREAKER AND THE MG SET OUTPUT BREAKER. THE TECHNICIANS TERMINATED TESTING AND RETURNED TO THE CONTROL ROOM AND EXPLAINED TO THE CONTROL ROOM PERSONNEL WHAT HAD OCCURRED. THE METER WHICH WAS USED WAS NO LONGER FUNCTIONING SO TESTING WAS RESUMED WITH A NEW METER. AN INSPECTION OF THE FAILED METER WAS PERFORMED BY THE VENDOR (FLUKE) WHO DETERMINED THE FAILURE WAS DUE TO THE LEADS BEING PLUGGED INTO THE WRONG TEST CONNECTIONS OF THE METER.

[44] HOPE CREEK 1 DOCKET 50-354 LER 92-002
OPERATION OF THE PLANT PROHIBITED BY TECHNICAL SPECIFICATION DUE TO CONCURRENT INOPERABILITY OF CONTROL ROOM VENTILATION SYSTEM.
EVENT DATE: 012892 REPORT DATE: 022692 NSSS: GE TYPE: BWR

(NSIC 224123) ON JANUARY 28, 1992 AT 2100 HOURS, CONTROL ROOM PERSONNEL WERE PERFORMING A MONTHLY SURVEILLANCE TEST ON THE "B" CONTROL ROOM VENTILATION TRAIN (CRV). AT 2007 THE "B" CRV TRAIN TRIPPED DURING TO A LOW EVAPORATOR PRESSURE TRIP ON THE "B" CHILLER. THE "B" CHILLER WAS RESTARTED AND TRIPPED AGAIN AT 2011. AS THE "B" CRV TRAIN WOULD NOT REMAIN IN SERVICE IT WAS DECIDED TO RETURN THE "A" CRV TRAIN TO SERVICE UNTIL THE CAUSE OF THE "B" CRV TRAIN MALFUNCTION COULD BE

DETERMINED AND CORRECTED. THE "A" CRV TRAIN WAS STARTED AT 2035 AND TRIPPED AT 2038. WITH BOTH CONTROL ROOM VENTILATION TRAINS INOPERABLE, TECHNICAL SPECIFICATION 3.0.3 WAS ENTERED AT 2038. EQUIPMENT OPERATORS (EO - NON LICENSED) MONITORING THE START LOCALLY NOTICED THAT CHILLER FREON LEVEL DROPPED TO AN ABNORMAL LOW LEVEL DURING THE START CYCLE RESULTING IN THE CHILLER TRIPPING ON LOW EVAPORATOR PRESSURE. THE CHILLER WAS SECURED AND APPROXIMATELY 3 BOTTLES OF FREON WERE ADDED TO THE CHILLER TO RESTORE THE FREON LEVEL TO A NORMAL PRESTART VALUE. THE "A" CRV TRAIN WAS RESTARTED AND REMAINED IN SERVICE FOR 30 MINUTES BEFORE DECLARING THE UNIT OPERABLE, CLEARING TECHNICAL SPECIFICATION 3.0.3. SUBSEQUENT INVESTIGATION REVEALED THE TRIPS WERE CAUSED BY A COMBINATION OF PRESSURE SWITCH SETPOINT DRIFT AND MARGINAL FREON LEVEL IN THE EVAPORATOR.

[45] HOPE CREEK 1 DOCKET 50-354 LER 92-003
ENGINEERED SAFETY SYSTEM ACTUATION DUE TO UNPLANNED START OF THE "A" CONTROL ROOM EMERGENCY FILTER SYSTEM DUE TO EQUIPMENT MALFUNCTION.
EVENT DATE: 012992 REPORT DATE: 022692 NSSS: GE TYPE: BWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 224131) ON 1/29/92 AT 1349 HOURS DURING PERFORMANCE OF A TECH SPEC SURVEILLANCE TEST AN INADVERTENT ISOLATION OF THE "A" CONTROL ROOM VENTILATION (CRV) SYSTEM AND START OF THE "A" CONTROL ROOM EMERGENCY FILTER (CREF) UNIT OCCURRED. THE NUCLEAR CONTROL OPERATOR (NCO RO -LICENSED) RECEIVED INDICATIONS OF A CHANNEL "C1" HIGH RADIATION SIGNAL WHICH GENERATED THE CRV ISOLATION AND CREF START. INSTRUMENT AND CONTROLS TECHNICIANS (I&C TECHS) WERE PERFORMING A SURVEILLANCE ON THE "D" CHANNEL RADIATION DETECTOR ASSOCIATED WITH THE "A" CRV AND CREF. THIS DETECTOR IS LOCATED IN THE SAME INLET PLENUM AS THE "C1" DETECTOR WHICH CAUSED THE ISOLATION OF THE "A" CRV AND START OF THE "A" CREF. THE TECHNICIANS WERE CONTACTED AND REPORTED TO THE MAIN CONTROL ROOM. INITIALLY IT WAS BELIEVED THAT AS THE TECHS WERE PASSING SOURCES NEAR THE "C1" DETECTOR DURING THE "D" DETECTOR TEST THEY HAD INADVERTENTLY CAUSED THE ACTUATION OF THE "C1" CHANNEL. THE CRV WAS RETURNED TO NORMAL OPERATION AND SURVEILLANCE TESTING WAS COMPLETED WITH NO FURTHER INCIDENTS. SUBSEQUENT INVESTIGATION REVEALED THAT THE FOIL COVER ON THE "C1" CHANNEL DETECTOR FAILED AND WHEN THE "D" CHANNEL DETECTOR WAS REMOVED FROM THE DUCT LIGHT ENTERING THE DUCT CAUSED A FALSE HIGH INDICATION ON THE "C1" DETECTOR. THE REPLACEMENT OF THE FOIL ON THE "C1" DETECTOR HAS BEEN SCHEDULED.

[46] INDIAN POINT 2 DOCKET 50-247 LER 92-002
REACTOR TRIP DUE TO MAIN FEEDWATER REGULATING VALVE GOING CLOSED.
EVENT DATE: 012792 REPORT DATE: 022692 NSSS: WE TYPE: PWR
VENDOR: WISCONSIN BRIDGE & IRON

(NSIC 224137) ON JANUARY 27, 1992 THE PLANT WAS OPERATING AT 100% POWER WHEN A REACTOR TRIP OCCURRED DUE TO STEAM FLOW-FEEDWATER FLOW MISMATCH. SUBSEQUENTLY, A HIGH STEAM FLOW-LOW REACTOR COOLANT SYSTEM (RCS) AVERAGE TEMPERATURE CONDITIONS RESULTED IN A SAFETY INJECTION (SI) SIGNAL. THE REACTOR TRIP WAS CAUSED BY A MAIN FEEDWATER REGULATING VALVE GOING TO THE CLOSED POSITION UNEXPECTEDLY. THE SI SIGNAL AROSE FROM A DECREASE IN RCS TEMPERATURE IN RESPONSE TO THE REACTOR TRIP COINCIDENT WITH HIGH STEAM FLOW SIGNALS FROM TWO STEAM GENERATORS (ALTHOUGH VERY HIGH STEAM FLOW CONDITIONS INDICATING A STEAM LINE BREAK DID NOT EXIST). FOLLOWING THE SI SIGNAL, CONTAINMENT ISOLATION, PHASE A, COULD NOT BE RESET WITHOUT USE OF INSTALLED KEYED BYPASS SWITCHES. IN THE SUBSEQUENT EVENT EVALUATION IT WAS DETERMINED THAT AN OPEN CIRCUIT IN THE SI INTERLOCK CIRCUITRY PREVENTED RESET. GIVEN THE INITIATING EVENT (CLOSURE OF THE FEEDWATER REGULATING VALVE) AND THE SI SIGNAL, ALL PLANT SAFETY SYSTEM REACTED IN ACCORDANCE WITH DESIGN. THE REACTOR TRIPPED AND THE APPROPRIATE CONTAINMENT VALVES CLOSED. THERE WAS NO IMPACT UPON THE HEALTH AND SAFETY OF THE PUBLIC.

[47] INDIAN POINT 3 DOCKET 50-286 LER 91-012 REV 01
UPDATE ON DESIGN BASES RECONSTITUTION REVEALED PROCEDURAL INADEQUACY THAT COULD HAVE RESULTED IN OVERLOADING VITAL BUSES DURING A LOCA.
EVENT DATE: 112191 REPORT DATE: 021892 NSSS: WE TYPE: PWR

(NSIC 224072) ON NOVEMBER 21, 1991, WITH THE REACTOR AT ONE HUNDRED PERCENT POWER, A VITAL BUS LOAD STUDY COMPLETED AS PART OF A DESIGN BASIS RECONSTITUTION EFFORT REVEALED A PROCEDURAL INADEQUACY THAT COULD POTENTIALLY RESULT IN AN OVERLOAD OF THE VITAL BUSES DURING A LOSS OF COOLANT ACCIDENT WITH OFFSITE POWER AVAILABLE. ON JANUARY 9, 1991 A ONE HOUR REPORT WAS MADE TO THE NRC REGARDING AN UNANALYZED CONDITION THAT COULD OCCUR WHILE OPERATING UNDER DEGRADED VOLTAGE CONDITIONS COINCIDENT WITH AN ENGINEERED SAFEGUARDS ACTUATION. THE ROOT CAUSE OF THE EVENTS WAS INSUFFICIENT CORRELATION BETWEEN PLANT DESIGN DOCUMENTS AND THE PLANT EMERGENCY PROCEDURES. EMERGENCY AND OPERATING PROCEDURES HAVE BEEN REVISED TO CORRECT THE PROCEDURAL INADEQUACY AND PRECLUDE THE POSTULATED EVENTS. LONG TERM CORRECTIVE ACTIONS ARE BEING DEVELOPED AND WILL BE IDENTIFIED IN A SUPPLEMENT TO THIS LER.

[48]   INDIAN POINT 3   DOCKET 50-286   LER 92-002
INOPERABLE SAFEGUARDS COMPONENTS DUE TO FAULTED INDICATING LAMP.
EVENT DATE: 012392   REPORT DATE: 022492   NSSS: WE   TYPE: PWR

(NSIC 224073) ON JANUARY [illegible], 1991, WITH THE REACTOR OPERATING AT 100 PERCENT POWER, AN INDICATING LAMP ON THE AUXILIARY BOILER FEED PUMP LOCAL CONTROL PANEL ELECTRICALLY FAULTED. AS A RESULT, A SIX AND ONE-QUARTER AMPERE CONTROL POWER FUSE BLEW, RENDERING THE AUTOMATIC INITIATION FEATURE OF SAFEGUARDS EQUIPMENT ON 480 VOLT VITAL BUS 5A INOPERABLE. THE TECHNICAL SPECIFICATION LIMITING CONDITIONS FOR OPERATION ACTION STATEMENT FOR CONTAINMENT COOLING AND IODINE REMOVAL SYSTEMS WAS ENTERED. A PLANT SHUTDOWN WAS STARTED AND AN UNUSUAL EVENT DECLARED. THE FAULTED LAMP SOCKET LEADS WERE LIFTED VIA A TEMPORARY MODIFICATION, CLEARING THE ELECTRICAL FAULT. THE FUSE WAS REPLACED. THE PLANT SHUTDOWN AND UNUSUAL EVENT WERE TERMINATED. CORRECTIVE ACTIONS INCLUDE REPLACING THE LAMP SOCKET DURING THE NEXT REFUELING OUTAGE, REEVALUATING QUALIFIED FUSE INVENTORIES, AND CONSIDERING USE OF LAMP SOCKETS WITH CURRENT LIMITERS OR BULB REPLACEMENT WITH LEDS.

[49]   KEWAUNEE   DOCKET 50-305   LER 92-001
TRIP TESTING OF WESTINGHOUSE TYPE EB MOLDED CASE CIRCUIT BREAKER REVEALS HIGH RATE OF OUT-OF-TOLERANCE TRIP VALUES.
EVENT DATE: 011092   REPORT DATE: 021092   NSSS: WE   TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 224035) ON 1/10/92, WITH THE PLANT AT 100% POWER, A MANAGEMENT REVIEW OF MOLDED CASE CIRCUIT BREAKER (MCCB) TEST RESULTS FOUND THAT A SIGNIFICANT NUMBER OF MCCBS WITH OUT-OF-TOLERANCE TRIP VALUES WERE IDENTIFIED WHILE TESTING WESTINGHOUSE TYPE EB MCCBS. THE TESTING INVOLVED TWO TRIP TESTS, 1) TO DETERMINE THE TIME REQUIRED FOR THERMAL TRIPPING AT 300% OF RATED CURRENT, AND 2) THE MULTIPLE OF RATED CURRENT REQUIRED FOR MAGNETIC TRIPPING IN LESS THAN 0.1 SECONDS. THIRTY-FIVE SPARE BREAKERS WERE SELECTED FOR TESTING, OF THESE 10 WERE FOUND TO HAVE OUT-OF-TOLERANCE (LONG) THERMAL TRIPPING, AND 1 OF THE 10 HAD AN OUT-OF-TOLERANCE (HIGH) CURRENT MULTIPLE FOR MAGNETIC TRIPPING. THE MAGNETIC TRIP WAS NOT TESTED ON 2 OF THE 10 DUE TO THE INABILITY TO RESET THE BREAKERS AFTER THE THERMAL TRIP. PRESENTLY, THE CAUSE OF THE OUT-OF-TOLERANCE TRIPS IS ATTRIBUTED TO THE INFREQUENT EXERCISING OF THE BREAKERS. SOME OF THESE BREAKERS MAY NOT HAVE BEEN EXERCISED SINCE INITIAL PLANT STARTUP. ACTIONS HAVE BEEN INITIATED TO ADDRESS THE CONCERNS ASSOCIATED WITH THE UNEXPECTED RATE OF OUT-OF-TOLERANCE TRIP VALUES.

[50]   LA SALLE 1   DOCKET 50-373   LER 91-015 REV 02
UPDATE ON INADEQUATE TESTING OF DIESEL GENERATORS DUE TO INADEQUATE PROCEDURES/TECHNICAL SPECIFICATION MISINTERPRETATION.
EVENT DATE: 110791   REPORT DATE: 030692   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 2 (BWR)

(NSIC 224158) ON 11/7/91 AT APPROX. 1500 HOURS, WITH UNIT 1 AND UNIT 2 IN OPERATONAL CONDITION 1 (RUN) AT 100% POWER, DURING A NUCLEAR REGULATORY COMMISSION (NRC) ELECTRICAL DISTRIBUTION SYSTEM FUNCTIONAL INSPECTION (EDSFI) IT WAS DETERMINED, THAT CERTAIN EMERGENCY SAFETY FEATURE (ESF) BUS UNDERVOLTAGE RELAY CONTACTS WERE NOT FUNCTIONALLY TESTED AS REQUIRED BY PLANT TECH SPEC

4.8.1.1.2.D.4. THE STATION WAS PLACED ON A 24 HOUR TIMECLOCK AND THE REQUIRED CONTACT TESTING WAS COMMENCED TO FULFILL THE TECH SPEC SURVEILLANCE REQUIREMENTS. ON 11/8/91, AT 0900 HOURS IT WAS DETERMINED THAT THE ALLOWANCE OF THE 24 HOUR PERIOD TO PERFORM THE REQUIRED SURVEILLANCES WAS INAPPROPRIATE AND A LATE NOTIFICATION WAS MADE. TECHNICAL SPECIFICATION 3.0.5 WAS ENTERED. ALL TESTING WAS COMPLETED AT 1030 HOURS ON 11/8/91. THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE TESTING PROCEDURES. THE SAFETY CONSEQUENCES OF THIS EVENT WERE MINIMAL. THE 0, 1A, 2A, 1B, AND 2B DIESEL GENERATORS WERE INOPERABLE SOLELY DUE TO THE MISSED SURVEILLANCE AND WERE FULLY FUNCTIONAL THROUGHOUT THE EVENT, EXCEPT WHILE EACH WAS UNDERGOING TESTING IN ACCORDANCE WITH STATION CORRECTIVE ACTIONS. THIS EVENT IS BEING REPORTED PURSUANT TO 10CFR50.73(A)(2)(I) DUE TO A DEVIATION FROM PLANT TECH SPECS.

[51] LIMERICK 1 DOCKET 50-352 LER 92-001
MANUAL ISOLATION OF THE MAIN CONTROL ROOM DUE TO A HIGH TOXIC CHEMICAL CONCENTRATION ALARM.
EVENT DATE: 012992 REPORT DATE: 021392 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 224089) ON 1/29/92, A HIGH TOXIC CHEMICAL CONCENTRATION ALARM WAS RECEIVED IN THE MAIN CONTROL ROOM (MCR) AS A RESULT OF A HIGH VINYL CHLORIDE CONCENTRATION IN THE MCR OUTSIDE AIR INTAKE PLENUM, AS DETECTED BY THE 'A' AND 'B' TOXIC GAS ANALYZERS. MCR PERSONNEL IMMEDIATELY IMPLEMENTED SPECIAL EVENT PROCEDURE SE-2, "TOXIC GAS" AND DONNED SELF-CONTAINED BREATHING APPARATUS (SCBA). ADDITIONALLY, MCR PERSONNEL MANUALLY INITIATED A MCR VENTILATION SYSTEM CHLORINE ISOLATION, AN ENGINEERED SAFETY FEATURE (ESF). THE 'A' TRAIN OF THE CONTROL ROOM EMERGENCY FRESH AIR SUPPLY (CREFAS) SYSTEM, ALSO AN ESF, INITIATED AS DESIGNED AND PROVIDED TOTAL RECIRCULATION OF THE MCR AIR WITHOUT ANY INTAKE FROM THE OUTSIDE ATMOSPHERE. THE 'A' AND 'B' MCR TOXIC GAS ANALYZERS DETECTED A VINYL CHLORIDE CONCENTRATION OF 21.29 PPM AND 21.25 PPM RESPECTIVELY (I.E., ALARM SETPOINT IS 10 PPM) WHICH IS WELL BELOW THE HAZARDOUS CONCENTRATION LIMIT OF 1000 PPM. CHEMISTRY PERSONNEL OBTAINED AND ANALYZED SAMPLES OF THE MCR AIR AND THE MCR OUTSIDE AIR PLENUM. THE SAMPLES SLOWED NO DETECTABLE VINYL CHLORIDE THEREFORE MCR PERSONNEL REMOVED SCBA AND THEN RESET THE CHLORINE ISOLATION. THIS EVENT WAS CAUSED BY THE PRESENCE OF VINYL CHLORIDE IN THE MCR OUTSIDE AIR INTAKE PLENUM FROM AN ATMOSPHERIC RELEASE OF THE CHEMICAL FROM THE OCCIDENTAL CHEMICAL CORPORATION (OCC) WHICH IS LOCATED IN THE VICINITY OF LIMERICK GENERATING STATION.

[52] LIMERICK 2 DOCKET 50-353 LER 92-002
REFUEL FLOOR ISOLATION DUE TO FAILED GEIGER MUELLER TUBE IN RADIATION DETECTOR.
EVENT DATE: 011992 REPORT DATE: 021492 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 224108) ON 1/19/92, A MOMENTARY SPIKE OCCURRED ON THE 'D' CHANNEL OF THE UNIT 2 REFUELING FLOOR EXHAUST DUCT RADIATION MONITOR. THIS MOMENTARY SPIKE RESULTED IN AUTOMATIC ACTUATIONS OF THE PRIMARY CONTAINMENT AND THE REACTOR VESSEL ISOLATION CONTROL SYSTEM (PCRVICS), AN ENGINEERED SAFETY FEATURE. THE ACTUATION CLOSED ISOLATION VALVES TO ONE OF THE UNIT 2 PRIMARY CONTAINMENT H2/O2 COMBUSTIBLE GAS ANALYZERS (CGA) AND THE 2B CONTAINMENT HYDROGEN RECOMBINER. THE SHIFT INSTRUMENTATION AND CONTROLS (IC) TECHNICIANS CLEANED THE CONNECTOR ON THE DETECTOR CABLE. FOLLOWING THE CONNECTOR CLEANING, THE RADIATION MONITOR WAS RETURNED TO SERVICE WITH NO ABNORMAL INDICATIONS. THE ISOLATIONS WERE RESET IN ACCORDANCE WITH PLANT PROCEDURES AND THE SYSTEMS WERE RESTORED EXPEDITIOUSLY BY OPERATORS, PREVENTING ANY ADVERSE IMPACTS ON PLANT SYSTEMS. THE SYSTEMS RESPONDED AS DESIGNED AND THE REDUNDANT CGA AND 2A CONTAINMENT HYDROGEN RECOMBINER REMAINED OPERABLE. THE CAUSE OF THIS EVENT WAS UNEXPECTED EQUIPMENT FAILURE. THE 'D' CHANNEL GEIGER-MUELLER (GM) TUBE DETECTOR EXPERIENCED A SPIKE DUE TO SOME FORM OF DEGRADATION SYMPTOMATIC OF END OF LIFE OF THE GM TUBE. THIS IS THE FIRST INSTANCE OF THIS TYPE OF FAILURE AT LIMERICK. THE RADIATION SENSOR/CONVERTER (WHICH INCLUDES THE GM TUBE) WAS REPLACED ON 1/31/92.

[53] LIMERICK 2 DOCKET 50-353 LER 92-003
A WATERTIGHT DOOR, WHICH SEPARATES THE RESIDUAL HEAT REMOVAL PUMP ROOMS, WAS DISCOVERED OPEN, RESULTING IN A CONDITION OUTSIDE OF THE MODERATE ENERGY PIPE BREAK DESIGN BASIS.
EVENT DATE: 020492 REPORT DATE: 022892 NSSS: GE TYPE: BWR

(NSIC 224148) ON 2/4/92, DURING PERFORMANCE OF THE DAILY FIRE DOOR POSITION VERIFICATION SURVEILLANCE TEST, A FIREWATCH DISCOVERED THAT WATERTIGHT DOOR NO. 75 WAS OPEN AND UNSUPERVISED. DOOR NO. 75 SEPARATES THE RESIDUAL HEAT REMOVAL (RHR) 2A/2C AND 2B/2D PUMP ROOMS. THE FIREWATCH IMMEDIATELY CLOSED AND DOGGED THE DOOR AND NOTIFIED THE MAIN CONTROL ROOM. AN EVALUATION CONCLUDED THAT THE DOOR WAS OPEN FOR A PERIOD OF 22 MINUTES. DOOR NO. 75 IS REQUIRED TO BE ALWAYS CLOSED AND DOGGED FOR MODERATE ENERGY PIPE BREAK (MEPB) CONSIDERATIONS. THEREFORE, WITH THE DOOR OPEN, THE MEPB BARRIER BETWEEN THE RHR PUMP ROOMS WAS OUTSIDE THE MEPB DESIGN BASIS. ADDITIONALLY, DOOR NO. 75 IS REQUIRED FOR FIRE PROTECTION CONSIDERATIONS PER THE TECHNICAL SPECIFICATIONS (TS) SECTION 3.7.7. HOWEVER, SINCE THERE WERE OPERABLE FIRE DETECTORS IN BOTH RHR PUMP ROOMS, AND THE DOOR WAS CLOSED IN LESS THAN ONE HOUR, THE ACTION ASSOCIATED WITH TS 3.7.7 WAS SATISFIED. THE ACTUAL CONSEQUENCES OF THIS EVENT WERE MINIMAL IN THAT NO FIRE OR MEPB OCCURRED IN EITHER RHR PUMP ROOM DURING THE 22 MINUTE TIME PERIOD IN WHICH THE DOOR WAS OPEN. THE PROXIMATE CAUSE OF THIS EVENT IS THAT DOOR NO. 75 WAS NOT PROPERLY CLOSED THE LAST TIME THE DOOR WAS USED, HOWEVER, THE ROOT CAUSE OF THIS EVENT CANNOT BE FULLY DETERMINED. NO DIRECT CORRECTIVE ARE PLANNED.

[54] MILLSTONE 3 DOCKET 50-423 LER 92-002
SEISMIC DEFICIENCIES FOUND IN SAFETY RELATED INSTRUMENT CABINETS FOXBORO SPEC 200 EQUIPMENT.
EVENT DATE: 012492 REPORT DATE: 022492 NSSS: WE TYPE: PWR
VENDOR: FOXBORO CO., THE

(NSIC 224127) ON 1/24/92, AT 1300 HOURS, WHILE SHUTDOWN IN MODE 5 (COLD SHUTDOWN), AN ENGINEERING EVALUATION DETERMINED THAT THE POTENTIAL EXISTED FOR SOME FOXBORO SPECIFICATION SPEC 200 INSTRUMENTATION TO BE INOPERABLE DUE TO INADEQUATE DOCUMENTATION TO SUPPORT SEISMIC QUALIFICATION. THE EVALUATION WAS INITIATED AS THE RESULT OF A FOXBORO ADVISORY ISSUED 10/22/91, ON POTENTIAL INSTALLATION ERRORS WHICH COULD COMPROMISE THE SEISMIC QUALIFICATION OF THE SPEC 200 EQUIPMENT. THE ADVISORY IDENTIFIED THAT DUMMY MODULES SHOULD BE INSTALLED IN SPARE SLOTS AND INPUT/OUTPUT MODULES SHOULD HAVE RAIL GUIDES AND BUMPER ASSEMBLIES. INSPECTION OF OUR SAFETY RELATED SPEC 200 EQUIPMENT FOUND DUMMY MODULES, RAIL GUIDES AND BUMPER ASSEMBLIES MISSING. IT WAS DETERMINED THAT THE MISSING BUMPER ASSEMBLIES FROM ACTIVE CARDS COULD HAVE ALLOWED CARD FAILURE DURING A SEISMIC EVENT AND PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION ASSOCIATED WITH THE AUXILIARY FEEDWATER SYSTEM. THE CAUSE OF THIS EVENT IS INSUFFICIENT VENDOR INFORMATION FROM FOXBORO. INSTRUCTIONS ADDRESSING THE NEED FOR BUMPER ASSEMBLIES HAD NOT BEEN RECEIVED PRIOR TO THE OCTOBER 1991 ADVISORY. THE SPEC 200 RACKS HAVE BEEN UPDATED WITH THE INSTALLATION OF DUMMY MODULES, RAIL GUIDES AND BUMPER ASSEMBLIES AND ARE IN FULL COMPLIANCE WITH FOXBORO QUALIFICATION REQUIREMENTS.

[55] NINE MILE POINT 1 DOCKET 50-220 LER 92-001
VIOLATION OF TECHNICAL SPECIFICATIONS DUE TO LACK OF CONFIGURATION MANAGEMENT CAUSED BY PERSONNEL ERROR.
EVENT DATE: 012292 REPORT DATE: 030292 NSSS: GE TYPE: BWR

(NSIC 224134) ON 1/10/92, AT 2309 HOURS, WITH THE MODE SWITCH IN "RUN" AND REACTOR POWER LEVEL AT 72% OF RATED, NINE MILE POINT UNIT ONE (NMP1) WAS IN A CONDITION PROHIBITED BY PLANT TECH SPECS (TS). THE CONDITION EXISTED FOR FIVE HOURS AND FIFTEEN MINUTES WITHOUT PLANT PERSONNEL BEING AWARE OF THE VIOLATION. ON 1/22/92, AT 0700 HOURS, WITH THE MODE SWITCH IN "RUN" AND REACTOR POWER LEVEL AT APPROX. 73% OF RATED, NMP1 WAS IN A CONDITION PROHIBITED BY PLANT TS. IT WAS DETERMINED THAT THE MINIMUM NUMBER OF OPERABLE INSTRUMENT CHANNELS PER OPERABLE TRIP SYSTEM REQUIRED BY PLANT TS WERE NOT MET. THE IMMEDIATE CAUSE FOR BOTH EVENTS WAS AN INSTRUMENTATION ROOT VALVE FOR THE REACTOR PROTECTION SYSTEM FOUND CLOSED. COMBINED WITH A LEAKING INSTRUMENT DRAIN VALVE. THE ROOT CAUSE WAS LACK

OF CONFIGURATION MANAGEMENT, I.E., THE ROOT VALVE WAS NOT IDENTIFIED ON STATION PRINTS OR VALVE LINEUP PROCEDURES. THE IMMEDIATE CORRECTIVE ACTION WAS TO RAISE REACTOR POWER LEVEL TO CLEAR THE CONDITION. ADDITIONAL CORRECTIVE ACTIONS INCLUDED: ISSUE A DEVIATION EVENT REPORT TO EVALUATE THE EVENT, IMPLEMENT A TEMPORARY MODIFICATION TO MAINTAIN THE REQUIRED NUMBER OF OPERABLE INSTRUMENT CHANNELS PER OPERABLE TRIP SYSTEM, AND ISSUE A WORK REQUEST TO INVESTIGATE THE INSTRUMENTATION. LONG TERM CORRECTIVE ACTIONS INCLUDE REVIEWING THE ROOT VALVE VERIFICATION PROGRAM FOR OTHER RPS INPUT INSTRUMENTATION.

[56] NINE MILE POINT 1 DOCKET 50-220 LER 92-002
BREACH OF SECONDARY CONTAINMENT INTEGRITY DUE TO INADEQUATE DESIGN.
EVENT DATE: 020492 REPORT DATE: 022892 NSSS: GE TYPE: BWR

(NSIC 224135) ON FEBRUARY 4, 1992, AT 1230 HOURS, WITH THE MODE SWITCH IN THE "RUN" POSITION AND THE REACTOR POWER LEVEL AT APPROXIMATELY 98 PERCENT OF RATED, SECONDARY CONTAINMENT INTEGRITY WAS MOMENTARILY BREACHED. THIS BREACH WAS THE RESULT OF BOTH REACTOR BUILDING AIRLOCK DOORS, PROVIDING ACCESS FROM THE TURBINE BUILDING TO THE REACTOR BUILDING, BEING OPEN SIMULTANEOUSLY. NINE MILE POINT UNIT 1 (NMP1) TECHNICAL SPECIFICATIONS (T.S.), SECTIONS 1.12 AND 3.4.3, STATE THAT AT LEAST ONE DOOR IN EACH OF THE DOUBLE DOOR ACCESS WAYS SHALL BE CLOSED WHENEVER SECONDARY CONTAINMENT INTEGRITY IS REQUIRED. THE CAUSE OF THIS EVENT IS MAN-MACHINE INTERFACE. THE LIMITED TIME COMBINED WITH SEVEN DISTINCT ACTIONS THE INDIVIDUAL MUST PERFORM TO OPEN THE DOOR, RESULTS IN AN INCREASED PROBABILITY THAT BOTH DOORS COULD BE OPENED AT THE SAME TIME. THE REACTOR BUILDING AIRLOCK DOORS ARE NOT PHYSICALLY INTERLOCKED TO PREVENT SIMULTANEOUS OPENING. INITIAL CORRECTIVE ACTIONS TAKEN WERE TO CLOSE AT LEAST ONE AIRLOCK DOOR TO RESTORE SECONDARY CONTAINMENT. ADDITIONAL CORRECTIVE ACTIONS INCLUDE THE EVALUATION OF THE INSTALLATION OF PHYSICAL INTERLOCKS AND THE SUBMITTAL OF A TECHNICAL SPECIFICATION AMENDMENT TO INCORPORATE A LIMITING CONDITION OF OPERATION (LCO) FOR A SPECIFIC TIME PERIOD TO ALLOW FOR THIS EVENT.

[57] NORTH ANNA 1 DOCKET 50-338 LER 92-005
EMERGENCY DIESEL GENERATOR LOAD SEQUENCING TIMER SETPOINT DRIFT.
EVENT DATE: 011392 REPORT DATE: 021292 NSSS: WE TYPE: PWR
VENDOR: AGASTAT RELAY CO.

(NSIC 224086) ON JANUARY 13, 1992, WITH UNIT 1 IN COLD SHUTDOWN (MODE 5), IT WAS DETERMINED DURING EMERGENCY DIESEL GENERATOR (EDG) LOAD SEQUENCING TIMER TESTING THAT FOUR TIMERS HAD DRIFTED OUTSIDE THEIR SETPOINT TOLERANCE LISTED UNDER TECHNICAL SPECIFICATION (TS) TABLE 4.8-1. SINCE THIS CONDITION WAS PROHIBITED BY THE TS, THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B). THE PROBABLE CAUSE OF THE EVENT IS SETPOINT DRIFT. AS AN IMMEDIATE CORRECTIVE ACTION, EACH TIMER WAS RESET AND SUCCESSFULLY RETESTED. ENGINEERING PERFORMED AN EVALUATION TO REVIEW THE IMPACT THAT THE SETPOINT DRIFT OF EACH AFFECTED TIMER WOULD HAVE ON THE OPERATION OF THE EDG LOAD SEQUENCING SCHEME AND DETERMINED THAT NO SIGNIFICANT SAFETY CONSEQUENCES RESULTED FROM THIS EVENT. EACH FAN STARTED BY THE TIMERS TRIPS ON A CDA SIGNAL AND DOES NOT CAUSE AN EDG CONCERN; THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED AT ANY TIME DUE TO THIS EVENT.

[58] NORTH ANNA 1 DOCKET 50-338 LER 92-003
RESIDUAL HEAT REMOVAL SYSTEM OVERPRESSURE PROTECTION.
EVENT DATE: 012192 REPORT DATE: 021992 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)

(NSIC 224085) ON JANUARY 21 , 1 992, WITH UNIT 1 IN MODE 5 AND UNIT 2 IN MODE 1, AN ENGINEERING EVALUATION PERFORMED IN RESPONSE TO WESTINGHOUSE LETTERS VRA 90-544 AND VRA 90-545 DETERMINED THAT THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM SUCTION RELIEF VALVE DISCHARGE PIPING ARRANGEMENT MAY NOT PASS ITS DESIGN FLOW RATE TO PROTECT THE RHR SYSTEM FROM OVERPRESSURIZATION WHEN IT IS NOT ISOLATED FROM THE REACTOR COOLANT SYSTEM AT OR NEAR 350XF DURING A CHARGING/LETDOWN MISMATCH EVENT THIS EVENT IA REPORTABLE PURAUANT TO 10CFR50.73 (A)(2)(V)( ) AS A CONDITION THAT ALONE COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION

OF A SYSTEM THAT IA NEEDED TO REMOVE RESIDUAL HEAT . A FOUR HOUR REPORT WAS MADE PURAUANT TO 10CFRSO.72(B)(2)(III)(B). THE CAUSE OF THE EVENT WAS A POTENTIAL DESIGN DEFICIENCY OF THE RHR SYSTEM SUCTION RELIEF VALVE DISCHARGE PIPING ARRANGEMENT. NO SIGNIFICANT SAFETY CONSEQUENCES WOULD RESULT FROM THIS EVENT BEYOND THOSE ANALYZED IN THE UFSAR. THE UFSAR EVALUATED A BREAK CAUSED BY AN OVERPREAAURIZATION EVENT IN THE LARGEST RHR LINE THAT COULD ADVERSELY IMPACT BOTH RHR TRAINS SIMULTANEOUSLY. RESULTS OF THE ANALYSIS CONFIRM THAT THE MAKEUP REQUIRED TO PRECLUDE AN UNSAFE CONDITION CAN BE PROVIDED. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED AT ANY TIME DUE TO THIS EVENT .

[59] OCONEE 1 DOCKET 50-269 LER 92-002
EQUIPMENT FAILURE IN EMERGENCY POWER SYSTEM AND INAPPROPRIATE ACTION RESULT IN TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 012992 REPORT DATE: 030592 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 224143) KEOWEE HYDRO STATION SUPPLIES EMERGENCY POWER TO THE OCONEE NUCLEAR STATION. ON 1/29/92, OCONEE UNIT 1 WAS AT 100% FULL POWER, UNIT 2 WAS AT COLD SHUTDOWN AND UNIT 3 WAS AT 99% FULL POWER. AT 2104 HOURS, KEOWEE HYDRO UNIT 1 FAILED TO START DURING A ROUTINE ATTEMPT TO SUPPLY POWER TO THE GRID. THE OPERATOR SHUT KEOWEE HYDRO UNIT 1 DOWN AND STARTED KEOWEE HYDRO UNIT 2 TO SUPPLY POWER TO THE GRID. THERE WAS A KNOWN PROBLEM WITH THE BREAKER ANTI-PUMP "X" RELAYS ON THE FIELD AND FIELD FLASHING BREAKERS SO THE OPERATOR INSPECTED THE RELAYS ASSOCIATED WITH REOWEE HYDRO UNIT 1. NONE OF THE RELAYS WERE FOUND TO BE OUT OF THE EXPECTED POSITION. KEOWEE HYDRO UNIT 1 WAS DETERMINED TO BE INOPERABLE FROM JANUARY 28, 1992, AT 2149 HOURS, WHEN THE RELAYS FAILED TO RESET AFTER THE LAST UNIT SHUTDOWN UNTIL JANUARY 29, 1992, AT 2116 HOURS WHEN THE UNIT WAS SUCCESSFULLY STARTED. THE ROOT CAUSE OF THIS EVENT IS EQUIPMENT FAILURE. A SECOND ROOT CAUSE IS INAPPROPRIATE ACTION, NO ACTION TAKEN WHEN REQUIRED BECAUSE THE NEED WAS NOT RECOGNIZED. KEOWEE HYDRO UNIT 2 WAS NOT OPERABILITY TESTED WITHIN 1 HOUR AS REQUIRED BY TECHNICAL SPECIFICATIONS. CORRECTIVE ACTION WILL INCLUDE DIAGNOSING THE SPECIFIC FAILURE MODE OF THE RELAY AND SUBMITTING A SUPPLEMENT TO THIS REPORT WHICH WILL OUTLINE THE CORRECTIVE ACTIONS TO PREVENT A RECURRENCE.

[60] OCONEE 3 DOCKET 50-287 LER 91-009 REV 01
UPDATE ON TECHNICAL SPECIFICATION REQUIRED CONTAINMENT INTEGRITY VALVE FOUND MISPOSITIONED DURING FORCED OUTAGE DUE TO UNKNOWN CAUSE, POSSIBLE INAPPROPRIATE ACTION.
EVENT DATE: 120191 REPORT DATE: 011?92 NSSS: BW TYPE: PWR

(NSIC 224074) THE UNIT 3 REACTOR BUILDING CONTAINMENT IS SUPPLIED WITH INSTRUMENT AIR (IA) THROUGH A THREE-INCH LINE WITH NORMALLY CLOSED ISOLATION VALVES (3IA90 AND 3IA-91) ON EITHER SIDE OF CONTAINMENT. ON DECEMBER 1, 1991 AT APPROXIMATELY 2130 HOURS AND WITH UNIT 3 AT COLD SHUTDOWN CONDITIONS, A NON-LICENSED OPERATOR WHO HAD BEEN SENT TO OPEN 3IA-91 (INSIDE THE REACTOR BUILDING) DISCOVERED THE VALVE IN THE OPEN POSITION. INVESTIGATION COULD NOT DETERMINE WHEN THE VALVE WAS LAST OPENED. THE OTHER ISOLATION VALVE, OUTSIDE THE RB, WAS FOUND CLOSED. IT WAS CONSERVATIVELY ASSUMED THAT 3IA91 HAD BEEN OPEN SINCE MARCH 22, 1991 AT 1550 HOURS. UNIT 3 OPERATED WITH THE REACTOR COOLANT SYSTEM (RCS) ABOVE 200 DEGREES F AND 300 PSIG, THE CONDITIONS REQUIRED BY TECHNICAL SPECIFICATIONS FOR CONTAINMENT INTEGRITY, FROM MARCH 22, 1991 AT 1550 HOURS TO MARCH 24, 1991 AT 1130 HOURS AND FROM MARCH 27, 1991 AT 1900 HOURS TO NOVEMBER 23, 19[illegible] AT 1720 HOURS. THE ROOT CAUSE OF THIS EVENT IS CONSIDERED UNKNOWN, POSSIBLE INAPPROPRIATE ACTION. CORRECTIVE ACTIONS INCLUDED CHANGES TO THE METHOD OF DOCUMENTATION OF THIS AND OTHER ROUTINELY OPERATED CONTAINMENT INTEGRITY VALVES.

[61] PALISADES DOCKET 50-255 LER 91-014 REV 01
UPDATE ON SAFETY RELATED CIRCUITS ROUTED WITH OPPOSITE CHANNEL CIRCUITS.
EVENT DATE: 070991 REPORT DATE: 021492 NSSS: CE TYPE: PWR

(NSIC 224059) THROUGH THE ON-GOING PALISADES CONFIGURATION CONTROL PROJECT A NUMBER OF APPARENT DISCREPANCIES IN CIRCUIT ROUTING HAVE BEEN IDENTIFIED. PREVIOUS NRC CORRESPONDENCE DATED SEPTEMBER 4, 1990 AND MARCH 12, 1991 PROVIDED DETAILS OF THE "CIRCUIT AND RACEWAY SCHEDULE ENHANCEMENT PROJECT" AND STATUS OF THE PROJECTS ACTIVITIES. THE ELECTRICAL CIRCUIT CHANNELIZATION AND SEPARATION DEFICIENCIES WHICH HAVE BEEN EVALUATED AT THIS TIME, WHILE NOT CONFORMING TO FSAR CRITERIA IN ALL RESPECTS, HAVE RESULTED IN IDENTIFICATION OF VERY FEW CONDITIONS INVOLVING A LOSS OF ELECTRICAL OR PROTECTION SYSTEM INDEPENDENCE, OR NONCOMPLIANCE WITH 10 CFR 50, APPENDIX R REQUIREMENTS. IN THOSE FEW CASES, TWO TO DATE, COMPENSATORY MEASURES AND CORRECTIVE ACTIONS HAVE BEEN TAKEN AS STIPULATED IN PLANT TECHNICAL SPECIFICATIONS AND/OR THE CONDITION WAS PROMPTLY CORRECTED. AT THE TIME OF THE EVENT THE PLANT WAS OPERATING AT ABOUT 7[illegible]% POWER AND ESCALATING TO FULL POWER FOLLOWING A SHORT OUTAGE. THIS EVENT IS REPORTABLE AS A CONDITION OUTSIDE THE DESIGN BASIS OF THE PLANT.

[62] PALISADES DOCKET 50-255 LER 92-001
LOSS OF AIR SIDE SEAL OIL IN THE ELECTRICAL GENERATOR RESULTS IN A TURBINE/REACTOR TRIP.
EVENT DATE: 120991 REPORT DATE: 010892 NSSS: CE TYPE: PWR
VENDOR: LIMITORQUE CORP.

(NSIC 223701) ON 12/9/91, AT 1630 HOURS, WITH THE PLANT OPERATING AT 100% POWER, THE TURBINE BUILDING AUXILIARY OPERATOR OBSERVED THAT SEAL OIL PRESSURE ON THE AIR SIDE OF THE ELECTRICAL GENERATOR HAD DECREASED FROM THE PREVIOUS SHIFT READING. AUTOMATIC AND MANUAL ACTION WAS TAKEN TO RESTORE THE SEAL OIL PRESSURE HOWEVER, PRESSURE CONTINUED TO DROP. IT WAS ALSO OBSERVED THAT HYDROGEN PRESSURE ON THE MAIN ELECTRICAL GENERATOR WAS DROPPING. AT 1715 HOURS, THE SHIFT SUPERVISOR ORDERED AN EMERGENCY POWER REDUCTION DUE TO A CONTINUING LOSS OF ELECTRICAL GENERATOR HYDROGEN PRESSURE. DURING THE EMERGENCY POWER REDUCTION, THE LEVEL IN THE "B" STEAM GENERATOR WENT TO ITS HIGH LEVEL SETPOINT WHICH CAUSED THE AUTOMATIC CLOSURE OF THE FEEDWATER REGULATING VALVE. STEAM GENERATOR LEVEL CONTINUED TO INCREASE. AT 1722 HOURS, CONTROL ROOM OPERATORS INITIATED BOTH A MANUAL REACTOR TRIP AND A MANUAL TURBINE TRIP DUE TO INCREASED STEAM GENERATOR LEVEL. AT 1722 HOURS, WITH THE PLANT OPERATING AT APPROX. 20% POWER, AN AUTOMATIC REACTOR TRIP ON LOSS OF LOAD OCCURRED WHEN THE TURBINE TRIP WAS INITIATED SLIGHTLY BEFORE THE REACTOR TRIP. THE CAUSE OF THE EVENT WAS A MALFUNCTION IN THE ELECTRICAL GENERATOR SEAL OIL SYSTEM DUE TO A PLUGGED FILTER. NO ROOT CAUSE FOR THE PLUGGING OF THE FILTER WAS DETERMINED.

[63] PALISADES DOCKET 50-255 LER 92-002 REV 01
UPDATE ON EQUIPMENT NOT CHANNEL CHECKED IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS TABLES 3.17.1 AND 4.1.1 AND SECTION 4.0.4.
EVENT DATE: 121491 REPORT DATE: 021492 NSSS: CE TYPE: PWR

(NSIC 224060) ON 12/16/91, WITH THE PLANT AT HOT SHUTDOWN, IT WAS DISCOVERED THAT SELECTED REACTOR PROTECTIVE SYSTEM (RPS) INSTRUMENT SURVEILLANCE REQUIREMENTS WERE NOT BEING PERFORMED AS REQUIRED BY THE TECH SPECS (TS). SPECIFICALLY, THE CHANNEL CHECKS REQUIRED FOR REACTOR COOLANT FLOW (AA,FI) AND WIDE RANGE NUCLEAR POWER INSTRUMENTATION (IG) WERE BEING PERFORMED AS THOUGH THE APPLICABILITY WAS "AT HOT STANDBY AND ABOVE" RATHER THAN THE SPECIFIED APPLICABILITY OF "IF ANY CLUTCH POWER SUPPLY IS ENERGIZED." NO PARTICULAR SAFETY SIGNIFICANCE IS ATTRIBUTED TO THIS EVENT BECAUSE THESE INSTRUMENTS WERE VERIFIED TO BE OPERABLE BY SHIFTLY (AT LEAST ONCE EACH 12 HOURS) CHANNEL CHECKS WHENEVER THE PLANT WAS AT HOT STANDBY OR ABOVE WITH NO DISCREPANCIES FOUND. ALL OTHER RPS INSTRUMENTATION SURVEILLANCE WAS BEING PERFORMED AS REQUIRED. THE ROOT CAUSE OF THIS EVENT WAS A PROCEDURAL WEAKNESS WHICH RESULTED FROM THE INADEQUATE IMPLEMENTATION OF TS REQUIREMENTS. CORRECTIVE ACTION FOR THIS EVENT INCLUDES (1) REVIEW AND REVISION OF TS SURVEILLANCE PROCEDURES AND SELECTED OPERATING PROCEDURES TO ADEQUATELY IMPLEMENT TS REQUIREMENTS, (2) A REVIEW OF THE TS CHANGE AND IMPLEMENTATION PROCESS AND (3) THE DEVELOPMENT AND PERFORMANCE OF TRAINING ON REVISED TS SURVEILLANCE PROCEDURES AND THEIR BASIS DOCUMENTS.

[64] PALISADES DOCKET 50-255 LER 92-004 REV 01
UPDATE ON LOSS OF CONTAINMENT INTEGRITY DUE TO THE FAILURE OF THE EMERGENCY ESCAPE AIRLOCK EQUALIZING VALVE.
EVENT DATE: 010792 REPORT DATE: 030692 NSSS: CE TYPE: PWR
VENDOR: WOOLLEY, W. J. COMPANY

(NSIC 224149) ON 1/7/92, AT 0435 HOURS, WITH THE PLANT OPERATING AT 100% POWER, IT WAS DETERMINED THAT CONTAINMENT INTEGRITY, AS DEFINED IN TECHNICAL SPECIFICATIONS (TS) 1.4 AND 3.6.1A, HAD BEEN VIOLATED. AT THE TIME OF DISCOVERY THE PLANT STAFF WAS PERFORMING TS SURVEILLANCE PROCEDURE (TSSP) SO-4B, "ESCAPE AIR LOCK PENETRATION LEAK TEST." THE ESCAPE AIR LOCK WAS LAST SATISFACTORILY TESTED ON JULY 10, 1991. THIS EVENT IS NOT CONSIDERED A SAFETY SIGNIFICANT ISSUE SINCE A SIGNIFICANT RADIOLOGICAL SOURCE TERM DID NOT EXIST WHEN CONTAINMENT INTEGRITY WAS VIOLATED ON 1/6/92. THEREFORE, THE IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC WAS INSIGNIFICANT. THIS IS FURTHER SUPPORTED BY THE FACT THAT THE OUTER DOOR OF THE ESCAPE AIR LOCK WAS OPERABLE AND ONLY ONE DOOR OF AN AIR LOCK IS REQUIRED FOR CONTAINMENT INTEGRITY. THE PROXIMATE CAUSE OF THIS EVENT WAS THAT THE ESCAPE AIR LOCK INNER DOOR EQUALIZING VALVE WAS STUCK IN THE OPEN POSITION, THEREBY DEFEATING THE ISOLATION CAPABILITY OF THE INNER DOOR. NO ROOT CAUSE FOR THE VALVE STICKING OPEN HAS BEEN DETERMINED AT THIS TIME. CORRECTIVE ACTION FOR THIS EVENT INCLUDES REVISING THE MAINTENANCE PROCEDURE FOR THE AIRLOCKS TO REQUIRE THAT THE EQUALIZING VALVES BE VISUALLY CHECKED FOLLOWING THE COMPLETION OF THE SEAL CONTACT CHECK, PERFORMING MAINTENANCE ON THE ESCAPE AIR LOCK INNER DOOR EQUALIZING VALVE DURING THE 1992 REFUELING OUTAGE.

[65] PALISADES DOCKET 50-255 LER 92-005
CLASS 1E PRESSURIZER LEVEL INDICATOR CABLE CONNECTED TO THE NON-CLASS 1E CRITICAL FUNCTIONS MONITORING SYSTEM COMPUTER WITHOUT ADEQUATE ELECTRICAL ISOLATION.
EVENT DATE: 011792 REPORT DATE: 021492 NSSS: CE TYPE: PWR

(NSIC 224061) ON JANUARY 15, 1992, AT APPROXIMATELY 1500 HOURS, WITH THE PLANT OPERATING AT 100% POWER, THE NRC NOTIFIED THE PALISADES PLANT STAFF OF AN APPARENT MISROUTING OF THE CLASS 1E PRESSURIZER LEVEL INSTRUMENT LOOP (LT-0103). THE PLANT STAFF SUBSEQUENTLY VERIFIED THAT THE INSTRUMENT LOOP IS CONNECTED TO THE NON-CLASS 1E CRITICAL FUNCTIONS MONITORING SYSTEM (CFMS) COMPUTER WITHOUT ADEQUATE ELECTRICAL ISOLATION. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO INADEQUATE DESIGN CONTROL DURING THE TIME THE EQUIPMENT MODIFICATION WAS ENGINEERED AND INSTALLED. THE IMMEDIATE CORRECTIVE ACTION FOR THIS EVENT WAS TO PERFORM A TEMPORARY MODIFICATION TO DISCONNECT THE LT-0103 LOOP FROM THE NON-CLASS 1E CFMS COMPUTER, ISOLATING ANY CABLE FAULTS FROM THE INSTRUMENT LOOP. ADDITIONAL CORRECTIVE ACTION INCLUDES A 1992 REFUELING OUTAGE MODIFICATION TO RE-ROUTE THE CABLE FROM THE LT-0103 INSTRUMENT LOOP TO THE CORRECT CLASS 1E CFMS COMPUTER INPUT TERMINATION CABINET AND A REVIEW OF ALL THE RG 1.97 INSTRUMENTATION LOOPS TO VERIFY ADEQUATE ELECTRICAL ISOLATION.

[66] PALISADES DOCKET 50-255 LER 92-006
LACK OF ENVIRONMENTAL QUALIFICATION OF RESIDUAL HEAT REMOVAL HEAT EXCHANGER TEMPERATURE ELEMENT.
EVENT DATE: 012092 REPORT DATE: 021992 NSSS: CE TYPE: PWR

(NSIC 224062) ON JANUARY 20, 1992, AT APPROXIMATELY 1000 HOURS, WITH THE PLANT OPERATING AT 100% POWER, IT WAS DETERMINED THAT TEMPERATURE ELEMENT TE-0351B WAS NOT ENVIRONMENTALLY QUALIFIED IN ACCORDANCE WITH REGULATORY GUIDE 1.97. TE-0351B IS LOCATED ON THE LOW PRESSURE SAFETY INJECTION (LPSI) HEADER, DOWNSTREAM OF THE RESIDUAL HEAT REMOVAL (RHR) HEAT EXCHANGER OUTLET VALVE. THIS EVENT WAS CAUSED BY A COMBINATION OF PERSONNEL ERROR AND INACCURATE DOCUMENTATION. IN 1986, THE TEMPERATURE ELEMENT WAS DETERMINED TO BE IN A LOCATION THAT WAS DESIGNATED AS A NON-HARSH ENVIRONMENT. THIS INFORMATION WAS NOT VERIFIED AND, THEREFORE, THE RESULT WAS THE TEMPERATURE ELEMENT WAS NOT REQUIRED TO BE ENVIRONMENTALLY QUALIFIED. IN ADDITION, A RECENTLY DISCOVERED TYPOGRAPHICAL ERROR IN THE ENVIRONMENTAL EQUIPMENT QUALIFICATION LIST LED SUBSEQUENT REVIEWERS TO CONCUR WITH THE 1986 DETERMINATION. THIS EVENT DID NOT INVOLVE THE FAILURE OF EQUIPMENT IMPORTANT TO SAFETY. CORRECTIVE ACTION FOR THIS EVENT INCLUDES COMPLETION OF AN OPERABILITY DETERMINATION FOR TE-0351B, THE PREPARATION OF A MODIFICATION PACKAGE

TO REPLACE TE-0351B INSTRUMENT LOOP WITH ENVIRONMENTALLY QUALIFIED EQUIPMENT, AND CONTINUED REVIEW OF THE EQ LIST FOR COMPLETENESS.

[67] PALISADES DOCKET 50-255 LER 92-007
MAIN STEAM ISOLATION VALVES INOPERABLE DUE TO AN UNQUALIFIED ELECTRICAL CIRCUIT.
EVENT DATE: 020592 REPORT DATE: 030692 NSSS: CE TYPE: PWR

(NSIC 224141) ON 2/5/92, THE PLANT WAS OPERATING AT 100% POWER. AS A RESULT OF AN ON GOING EQUIPMENT CLASSIFICATION (Q-LIST) REVIEW PROGRAM IT WAS DETERMINED THAT THE MAIN STEAM ISOLATION VALVE (MSIV) (SB;ISV) ACTUATOR SOLENOID VALVES (SB;PSV), COULD BE RENDERED INOPERABLE BY A MAIN STEAM LINE BREAK OUTSIDE OF CONTAINMENT. AT APPROX. 1600 HOURS THE MSIVS WERE DECLARED INOPERABLE, AN UNUSUAL EVENT WAS DECLARED, AND TECHNICAL SPECIFICATION 3.5.3 WAS ENTERED REQUIRING THE REACTOR TO BE IN HOT STANDBY IN 6 HOURS. AS A RESULT OF ENGINEERING AND MANAGEMENT REVIEWS, IT WAS CONSIDERED FEASIBLE THAT A MODIFICATION COULD BE MADE TO RESOLVE THE DEFICIENCY. AN ORAL REQUEST FOR A 72 HOUR TEMPORARY WAIVER OF COMPLIANCE WAS MADE TO THE NRC AND WAS GRANTED. THE UNUSUAL EVENT WAS THEN TERMINATED. A WRITTEN FOLLOW-UP TO THE ORAL REQUEST WAS MADE ON 2/6/92. AT APPROX. 1842 HOURS, ON 2/6/92, PLANT MANAGEMENT DECIDED THAT MODIFICATIONS TO CORRECT THE DEFICIENCY COULD NOT BE SAFELY ACCOMPLISHED IN THE 72 HOUR TIME INTERVAL. THUS, THE DECISION WAS THEN MADE TO EXIT THE TEMPORARY WAIVER OF COMPLIANCE AND CONTINUE WITH PLANT SHUTDOWN IN ACCORDANCE WITH THE TS REQUIREMENTS. AN UNUSUAL EVENT WAS RE-DECLARED AND EXITED WHEN THE MSIVS WERE CLOSED WHEN THE PLANT WAS PUT IN HOT STANDBY. THE MAIN STEAM LINE ISOLATION VALVE ACTUATION SOLENOID VALVES WILL BE RELOCATED TO A NON-HARSH ENVIRONMENT.

[68] PALISADES DOCKET 50-255 LER 92-008
BOTH CONTROL ROOM HVAC TRAINS INOPERABLE DUE TO EQUIPMENT FAILURE.
EVENT DATE: 020692 REPORT DATE: 030692 NSSS: CE TYPE: PWR

(NSIC 224160) ON FEBRUARY 6, 1992, AT 2144 HRS, THE PLANT WAS AT APPROXIMATELY 32% POWER AND IN THE PROCESS OF SHUTTING DOWN. BOTH TRAINS OF CONTROL ROOM HEATING, VENTILATION, AND AIR CONDITIONING (HVAC)(VI) BECAME INOPERABLE WHEN THE OPERATING AIR CONDITIONING CONDENSING UNIT VC-11 (VI;CDU), DEVELOPED A LEAK IN THE HOT GAS BYPASS LINE WHILE THE REDUNDANT TRAIN CONDENSING UNIT, VC-10, WAS OUT OF SERVICE FOR NORMAL PREVENTATIVE MAINTENANCE. AN UNUSUAL EVENT WAS DECLARED AND APPROPRIATE NOTIFICATIONS MADE. MAINTENANCE WAS INITIATED TO REPAIR THE LEAK AND VC-11 WAS RETURNED TO SERVICE. THE UNUSUAL EVENT WAS TERMINATED AT 0325 HOURS ON FEBRUARY 7, 1992. THE CONTROL ROOM TEMPERATURE WAS OBSERVED TO REACH 81 F AT APPROXIMATELY FOUR HOURS INTO THE EVENT. WITHIN A HALF-HOUR OF RETURNING VC-11 TO SERVICE THE CONTROL ROOM TEMPERATURE RETURNED TO 74 F. THE EVENT WAS CAUSED BY A LACK OF CLEARANCE BETWEEN THE VC-11 CONDENSING UNIT'S HOT GAS BYPASS LINE AND IT'S PENETRATION THROUGH THE FLOORING. MAINTENANCE WILL INSPECT ALL THE ASSOCIATED PIPING ON EACH CONDENSING UNIT AND REVISIONS WILL BE MADE TO THE PREVENTATIVE MAINTENANCE PROCEDURES TO CHECK THE TUBING AND CONDENSING UNIT HOLD DOWN BOLTING TO PRECLUDE FUTURE OCCURRENCES OF THIS KIND.

[69] PALISADES DOCKET 50-255 LER 92-009
INADVERTENT ACTUATION OF THE CONTROL ROOM HVAC SYSTEM DUE TO DAMAGED ELECTRICAL CABLE.
EVENT DATE: 021392 REPORT DATE: 030692 NSSS: CE TYPE: PWR

(NSIC 224142) ON FEBRUARY 13, 1992, AT 1920 HOURS, THE CONTROL ROOM VENTILATION SYSTEM INADVERTENTLY SWITCHED TO THE EMERGENCY MODE. AT THE TIME OF OCCURRENCE, THE PLANT WAS IN COLD SHUTDOWN AND THE PRIMARY COOLANT SYSTEM WAS DEPRESSURIZED. THE CONTROL ROOM OPERATORS IMMEDIATELY VERIFIED THAT THERE WAS NOT A VALID CONTAINMENT HIGH PRESSURE (CHP) OR A CONTAINMENT HIGH RADIATION (CHR) SIGNAL PRESENT. IT WAS SUBSEQUENTLY DETERMINED THAT ELECTRICAL MAINTENANCE WAS REPLACING DAMAGED FLEX CONDUIT ON THE CHR RELAY 5R-6 CIRCUIT. WHEN THE WIRE ON RELAY 5R-6, POINT 16, WAS DISCONNECTED, THE ELECTRICAL CIRCUIT SCHEME S101 DE-ENERGIZED CAUSING THE "A" TRAIN OF CONTROL ROOM HVAC TO AUTOMATICALLY SWITCH TO THE EMERGENCY MODE. THIS EVENT WAS CAUSED BY A COMBINATION OF INADEQUATE JOB PLANNING AND PERSONNEL ERROR. CORRECTIVE ACTION FOR THIS EVENT INCLUDES (1) IN-PLANT

TRAINING FOR ELECTRICIANS, I&C STAFF AND I&C TECHNICIANS ON THE IMPORTANCE OF JOB PLAN AND SCHEMATICS REVIEWS PRIOR TO INITIATING WORK AND (2) A REVIEW OF JOB PLANNING REQUIREMENTS AND THE IMPORTANCE OF POST-MAINTENANCE TESTING WITH THE JOB PLANNERS.

[70] PALO VERDE 1 DOCKET 50-528 LER 92-002
MISSED TECHNICAL SPECIFICATION ACTION WHILE RADIATION MONITOR WAS INOPERABLE.
EVENT DATE: 011792 REPORT DATE: 022492 NSSS: CE TYPE: PWR

(NSIC 224128) ON 1/17/92, PALO VERDE UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT APPROX. 29% POWER. AT APPROX. 1000 MST ON 1/24/92, WHILE PERFORMING A ROUTINE REVIEW OF COMPLETED WORK DOCUMENTS, A UNIT 1 CHEMISTRY SUPERVISOR DISCOVERED THAT ON 1/17/92, A CONTAINMENT AIR GRAB SAMPLE HAD NOT BEEN TAKEN WITHIN THE TECH SPEC ALLOWED TIME LIMIT WHILE THE CONTAINMENT BUILDING ATMOSPHERE RADIATION MONITOR (RU-1) WAS INOPERABLE. RU-1 WAS TAKEN OUT OF SERVICE FOR CORRECTIVE MAINTENANCE AND DECLARED INOPERABLE AT APPROX. 0825 MST ON 1/17/92. WITH RU-1 INOPERABLE, TS LIMITING CONDITION FOR OPERATION (LCO) 3.4.5.1, "REACTOR COOLANT SYSTEM LEAKAGE," ACTION A, REQUIRES THAT GRAB SAMPLES OF THE CONTAINMENT ATOMPSHERE BE OBTAINED AT LEAST ONCE PER 12 HOURS. THE VALID CONTAINMENT AIR GRAB SAMPLES WERE TAKEN APPROX. 12 HOURS AND 51 MINUTES APART. THE CAUSE OF THIS EVENT WAS A PERSONNEL ERROR BY THE DAY SHIFT CONTROL ROOM ASSISTANT SHIFT SUPERVISOR. AS IMMEDIATE CORRECTIVE ACTION, THE CONTAINMENT ISOLATION VALVES WERE OPENED AND A CONTAINMENT AIR GRAB SAMPLE WAS TAKEN AND ANALYZED. THE CONTAINMENT AIR GRAB SAMPLES TAKEN AT THE BEGINNING AND END OF THIS INTERVAL DEMONSTRATED THAT CONTAINMENT ATMOSPHERE RADIATION LEVELS WERE NORMAL DURING THIS EVENT. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED PURSUANT TO 10CFR50.73.

[71] PALO VERDE 1 DOCKET 50-528 LER 92-003
MISSED TECHNICAL SPECIFICATION ACTION DUE TO PERSONNEL ERROR.
EVENT DATE: 020492 REPORT DATE: 030592 NSSS: CE TYPE: PWR

(NSIC 224152) AT APPROX. 1754 ON 2/4/92, PALO VERDE UNIT 1 WAS IN MODE 1 AT APPROX. 100% POWER WHEN THE UNIT 1 CONTROL ROOM ASSISTANT SHIFT SUPERVISOR DISCOVERED THAT THE CHANNEL "B" LOW STEAM GENERATOR PRESSURE TRIP SETPOINT FOR THE NUMBER 2 STEAM GENERATOR HAD BEEN AT APPROX. 872 POUNDS PER SQUARE INCH ABSOLUTE (PSIA) FOR APPROX. ONE (1) HOUR AND 27 MINUTES. TECH SPEC (TS) 2.2.1, TABLE 2.2-1 AND TS 3.3.2, TABLE 3.3-4 REQUIRE THAT THE LOW STEAM GENERATOR PRESSURE TRIP SETPOINT BE GREATER THAN OR EQUAL TO 919 PSIA WITH A MINIMUM ALLOWABLE VALUE OF 912 PSIA. THE TS 2.2.1 ACTION REQUIRES THAT THE CHANNEL BE DECLARED INOPERABLE AND THAT THE TS 3.3.1 ACTION BE APPLIED UNTIL THE CHANNEL IS RESTORED TO OPERABLE STATUS. TS 3.3.1 ACTION 2 AND TS 3.3.2 ACTION 13 STATE THAT WITH ONE (1) CHANNEL INOPERALLE, POWER OPERATION MAY CONTINUE PROVIDED THE INOPERABLE CHANNEL IS PLACED IN THE BYPASSED OR TRIPPED CONDITION WITHIN ONE (1) HOUR. THEREFORE, TS 3.3.1 ACTION 2 AND TS 3-3.2 ACTION 13 WERE NOT MET WHILE THE CHANNEL "B" LOW STEAM GENERATOR PRESSURE TRIP FOR THE NUMBER 2 STEAM GENERATOR WAS INOPERABLE. THE CAUSE OF THIS EVENT WAS A COGNITIVE PERSONNEL ERROR BY AN INSTRUMENTATION AND CONTROLS TECHNICIAN WHO DID NOT RESET THE TRIP SETPOINT IN ACCORDANCE WITH AN APPROVED PROCEDURE FOLLOWING SURVEILLANCE TESTING. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED PERSUANT TO 10CFR50.73.

[72] PALO VERDE 3 DOCKET 50-530 LER 92-001
REACTOR TRIP FOLLOWING REACTOR POWER CUTBACK DUE TO LOSS OF MAIN FEEDWATER PUMP.
EVENT DATE: 012492 REPORT DATE: 021892 NSSS: CE TYPE: PWR

(NSIC 224107) ON JANUARY 24, 1992, AT APPROXIMATELY 1750 MST, PALO VERDE UNIT 3 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 100 PERCENT POWER WHEN A REACTOR TRIP OCCURRED FOLLOWING A REACTOR POWER CUTBACK WHICH WAS INITIATED WHEN CONTROL ROOM PERSONNEL MANUALLY TRIPPED THE MAIN FEEDWATER PUMP "B" (MFWP). THE MFWP "B" WAS TRIPPED IN AN ATTEMPT TO STABILIZE THE SUCTION PRESSURE FOR THE MFWP "A" FOLLOWING THE RECEIPT OF MFWP LOW SUCTION PRESSURE TRIP ALARMS FOR BOTH OPERATING MFWPS. FOLLOWING THE REACTOR TRIP, THE PLANT WAS STABILIZED IN MODE 3 (HOT STANDBY) AT NORMAL OPERATING TEMPERATURE AND PRESSURE. AT APPROXIMATELY 1816 MST, THE EVENT WAS CLASSIFIED AS AN UNCOMPLICATED REACTOR TRIP. ALL PLANT EQUIPMENT

RESPONDED AS EXPECTED. NO ENGINEERED SAFETY FEATURE ACTUATION SYSTEM ACTUATIONS OCCURRED AND NONE WERE REQUIRED. THE MFWP LOW SUCTION PRESSURE OCCURRED WHEN THE MFUP MINI-FLOW RECIRCULATION VALVES FAILED OPEN ON LOSS OF INSTRUMENT AIR PRESSURE. AS CORRECTIVE ACTION, THE INSTRUMENT AIR LINE WAS REPAIRED. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED PURSUANT TO 10CFR50.73.

[73] PEACH BOTTOM 3 DOCKET 50-278 LER 92-002
DRYWELL OXYGEN CONCENTRATION LEVEL EXCEEDED THE VALUE SPECIFIED IN THE TECHNICAL SPECIFICATION DUE TO AN ANALYZER FAILURE.
EVENT DATE: 011792 REPORT DATE: 021892 NSSS: GE TYPE: BWR

(NSIC 224068) ON 01/17/92, IT WAS DISCOVERED THAT THE DRYWELL (DW) OXYGEN (02) CONCENTRATION LEVEL EXCEEDED THE 4% LIMIT SPECIFIED IN THE TECHNICAL SPECIFICATIONS (TECH SPEC) THIS RESULTED IN A TECH SPEC VIOLATION AND A CONDITION OUTSIDE DESIGN BASIS. DW 02 CONCENTRATION SAMPLES WERE OBTAINED USING A PORTABLE MONITOR AND THIS SAMPLE INDICATED THAT THE ACTUAL CONCENTRATION LEVEL WAS ABOUT 8.0%. THE CAUSE OF THE EVENT WAS THAT THE INSTRUMENT AIR BYPASS MANUAL VALVE WAS FOUND OPEN WHICH ALLOWED INSTRUMENT AIR TO LEAK INTO THE DW. A CONTRIBUTING FACTOR TO THIS EVENT HAS BEEN DETERMINED TO BE THAT THE ANALYZER WAS GIVING AN INDICATED LOW READING. THE FAILURE TO IDENTIFY THAT THE DW 02 ANALYZER WAS NOT FUNCTIONING PROPERLY HAS BEEN ATTRIBUTED TO THE FACT THAT THE ACCEPTANCE RANGE ON THE SURVEILLANCE TEST (ST) WAS TOO LOW. AFTER DISCOVERY OF THE EVENT, THE DW 02 CONCENTRATION LEVELS WERE REDUCED AND THE 02 ANALYZER ON THE OTHER UNIT WAS VERIFIED TO BE OPERATIONAL. ADDITIONALLY, ROUND SHEETS WILL BE REVISED TO INCLUDE INSTRUMENT NITROGEN COMPRESS OR RUN TIMES. INDEPENDENT CONTAINMENT SAMPLES ARE CURRENTLY BEING ANALYZED ON A PERIODIC BASIS AS A COMPENSATORY ACTION. THE ST USED TO RECORD THE 02 CONCENTRATION LEVELS WILL BE REVISED.

[74] PILGRIM 1 DOCKET 50-293 LER 91-008 REV 01
UPDATE ON THREE AUTOMATIC GROUP 1 ISOLATIONS DUE TO FALSE HIGH REACTOR WATER LEVEL SIGNALS WHILE SHUTDOWN.
EVENT DATE: 043091 REPORT DATE: 022492 NSSS: GE TYPE: BWR

(NSIC 224117) ON APRIL 30, 1991, THREE AUTOMATIC PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM (PCIS) GROUP 1 ISOLATIONS OCCURRED WHILE SHUTDOWN AT 0116 HOURS, 0933 HOURS, AND 1037 HOURS, RESPECTIVELY, DUE TO A FALSE HIGH REACTOR VESSEL (RV) WATER LEVEL SIGNAL. THE ACTUATIONS RESULTED IN THE AUTOMATIC CLOSING OF THE RELATED PRIMARY CONTAINMENT SYSTEM ISOLATION VALVES. THE GROUP 1 ISOLATIONS WERE INITIATED BY THE REACTOR WATER LEVEL TRIP UNITS DOWN STREAM OF REFERENCE LEG CONDENSING CHAMBER 12B. THE FALSE HIGH RV WATER LEVEL SIGNALS WERE PRIMARILY DUE TO UNDERSIZED RV WATER LEVEL HEAD EQUALIZING LINES. CONTRIBUTING CAUSES INCLUDED SENSING LINE HANGER INTERFERENCE, AIR ENTRAPMENT AND MARGINAL SENSING LINE SLOPE. CORRECTIVE ACTIONS TAKEN INCLUDE: INCREASING THE HEAD EQUALIZING LINE SIZE FROM ONE INCH TO TWO INCH, ELIMINATING THE HANGER INTERFERENCE, BACKFILLING SENSING LINES AT A HIGHER FLUSH VELOCITY AND IMPROVING SENSING LINE SLOPE. THESE EVENTS OCCURRED WHEN IN THE HOT SHUTDOWN MODE OF OPERATION WITH THE REACTOR MODE SELECTOR SWITCH IN THE SHUTDOWN POSITION. THE REACTOR POWER LEVEL WAS ZERO PERCENT. THE RV PRESSURES AND RV WATER TEMPERATURES FOR THE THREE EVENTS WERE AS FOLLOWS: FIRST EVENT, 60 PSIG AND 308 DEGREES FAHRENHEIT; SECOND EVENT, 12 PSIG AND 248 DEGREES FAHRENHEIT; THIRD EVENT, 3 PSIG AND 168 DEGREES FAHRENHEIT.

[75] PILGRIM 1 DOCKET 50-293 LER 92-001
CLASS I PIPING SEISMIC DAMPING RATIOS.
EVENT DATE: 121991 REPORT DATE: 022192 NSSS: GE TYPE: BWR

(NSIC 224118) ON DECEMBER 19, 1991, THE CLASS I PIPING SEISMIC DAMPING RATIOS PERMITTED FOR USE SINCE 1982 WERE FOUND TO NOT CONFORM WITH THE DAMPING RATIOS APPROVED BY THE NRC FOR USE AT PILGRIM STATION. THE DAMPING RATIOS WERE INCREASED IN TWO CHANGES TO THE FINAL SAFETY ANALYSIS REPORT (FSAR). THE CHANGES INCORPORATED THE DAMPING RATIOS FROM NRC REGULATORY GUIDE 1.61 AND ASME SECTION III CODE CASE N-411. NRC APPROVAL FOR THE DAMPING RATIO CHANGES WAS NOT SOUGHT UNDER 10 CFR 50.59 BECAUSE IT WAS NOT RECOGNIZED THAT THE CHANGES COULD CONSTITUTE AN UNREVIEWED SAFETY QUESTION. THIS CONDITION WAS DISCOVERED DURING A

REVIEW OF THE MAIN STEAM LINE/SAFETY RELIEF VALVE TAILPIPE STRESS ANALYSIS. THE PLANT WAS OPERATING WITH THE REACTOR MODE SWITCH IN THE RUN POSITION. REACTOR PRESSURE WAS 1035 PSIG AND REACTOR WATER TEMPERATURE WAS APPROXIMATELY 548 DEGREES FAHRENHEIT. THE USE OF THE HIGHER DAMPING RATIOS WAS DETERMINED NOT TO BE REPORTABLE IN ACCORDANCE WITH 10 CFR 50.73. THIS REPORT IS SUBMITTED VOLUNTARILY. THE CONDITION POSES NO THREAT TO THE PUBLIC HEALTH AND SAFETY. ALL PIPING SYSTEMS DESIGNED OR MODIFIED USING SEISMIC DAMPING RATIOS GREATER THAN THE NRC APPROVED FSAR RATIOS ARE OPERABLE AS DETERMINED IN AN OPERABILITY EVALUATION APPROVED ON JANUARY 13, 1992. LONG TERM CORRECTIVE ACTIONS ARE UNDER DEVELOPMENT.

[76] POINT BEACH 1 DOCKET 50-266 LER 92-001
TURBINE RUNBACK CAUSED BY IMPROPER POST-MAINTENANCE TESTING.
EVENT DATE: 012092 REPORT DATE: 021892 NSSS: WE TYPE: PWR

(NSIC 224064) AT 1353 ON JANUARY 20, 1992, WITH BOTH UNITS OPERATING AT 100% REACTOR POWER, AN AUTOMATIC RUNBACK OF THE UNIT 1 TURBINE OCCURRED. THE TURBINE RUNBACK WAS INITIATED AS A RESULT OF THE ACTUATION OF ROD BOTTOM BISTABLE WHICH GENERATED A ROD BOTTOM SIGNAL. THE ACTUATION OF THE BISTABLES OCCURRED WHEN POWER WAS LOST TO THE ROD POSITION INDICATION (RPI) CIRCUITRY DUE TO THE LOSS OF 1B03, A 480 VOLT SAFEGUARDS BUS. ACTUATION OF THE ROD BOTTOM BISTABLES CAUSED A TURBINE RUNBACK TO OCCUR. THE RUNBACK ALSO RESULTED IN THE AXIAL FLUX DIFFERENCE BEING OUT OF BAND FOR SEVENTEEN MINUTES BY COMPUTER INDICATION. THIS IS A VIOLATION OF TECHNICAL SPECIFICATION SECTION 15.3.10.B.2.B. POWER WAS RESTORED TO 1B03 AT 1356 AND THE POWER INCREASE WAS COMMENCED AT 1359. FULL POWER WAS REACHED AT APPROXIMATELY 1600.

[77] POINT BEACH 1 DOCKET 50-266 LER 92-002
MISSED VISUAL EXAMINATION OF REACTOR VESSEL INTERIOR.
EVENT DATE: 012292 REPORT DATE: 022192 NSSS: WE TYPE: PWR

(NSIC 224065) ON JANUARY 22, 1992, DURING OUR REVIEW OF THE IN-SERVICE INSPECTION LONG TERM PLAN AND ASSOCIATED RECORDS FOR EXAMINATIONS PERFORMED DURING THE SECOND TEN-YEAR INTERVAL FOR POINT BEACH NUCLEAR PLANT UNIT 1, WE DETERMINED THAT A VISUAL EXAMINATION (VT-3) OF ACCESSIBLE PORTIONS OF THE INSIDE OF THE REACTOR VESSEL WAS NOT PERFORMED AT THE REQUIRED PERIODICITY. THIS EXAMINATION IS REQUIRED BY THE 1977 EDITION, SUMMER 1979 ADDENDUM OF THE ASME SECTION XI CODE, ARTICLE IWB-2500, CATEGORY BN-1. THE ASME SECTION XI CODE REQUIRES THIS EXAMINATION BE PERFORMED ONCE EACH FORTY-MONTH PERIOD DURING THE SECOND TEN-YEAR INTERVAL. CONTRARY TO THIS REQUIREMENT, THE EXAMINATION WAS NOT PERFORMED DURING THE LAST FORTY-MONTH PERIOD OF THE SECOND TEN-YEAR INTERVAL. THE SECOND TEN-YEAR INTERVAL FOR POINT BEACH NUCLEAR PLANT UNIT 1 ENDED ON DECEMBER 20, 1990.

[78] PRAIRIE ISLAND 1 DOCKET 50-282 LER 92-001
FIRE DOOR LEFT OPEN AS RESULT OF PERSONNEL ERROR.
EVENT DATE: 011792 REPORT DATE: 021392 NSSS: WE TYPE: PWR

(NSIC 224070) ON JANUARY 17, 1992, BOTH UNITS WERE AT FULL POWER. AT ABOUT 1520 THE NRC RESIDENT INSPECTOR CALLED THE CONTROL ROOM TO REPORT THAT DOOR NO. 169 BETWEEN BUS ROOMS 25 AND 16 WAS OPEN AND NO FIRE WATCH WAS PRESENT. AN OPERATOR WAS DISPATCHED TO INVESTIGATE; THE OPERATOR CLOSED THE DOOR. THE DOOR HAD NOT BEEN BLOCKED OPEN, NOR WAS ANY OBSTRUCTION PRESENT TO HOLD THE DOOR OPEN. THE DOOR HAD BEEN HELD IN THE OPEN POSITION BY A DETENT WHICH WAS INTEGRAL WITH THE DOOR CLOSER. SINCE CONSTRUCTION PERSONNEL HAD BEEN WORKING IN THE AREA, THE PROJECT ENGINEER WAS INFORMED OF THE INCIDENT. THE DOOR CLOSER IS EQUIPPED WITH A FUSIBLE LINK INTENDED TO PROVIDE AUTOMATIC CLOSURE IN THE EVENT OF A FIRE. TESTING SHOWED THAT WITH THE FUSIBLE LINK REMOVED, THE DOOR WOULD NOT CLOSE, BUT WAS STILL HELD OPEN BY ITS DETENT. DOOR NO. 169 WAS DETERMINED TO BE INOPERABLE WHEN HELD IN THE OPEN POSITION BY THE DETENT. THERE WAS A PERSON PERFORMING QUALITY INSPECTIONS IN THE ROOM, AT THE TIME THE DOOR WAS FOUND OPEN, BUT HE HAD NOT BEEN DESIGNATED AS A FIRE WATCH. SINCE NO FIRE WATCH WAS PRESENT, TECHNICAL SPECIFICATION 3.14.G WAS VIOLATED.

[79] QUAD CITIES 1 DOCKET 50-254 LER 92-001
REACTOR WATER CLEANUP ISOLATION ON NON-REGENERATIVE HEAT EXCHANGER HIGH OUTLET TEMPERATURE.
EVENT DATE: 010692 REPORT DATE: 020392 NSSS: GE TYPE: BWR

(NSIC 223905) ON JANUARY 6, 1992, AT 1115 HOURS, UNIT ONE WAS IN THE RUN MODE AT 100 PERCENT O F RATED CORE THERMAL POWER. WHILE RETURNING THE REACTOR WATER CLEAN-UP (RWCU) SY STEM TO SERVICE, A NON-REGENERATIVE HEAT EXCHANGER (NRHX) HIGH TEMPERATURE ALARM WAS RECEIVED. ALTHOUGH NOT AN ENGINEERED SAFETY FEATURE (ESF) GROUP III ISOLATI ON, THIS RESULTED IN A CHALLENGE TO THE ESF LOGIC AND A SYSTEM ISOLATION. THE 1- 1299-9 VALVE, RWCU HOTLEG SUCTION ISOLATION VALVE WAS CLOSED. THE RWCU SYSTEM WA S RETURNED TO SERVICE AT 1445 HOURS. AN EMERGENCY NOTIFICATION SYSTEM (ENS) PHON E NOTIFICATION WAS MADE AT 1447 HOURS IN ACCORDANCE WITH 10CFR50.72(B)(II). THE PRIMARY CAUSE WAS DETERMINED TO BE PERSONNEL ERROR WITH A MECHANICAL PROBLEM AS A CONTRIBUTING FACTOR. THIS REPORT IS BEING SUBMITTED TO COMPLY WITH THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV).

[80] QUAD CITIES 1 DOCKET 50-254 LER 92-003
HPCI SUCTION LUBE OIL PRESSURE RELIEF LINE PIPE HANGER FOUND OUTSIDE FSAR COMPLIANCE DUE TO UNKNOWN CAUSES.
EVENT DATE: 020692 REPORT DATE: 030492 NSSS: GE TYPE: BWR

(NSIC 224139) AT 1230 HOURS ON FEBRUARY 6, 1992, UNIT ONE WAS IN THE RUN MODE AT 100 PERCENT RATED CORE THERMAL POWER. THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM WAS ALREADY IN A 14 DAY LIMITED CONDITION OF OPERATION (LCO) PER TECHNICAL SPECIFICATIONS. AT THIS TIME, BOILING WATER REACTOR SITE ENGINEERING (BWRSE) PERSONNEL INFORMED THE STATION THAT THE HPCI LUBE OIL COOLER INLET PRESSURE RELIEF LINE PIPE HANGER SUPPORT, M-Z84D-251, WAS OUTSIDE THE FINAL SAFETY ANALYSIS REPORT (FSAR) ALLOWABLES FOR THERMAL AND SEISMIC LOADINGS. THE PIPE HANGER SUPPORT WAS FOUND TO BE ACCEPTABLE FOR OPERABILITY, THUS NO IMMEDIATE ACTION WAS REQUIRED BY THE STATION. THE EXACT CAUSE OF THIS EVENT IS UNKNOWN. A MINOR DESIGN CHANGE, P04-1-92-022, WAS INITIATED TO CORRECT THE PROBLEM. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(II)(B).

[81] QUAD CITIES 2 DOCKET 50-265 LER 91-015
REACTOR BUILDING VENTILATION ISOLATION DURING RETURN TO SERVICE VERIFICATION.
EVENT DATE: 122491 REPORT DATE: 011892 NSSS: GE TYPE: BWR

(NSIC 223854) ON DECEMBER 24, 1991 AT 0445 HOURS UNIT TWO WAS IN THE RUN MODE AT 65% OF RATED CORE THERMAL POWER. WHILE PERFORMING A RETURN TO SERVICE VERIFICATION ON OUT OF SERVICE 3639, THE UNIT 2 EQUIPMENT OPERATOR (EO) INADVERTENTLY REMOVED FUSE F-3 AND A WHITE PLASTIC FUSE HOLDER OBSTRUCTING VIEW OF THE FUSE, FROM THE 2252-24X PANEL, UNIT TWO (U2) REACTOR (RX) BUILDING (BLDG) VENT CONTROL PANEL. THE FUSE REMOVAL CAUSED THE U2 RX BLDG VENT ISOLATION DAMPERS TO FAIL CLOSE AND TRIPPED THE SUPPLY AND EXHAUST FANS. THE IMMEDIATE CORRECTIVE ACTIONS WAS TO NOTIFY THE CONTROL ROOM OF THE EVENT AND REINSTALL THE FUSE. THE RETURN TO SERVICE WAS COMPLETED AND THE U2 RX BLDG VENTS TURNED BACK ON. FURTHER CORRECTIVE ACTIONS WILL BE TO REMOVE THE WHITE PLASTIC FUSE HOLDERS COMPLETELY. THE APPARENT CAUSE OF THE EVENT WAS A DESIGN DEFICIENCY FROM A HUMAN FACTORS STANDPOINT. THIS EVENT IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV).

[82] QUAD CITIES 2 DOCKET 50-265 LER 92-005
STANDBY GAS TREATMENT SYSTEM DUE TO DISLODGED FUEL POOL RADIATION MONITOR BYPASS SWITCH CAUSED BY WORK ON 902-2 PANEL.
EVENT DATE: 012592 REPORT DATE: 022092 NSSS: GE TYPE: BWR

(NSIC 224063) ON JANUARY 25, 1992, AT 1103 HOURS, UNIT TWO WAS IN THE REFUEL MODE. DURING INSTALLATION OF A MOUNTING PLATE ON THE 902-2 PANEL, THE "A" FUEL POOL RADIATION MONITOR BYPASS SWITCH (33)(IL) FELL FORWARD FROM THE 902-10 PANEL, CAUSING A SHORT CIRCUIT AND BLOWING THE RADIATION MONITOR UPSCALE TRIP RELAY FUSE (FU). THIS LED TO AN INITIATION OF THE "A" TRAIN OF STANDBY GAS TREATMENT SYSTEM (SBGTS) (BH) AND AN ISOLATION OF THE REACTOR BUILDING VENTILATION SYSTEM (VA). THE ROOT CAUSE OF THIS EVENT WAS THE FAILURE OF THE BYPASS SWITCH MOUNTING

BRACKET DUE TO VIBRATIONS. IMMEDIATE CORRECTIVE ACTIONS INCLUDED REMOUNTING THE BYPASS SWITCH, REPLACEMENT OF THE UPSCALE TRIP FUSE, AND THE REESTABLISHMENT OF THE REACTOR BUILDING VENTILATION SYSTEM AND THE SHUTDOWN OF THE "A" TRAIN OF SBGTS. A MINOR DESIGN CHANGE WILL RELOCATE AND SEISMICALLY MOUNT THE BYPASS SWITCHES IN THE 90X-10 PANELS. IN THE INTERIM, INSTRUMENT MAINTENANCE TECHNICIANS WILL CHECK THE MOUNTING SCREWS DURING EACH MONTHLY PERFORMANCE OF QIS 35-2.

[83] RIVERBEND 1 DOCKET 50-458 LER 91-008 REV 01
UPDATE ON FIRE HAZARDS ANALYSIS DEFICIENCIES INCLUDING LACK OF FIRE WRAP/INADEQUATE FIRE BARRIER.
EVENT DATE: 041591 REPORT DATE: 021892 NSSS: GE TYPE: BWR

(NSIC 224100) AT 1345 HOURS ON 4/15/91, WITH THE REACTOR AT FULL POWER IN OPERATIONAL CONDITION 1, IT WAS DISCOVERED THAT ELECTRICAL CABLES LOCATED IN FIRE AREA ET-2, WHICH MAY CAUSE SPURIOUS OPERATION OF VALVES 1E51*MOVF063 (RCIC INBOARD STEAM ISOLATION VALVE) AND 1E51*MOVF078 (RCIC VACUUM BREAKER VALVE), DID NOT HAVE FIRE WRAP CONTRARY TO FIRE HAZARDS ANALYSIS (FY) REQUIREMENTS. AT 1300 ON 4/23/91, ADDITIONAL CABLES, WHICH COULD CAUSE THE SAME PROBLEM WERE FOUND IN FIRE AREAS AB-2, C-2 AND C-6. RCIC IS REQUIRED BY THE FHA 224100FE SHUTDOWN IN THESE FIRE AREAS. SINCE THESE VALVES ARE REQUIRED NOT TO CHANGE POSITION FOR OPERATION OF RCIC AND FIRE DAMAGE TO THESE CABLES MAY CAUSE LOSS OF RCIC, THE CABLES WOULD REQUIRE WRAPPING IN THESE FIRE AREAS. UPON DISCOVERY OF THIS CONDITION, THE AFFECTED CABLES WERE TREATED AS HAVING MISSING FIRE BARRIERS AND THE ACTION STATEMENT PRESCRIBED IN TECHNICAL SPECIFICATION 3/4.7.7, "FIRE RATED ASSEMBLIES", WAS IMPLEMENTED FOR AREAS CONTAINING THESE CABLES. ERRORS MADE DURING THE ORIGINAL DEVELOPMENT OF THE FHA WERE THE CAUSE FOR THE IDENTIFIED CABLES NOT BEING WRAPPED IN THE IDENTIFIED FIRE AREAS. ADDITIONAL DEFICIENCIES HAVE BEEN DISCOVERED DURING THE FHA REVIEW. THESE RECENTLY DISCOVERED DEFICIENCIES CONCERN APPENDIX R SEPARATION AND A FIRE AREA THAT WAS NOT PREVIOUSLY IDENTIFIED.

[84] ROBINSON 2 DOCKET 50-261 LER 92-002
FAILURE TO TEST ALL CIRCUITRY ASSOCIATED WITH AUXILIARY FEEDWATER AUTO START.
EVENT DATE: 012792 REPORT DATE: 021292 NSSS: WE TYPE: PWR

(NSIC 224027) AT ABOUT 1045 HOURS ON 1/27/92, WITH H.B. ROBINSON UNIT NO. 2 OPERATING AT 100% POWER, IT WAS DETERMINED THAT THE CONDITION IDENTIFIED IN ACR 92-013, DATED 1/15/92, REPRESENTED A CONDITION THAT WAS REPORTABLE, BUT DID NOT INVOLVE INOPERABLE EQUIPMENT. THE ACR IDENTIFIED THAT A SET OF NORMALLY CLOSED CONTACTS, TX, AND INTERCONNECTING WIRING, SHOWN ON CONTROL WIRING DIAGRAM, B-190628 SHEETS 651 AND 655, COULD NOT BE SHOWN TO BE TESTED DURING THE PERFORMANCE OF MST-202, "4KV MAIN FEEDWATER BREAKERS OPEN-AUTO-START TEST OF MOTOR DRIVEN AUXILIARY FEEDWATER (MDAFW) SYSTEM". MST-202 IS CONDUCTED TO SATISFY THE REQUIREMENTS OF TECH SPECS TABLE 4.8-1 ITEM E, TRIP MAIN FEEDWATER PUMPS. THE ACR ALSO STATED THAT THE OPERABILITY OF THE MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS. THE ACR ALSO STATED THAT THE OPERABILITY OF THE MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS WAS NOT A CONCERN BECAUSE BOTH PUMPS AUTO STARTED ON A STEAM GENERATOR LOW LEVEL SIGNAL DURING A REACTOR TRIP ON 8/30/91. THIS MDAFW AUTO START ON STEAM GENERATOR LOW LEVEL SIGNAL UTILIZED THE SAME TX CONTACTS THAT WERE IDENTIFIED IN THE ACR, AND HAS THEREBY VERIFED THAT THE UNTESTED TX CONTACTS AND INTERCONNECTING WIRING ARE INTACT AND FUNCTIONAL.

[85] SALEM 1 DOCKET 50-272 LER 92-002
CONTAINMENT SPRAY FLUID TRAVEL TIME GREATER THAN ORIGINALLY ASSUMED.
EVENT DATE: 011692 REPORT DATE: 021892 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 224066) ON 1/16/92, AN ENGINEERING ASSESSMENT CONCLUDED THAT THE SALEM UNIT 1 (SU1) CONTAINMENT SPRAY SYSTEM CALCULATED FLUID TRAVEL TIME (FROM THE TIME THE CONTAINMENT SPRAY PUMPS REACH FULL SPEED TO WHEN FLUID EXITS THE SPRAY NOZZLES) EXCEEDS THE ORIGINAL ASSUMED VALUE OF 28 SECONDS. THE CORRECT FLUID TRAVEL TIME, PER VERIFIED CALCULATIONS, IS 47 SECONDS. THIS ADDITIONAL TRAVEL TIME EXCEEDS THE UPDATED FINAL SAFETY ANALYSIS (UFSAR) LICENSING BASIS ASSUMPTION OF 59 SECONDS.

THE ROOT CAUSE OF THIS EVENT IS INADEQUATE DESIGN REVIEW. THE ORIGINAL CALCULATION WAS DOCUMENTED IN A PSE&G MEMORANDUM TO WESTINGHOUSE, DATED 6/29/78. THE MEMO PROVIDED A 28 SECOND FLUID TRAVEL TIME WHICH WAS APPARENTLY BASED ON AN INCORRECT APPROXIMATION OF CONTAINMENT SPRAY SYSTEM PIPING VOLUME. A JUSTIFICATION FOR CONTINUED OPERATION REPORT WAS COMPLETED TO SUPPORT SU1 CONTINUED OPERATION. THE CURRENT PROGRAM FOR IMPLEMENTATION OF NEW OR REVISED DESIGN CALCULATIONS REQUIRES DETAILED REVIEW AND VERIFICATION OF THE CALCULATIONS. THE UFSAR WILL BE REVISED AS APPROPRIATE. A REVIEW OF THE TECH SPECS (TS) WAS COMPLETED. A LICENSE CHANGE REQUEST (LCR) WILL BE SUBMITTED TO LOWER THE CONTAINMENT SPRAY ESF RESPONSE TIME TO BE CONSISTENT WITH THE SAFETY ANALYSIS ASSUMPTION(S). PROCEDURE REVISIONS HAVE BEEN INITIATED, TO PREVENT INADVERTENT APPLICATION OF THE TS LIMIT, UNTIL THIS LCR IS APPROVED AND IMPLEMENTED.

[86]   SALEM 1   DOCKET 50-272   LER 92-003
CONTROL ROOM VENTILATION SWITCH DUE TO 1R1B RMS CHANNEL CABLE FAILURE.
EVENT DATE: 011792   REPORT DATE: 021492   NSSS: WE   TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 224067) ON JANUARY 17, 1992 AT 1047 HOURS, DURING NORMAL POWER OPERATION, THE CONTROL ROOM VENTILATION AUTOMATICALLY SWITCHED FROM THE NORMAL TO THE ACCIDENT MODE OF OPERATION (100% RECIRCULATION) FOR BOTH SALEM UNIT 1 AND SALEM UNIT 2 (BY DESIGN). THIS WAS DUE TO A HIGH CHANNEL SPIKE INITIATING AN ALARM SIGNAL ON THE CONTROL ROOM AIR INTAKE RADIATION MONITORING SYSTEM (RMS) CHANNEL, 1R1B. THE SWITCHING OF THE CONTROL ROOM VENTILATION SYSTEM TO THE EMERGENCY MODE OF OPERATION IS AN ENGINEERED SAFETY FEATURE (ESF). THE ROOT CAUSE OF THE 1R1B CHANNEL ESF ACTUATION IS EQUIPMENT FAILURE. INVESTIGATION IDENTIFIED THAT THE 1R1B CHANNEL FAILURE WAS THE RESULT OF A BROKEN DETECTOR SHIELD WIRE SOLDERED CONNECTION IN THE MULTI-PRONG DETECTOR CABLE CONNECTOR CONNECTED TO THE BACK OF THE CONTROL ROOM 1R1B CHANNEL RACK. THE CHANNEL SPIKING WAS REPRODUCED, BY THE MAINTENANCE-I&C TECHNICIAN, WHEN THE CABLE WAS MANIPULATED. THIS CABLE IS DISCONNECTED TO SUPPORT CHANNEL CALIBRATION. THE SHIELD WIRE SOLDERED CONNECTION HAD APPARENTLY BROKEN DUE TO THE PERIODIC DISCONNECTION/RECONNECTION OF THE CABLE. THE CABLE SHIELD WIRE WAS REPAIRED. THIS EVENT WILL BE REVIEWED BY SYSTEM ENGINEERING TO MODIFY THE PREVENTIVE MAINTENANCE PROGRAM AS APPLICABLE.

[87]   SALEM 1   DOCKET 50-272   LER 92-005
ESF ACTUATION SIGNALS DUE TO LOSS OF POWER TO RADIATION MONITORING SYSTEM CHANNELS.
EVENT DATE: 013092   REPORT DATE: 030292   NSSS: WE   TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 224144) THIS LER ADDRESSES TWO (2) ENGINEERED SAFETY FEATURE (ESF) ACTUATION SIGNALS INITIATED THROUGH THE RADIATION MONITORING SYSTEM (RMS): ONE FOR CONTAINMENT PURGE/PRESSURE-VACUUM RELIEF (CP/P-VR) SYSTEM ISOLATION AND ONE FOR CONTROL ROOM VENTILATION SWITCH TO THE EMERGENCY MODE OF OPERATION. ON 1/30/92, DURING NORMAL PLANT OPERATIONS, THE 120 VC 15 AMP BREAKER FOR RMS RACK 120 TRIPPED OPEN RESULTING IN THE ASSOCIATED RMS CHANNELS LOSING POWER. THESE CHANNELS INCLUDE: 1. THE 1R1A CONTROL ROOM GENERAL AREA MONITOR CHANNEL; 2. THE 1R11A CONTAINMENT PARTICULATE RMS MONITOR CHANNEL; 3. THE 1R13A NO. 11 CONTAINMENT FAN COIL UNIT PROCESS MONITOR CHANNEL; 4. THE 1R33 REACTOR COOLANT FILTERS PROCESS MONITOR CHANNEL; AND 5. THE 1R19A NO. 11 STEAM GENERATOR BLOWDOWN PROCESS MONITOR CHANNEL. SUBSEQUENTLY: THE 1R1A CHANNEL INITIATED A SIGNAL FOR CONTROL ROOM VENTILATION SWITCH TO THE EMERGENCY MODE OF OPERATION; THE 1R11A CHANNEL INITIATED A CP/P-VR SYSTEM ISOLATION SIGNAL; AND THE 1R19A CHANNEL, UPON LOSING POWER INITIATED A STEAM GENERATOR BLOWDOWN ISOLATION SIGNAL. THE EQUIPMENT REQUIRED TO CHANGE STATE AS A RESULT OF THE 1R1A, 1R11A AND 1R19A SIGNALS RESPONDED AS PER DESIGN. THE CAUSE OF THE LOSS OF POWER TO THE 120 VAC RMS RACK NO. 120 WAS THE BREAKER TRIPPING. INVESTIGATION AS TO THE CAUSE OF THE BREAKER TRIPPING IS CONTINUING.THE SUBJECT RMS CHANNELS WERE RETURNED TO SERVICE ON 1/30/92.

[88] SALEM 2 DOCKET 50-311 LER 92-003
CONTAINMENT ROOM VENTILATION SWITCH TO EMERGENCY MODE DUE TO 2R1B CHANNEL FAILURE.
EVENT DATE: 012392 REPORT DATE: 022092 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 1 (PWR)
VENDOR VICTOREEN INC

(NSIC 224076) ON JANUARY 23, 1992 AT 1622 HOURS, 1701 HOURS AND 1720 HOURS, SALEM UNITS 1 AND 2 CONTROL ROOM VENTILATION AUTOMATICALLY SWITCHED FROM THE NORMAL TO THE ACCIDENT MODE OF OPERATION (100% RECIRCULATION). NO POSITIVE INDICATION IN EITHER CONTROL ROOM SHOWED WHICH RADIATION MONITORING SYSTEM (RMS) CHANNEL CAUSED THE EVENT. THE SWITCHING OF THE CONTROL ROOM VENTILATION SYSTEM TO THE EMERGENCY MODE OF OPERATION IS AN ENGINEERED SAFETY FEATURE (ESF). THE ROOT CAUSE OF THE THREE (3) 2RL B CHANNEL ESF ACTUATIONS IS DESIGN INADEQUACY RESULTING IN AN EQUIPMENT FAILURE. THE MAJORITY OF THE SALEM UNIT 2 RMS CHANNEL MONITORS ARE MANUFACTURED BY VICTOREEN AS IS THE 2R1B CHANNEL. PERIODIC PROBLEMS WITH THE VICTOREEN SYSTEM (I.E. PIN CONNECTIONS) HAD BEEN EXPERIENCED AS INDICATED IN PRIOR LERS. INVESTIGATION IDENTIFIED THAT THE 2R1B CHANNEL FAILURE(S) WERE THE RESULT OF TWO (2) BENT PINS ON THE CONTROL CIRCUITRY SCALAR MODULE RESULTING IN POOR CONTACT WITH THE CHANNEL BACKPLANE. THE MAINTENANCE-I&C TECHNICIAN WAS ABLE TO REPRODUCE THE EVENT BY MANIPULATION OF THE SCALAR MODULE. THE 2R1B R CHANNEL CONTROLLER, SCALAR AND BACK PLANE PINS WERE CLEANED. THE BENT SCALAR PINS WERE STRAIGHTENED. UPON COMPLETION OF REPAIRS, THE CHANNEL WAS SUCCESSFULLY CALIBRATED AND RETURNED TO SERVICE. ENGINEERING HAS INVESTIGATED THE CONCERNS WITH THE UNIT 2 RMS CHANNELS.

[89] SAN ONOFRE 1 DOCKET 50-206 LER 91-013 REV 02
UPDATE ON MISASSEMBLY OF THE 4160 VOLT ROOM HALON SYSTEM ACTUATION LINES.
EVENT DATE: 070191 REPORT DATE: 030592 NSSS: WE TYPE: PWR

(NSIC 224133) AT 0524 ON 7/1/91, AN INADVERTENT ACTUATION OF THE HALON FIRE SUPPRESSION SYSTEM IN THE UNIT 1 4160 VOLT (4 KV) SWITCHGEAR ROOM OCCURRED. MOISTURE INTRUSION INTO THE HALON CONTROL PANEL ACTUATION CIRCUITRY CAUSED THE INADVERTENT ACTUATION. CONTRARY TO DESIGN, THE DISCHARGE WAS LIMITED SOLELY TO THE MAIN BANK MASTER BOTTLE. THE MAIN BANK SLAVE BOTTLES FAILED TO PROPERLY ACTUATE AS DESIGNED DUE TO AN INCORRECTLY CONNECTED ACTUATION LINE BETWEEN THE MASTER BOTTLE AND THE SLAVE BOTTLES. THE RESERVE BANK (REDUNDANT) HALON SYSTEM WAS SIMILARLY INCORRECTLY CONFIGURED AND WAS THEREFORE ALSO INCAPABLE OF COMPLETE DISCHARGE. AS A RESULT, BOTH BANKS OF THE 4 KV ROOM HALON SYSTEM WERE INOPERABLE. INVESTIGATION HAS CONCLUDED THAT THE ACTUATION LINES WERE INCORRECTLY CONNECTED DURING MAINTENANCE ACTIVITIES THAT OCCURRED EITHER IN 6/88 OR IN 4-5/89, WHEN THE MAIN AND RESERVE BANK MASTER BOTTLES WERE DISCONNECTED AND TRANSPORTED TO AN OFF-SITE VENDOR FOR SERVICING, AND REINSTALLED BY SCE. THIS EVENT REPRESENTS A CONDITION PROHIBITED BY TS 3.14.4, PART B. SCE'S INVESTIGATION HAS IDENTIFIED DEFICIENCIES IN: 1) MAINTENANCE PROCEDURES, 2) POST-MAINTENANCE VERIFICATION, 3) PRACTICES RELATED TO MARKING OF PARTS FOR CORRECT REASSEMBLY, 4) HALON SYSTEM KNOWLEDGE.

[90] SAN ONOFRE 2 DOCKET 50-361 LER 91-018 REV 01
UPDATE ON SAFETY RELATED INSTRUMENTATION NOT INSTALLED IN A SEISMICALLY QUALIFIED CONFIGURATION.
EVENT DATE: 101591 REPORT DATE: 022592 NSSS: CE TYPE: PWR
VENDOR: FOXBORO CO., THE

(NSIC 224124) ON 10/15/91, WHILE UNIT 2 WAS IN MODE 5 DURING A REFUELING OUTAGE, AN INSPECTION OF THE SAFETY RELATED FOXBORO SPEC 200 PROCESS INSTRUMENTATION CABINETS (CAB) DETERMINED THAT SEVERAL INSTRUMENT MODULES WERE MISSING THE UPPER AND/OR LOWER GUIDE RAILS AND/OR VIBRATION DAMPENING MATERIAL (BUMPERS). WITHOUT THE GUIDE RAILS AND BUMPERS 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 AFFECTED INSTRUMENT LOOPS TO HAVE BEEN EXCEEDED DURING PREVIOUS PERIODS OF OPERATION IN MODES 1, 2, 3, AND 4. SUBSEQUENT SEISMIC TESTING DEMONSTRATED THAT WITHOUT BUMPERS INSTALLED, THE AFFECTED INSTRUMENT MODULES WOULD HAVE FUNCTIONED PROPERLY DURING AND AFTER THE SONGS DESIGN BASIS SEISMIC

EVENTS. BY 11/4/91, ALL MISSING GUIDE RAILS WERE INSTALLED, AND THE AFFECTED INSTRUMENTATION WAS RETURNED TO AN OPERABLE CONDITION. THE PRIMARY CAUSE OF THIS DEFICIENCY WAS DETERMINED TO BE INADEQUATE VENDOR INSTALLATION INFORMATION ADDITIONALLY, ALTHOUGH THE INFORMATION PROVIDED IN THE VENDOR QUALIFICATION REPORT PRESENTED AN OPPORTUNITY FOR SCE TO IDENTIFY THE INSTALLATION DEFICIENCY, THIS SUBTLETY WAS NOT DETECTED.

[91] SAN ONOFRE 3 DOCKET 50-362 LER 90-014 REV 02
UPDATE ON CONTAINMENT SPRAY SYSTEM PRESSURE INDICATOR INOPERABLE DUE TO FAILED PRESSURE TRANSMITTER.
EVENT DATE: 122390 REPORT DATE: 010392 NSSS: CE TYPE: PWR
VENDOR: FOXBORO CO., THE

(NSIC 223835) ON 12/23/90, WITH UNIT 3 AT 100% POWER, CONTAINMENT SPRAY SYSTEM (CSS) TRAIN "B" PUMP DISCHARGE PRESSURE INDICATOR 3PI-0303-2 WAS DETERMINED TO BE INOPERABLE DUE TO ITS FAILURE TO DISPLAY THE CORRECT PRESSURE READING. THIS INSTRUMENT IS A POST-ACCIDENT MONITORING INSTRUMENTATION (PAMI) COMPONENT AND AS SUCH IS SUBJECT TO THE REQUIREMENTS OF TECHNICAL SPECIFICATION (TS) 3.3.3.6 WHICH ALLOWS FOR THE INOPERABILITY OF PAMI EQUIPMENT FOR U[illegible] INVESTIGATION CONCLUDED THAT 3PI-0303-2 BECAME INOPERABLE [illegible] ASSOCIATED PRESSURE TRANSMITTER 3PT-0303-2, WHICH PROVI[illegible] 3PI-0303-2, FAILED DURING SUBGROUP RELAY TESTING AS[illegible]. THIS REPRESENTS A CONDITION PROHIBITED BY TS 3.3.3.6, S[illegible] INOPERABLE FOR GREATER THAN 7 DAYS. THE TRANSMITTER FAILURE [illegible] JAMMING OF THE FEEDBACK COIL BY A SMALL BLACK, FERROUS PARTICLE, [illegible] VI[illegible]NITY OF THE GAP BETWEEN THE FEEDBACK COIL (MOVING PART) AND I[illegible] ST[illegible]IONARY PART). THE PARTICLE WAS MOST LIKELY INTRODUCED DURING A CA[illegible]TIVITY. JAMMING OF THE FEEDBACK COIL PRODUCED A PRESSURE INDICATION WHICH WAS SUBSTANTIALLY GREATER THAN ACTUAL PRESSURE (APPROXIMATELY 400 PSIG VERSUS 25 PSIG). THE TRANSMITTER WAS REPLACED, AND THE INDICATOR WAS RETURNED TO OPERABLE STATUS ON 12/28/90.

[92] SEQUOYAH 1 DOCKET 50-327 LER 91-009 REV 01
UPDATE ON OPERATION WITH INOPERABLE AUXILIARY BUILDING FIRE SUPPRESSION SYSTEM BECAUSE OF INADEQUATE TEST PERFORMANCE AND REVIEW.
EVENT DATE: 050691 REPORT DATE: 030292 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 224155) THIS LER PROVIDES DETAILS CONCERNING THE CAUSE OF THE AUXILIARY BUILDING SPRINKLER SYSTEM'S REDUCED PERFORMANCE AND ALSO TO REPORT THE CORRECTIVE ACTIONS TAKEN TO RETURN THE SYSTEM TO OPERABLE STATUS. ON MAY 6, 1991, AT 1600 EDT, WITH UNITS 1 AND 2 OPERATING IN MODE 1, LCO 3.7.11.1 WAS ENTERED WHEN THE FIRE SUPPRESSION SYSTEM FOR THE AUXILIARY BUILDING WAS DECLARED INOPERABLE. THE SURVEILLANCE TEST THAT DEMONSTRATES THE OPERABILITY OF THE SYSTEM WAS PERFORMED ON 4/2/91. ON 5/6/91, THE FIRE PROTECTION ENGINEER DETERMINED THAT THE TEST DATA DID NOT SATISFY THE ACCEPTANCE CRITERIA. THE TEST WAS INVALIDATED, THE SYSTEM WAS DECLARED INOPERABLE, AND LCO 3.7.11.1 WAS ENTERED. INADEQUATE MANAGERIAL SUPERVISION HAD RESULTED IN A TEST DIRECTOR BEING ASSIGNED TO CONDUCT THIS TEST IN APRIL WHO HAD NOT BEEN PROPERLY TRAINED. THE TEST DIRECTOR HAD INCORRECTLY CONSIDERED THE TEST ACCEPTABLE. FOLLOWING SYSTEM ADJUSTMENTS AND TESTING, THE EXISTING SYSTEM WAS CONSIDERED ACCEPTABLE TO SERVE AS THE BACKUP FIRE SUPPRESSION SYSTEM IN ACCORDANCE WITH ACTION STATEMENT (B)(1) FOR LCO 3.7.11.1. THE CAUSE OF THE REDUCTION IN SYSTEM PERFORMANCE HAS BEEN DETERMINED TO BE INTERNAL PIPING CORROSION DEPOSITS, INCRUSTATION, AND A BUILDUP OF RIVER SEDIMENT. THE FIRE PROTECTION SYSTEM WAS RESTORED TO OPERABLE STATUS ON 11/30/91.

[93] SEQUOYAH 1 DOCKET 50-327 LER 91-016 REV 01
UPDATE ON OPERATIONS WITH UNQUALIFIED PENETRATION SEALS CAUSED BY THERMAL MOVEMENTS.
EVENT DATE: 071191 REPORT DATE: 021492 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
VENDOR: DOW CORNING CORP.

(NSIC 224036) ON JULY 11, 1991, WITH BOTH UNITS OPERATING IN MODE 1, A CONDITION

WAS IDENTIFIED INVOLVING UNQUALIFIED PENETRATION SLEEVE SEALS BECAUSE OF PIPING THERMAL MOVEMENTS. THE UNQUALIFIED SLEEVE SEALS CONSIST OF DOW CORNING 3-6548 ROOM TEMPERATURE VULCANIZING (RTV) SILICONE FOAM. PIPING THROUGH FOUR UNIT 1 AND FIVE UNIT 2 PENETRATIONS EXCEEDED THE AXIAL MOVEMENT CRITERIA OF 15 PERCENT OF THE MINIMUM ANNULAR DISTANCE AND THREE UNIT 1 AND THREE UNIT 2 PENETRATIONS EXCEEDED THE 1/4-INCH LIMIT FOR RADIAL MOVEMENTS. THE CAUSE OF THE EXISTING UNQUALIFIED PENETRATIONS IS THAT ORIGINAL DESIGN REQUIREMENTS RELATIVE TO RTV FOAM SEALS WERE INADEQUATE. CORRECTIVE ACTIONS INCLUDE MAINTAINING REQUIRED FIRE WATCH COVERAGE IN THE INTERIM, MODIFYING THE UNQUALIFIED SEALS AND ESTABLISHING AN ACTION PLAN TO ADDRESS THE REMAINING SLEEVE SEAL ISSUES. THIS REPORT IS BEING SUBMITTED AS REQUIRED BY 10 CFR 50.73(A)(2)(I)(B) AND IN FOLLOW-UP TO SPECIAL REPORT 91-11.

[94] SEQUOYAH 1 DOCKET 50-327 LER 92-001
FAILURE TO PERFORM A SURVEILLANCE REQUIREMENT BECAUSE OF PROCEDURAL INADEQ[illegible]ES
EVENT DATE: 011792 REPORT DATE: 021892 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 224080) ON JANUARY 17, 1992, AT 0800, TVA DETERMINED THAT THE 2B-[illegible] GENERATOR LOAD SEQUENCE TIMER ASSOCIATED WITH THE B TRAIN ELECTRIC BOARD [illegible] HANDLING UNIT HAD NOT BEEN CALIBRATED WITHIN THE FREQUENCY REQUIRED BY TECHNICAL SPECIFICATI ON (TS) SURVEILLANCE REQUIREMENT (SR) 4.8.1.1.2.D.10. ON JANUARY 17, THE TIMER WAS SUBSEQUENTLY FOUND TO BE WITHIN TOLERANCE. ON MARCH 8, 1991, THE TIMER SURVEILLANCE INSTRUCTION (SI) WAS REVISED TO INCLUDE CALIBRATION OF THESE SEQUENCE TIMERS. THE REVISION WAS NOT REVIEWED BY THE PERIODIC TEST COORDINATOR AS INTENDED BY THE STANDARD GOVERNING PROCEDURE REVISIONS BECAUSE THIS INTENT WAS NOT CLEARLY CONVEYED; THEREFORE, THE SCHEDULING MECHANISM FOR ENSURING THAT SRS ARE PERFORMED WAS BYPASSED. ADDITIONALLY, WORK REQUESTS (WRS) WRITTEN TO CALIBRATE THE TIMERS WITHIN FREQUENCY WERE NOT COORDINATED WITH THE PERIODIC TEST SECTION. THE SITE STANDARD GOVERNING WRS WAS FOUND TO NOT REQUIRE THIS COORDINATION. CORRECTIVE ACTION INCLUDES REVISING THE SITE STANDARDS GOVERNING CONTROL OF SITE PROCEDURES AND WRS TO ADEQUATELY CONVEY THE INTENT OF THE PERIODIC TEST COORDINATOR'S REVIEW OF PROCEDURES, AND TO ENSURE THE WRS SATISFYING SRS ARE COORDINATED WITH THE PERIODIC TEST SECTION.

[95] SEQUOYAH 1 DOCKET 50-327 LER 92-002
A CONTAINMENT VENTILATION ISOLATION OCCURRED AS A RESULT OF A SPURIOUS ACTUATION OF A RADIATION ALARM.
EVENT DATE: 011892 REPORT DATE: 021892 NSSS: WE TYPE: PWR

(NSIC 224081) AT 1358 EASTERN STANDARD TIME (EST) ON JANUARY 18, 1992, WITH UNIT 1 OPERATING AT 100 PERCENT POWER, A B TRAIN CONTAINMENT VENTILATION ISOLATION (CVI) OCCURRED BECAUSE OF A SPURIOUS ACTUATION OF CONTAINMENT PURGE EXHAUST RADIATION MONITOR (RM) 1-RM-90-131. A PURGE WAS NOT BEING PERFORMED AT THE TIME OF THE ISOLATION. OPERATIONS VERIFIED THAT HIGH RADIATION CONDITIONS DID NOT EXIST AND RECOVERED FROM THE CVI. NO PROBLEMS WITH THE RM WERE IDENTIFIED DURING SUBSEQUENT TROUBLESHOOTING AND THE RM WAS RETURNED TO SERVICE. THE ROOT CAUSE OF THIS EVENT IS INDETERMINATE.

[96] SEQUOYAH 1 DOCKET 50-327 LER 92-003
FIRE SUPPRESSION VALVE POSITIONS INSIDE CONTAINMENT NOT VERIFIED BECAUSE OF A DEFICIENT PROCEDURE.
EVENT DATE: 012792 REPORT DATE: 022692 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 2241[illegible]) ON JANUARY 27, 1992, SEQUOYAH (SQN) DETERMINED THAT SURVEILLANCE REQUIREMENT (SR) 4.7.11.2.A WAS NOT SATISFIED FOR FIRE SUPPRESSION SYSTEM VALVES INSIDE CONTAINMENT WITH THE UNIT AT POWER. A REVIEW OF THE SURVEILLANCE PROCEDURE'S REVISION HISTORY INDICATED THAT THIS CONDITION HAS EXISTED SINCE THE INITIAL ISSUE OF THE PROCEDURE. THE ROOT CAUSE OF THIS EVENT APPEARS TO BE THAT THE INTENT OF THE SR WAS CONSIDERED TO BE FULFILLED WITHOUT VERIFYING THE POSITION OF THE VALVES IN CONTAINMENT WHILE THE UNIT WAS OPERATING BECAUSE OF ACCESSIBILITY CONSIDERATIONS. THE VALVES WERE REQUIRED TO BE LOCKED OR SEALED IN

THE OPEN POSITION THROUGHOUT THIS TIMEFRAME. THE SURVEILLANCE INSTRUCTION (SI) IMPLEMENTING THE SR WAS REVISED ON JANUARY 28, 1992, AND PERFORMED ON JANUARY 29, 1992. THE VALVES WERE FOUND IN THE CORRECT POSITION.

[97] SHEARON HARRIS 1 DOCKET 50-400 LER 91-021
TECHNICAL SPECIFICATION VIOLATION DUE TO DEFICIENT CONTAINMENT SUMP LEAK RATE COMPUTER PROGRAM.
EVENT DATE: 121891 REPORT DATE: 011692 NSSS: WE TYPE: PWR

(NSIC 223803) ON 12/18/91, AN ERROR WAS DISCOVERED IN THE PROGRAM UTILIZED BY THE PLANT PROCESS COMPUTER FOR DETERMINATION OF CONTAINMENT SUMP INLEAKAGE FLOWRATE. THIS ERROR WOULD HAVE PREVENTED THE COMPUTER FROM DETECTING A ONE GALLON PER MINUTE INCREASE IN FLOW UNDER CERTAIN CONDITIONS, AS REQUIRED BY TECHNICAL SPECIFICATIONS (TS). THIS CONDITION WAS DISCOVERED BY THE CONTROL ROOM STAFF DURING AN EVOLUTION THAT INCLUDED DISCHARGING WATER INTO THE CONTAINMENT SUMP. THE EXPECTED LEVEL INCREASE WAS OBSERVED, BUT THE COMPUTER GENERATED FLOW RATE REMAINED UNCHANGED AT ZERO. THE COMPUTER PROGRAM WAS DECLARED INOPERABLE AT THIS POINT AND COMPENSATORY MEASURES WERE COMMENCED TO COMPLY WITH TS REQUIREMENTS. INVESTIGATION REVEALED THAT THIS CONDITION WAS CAUSED BY PERSONNEL ERROR DURING THE DEVELOPMENT OF THE PLANT MODIFICATION THAT INSTALLED THIS CAPABILITY INTO THE PLANT COMPUTER IN SEPTEMBER OF 1988. SUBSEQUENT CORRECTIVE ACTIONS INCLUDED CORRECTING THE ERROR IN THE COMPUTER PROGRAM AND RESTORING IT TO SERVICE. THIS IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(I)(B) AS A TS VIOLATION. NO SIMILAR EVENTS HAVE BEEN REPORTED

[98] SHEARON HARRIS 1 DOCKET 50-400 LER 92-002
TECHNICAL SPECIFICATION VIOLATION DUE TO UNDETECTED FAILURE OF PLANT PROCESS COMPUTER.
EVENT DATE: 011992 REPORT DATE: 021892 NSSS: WE TYPE: PWR
VENDOR: GOULD INC.

(NSIC 224095) AT 0600 ON 1/19/92, PROBLEMS WERE ENCOUNTERED WITH THE PLANT PROCESS COMPUTER. A SUBROUTINE OF THIS COMPUTER PERFORMS THE CONTAINMENT SUMP LEAKRATE CALCULATION TO SATISFY TECHNICAL SPECIFICATION 3.4.6.1.B. BY 1100, CONTROL ROOM PERSONNEL AN D A PLANT COMPUTER TECHNICIAN COMPLETED EFFORTS THAT WERE THOUGHT TO RESTORE THE SUMP LEAKRATE FUNCTION. WITH THESE EFFORTS COMPLETE AND INDICATIONS OF NORMAL S YSTEM OPERATION, MANUAL LOGGING OF SUMP LEVEL AND THE CORRESPONDING LEAKRATE CAL CULATION WAS SECURED. AT 0710, THE FOLLOWING MORNING, THE SYSTEM ENGINEER FOR TH E COMPUTER WAS INFORMED OF THE PROBLEMS FROM THE PREVIOUS DAY AND BEGAN ADDITION AL INVESTIGATION. THIS INVESTIGATION REVEALED THAT THE PROGRAM THAT PERFORMS THE SUMP LEAKRATE CALCULATION, HAD NOT RESTARTED AS REQUIRED. THIS PREVENTED THE PR OGRAM FROM OPERATING PROPERLY. THE SHIFT SUPERVISOR WAS IMMEDIATELY INFORMED OF THIS CONDITION. THE PROGRAM WAS DECLARED INOPERABLE AND MANUAL LOGGING WAS COMMENCED. THIS EVENT HAD TWO CAUSES. THE FIRST WAS A DATA TRANSMISSION ERROR THAT WAS SENSED BY THE COMPUTER, AS A DISK FAILURE ON THE "A" CENTRAL PROCESSING UNIT. THE SECOND WAS A PROCEDURAL INADEQUACY THAT ALLOWED THE PROGRAM TO BE RESTORED TO SERVICE WHILE IT WAS NOT ACTUALLY FUNCTIONAL. THE PROBLEM WITH THE PROGRAM WAS CORRECTED BY THE SYSTEM ENGINEER AND IT WAS DECLARED OPERABLE AT 1115.

[99] SHEARON HARRIS 1 DOCKET 50-400 LER 92-003
DISCREPANCY IN REQUIRED HOT LEG RECIRCULATION SWITCH-OVER TIME CREATED POSSIBLE CONDITION THAT COULD HAVE PREVENTED RESIDUAL HEAT REMOVAL.
EVENT DATE: 012892 REPORT DATE: 022692 NSSS: WE TYPE: PWR

(NSIC 224096) ON JANUARY 28, 1992, A DISCREPANCY WAS DISCOVERED INVOLVING THE SAFETY INJECTION SWITCH-OVER TIME FROM COLD LEG RECIRCULATION TO HOT LEG RECIRCULATION DURING THE LONG-TERM COOLING PHASE OF A LOSS OF COOLANT ACCIDENT. THE CURRENT WESTINGHOUSE DESIGN ANALYSIS REQUIRES SWITCH-OVER TO HOT LEG RECIRCULATION AFTER 18 HOURS TO REDUCE BORON PRECIPITATION ACCUMULATION IN THE UPPER REACTOR CORE REGION. THE VALUE SPECIFIED IN THE FSAR AND EMERGENCY OPERATING PROCEDURES WAS 24 HOURS. THIS ERROR OCCURRED DURING EARLY FSAR AMENDMENTS PROVIDED BY THE PLANTS ARCHITECT ENGINEER AND NSSS SUPPLIER DURING THE

LICENSING PROCESS. IMMEDIATE CORRECTIVE ACTIONS WERE TO ISSUE A NIGHT ORDER THAT DIRECTS RECIRCULATION SWITCH-OVER AT THE 18 HOUR POINT. THE FSAR AND APPLICABLE EMERGENCY OPERATING PROCEDURES HAVE BEEN REVISED TO REFLECT THE CORRECT VALUE. THIS EVENT IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(V)(B) AS A CONDITION THAT COULD HAVE PREVENTED FULFILLMENT OF THE SAFETY FUNCTION OF A SYSTEM NEEDED TO REMOVE RESIDUAL HEAT.

[100] SOUTH TEXAS 1 DOCKET 50-498 LER 92-002
CONTAINMENT INTEGRITY TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 101891 REPORT DATE: 022092 NSSS: WE TYPE: PWR

(NSIC 224104) ON JANUARY 24, 1992, UNIT 1 WAS IN MODE 1 AT 100% WHEN IT WAS DISCOVERED THAT CONTAINMENT INTEGRITY REQUIREMENTS WERE VIOLATED BEGINNING ON OCTOBER 18, 1991, AND LASTING APPROXIMATELY 47 HOURS. REPAIRS WERE MADE TO A LEAKING HANDHOLE COVER ON THE SECONDARY SIDE OF STEAM GENERATOR 1C, WHILE THE UNIT WAS IN MODE 4, IN VIOLATION OF THE CONTAINMENT INTEGRITY TECHNICAL SPECIFICATION. THIS EVENT WAS CAUSED BY A MISINTERPRETATION OF THE REQUIREMENTS OF THE CONTAINMENT INTEGRITY TECHNICAL SPECIFICATIONS. CORRECTIVE ACTIONS INCLUDE DISSEMINATION OF INFORMATION REGARDING THIS EVENT TO PLANT MANAGEMENT AND APPROPRIATE OPERATIONS, LICENSING, AND SCHEDULING PERSONNEL. THIS EVENT WILL ALSO BE REVIEWED WITH APPROPRIATE PLANT PERSONNEL DURING LICENSED OPERATOR REQUALIFICATION TRAINING AND THROUGH A MANAGEMENT AND TECHNICAL STAFF TRAINING BULLETIN. ADDITIONALLY, MAINTENANCE WILL ADD GUIDANCE TO APPROPRIATE PROCEDURES, THAT CONTAINMENT INTEGRITY IS REQUIRED IN MODES 1 THROUGH 4 AND THAT OPENING SECONDARY STEAM GENERATOR COVERS BREACHES CONTAINMENT INTEGRITY.

[101] SOUTH TEXAS 2 DOCKET 50-499 LER 92-001
REACTOR TRIP DUE TO A DROPPED CONTROL ROD.
EVENT DATE: 012292 REPORT DATE: 022092 NSSS: WE TYPE: PWR

(NSIC 224105) ON JANUARY 22, 1992, UNIT 2 WAS IN MODE 1 AT 100% POWER. AT 0909 HOURS, UNIT 2 EXPERIENCED A REACTOR TRIP DUE TO POWER RANGE HIGH NEUTRON FLUX NEGATIVE RATE. THE PLANT WAS BROUGHT TO A STABLE CONDITION IN MODE 3 WITH NO UNEXPECTED POST-TRIP TRANSIENTS. THE CAUSE OF THE POWER RANGE HIGH NEUTRON FLUX NEGATIVE RATE TRIP WAS DROPPING OF CONTROL ROD H-6 INTO THE REACTOR CORE. THE CONTROL ROD DROPPED WHEN THE BLOCKING DIODE AND ITS ASSOCIATED STATIONARY GRIPPER COIL'S POWER CIRCUIT FAILED OPEN, RESULTING IN AN INTERRUPTION OF CURRENT TO THE STATIONARY GRIPPER COIL. THE CAUSE OF THE DIODE FAILURE REMAINS UNKNOWN. THE FAULTY DIODE WAS REPLACED, ALONG WITH ALL OTHER BLOCKING DIODES SHARING THE FAULTY DIODE'S MANUFACTURER'S DATE CODE. HL&P HAS SENT THE FAULTY DIODE AND THE OTHER SELECTED DIODES TO AN INDEPENDENT LABORATORY FOR ANALYSIS. HL&P WILL EVALUATE THE RESULTS OF THE ANALYSIS AND INITIATE FURTHER CORRECTIVE ACTIONS AS NEEDED. ADDITIONALLY HL&P, IN COOPERATION WITH WESTINGHOUSE, WILL PERFORM TESTING TO DETERMINE IF THE BLOCKING DIODES CAN BE ELIMINATED FROM THE PRESENT ROD-CONTROL SYSTEM DESIGN.

[102] SOUTH TEXAS 2 DOCKET 50-499 LER 92-002
SAFETY ANALYSIS DEFICIENCY DUE TO VERITRAK TRANSMITTER UNCERTAINTIES.
EVENT DATE: 012292 REPORT DATE: 022092 NSSS: WE TYPE: PWR

(NSIC 224106) ON JANUARY 22, 1992, IT WAS DETERMINED THAT STP UNIT 2 HAD BEEN OPERATED IN A CONFIGURATION WHICH RESULTED IN AN OVER TEMPERATURE DELTA TEMPERATURE (OTDT) TRIP SETPOINT WHICH WAS NOT CONSERVATIVE RELATIVE TO THE UFSAR SAFETY ANALYSIS. FOR A PERIOD OF APPROXIMATELY ONE MONTH BEGINNING ON SEPTEMBER 19, 1990, UNIT 2 WAS OPERATED WITH A FAILED THOT RESISTANCE TEMPERATURE DETECTOR (RTD) WHICH WAS BYPASSED UNTIL THE UNIT ENTERED A REFUELING OUTAGE. ALTHOUGH WITHIN THE LIMITS OF THE TECHNICAL SPECIFICATIONS, OPERATION WITH THE FAILED RTD COINCIDENT WITH THE NONCONSERVATIVE OTDT SETPOINT, WHICH SHOULD HAVE INCORPORATED VERITRAK TRANSMITTER UNCERTAINTIES, REPRESENTED A REPORTABLE CONDITION PURSUANT TO 10CFR50.73 FOR OPERATION IN AN UNANALYZED CONDITION. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR THROUGH A LACK OF ATTENTION TO DETAIL IN THE REVIEW AND RESOLUTION OF NSSS VENDOR RECOMMENDATIONS. ADMINISTRATIVE COMPENSATORY ACTIONS ALLOW STP UNITS 1 AND 2 TO CONTINUE NORMAL OPERATION WITHIN THE PRESENTLY DEFINED

SAFETY LIMITS UNTIL THE PLANT SAFETY ANALYSIS IS REVISED AND ANY NECESSARY TECHNICAL SPECIFICATION CHANGES ARE APPROVED.

[103] ST. LUCIE 1 DOCKET 50-335 LER 92-001
FUEL HANDLING BUILDING VENTILATION RADIATION MONITOR OUT OF SERVICE RESULTS IN A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS DUE TO A PERSONNEL ERROR.
EVENT DATE: 011692 REPORT DATE: 021892 NSSS: CE TYPE: PWR

(NSIC 22 084) ON 1/17/92 AT 0730, A PLANT CHEMISTRY SUPERVISOR NOTED THAT THE UNIT 1 FUEL HANDLING BUILDING (FHB) VENTILATION STACK RADIATION MONITOR WAS NOT IN SERVICE DURING A ROUTINE CHECK OF THAT SYSTEMS OPERABILITY. THIS FHB MONITOR IS REQUIRED TO BE IN SERVICE BY PLANT TECHNICAL SPECIFICATIONS, OR ELSE TO HAVE IN PLACE ALTERNATE MEANS OF SAMPLING AND MONITORING THE FHB VENTILATION EFFLUENT. A REVIEW OF THIS EVENT INDICATED THAT THE RADIATION MONITOR WAS PROBABLY PLACED OUT OF SERVICE DURING ROUTINE GRAB SAMPLING OF THE FHB EFFLUENT AT 0820 ON 16 JANUARY. BY PROCEDURE, THE RADIATION MONITORS SAMPLE PUMP IS SECURED DURING GRAB SAMPLING AND SHOULD BE RESTARTED FOLLOWING SAMPLING COMPLETION. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO AN OVERSIGHT BY CHEMISTRY PERSONNEL FOR NOT RESTARTING THE FHB STACK RADIATION MONITOR SAMPLE PUMP. A CONTRIBUTING FACTOR TO THIS EVENT IS THAT THE SAMPLE PUMPS LOW FLOW ALARM WAS IMPROPERLY OVERRIDDEN DURING THE GRAB SAMPLE SURVEILLANCE. ANOTHER CONTRIBUTING FACTOR IS THAT OPERATIONS PERSONNEL WHO FOUND THE SAMPLE PUMP SECURED DURING THE MIDNIGHT SHIFT EQUIPMENT CHECKS ON JANUARY 17TH DID NOT CONTACT CHEMISTRY PERSONNEL AND IMPROPERLY RESTARTED THE MONITOR. DURING THE 23 HOUR PERIOD THAT THE MONITOR WAS OUT OF SERVICE, THERE WERE NO WORK ACTIVITIES TAKING PLACE IN THE FHB, AND SUBSEQUENTLY NO UN

[104] SUMMER 1 DOCKET 50-395 LER 92-002
IMPROPER OPERATION OF CONTAINMENT ISOLATION VALVE DUE TO PERSONNEL ERROR.
EVENT DATE: 012892 REPORT DATE: 022892 NSSS: WE TYPE: PWR

(NSIC 224125) ON JANUARY 28, 1992, AT APPROXIMATELY 1930 HOURS, OPERATIONS PERSONNEL IDENTIFIED A NONCOMPLIANCE WITH THE ACTION REQUIREMENTS OF TECHNICAL SPECIFICATION 3.6.4, "CONTAINMENT ISOLATION VALVES." A STATUS REVIEW OF INOPERABLE EQUIPMENT LOGS FOUND THAT ON THREE (3) SEPARATE OCCASIONS CONTAINMENT ISOLATION VALVE XVA-9312B, "SAMPLE LINE RETURN FROM RADIATION MONITOR," HAD BEEN TEMPORARILY OPENED TO FACILITATE THE PERFORMANCE OF A REQUIRED SURVEILLANCE ACTIVITY. THIS VALVE HAD BEEN DECLARED INOPERABLE ON JANUARY 27, 1992, AT 0400 HOURS, WHEN IT DID NOT APPEAR TO CLOSE DURING A SLAVE RELAY SURVEILLANCE TEST THAT SIMULATED A CONTAINMENT PHASE A ISOLATION. VALVE XVA-9312B IS LOCATED IN THE SAMPLE LINE RETURN FOR RADIATION MONITOR RM-A2, WHICH PROVIDES MEASUREMENT OF THE AIRBORNE ACTIVITY (PARTICULATE, IODINE, GAS) INSIDE THE REACTOR BUILDING. THIS EVENT RESULTED FROM PERSONNEL ERROR. DURING THE COORDINATION OF A REACTOR BUILDING ATMOSPHERE SURVEILLANCE UNDER TECHNICAL SPECIFICATION 3.4.6.2, "OPERATIONAL LEAKAGE," OPERATIONS PERSONNEL FAILED TO RECOGNIZE THAT THE TEMPORARY RESTORATION OF THE INOPERABLE VALVE WOULD BE IN CONFLICT WITH THE INTENT OF TECHNICAL SPECIFICATIONS AND PLANT PROCEDURES. OPERATIONS AND HEALTH PHYSICS PERSONNEL DISCONTINUED THE USE OF THE SAMPLE LINE RETURN FOR SUBSEQUENT SURVEILLANCES OF THE REACTOR BUILDING ATMOSPHERE.

[105] SURRY 1 DOCKET 50-280 LER 91-014 REV 01
UPDATE ON MAIN STEAM FLOW SETPOINTS IN EXCESS OF TECHNICAL SPECIFICATIONS DUE TO FLOW TRANSMITTER SCALING INACCURACIES RESULTING FROM INCORRECT SCALING METHODOLOGY.
EVENT DATE: 072491 REPORT DATE: 021192 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 224069) ON JULY 24, 1991, WITH UNIT 1 AND UNIT 2 OPERATING AT 100% POWER, IT WAS DETERMINED AFTER ANALYZING RECENT SPECIAL TEST DATA THAT THE SCALING FOR THE UNIT 2 MAIN STEAM (MS) FLOW TRANSMITTERS WAS INCORRECT. THESE FLOW TRANSMITTERS PROVIDE INPUT TO CERTAIN REACTOR PROTECTION SYSTEM (RPS) AND ENGINEERED SAFETY FEATURES (ESF) ACTUATION SIGNALS. IT WAS DETERMINED THAT THE SCALING RESULTED IN UNIT 2 SET POINTS IN EXCESS OF THE MAXIMUM ALLOWED VALUE

110% (AT FULL LOAD) OF FULL STEAM FLOW SPECIFIED IN TECHNICAL SPECIFICATION (TS) 3.7.D. ON JULY 22, 1991, VOLTAGE BIASES WERE PLACED ON THE UNCOMPENSATED UNIT 2 STEAM FLOW SIGNALS TO REDUCE THE HI STEAM FLOW ESF FUNCTION ACTUATION TO A VALUE BELOW 110%. THIS EVENT WAS CAUSED BY THE USE OF INCORRECT SCALING METHODOLOGY, EMPLOYED IN 1977, TO RESCALE THE MS FLOW TRANSMITTERS TO CORRECT AN OBSERVED DIFFERENCE IN THE MS AND FEEDWATER FLOW INDICATIONS. SINCE THE SAME INCORRECT SCALING INFORMATION WAS USED IN 1977 FOR UNIT 1, SIMILAR VOLTAGE BIASES WERE INTRODUCED ON UNIT 1 ON JULY 24, 1991. FOLLOWING AN EVALUATION OF SPECIAL TEST DATA, THE UNIT 2 STEAM FLOW AND FEEDWATER FLOW TRANSMITTERS WERE RESPANNED AND THE STEAM FLOW CIRCUITRY RESCALED. INDEPENDENT TESTING IS BEING PERFORMED TO ENABLE THESE MEASURES TO BE IMPLEMENTED FOR UNIT 1. THE EVENT IS BEING REPORTED, PURSUANT TO

[106] SURRY 2 DOCKET 50-281 LER 92-001
LESS THAN ONE OPERABLE CHARGING PUMP DUE TO MECHANICAL FAILURE AND PERSONNEL ERROR.
EVENT DATE: 013092 REPORT DATE: 022692 NSSS: WE TYPE: PWR
VENDOR: BYRON JACKSON PUMPS, INC.
RAYMOND CONTROLS

(NSIC 224113) ON JANUARY 30, 1992 AT 1345 HOURS, WITH UNIT 1 AT 98% POWER AND UNIT 2 AT 100% POWER, UNIT 2 CHARGING PUMP 2-CH-P-1A WAS DECLARED INOPERABLE DUE TO AN OIL LEAK. A 24 HOUR TO HOT SHUTDOWN WAS INITIATED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.3.B.2 DUE TO LESS THAN TWO CHARGING PUMPS OPERABLE. THE "C" CHARGING PUMP WAS BEING MAINTAINED IN THE "PULL TO LOCK" POSITION IN ACCORDANCE WITH OPERATING PROCEDURE-S. AT 1515, THE STANDBY CHARGING PUMP "B" WAS STARTED. WHILE PERFORMING POST START CHECKS, IT WAS NOTED THAT VENTILATION DAMPER 2-VSMOD-201B FOR THE "B" CHARGING PUMP CUBICLE HAD FAILED TO OPEN AS DESIGNED. PROPER DAMPER ALIGNMENT IS REQUIRED FOR LONG TERM PUMP OPERABILITY. THEREFORE, AT 1520 THE "B" CHARGING PUMP WAS DECLARED INOPERABLE. DUE TO LESS THAN ONE OPERABLE CHARGING PUMP, THIS OCCURRENCE WAS CONSIDERED TO BE A CONDITION NOT ALLOWED BY THE TECHNICAL SPECIFICATIONS AND A SIX HOUR CLOCK WAS ENTERED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.0.1. SINCE AT LEAST ONE CHARGING PUMP WAS CAPABLE OF PERFORMING ITS AUTOMATIC FUNCTION, IT WAS CONCLUDED THAT NO ACTUAL OR POTENTIAL CONSEQUENCES TO PUBLIC HEALTH AND SAFETY WERE CREATED BY THE EVENT. THIS REPORT IS REQUIRED BY 10CFR50.73(A)(2)(I)(B).

[107] THREE MILE ISLAND 1 DOCKET 50-289 LER 91-004 REV 01
UPDATE ON MOVEMENT OF IRRADIATED FUEL ASSEMBLY WITHOUT CONTAINMENT ISOLATION DUE TO PERSONNEL ERRORS AND PROCEDURAL INADEQUACY.
EVENT DATE: 100891 REPORT DATE: 022692 NSSS: BW TYPE: PWR

(NSIC 224116) 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 1303-11.4 REQUIRES FUEL MOVEMENT; HOWEVER, IT ALSO REQUIRES, BY REFERENCE, THAT NO MOVEMENT OF FUEL TAKE PLACE UNTIL THE PREREQUISITES OF 1505-1 WERE COMPLETE. IN THIS EVENT, THE BRIDGE CREW DID NOT ADEQUATELY PREPARE FOR THE EVOLUTION ABOUT TO TAKE PLACE AND MOVED FUEL TO TEST THE HOIST INTERLOCKS WITHOUT HAVING VERIFIED CONTAINMENT ISOLATION. ALL OF THE 1505-1 PREREQUISITES FOR CONTAINMENT ISOLATION 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 PERSONNEL ERROR, DUE TO A LACK OF UNDERSTANDING THAT THE TEST INVOLVED FUEL MOVEMENT; AND, PROCEDURAL INADEQUACY, BECAUSE 1303-11.4 DID NOT CLEARLY CAUTION THE OPERATORS THAT CONTAINMENT ISOLATION WAS REQUIRED PRIOR TO TEST OF THE INTERLOCKS. THE CAUSES OF THE EVENT WERE REVIEWED WITH ALL FUEL HANDLING PERSONNEL PRIOR TO THE COMMENCEMENT OF THE FUEL SHUFFLE.

[108] THREE MILE ISLAND 1 DOCKET 50-289 LER 92-001
INADVERTENT EMERGENCY FEEDWATER INITIATION DURING PLANNED MAINTENANCE DUE TO INSTALLATION ERROR.
EVENT DATE: 012292 REPORT DATE: 022192 NSSS: BW TYPE: PWR

(NSIC 224075) INADVERTENT EMERGENCY FEEDWATER (EFW) ACTUATION DURING PLANNED MAINTENANCE DUE TO INSTALLATION ERROR TMI-1 WAS OPERATING AT 100% POWER. DURING A PLANNED MAINTENANCE ACTIVITY, EMERGENCY FEEDWATER (EFW) WAS INADVERTENTLY INITIATED FOR A SHORT TIME. THIS EVENT IS REPORTABLE IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(IV). THE ACTUATION OCCURRED DUE TO A CONSTRUCTION WIRING ERROR RESULTING FROM MODIFICATIONS DURING A PREVIOUS OUTAGE. THIS EVENT WAS CAUSED BY LACK OF DRAWING CLARITY, AND INADEQUATE SUPERVISORY OVERSIGHT. ALL EQUIPMENT FUNCTIONED AS EXPECTED CONSIDERING THE WIRING ERROR . THERE WAS NO ADVERSE IMPACT ON NUCLEAR SAFETY. WIRING ERRORS WHICH COULD AFFECT SYSTEM OPERATION HAVE NOW BEEN CORRECTED. A DETAILED WALKDOWN WILL BE PERFORMED AND THE APPLICABLE DRAWINGS WILL BE REVISED TO REFLECT THE AS BUILT CONFIGURATION. TMI-1 HAS IN PLACE SUFFICIENT PROCEDURAL CONTROLS TO PRECLUDE OR IDENTIFY WIRING ERRORS AND AS SUCH THIS EVENT IS CONSIDERED TO BE AN ISOLATED CASE. THE POTENTIAL FOR SINGLE FAILURES TO CAUSE AN INADVERTENT EFW ACTUATION HAD BEEN PREVIOUSLY EVALUATED AND WITH THE NRC'S CONCURRENCE THIS WAS DETERMINED TO BE ACCEPTABLE. NO ADDITIONAL ACTION IS CONSIDERED TO BE NECESSARY.

[109] TROJAN DOCKET 50-344 LER 92-001
FAILURE TO RECOGNIZE A FIRE SUPPRESSION SYSTEM DELUGE NOZZLE LED TO ERECTION OF SCAFFOLD WHICH OBSTRUCTED THE NOZZLE SPRAY PATTERN.
EVENT DATE: 011692 REPORT DATE: 021792 NSSS: WE TYPE: PWR

(NSIC 224088) ON JANUARY 16, 1992, THE TROJAN NUCLEAR PLANT WAS IN MODE 5 (COLD SHUTDOWN) DURING AN EXTENDED OUTAGE. THE "A" EMERGENCY DIESEL GENERATOR (EDG) WAS OUT OF SERVICE. BOTH "A" AND "B" SERVICE WATER SYSTEMS WERE OPERATING. AT APPROXIMATELY 0100, SCAFFOLD WAS ERECTED BETWEEN FIRE DELUGE SYSTEM NOZZLES AND THE "B" SERVICE WATER BOOSTER PUMPS. THE SCAFFOLD OBSTRUCTED THE NOZZLE SPRAY PATTERN. CONTRARY TO TECHNICAL SPECIFICATION 3.7.8.2, "SPRAY, SPRINKLER, AND/OR DELUGE SYSTEMS", A CONTINUOUS FIRE WATCH WITH BACKUP FIRE SUPPRESSION EQUIPMENT WAS NOT ESTABLISHED WITHIN ONE HOUR OF OBSTRUCTING THE SPRAY PATTERN. TECHNICAL SPECIFICATION 3.7.8.2 REQUIRES THIS DELUGE SYSTEM TO BE OPERABLE WHENEVER THE SERVICE WATER BOOSTER PUMPS ARE REQUIRED TO BE OPERABLE. THE "B" TRAIN SERVICE WATER BOOSTER PUMPS WERE REQUIRED TO BE OPERABLE TO SUPPORT OPERATION OF THE "B" EDG. THE CAUSE WAS FAILURE TO RECOGNIZE THAT THESE DELUGE SYSTEM NOZZLES WERE PART OF A FIRE PROTECTION SYSTEM. PIPING ADJACENT TO THESE NOZZLES WAS NOT PAINTED RED. A CONTINUOUS FIRE WATCH WAS ESTABLISHED AND BACKUP FIRE SUPPRESSION EQUIPMENT WAS VERIFIED AVAILABLE WHEN THE CONDITION WAS DISCOVERED. A WALKDOWN OF FIRE PROTECTION PIPING HAS BEEN COMPLETED TO IDENTIFY PIPING THAT IS ADJACENT TO NOZZLES AND NOT PAINTED RED. PIPING IDENTIFIED IN THE WALKDOWN WILL BE PAINTED RED TO IDENTIFY IT AS PART OF A FIRE

[110] TURKEY POINT 3 DOCKET 50-250 LER 91-012
USE OF DESIGN DELTA-TEMPERATURE (DT) RATHER THAN INDICATED DT AS REQUIRED BY TECHNICAL SPECIFICATIONS FOR UNITS 3 AND 4.
EVENT DATE: 100791 REPORT DATE: 121991 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)

(NSIC 223671) ON OCTOBER 7, 1991, WITH UNIT 3 AT 50% OF RATED POWER, DURING CALIBRATION OF A NEW DIGITAL INSTRUMENTATION RACK, THE DELTA-TO FACTOR IN OVERPOWER DELTA-T AND OVERTEMPERATURE (OT) DELTA-T SETPOINT FORMULA WAS FOUND TO BE SET AT THE DESIGN VALUE (56.1F) INSTEAD OF THE INDICATED VALUE AS REQUIRED BY TECHNICAL SPECIFICATIONS. SAFETY EVALUATION WAS PERFORMED TO ASSESS THE EFFECT OF THE HIGHER DELTA-TO FACTOR. THE EVALUATION CONCLUDED THAT THE PLANT HAD BEEN OPERATING WITHIN ITS DESIGN BASES DURING ALL PAST CYCLES. INDICATED DELTA-T VALUES AT RATED THERMAL POWER WERE OBTAINED FOLLOWING CALIBRATION AND INSERTED FOR THE DELTA-T FACTOR DELTA-T AND OT DELTA-T SETPOINT FORMULA. THE EVENT WAS DETERMINED TO BE REPORTABLE, IN ACCORDANCE WITH O CFR 50.73 (A)(2)(I)(B), DURING A DETAILED REVIEW OF A SAFETY EVALUATION WHICH AS SUBMITTED TO THE PLANT NUCLEAR SAFETY COMMITTEE FOR REVIEW ON NOVEMBER 29, 1991.

[111] TURKEY POINT 4 DOCKET 50-251 LER 92-001
TECHNICAL SPECIFICATION 3.0.3 ENTRY DUE TO ROD CLUSTER CONTROL POSITION INDICATORS NOT IN AGREEMENT WITH DEMAND INDICATION.
EVENT DATE: 020292 REPORT DATE: 030292 NSSS: WE TYPE: PWR

(NSIC 224138) AT APPROXIMATELY 1540, ON FEBRUARY 2, 1992, WITH TURKEY POINT UNIT 4 IN MODE 3 AND REACTOR COOLANT TEMPERATURE AT 547 DEGREES F, WHILE PULLING RODS DURING UNIT 4 REACTOR STARTUP, RODS J-11 AND L-7 IN CONTROL BANK A AND ROD H-6 IN SHUTDOWN BANK B INDICATED GREATER THAN 12 STEPS DIFFERENCE FROM THE GROUP STEP COUNTERS. THE REACTOR WENT CRITICAL AT 1600. AFTER WAITING THE 1 HOUR SOAK PERIOD, AS ALLOWED BY TECHNICAL SPECIFICATION 3.1.3.2, THE RODS STILL INDICATED MISALIGNMENT. TECHNICAL SPECIFICATION 3.0.3 WAS ENTERED AT 1640 DUE TO MORE THAN ONE ROD POSITION INDICATION (RPI) BEING OUT OF SERVICE ON A CONTROL BANK (CONTROL BANK A) AS SPECIFIED BY TECHNICAL SPECIFICATION 3.1.3.2. INCORE FLUX MAPS DETERMINED ALL RODS TO BE ALIGNED PROPERLY. THE RPIS WERE ADJUSTED TO INDICATE THE ROD POSITIONS PROPERLY AND TECHNICAL SPECIFICATION 3.0.3 FOR RODS J-11 AND L-7 WAS EXITED AT 1725. AT 1736, THE RPI FOR ROD H-6 WAS ADJUSTED TO INDICATE PROPERLY IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.1.3.2. TECHNICAL SPECIFICATION 3.1.3.2 ACTION STATEMENT WAS EXITED AT 1736.

[112] VOGTLE 2 DOCKET 50-425 LER 91-012 REV 01
UPDATE ON RCP THERMAL BARRIER ISOLATION VALVES DECLARED INOPERABLE DUE TO TORQUE SWITCH SETTINGS.
EVENT DATE: 121191 REPORT DATE: 021392 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: VOGTLE 1 (PWR)
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 224046) ON 12-11-91, DURING A REVIEW OF MINIMUM REQUIRED THRUST VALUES AGAINST FIELD MEASURED THRUST VALUES FOR SAFETY-RELATED MOTOR-OPERATED VALVES, OFFSITE ENGINEERING PERSONNEL DETERMINED THAT THE AS-LEFT TORQUE SWITCH SETTINGS FOR VALVES 2HV-19051, 2HV-19053, 2HV-19055, AND 2HV-19057 WERE INADEQUATE TO ENSURE THE DESIGN FUNCTION OF THESE VALVES. THESE VALVES ARE DESIGNED TO PROVIDE A REDUNDANT ISOLATION FUNCTION TO VALVE 2HV-2041 FOR ISOLATING A POSTULATED REACTOR COOLANT PUMP THERMAL BARRIER TUBE RUPTURE. THE SITE WAS NOTIFIED AND TECHNICAL SPECIFICATION (TS) ACTION REQUIREMENTS PERTAINING TO THE OPERABILITY OF THE THERMAL BARRIER ISOLATION FUNCTION WERE ENTERED WHILE CORRECTIVE ACTIONS WERE TAKEN. THIS INCLUDED INCREASING THE TORQUE SWITCH SETTINGS FOR 2HV-19051, 2HV-19053, AND 2HV-19055 AND INSTALLING A JUMPER TO TEMPORARILY BYPASS THE CLOSE TORQUE SWITCH FOR 2HV-19057. ON 1-17-92, THE AS-LEFT TORQUE SWITCH SETTINGS FOR VALVES 2HV-2041 AND 1HV-2041 WERE ALSO DETERMINED TO BE INADEQUATE. DUE TO THIS DISCOVERY, THE NRC WAS NOTIFIED PURSUANT TO 10 CFR 50.72(B)(2)(III), AND THE TS ACTION REQUIREMENTS WERE AGAIN ENTERED WHILE ACTION WAS TAKEN TO INCREASE THE TORQUE SWITCH SETTINGS FOR 2HV-2041 AND 1HV-2041. THE EVENT WAS CAUSED BY INADEQUATE ORIGINAL VENDOR SPECIFIED INFORMATION.

[113] WOLF CREEK 1 DOCKET 50-482 LER 92-001
FAILURE TO FOLLOW PROCEDURES COULD HAVE CAUSED BOTH INTERMEDIATE RANGE CHANNELS TO BE INOPERABLE DURING PHYSICS TESTING.
EVENT DATE: 011292 REPORT DATE: 021192 NSSS: WE TYPE: PWR

(NSIC 224049) ON JANUARY 13, 1992, AT APPROXIMATELY 0735 CST, WHILE PERFORMING LOW POWER PHYSICS TESTING, CONTROL ROOM OPERATORS WERE NOTIFIED THAT THE ANALOG CHANNEL OPERATIONAL TESTS OF THE INTERMEDIATE RANGE CHANNELS HAD NOT BEEN PERFORMED PROPERLY ON JANUARY 11, 1992 RESULTING IN BOTH CHANNELS BEING DECLARED INOPERABLE AND ENTRY INTO TECHNICAL SPECIFICATION 3.0.3. SUBSEQUENT EVALUATION OF THE SETPOINT VALUES USED ON JANUARY 11, 1992, HAS CONCLUDED THAT THE VALUES WERE WITHIN THE TECHNICAL SPECIFICATION ALLOWABLE VALUES AND THEREFORE, THE INTERMEDIATE RANGE CHANNELS WERE OPERABLE. SEVERAL FACTORS CONTRIBUTED TO THIS EVENT'S OCCURRENCE INCLUDING FAILURE TO PROPERLY REFERENCE TEMPORARY PROCEDURE CHANGES AT THE AFFECTED PROCEDURE STEPS PRIOR TO PROCEDURE USAGE AND THE ASSUMPTION BY THE INSTRUMENTATION AND CONTROLS (I&C) TEST PERFORMERS THAT THE TEMPORARY PROCEDURE CHANGES HAD BEEN PROPERLY INCORPORATED. TO PREVENT RECURRENCE, AN INDEPENDENT VERIFICATION OF THE USE OF THE PROPER SETPOINTS PRIOR TO PHYSICS TESTING WILL BE ADDED TO THE REACTOR ENGINEERING PHYSICS TESTING

PROCEDURE. ADDITIONALLY, THE DETAILS OF THIS EVENT ARE BEING ISSUED AS I&C REQUIRED READING TO EMPHASIZE THE IMPORTANCE OF ENSURING THAT ALL ASPECTS OF PROPER PROCEDURE PERFORMANCE HAVE BEEN COMPLETED.

[114] WPPSS 2 DOCKET 50-397 LER 91-036 REV 01
UPDATE ON MISSED ASME SECTION XI SURVEILLANCE.
EVENT DATE: 122691 REPORT DATE: 021292 NSSS: GE TYPE: BWR

(NSIC 224041) ON DECEMBER 26, 1991, A MISSED ASME SECTION XI INSERVICE TEST FOR FUEL POOL COOLING (FPC) RELIEF VALVE, PC-RV-117B, WAS DETERMINED TO VIOLATE THE REQUIREMENTS OF TECHNICAL SPECIFICATION SECTIONS 4.0.2 AND 4.0.5. THIS OVERSIGHT WAS IDENTIFIED BY THE IN-SERVICE TESTING (IST) PROGRAM LEADER ON DECEMBER 20, 1991. THE INITIAL CORRECTIVE ACTION WAS TO DECLARE FPC-RV-117B INOPERABLE. THERE ARE NO TECHNICAL SPECIFICATION OPERABILITY REQUIREMENTS FOR THE FUEL POOL COOLING SYSTEM, THEREFORE NO ACTION STATEMENTS WERE REQUIRED TO BE ENTERED. THE ROOT CAUSES FOR THIS EVENT WERE IDENTIFIED TO BE MANAGERIAL METHODS DID NOT ENSURE SUFFICIENT INTERDEPARTMENTAL COMMUNICATION, AND INADEQUATE DOCUMENTATIONAL PROVISIONS IN THE PROCEDURE RESULTED IN ADEQUATE VERIFICATION THAT PROGRAM REQUIREMENTS WERE SATISFIED. THE CORRECTIVE ACTIONS INCLUDE PERFORMING THE REQUIRED ASME SURVEILLANCE TESTING ON FPC-RV-117B, AND REVISING THE AFFECTED PLANT SURVEILLANCE PROCEDURE TO INCLUDE A DESCRIPTION OF THE PROGRAM REQUIREMENTS, ASSIGNMENT OF RESPONSIBILITIES, AND DOCUMENTATION VERIFICATION PROVISIONS. THERE IS NO SAFETY SIGNIFICANCE ASSOCIATED WITH THIS EVENT. THERE WAS NO CONDITION THAT CHALLENGED THE OPERATION OF THIS RELIEF VALVE. TESTING ON JANUARY 27, 1992 VERIFIED THAT THE VALVE WAS CAPABLE OF PERFORMING ITS SAFETY FUNCTION.

[115] WPPSS 2 DOCKET 50-397 LER 92-002
MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM INOPERABLE DUE TO INADEQUATE TESTING AND CALCULATIONS.
EVENT DATE: 011392 REPORT DATE: 021292 NSSS: GE TYPE: BWR

(NSIC 224094) ON JANUARY 13, 1992 IT WAS DETERMINED THAT THE MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL (MSLC) SYSTEM FAN TESTING REQUIRED BY TECHNICAL SPECIFICATION 4.6.1.4.C.2 WAS NOT CORRECTLY PERFORMED IN THAT CFM WAS BEING MEASURED AND RECORDED INSTEAD OF THE REQUIRED SCFM. ADDITIONALLY, ON JANUARY 27, 1992 A REPORTABILITY EVALUATION WAS COMPLETED AND IT WAS CONCLUDED THAT THE SETPOINT FOR PRESSURE SWITCH MSLC-PS-25, THE SWITCH WHICH PROVIDES AUTOMATIC TRANSFER OF THE OUTBOARD MSLC TRAIN FROM THE DEPRESSURIZATION TO THE BLEED MODE OF OPERATION, DID NOT ENSURE PROPER SYSTEM OPERATION. A TECHNICAL SPECIFICATION WAIVER WAS OBTAINED TO ALLOW VERIFICATION OF MSLC FAN OPERABILITY AT CFM INSTEAD OF SCFM. A TECHNICAL SPECIFICATION AMENDMENT REQUEST WAS SUBMITTED TO CHANGE THE FAN TESTING REQUIREMENT FROM SCFM TO CFM. PRESSURE SWITCH MSLC-PS-25 WAS RECALIBRATED USING THE INFORMATION FROM THE NEW SETPOINT CALCULATION. FINALLY A QUALITY ACTION TEAM HAS BEEN AUTHORIZED TO ADDRESS POTENTIAL IMPROVEMENTS IN TECHNICAL SPECIFICATION COMPLIANCE.

[116] WPPSS 2 DOCKET 50-397 LER 92-003
CONTAINMENT ATMOSPHERE CONTROL SYSTEM.
EVENT DATE: 012392 REPORT DATE: 022192 NSSS: GE TYPE: BWR

(NSIC 224126) ON JANUARY 23, 1992 CAC CIRCULATING FAN (CAC-FN-LB) FAILED TO START DURING THE PERFORMANCE OF A ROUTINE SURVEILLANCE TEST. PLANT OPERATORS DISCOVERED THE FAN DID NOT OPERATE BECAUSE OF TRIPPED OVERLOADS. IT WAS THEN DETERMINED THAT TRAIN B OF THE CONTAINMENT ATMOSPHERE CONTROL (CAC) SYSTEM HAD BEEN INOPERABLE LONGER THAN THE 30 DAYS ALLOWED BY THE TECHNICAL SPECIFICATIONS. DURING THE LER INVESTIGATION IT WAS DETERMINED THAT BOTH CAC DIVISIONS WERE OUT OF SERVICE FOR A 5-1/2 HOUR PERIOD DUE TO TESTING OF TRAIN A. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO RESET THE THERMAL OVERLOADS AND THE SURVEILLANCE TEST WAS PERFORMED. THE ROOT CAUSE OF THIS EVENT WAS INSUFFICIENT COMPONENT MONITORING. A SECOND ROOT CAUSE WAS THE FACT THE ELEMENTARY DRAWING CONTAINED TECHNICAL INACCURACIES. FURTHER CORRECTIVE ACTION WILL BE TAKEN TO CORRECT THE DESIGN AND THE DESIGN DRAWINGS ASSOCIATED WITH THE CAC STEM. IN ADDITION A REVIEW WILL BE PERFORMED OF

SIMILAR EQUIPMENT/SYSTEMS TO PREVENT REOCCURRENCE. THIS EVENT HAS MINOR SAFETY SIGNIFICANCE.

[117] YANKEE ROWE DOCKET 50-029 LER 91-006
MISSED SURVEILLANCE DUE TO PERSONNEL ERROR AND INSUFFICIENT INFORMATION.
EVENT DATE: 111391 REPORT DATE: 121391 NSSS: WE TYPE: PWR

(NSIC 223592) ON NOVEMBER 13, 1991, WITH THE PLANT IN MODE 5 AT 0% POWER, A SUPERVISORY REVIEW OF COMPLETED TECHNICAL SPECIFICATION (TS) REQUIRED SURVEILLANCES DETERMINED THE 7 DAY VALVE OPERABILITY TEST FOR THE REACTIVITY CONTROL SYSTEM FLOW PATH (SURVEILLANCE REQUIREMENT 4.1.2.2.A.1) WAS NOT PERFORMED ON NOVEMBER 4, 1991. IT WAS SATISFACTORILY PERFORMED ON NOVEMBER 1?. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO PERSONNEL ERROR. A CONTRIBUTING FACTOR WAS THE INCLUSION OF THE 7 DAY VALVE OPERABILITY SURVEILLANCE AT A STEP IN THE CHARGING PUMP OPERABILITY TEST PROCEDURE, A 31 DAY SURVEILLANCE, WITH NO INFORMATION TO DIRECT THE OPERATOR TO THE VALVE OPERABILITY SURVEILLANCE REQUIREMENT. THE SURVEILLANCE PROCEDURE WILL BE REVISED TO INCLUDE ADDITIONAL DETAIL TO ENSURE THAT PERSONNEL RESPONSIBLE FOR ITS PERFORMANCE ARE AWARE OF TS SURVEILLANCE SCHEDULES. THROUGHOUT THE FOURTEEN DAY PERIOD (OCTOBER 28 THROUGH NOVEMBER 11) WHEN THE VALVES WERE NOT CYCLED, THERE WAS FLOW THROUGH THE FLOW PATH FROM THE PURIFICATION SYSTEM. THERE WAS NO ADVERSE IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT. PREVIOUS MISSED SURVEILLANCES HAVE BEEN REPORTED IN LERS 86-11, 87-01, 8? 08. 90-0 9, AND 91-03.

[118] YANKEE ROWE DOCKET 50-029 LER 92-001
MISSED SURVEILLANCE CAUSES TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 011792 REPORT DATE: 021392 NSSS: WE TYPE: PWR

(NSIC 224023) ON 1/17/9' WHILE IN MODE 5, A SUPERVISORY REVIEW OF COMPLETED TECHNICAL SPECIFICATION (TS) REQUIRED SURVEILLANCES DETERMINED THAT THE SEVEN DAY VALVE OPERABILITY TEST FOR THE REACTIVITY CONTROL SYSTEM FLOW PATH (TS 4.1.2.2.A.1) HAD NOT BEEN PERFORMED ON 1/8/92. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO PERSONNEL ERROR IN THAT OPERATIONS PERSONNEL DID NOT RECOGNIZE THE REQUIREMENT TO CYCLE VALVE CS-MOV-540 WHILE IN THIS PLANT CONFIGURATION. OPERATIONS PERSONNEL WERE AWARE THAT PROCEDURAL SURVEILLANCES HAD NOT BEEN COMPLETED DUE TO EQUIPMENT BEING OUT OF SERVICE, BUT THEY FAILED TO RECOGNIZE THE EMBEDDED TS REQUIREMENT TO PERFORM THE VALVE OPERABILITY TEST CONTAINED IN THE PROCEDURE. THE REQUIRED SURVEILLANCE WAS SATISFACTORILY PERFORMED ON 1/18/92. OTHER CORRECTIVE ACTION INCLUDED REVISING AN OPERATIONS DEPARTMENT ADMINISTRATIVE PROCEDURE TO REQUIRE THE FOLLOWING ACTIONS WHEN SURVEILLANCES ARE SCHEDULED ON PLANT EQUIPMENT THAT IS NOT IN SERVICE: REVIEW PROCEDURE FOR EMBEDDED TS REQUIREMENTS, REVIEW TS REFERENCED IN PROCEDURE TO DETERMINE IF PARTS OF PROCEDURE NEED TO BE PERFORMED, AND REQUEST PERMISSION FROM OPERATIONS MANAGEMENT TO NOT PERFORM A SURVEILLANCE. THERE WAS NO ADVERSE EFFECT TO THE PUBLIC HEALTH OR SAFETY AS A RESULT OF THIS EVENT. SIMILAR LERS: 87-01 AND 91-06.

[119] YANKEE ROWE DOCKET 50-029 LER 92-002
FAILURE TO MAINTAIN BORIC ACID MIX TANK LINE TEMPERATURE.
EVENT DATE: 012092 REPORT DATE: 021992 NSSS: WE TYPE: PWR

(NSIC 224054) ON JANUARY 20, 1992, AT 1500 HOURS, DURING REPAIRS TO THE BORIC ACID MIX TANK (BAMT) HEAT TRACE, IT WAS DETERMINED THAT THE BAMT GRAVITY FEED LINE TEMPERATURE COULD NOT BE MAINTAINED AT 150 DEGREES FAHRENHEIT, OR GREATER, AS REQUIRED BY TECHNICAL SPECIFICATIONS. THIS DETERMINATION WAS MADE AFTER DISCOVERY THAT FOUR OF THE TWELVE THERMOCOUPLES ON THE BAMT GRAVITY FEED LINE WERE NOT CONNECTED TO THE TEMPERATURE INDICATOR. THE ROOT CAUSE OF THE BAML GRAVITY FEED LINE NOT BEING MAINTAINED AT 150 DEGREES FAHRENHEIT, OR GREATER IS THE RESULT OF TWO CHECK VALVES ALLOWING MIXING WHEN THERE IS NOT SUFFICIENT SYSTEM PRESSURE TO SEAT THE VALVE DISCS. SHORT TERM CORRECTIVE ACTION CONSISTED OF REPLACEMENT OF THE HEAT TRACE, AS WELL AS CONNECTING ALL TWELVE THERMOCOUPLES. OP-4214 WAS REVISED TO INCLUDE THE REQUIREMENT FOR MONITORING AND RECORDING ALL TWELVE THERMOCOUPLE READINGS. LONG TERM CORRECTIVE ACTION WILL CONSIST OF SUBMITTING A PROPOSED TS CHANGE PRIOR TO STARTUP FROM THE PRESENT OUTAGE. THERE

WAS NO ADVERSE IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT. RELATED EVENTS INVOLVING TEMPERATURE MONITORING IN THE BAMT GRAVITY FEED LINE HAVE OCCURRED PREVIOUSLY AT THE YANKEE NUCLEAR POWER STATION.

[120] ZION 1 DOCKET 50-295 LER 92-001
PROPER SURVEILLANCE WAS NOT PERFORMED WITH BOTH BATTERY ROOM VENTILATION FANS INOPERABLE.
EVENT DATE: 010992 REPORT DATE: 021092 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 2 (PWR)

(NSIC 224032) ON 10/28/91 DURING A TECHNICAL STAFF SYSTEM WALKDOWN, THE OA COMPUTER AND MISCELLANEOUS HVAC EXHAUST FAN (VI), OOV-007, WOULD NOT RESTART AFTER IT WAS SECURED BY THE OPERATING DEPARTMENT. A PERIODIC TEST (PT)-14, APPENDIX E, DEGRADED EQUIPMENT LOG, WAS NOT INITIATED TO TRACK THE INOPERABLE EQUIPMENT. ON 01/09/92 AT APPROXIMATELY 0800, DURING THE SHIFTLY PERFORMANCE OF THE EQUIPMENT OPERATOR CHECKLIST (PT-0, APPENDIX S1 & S2), THE EQUIPMENT OPERATOR NOTICED THAT THE BATTERY ROOM 211 (EJ) VENT FLOW METER WAS INDICATING ZERO FLOW. DUE TO THE INOPERABILITY OF BOTH THE OA AND OB COMPUTER AND MISCELLANEOUS HVAC EXHAUST FANS PER TECH SPEC GENERAL LIMITING CONDITION FOR OPERATION 3.0.3, UNIT 1 AND UNIT 2 WERE PLACED ON A 5 HOUR CLOCK TO HOT SHUTDOWN. THIS EVENT WAS CAUSED BY A PERSONNEL ERROR. THE SHIFT PERSONNEL INVOLVED IN THIS EVENT DID NOT INITIATE A PT-14 APPENDIX E ON 10/28/91 OR A PT-14 APPENDIX A, INOPERABLE EQUIPMENT SURVEILLANCE, ON 01/09/92 BECAUSE IT WAS NOT APPARENT TO THE SHIFT PERSONNEL THAT THE OA AND OB COMPUTER AND MISCELLANEOUS EXHAUST FANS AFFECTED BATTERY OPERABILITY. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL. CORRECTIVE ACTIONS INCLUDE REVISING THE PT-14 PROCESS, REVIEWING THE EVENT WITH OPERATING PERSONNEL, REVIEWING THE ADEQUACY OF THE VENTILATION SYSTEM TRAINING, AND TRAINING THE OPERATING DEPARTMENT ON THE NEW PT-14 PROCESS.

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

VENDOR CODE INDEX

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-2000
ORNL/NSIC-200
Vol. 11, No. 3

2. TITLE AND SUBTITLE

Licensee Event Report (LER) Compilation for month of March 1992

3. DATE REPORT PUBLISHED

MONTH	YEAR
May	1992

4. FIN OR GRANT NUMBER

A9135

5. AUTHOR(S)

6. TYPE OF REPORT

Monthly Report

7. PERIOD COVERED (Inclusive Dates)

March 1992

8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

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

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

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

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Operations Analysis Center (NOAC) during the one month period identified on the cover of the document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees 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.

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

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

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page) Unclassified

(This Report) Unclassified

15. NUMBER OF PAGES

16. PRICE

NRC FORM 335 (2-89)

120555139531 1 1AN1CV1NJ11M1
US NRC-OADM
DIV FOIA & PUBLICATIONS SVCS
TPS-PDR-NUREG
P-211
WASHINGTON DC 20555