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Licensee Event Report (LER) Compilation

For month of September 1987

Oak Ridge National Laboratory

Prepared for
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Oak Ridge National Laboratory
Nuclear Safety Information Center
Oak Ridge, TN 37831

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Abstract

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one month period identified on the cover of the document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting for revisions to those events occurring prior to 1984 are described in NRC Regulatory Guide 1.16 and NUREG-1061, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports. For those events occurring on and after January 1, 1984, LERs are being submitted in accordance with the revised rule contained in Title 10 Part 50.73 of the Code of Federal Regulations (10 CFR 50.73 - Licensee Event Report System) which was published in the Federal Register (Vol. 48, No. 144) on July 26, 1983. NUREG-1022, Licensee Event Report System - Description of Systems and Guidelines for Reporting, provides supporting guidance and information on the revised LER rule.

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

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[1] ARNOLD DOCKET 50-331 LER 87-009
EMERGENCY DIESEL GENERATOR TRIP DUE TO INCORRECT RELAY SETTING.
EVENT DATE: 052787 REPORT DATE: 071787 NSSS: GE TYPE: BWR
VENDOR: BROWN BOVERI
COLT INDUSTRIES, INC.

(NSIC 205452) ON MAY 27, 1987, WITH THE REACTOR SHUTDOWN FOR REFUELING, THE EMERGENCY DIESEL GENERATOR ('B' EDG) AUTOMATICALLY SHUT DOWN WHEN A LARGE PUMP WAS STARTED DURING PERFORMANCE OF A SIMULATED AUTOMATIC ACTUATION TEST. THE INTERMEDIATE CAUSE WAS AN INCORRECT SETPOINT ON A NEWLY INSTALLED PHASE DIFFERENTIAL OVERCURRENT (PDO) RELAY. THE ROOT CAUSE WAS AN INADEQUATE CONSTRUCTION ACCEPTANCE PROCEDURE, WITHIN WHICH A TEST PROVIDED IN THE MANUFACTURER'S INSTRUCTION MANUAL ADJUSTED THE RELAY SETPOINT DURING TESTING BUT DID NOT CALL FOR RETURNING IT TO THE DESIGN VALUE. AS CORRECTIVE ACTION, THE PROCEDURE WAS MODIFIED AND PERFORMED AGAIN SUCH THAT THE SETPOINTS ON THE PDO RELAYS FOR BOTH EDGS WERE LEFT AT THE CORRECT VALUE. THE OTHER EDG HAD NOT YET BEEN TESTED. SUBSEQUENT RESEARCH DETERMINED THE EDGS WOULD HAVE BEEN ABLE TO SUPPORT REQUIRED SAFEGUARD LOADS DURING THE TIME THE PDO RELAYS HAD INCORRECT SETPOINTS. FOLLOWING A RE-EXAMINATION OF IT'S REPORTABILITY, THIS EVENT IS BEING REPORTED PURSUANT TO 10 CFR 50.73(A)(2)(V)(D).

[2] ARNOLD DOCKET 50-331 LER 87-018
PARTIAL GROUP IV ISOLATION DUE TO DE-ENERGIZATION OF RPS BUS 'B' DURING MAINTENANCE.
EVENT DATE: 060187 REPORT DATE: 071787 NSSS: GE TYPE: BWR

(NSIC 205427) ON JUNE 1, 1987, WITH THE PLANT IN COLD SHUTDOWN, A PARTIAL GROUP IV ISOLATION OCCURRED, PER DESIGN, WHEN POWER WAS LOST TO THE 'B' PRIMARY CONTAINMENT ISOLATION SYSTEM LOGIC. THIS OCCURRED DURING REINSTALLATION OF A RELAY IN THE RPS POWER SUPPLY LOGIC. THE 'B' LOOP RHR SHUTDOWN COOLING INJECT VALVE WAS THE ONLY GROUP IV VALVE THAT WAS OPEN AND NOT TAGGED OUT, THEREFORE IT WAS THE ONLY VALVE TO ACTUATE. THE INTERMEDIATE CAUSE OF THE ISOLATION (M01905 CLOSING) WAS FAILURE TO TAGOUT THIS VALVE. THE ROOT CAUSE WAS A LACK OF PROCEDURAL GUIDANCE FOR THE PLANNING OF MAINTENANCE. AT THE PRESENT TIME THERE IS NO PROCEDURE SPECIFICALLY AIMED AT PLANNING MAINTENANCE TO AVOID ADVERSE AFFECTS ON SYSTEMS OTHER THAN THOSE BEING WORKED ON DIRECTLY. THIS EVENT DID NOT AFFECT THE SAFE OPERATION OF THE PLANT BECAUSE IT WAS ONLY A MOMENTARY ISOLATION CONDITION, WHICH WAS IMMEDIATELY RESET. AS A CORRECTIVE ACTION, THE CMAR, PMAR, JUMPER AND LIFTED LEAD, AND TAGOUT PROCEDURES WILL BE UPDATED TO PROVIDE MORE SPECIFIC GUIDANCE FOR MINIMIZING THE POSSIBILITY OF INADVERTENT ESF ACTUATIONS. THIS EVENT, WHICH OCCURRED ON JUNE 1, 1987, IS BEING REPORTED PURSUANT TO 10 CFR 50.73(A)(2)(IV). A 30 DAY EXTENSION ON THIS LER WAS GRANTED BY THE RESIDENT INSPECTOR.

[3] ARNOLD DOCKET 50-331 LER 87-021
REPLACEMENT OF SELECTED RELAYS DUE TO INADEQUATE SEISMIC QUALIFICATION.
EVENT DATE: 062487 REPORT DATE: 072487 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 205530) ON JUNE 24, 1987 THE REACTOR WAS IN COLD SHUTDOWN FOR A REFUEL OUTAGE ELECTRIC (GE) HGA11 AND HGA111 RELAYS MAY NOT BE SEISMICALLY QUALIFIED TO THE SPECIFICATIONS IN SECTION 3.10.1.1 OF THE UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR). THIS REPORT WAS BASED ON INFORMATION IN GE SERVICE ADVISORY LETTER (SAL) 721-PSM-174.1. GE SAL 721-PSM-174.1 INDICATED THE GE HGA11 AND HGA111 RELAYS ARE SUBJECT TO CONTACT CHATTERING WITH ACCELERATIONS ABOVE ABOUT 0.35G. THE CONCERN IS FOR RELAYS IN THE DE-ENERGIZED STATE UTILIZING THE NORMALLY CLOSED CONTACTS FOR A SAFETY RELATED FUNCTION. THESE CONTACTS CANNOT BE RELIED UPON TO MAINTAIN CONTINUITY DURING A SEISMIC EVENT INVOLVING HORIZONTAL ACCELERATIONS OF GREATER THAN 0.35G. AFTER REVIEWING THE APPLICATIONS OF HGA11

AND HGA111 RELAYS, IT WAS DETERMINED THAT CONTACT CHATTER IN 14 RELAYS COULD HAVE POTENTIALLY PREVENTED OR DELAYED THE FULFILLMENT OF A SAFEY FUNCTION DURING A CONCURRENT SEISMIC EVENT/DESIGN BASIS ACCIDENT. THEREFORE, EACH OF THE 14 RELAYS WERE REPLACED WITH GE CENTURY TYPE HFA RELAYS BEFORE STARTUP FROM THE 1987 REFUEL OUTAGE. THIS EVENT IS BEING REPORTED PURSUANT TO 10 CFR 50.73(A)(2)(V)(A) AND 10 CFR 21.

[4] BEAVER VALLEY 1 DOCKET 50-334 LER 87-014
SPURIOUS ACTUATION OF THE CONTROL ROOM EMERGENCY BOTTLED AIR PRESSURIZATION SYSTEM.
EVENT DATE: 061687 REPORT DATE: 071587 NSSS. WE TYPE: PWR
OTHER UNITS INVOLVED: BEAVER VALLEY 2 (PWR)

(NSIC 205453) ON 6/16/87, UNIT 2 OPERATIONS PERSONNEL WERE RESTORING AN EQUIPMENT CLEARANCE ON 480 VAC MOTOR CONTROL CENTER, MCC*2-ELO. AT 1653 HOURS, DURING THE RESTORATION, AN ACTUATION OF THE CONTROL ROOM EMERGENCY BOTTLED AIR PRESSURIZATION SYSTEM (CREBAPS) TRAIN B OCCURRED. THE SYSTEM ACTUATES ON HIGH-HIGH CONTROL ROOM RADIATION ALARMS, A CONTAINMENT ISOLATION PHASE B SIGNAL, A CHLORINE DETECTION SIGNAL, OR A MANUAL ACTUATION AT EITHER UNIT AND A LOSS OF POWER TO THE CONTROL ROOM RADIATION MONITORS AT UNIT 2 ONLY. AN INVESTIGATION DETERMINED THAT DUE TO AN INCORRECT SEQUENCE USED IN RESTORING MCC*2-ELO, A UNIT 2 CONTROL ROOM RADIATION MONITOR LOSS OF POWER OCCURRED. THE OPERATORS THEN ATTEMPTED TO RESET THE SIGNAL BUT WERE UNSUCCESSFUL. AN OPERATOR WAS THEN DISPATCHED TO ISOLATE THE CONTROL ROOM EMERGENCY BOTTLED AIR SYSTEM, PLACING UNIT 1 IN TECHNICAL SPECIFICATION 3.0.3. AT 1722 HOURS, AN OPERATOR WAS DISPATCHED TO OPEN THE 125 VDC CONTROL POWER BREAKER, WHICH DEENERGIZES THE UNIT 1/UNIT 2 INTERFACING CONTACT. A RESET OF THE SYSTEM WAS THEN ACCOMPLISHED. AT 1730 HOURS, THE CONTROL ROOM EMERGENCY BOTTLED AIR SYSTEM WAS UNISOLATED. THE CAUSE WAS DETERMINED TO BE PERSONNEL ERROR. THERE WERE NO SAFETY IMPLICATIONS AS A RESULT OF THIS INCIDENT. THE SYSTEM OPERATED AS DESIGNED TO INITIATE CREBAPS UPON A LOSS OF POWER TO THE UNIT 2 CONTROL ROOM RADIATION MONITORS.

[5] BEAVER VALLEY 1 DOCKET 50-334 LER 80-001 REV 01
UPDATE ON SUCTION PIPING OF CONTAINMENT VACUUM PUMP NOT ADEQUATELY SUPPORTED.
EVENT DATE: 062687 REPORT DATE: 071787 NSSS: WE TYPE: PWR

(NSIC 205580) ON JUNE 26, 1987, IT WAS DISCOVERED THAT THE SUCTION PIPING OF THE "A" CONTAINMENT VACUUM PUMP WAS NOT ADEQUATELY SUPPORTED FOR A SEISMIC EVENT. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT AS: 1) THE CONTAINMENT VACUUM PUMPS ARE NOT A SAFETY GRADE SYSTEM AND 2) THE PIPING SECTION WITH THE SEISMIC CONCERN WAS DOWNSTREAM OF TWO (2) FULLY OPERABLE CONTAINMENT ISOLATION VALVES AND THE FIRST PIPING SUPPORT AFTER THESE VALVES. DURING THE ORIGINAL SEISMIC ANALYSIS OF BEAVER VALLEY, AN INACCURATE COMPUTER CODE WAS USED. A RE-ANALYSIS WAS LATER PERFORMED IN RESPONSE TO I.E. BULLETIN 79-14, AND ADDITIONAL SUPPORTS WERE ADDED AS REQUIRED. HOWEVER, NOT ALL PIPES WITH DIAMETERS LESS THAN OR EQUAL TO TWO (2) INCHES WERE RE-ANALYZED.

[6] BEAVER VALLEY 1 DOCKET 50-334 LER 87-015
INOPERABLE CHLORINE DETECTORS CAUSED BY OXIDATION OF THE ELECTRODE COIL AND LOW PROBE SENSITIVITY.
EVENT DATE: 070387 REPORT DATE: 080387 NSSS: WE TYPE: PWR

(NSIC 205616) ON 7/3/87, WITH THE STATION AT NORMAL FULL POWER OPERATION, SENSITIVITY DURATION TESTING WAS BEING PERFORMED ON THE CONTROL ROOM CHLORINE DETECTORS AS A FOLLOWUP ACTION TO DESIGN CHANGE INSTALLATION. THE TESTING SHOWED THAT ALL THREE DETECTOR PROBES FAILED TO MEET THE TEN (10) SECOND RESPONSE CRITERIA. THE PROBES WERE RECALIBRATED AND RETURNED TO SERVICE ON 7/3/87. THE PROBES WERE RETESTED ON 7/9/87. TESTING SHOWED THAT ALL THREE DETECTOR PROBES

FAILED TO MEET THE RESPONSE TIME CRITERIA AND TWO OF THE THREE PROBES FAILED TO REACH THE VOLTAGE TRIP SETPOINT. THE PROBES WERE RECALIBRATED AND RETURNED TO SERVICE. THE CAUSE FOR THESE TWO EVENTS WAS ATTRIBUTED TO IMPROPER APPLICATION OF VENDOR TEST PROCEDURES, OXIDATION OF THE ELECTRODE COIL AND LOW PROBE SENSITIVITY. DISCUSSIONS WITH THE VENDOR RESULTED IN MODIFYING THE CALIBRATION PROCEDURE TO IMPROVE THE DETECTOR TIME RESPONSE. ON 7/14/87, ALL THREE PROBES WERE CHECKED AND FOUND TO BE SATISFACTORY. ON 7/17/87, ALL THREE PROBES WERE CHECKED AND ONE DID NOT MEET THE RESPONSE TIME CRITERIA. THE CAUSE FOR THIS FAILURE WAS ATTRIBUTED TO COIL OXIDATION. THE DETECTOR PROBE WAS CLEANED AND RETURNED TO SERVICE. THERE WERE NO SAFETY IMPLICATIONS AS THERE WERE NO ACTUAL CHLORINE RELEASES DURING THE PERIODS OF DETECTOR INOPERABILITY. MANUAL ACTUATION OF THE CONTROL ROOM EMERGENCY BOTTLED AIR PRESSURIZATION SYSTEM WAS AVAILABLE.

[7] BEAVER VALLEY 2 DOCKET 50-412 LER 87-001
REACTOR TRIP DUE TO LOSS OF VITAL BUS.
EVENT DATE: 062887 REPORT DATE: 072487 NSSS: WE TYPE: PWR
VENDOR: GOULD BROWN BOVERI COMPANY

(NSIC 205543) AT 1647 HOURS ON 6/28/87, WITH THE UNIT IN COLD SHUTDOWN, A BREAKER SUPPLYING A 480 VAC EMERGENCY MOTOR CONTROL CENTER (MCC) TRIPPED ON OVERCURRENT, THUS DE-ENERGIZING THE BREAKER. THE NO. 1 120 VAC VITAL BUS WAS BEING SUPPLIED FROM THE MCC AT THE TIME BECAUSE ITS NORMAL UNINTERRUPTIBLE POWER SUPPLY (UPS) WAS OUT OF SERVICE FOR MAINTENANCE. THEREFORE, THE LOSS OF THE MCC RESULTED IN THE LOSS OF THE VITAL BUS, WHICH DE-ENERGIZED SOURCE RANGE NEUTRON FLUX DETECTOR N-31 AND CAUSED A REACTOR TRIP ON HIGH SOURCE RANGE FLUX. A TRIP SIGNAL WAS ALSO GENERATED ON LOW-LOW 'C' STEAM GENERATOR (SG) LEVEL. THIS TRIP SIGNAL RESULTED FROM THE LOSS OF THE 'C' SG CHANNEL 1 LEVEL TRANSMITTER (WHICH WAS BEING SUPPLIED WITH AN ARTIFICIALLY HIGH LEVEL SIGNAL TO PERMIT REACTOR TRIP BREAKER CLOSURE FOR CONTROL ROD TESTING) IN COINCIDENCE WITH AN ACTUAL LOW LEVEL SIGNAL ON CHANNEL 3. NO SAFETY IMPLICATIONS RESULTED BECAUSE THE UNIT WAS ALREADY IN COLD SHUTDOWN WITH AN ADEQUATE MARGIN OF NEGATIVE REACTIVITY. THE OVERCURRENT CONDITION WAS CLEARED AND THE SUPPLY BREAKER CLOSED AT 1657 HOURS ON 6/28/87. THE UPS WAS RETURNED TO SERVICE ON 7/2/87, THUS RESTORING THE NORMAL POWER SUPPLY CONFIGURATION FOR THE VITAL BUS.

[8] BEAVER VALLEY 2 DOCKET 50-412 LER 87-002
INADVERTENT PRE-OPERATIONAL SAFETY INJECTION.
EVENT DATE: 062987 REPORT DATE: 072487 NSSS: WE TYPE: PWR

(NSIC 205544) ON 6/29/87, INITIAL OPERABILITY TESTING OF THE MAIN STEAM ISOLATION AND BYPASS VALVES WAS IN PROGRESS. AS PART OF THIS TEST, IT WAS VERIFIED THAT CONTROL OF THE TRAIN B LOW STEAMLINE PRESSURE SAFETY INJECTION BLOCK COULD BE TRANSFERRED FROM THE CONTROL ROOM TO THE EMERGENCY SHUTDOWN PANEL. AFTER SUCCESSFULLY VERIFYING THIS TRANSFER, THE OPERATORS (AS PER PROCEDURE) RETURNED CONTROL OF THE SAFETY INJECTION BLOCK TO THE CONTROL ROOM. THE SOLID STATE PROTECTION SYSTEM (SSPS) WAS THEN RETURNED TO NORMAL. AT THIS TIME A TRAIN B SAFETY INJECTION (SI) SIGNAL WAS INITIATED. ALL TRAIN B SI PUMPS WERE IN PULL-TO-LOCK FOR THE TEST AND DID NOT ACTUATE. ALL TRAIN B SI VALVES DID STROKE TO THEIR REQUIRED POSITION. THE 'B' DIESEL GENERATOR DID START. OPERATORS VERIFIED A SAFETY INJECTION WAS NOT REQUIRED AND SECURED THE PLANT. INVESTIGATION DETERMINED THAT DURING THE TESTING, THE PRESSURIZER LOW PRESSURE SI BLOCK HAD BEEN UNBLOCKED. WHEN THE SSPS WAS RESTORED, THIS CAUSED AN SI. A REVISION TO THIS PROCEDURE HAS BEEN INITIATED TO RESET THIS SECOND BLOCK PRIOR TO RESTORING SSPS. THERE WERE NO SAFETY CONCERNS DUE TO THIS EVENT AS ALL ESF EQUIPMENT FUNCTIONED AS PER DESIGN.

[9] BEAVER VALLEY 2 DOCKET 50-412 LER 87-003
REACTOR TRIP DUE TO LOSS OF SIMULATED ELECTRONIC STEAM GENERATOR LEVEL SIGNAL.
EVENT DATE: 070187 REPORT DATE: 072487 NSSS: WE TYPE: PWR

(NSIC 205568) ON 7/4/87, THE STATION WAS IN COLD SHUTDOWN WITH THE REACTOR TRIP BREAKERS CLOSED TO SUPPORT ROD DROP TESTING. TWO OF THREE STEAM GENERATORS (S/G) HAD SIMULATED LEVEL SIGNALS JUMPERED IN ON TWO OF THREE NARROW RANGE CHANNELS. THE JUMPERED SIGNALS WERE IN BECAUSE OF ACTUAL LOW LEVEL CONDITIONS AND TESTING ACTIVITIES ON THE LEVEL TRANSMITTERS. METER AND CONTROL REPAIRMEN (MCR) WERE PERFORMING A SURVEILLANCE PROCEDURE IN THE PROCESS RACKS TO CALIBRATE THE LOOP 21 DELTA-T/TAVG PROTECTION CHANNEL I. DURING THE CALIBRATION, ONE MCR ACCIDENTLY BUMPED ONE OF THE JUMPERS INSTALLED ON THE 21B STEAM GENERATOR (LEVEL CHANNEL 2FWS*LT485), CAUSING AN INTERRUPTION OF THE SIMULATED LEVEL SIGNAL. SINCE THE ACTUAL LEVEL IN THE 21B S/G WAS 0%, THIS RESULTED IN THE GENERATION OF A 2/3 S/G LO-LO LEVEL REACTOR TRIP SIGNAL AT 0919 HOURS. THE REACTOR TRIP BREAKERS OPENED; HOWEVER, ALL SHUTDOWN AND CONTROL ROD BANKS WERE ON THE BOTTOM OF THE CORE PRIOR TO THE BREAKERS OPENING. THE MCRS WERE COUNSELED TO UTILIZE ADDITIONAL CAUTION WHEN PERFORMING EVOLUTIONS IN THE PROCESS RACKS. TO PREVENT FUTURE OCCURRENCES, THE JUMPERS WERE SECURED OUT OF THE GENERAL AREA TO PREVENT ANY ADDITIONAL ACCIDENTAL CONTACT. THERE WERE NO SAFETY IMPLICATIONS AS A RESULT OF THIS EVENT. THE REACTOR PROTECTION SYSTEM FUNCTIONED AS DESIGNED ON A LO-LO LEVEL CONDITION IN A SINGLE STEAM GENERATOR.

[10] BIG ROCK POINT DOCKET 50-155 LER 87-003 REV 02
UPDATE ON INOPERABLE PRIMARY SYSTEM SAFETY VALVES DUE TO INADEQUATE CLEANING PROCEDURES.
EVENT DATE: 012787 REPORT DATE: 071487 NSSS: GE TYPE: BWR
VENDOR: CROSBY VALVE & GAGE CO.

(NSIC 205436) ON JANUARY 27, 1987 SURVEILLANCE TESTING RESULTS SHOWED THAT THREE (3) PRIMARY SYSTEM SAFETY RELIEF VALVES INDICATED CONSISTENTLY HIGH INITIAL AS-FOUND SETPOINTS. TESTING AND EVALUATION SHOWED FAILURES OCCUR ON THE FIRST "LIFT" ONLY AND SUBSEQUENT ACTUATIONS ARE SATISFACTORY. FURTHER INVESTIGATION SHOWED THAT THE APPARENT CAUSE OF THE "STICKING" WAS DUE TO THE BONDING EFFECT OF LAPPING COMPOUND REMAINING ON THE SEAT AND DISK SURFACES FOLLOWING PREVIOUS REBUILDS. INADEQUATE PROCEDURES WITH REGARD TO CLEANING OF THE VALVE FOLLOWING LAPPING IS BELIEVED TO BE THE CAUSE OF THE FAILURES. TO CORRECT THE PROBLEMS, ALL SIX SAFETY VALVES HAVE BEEN REMOVED, REBUILT, CLEANED, AND TESTED PRIOR TO START-UP FROM THE PRESENT REFUELING OUTAGE. TO AVOID RECURRENCE, PROCEDURES WILL BE MODIFIED TO PROVIDE CLEANING REQUIREMENTS FOR THE VALVES FOLLOWING LAPPING REPAIRS. ON JUNE 1, 1987 FOLLOW-UP TESTING RESULTS INDICATE THAT "STICKING" IS STILL OCCURRING AS EVIDENCED BY HIGH "FIRST LIFT" PRESSURES. ALL SIX VALVES WERE REBUILT USING NEW 17-4 PH STAINLESS STEEL DISC INSERTS AND CLEANED USING ACETONE. VALVES WERE THEN SET USING DIRECT NITROGEN BOTTLE PRESSURE WITHOUT THE HYDRAULIC UNIT TO AVOID THE INTRODUCTION OF CONTAMINANTS DURING TESTING.

[11] BIG ROCK POINT DOCKET 50-155 LER 87-008
NON-QUALIFIED EEQ SPLICES.
EVENT DATE: 061987 REPORT DATE: 072087 NSSS: GE TYPE: BWR

(NSIC 205437) ON JUNE 6, 1987 DURING A PLANT OUTAGE, WALKDOWNS OF EEQ CIRCUITS REVEALED TWENTY-THREE (23) UNQUALIFIED SPLICES ON VALVE POSITION INDICATING CIRCUITS. THESE 23 SPLICES DID NOT HAVE THE "3M" TAPE SYSTEM IN PLACE, THE METHOD QUALIFIED BY CONSUMERS POWER COMPANY/WYLE TESTS. CAUSE OF THE DEFICIENCY IS ATTRIBUTED TO A FAILURE TO RECOGNIZE THE NEED TO UPGRADE THE SPLICES ON THESE CIRCUITS OUTSIDE CONTAINMENT. FOLLOWING IDENTIFICATION OF THE PROBLEM, ALL 23 SPLICES WERE REPLACED WITH QUALIFIED RAYCHEM SPLICES PRIOR TO START-UP. TO PREVENT RECURRENCE, ADDITIONAL VERIFICATIONS AND WALKDOWNS WERE COMPLETED ON OTHER EEQ CIRCUITS PRIOR TO START-UP.

[12]   BRAIDWOOD 1   DOCKET 50-456   LER 87-029
MISSED TECHNICAL SPECIFICATION SURVEILLANCE DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 032087   REPORT DATE: 071487   NSSS: WE   TYPE: PWR

(NSIC 205429) ON 12/15/86 A GENERAL SURVEILLANCE (GSRV) REQUEST FORM WAS PROCESSED FOR AN INSTRUMENTATION SLAVE RELAY SURVEILLANCE PRESSURE INTERLOCK. THE BASE FREQUENCY WAS IMPROPERLY LISTED AS 550 DAYS. THIS ERROR WAS IDENTIFIED ON 2/9/87 AND THE BASE FREQUENCY WAS CHANGED TO 92 DAYS. HOWEVER, THE DUE DATE WAS NOT CHANGED AND AS A RESULT THE SURVEILLANCE WAS MISSED. THE SURVEILLANCE WAS SATISFACTORILY PERFORMED UPON DISCOVERY OF THIS ERROR ON 6/17/87. THE PROCEDURE WILL BE REVISED TO REQUIRE THE DUE DATE TO BE CHANGED IF THE BASE FREQUENCY IS CHANGED. GSRV WILL BE REVISED TO PROVIDE A WARNING TO CHECK DUE DATES IF FREQUENCIES ARE CHANGED. ADDITIONALLY, THIS EVENT WILL BE REVIEWED WITH ALL DEPARTMENT SURVEILLANCE COORDINATORS.

[13]   BRAIDWOOD 1   DOCKET 50-456   LER 87-036
MISSED CIRCULATION WATER BLOWDOWN COMPOSITE SAMPLE DUE TO INADEQUATE PROCEDURE.
EVENT DATE: 070487   REPORT DATE: 072987   NSSS: WE   TYPE: PWR

(NSIC 205698) AT 0900 ON JULY 4, 1987, A RADIATION CHEMISTRY TECHNICIAN (RCT) DISCOVERED THAT THE COMPOSITE SAMPLE FOR THE CIRCULATION BLOWDOWN LINE WAS NOT AVAILABLE. HE FOUND THAT THE SWITCH ON THE COMPOSITOR WAS IN THE METER POSITION, RATHER THAN IN THE TIMER POSITION. HE IMMEDIATELY RESET THE SWITCH TO THE TIMER POSITION, AND OBTAINED A GRAB SAMPLE. A NEW PROCEDURE, BWCP 603-10, OPERATION OF COMPOSITOR, WILL BE WRITTEN TO PREVENT RECURRENCE. ADDITIONALLY, THIS EVENT WILL BE INCLUDED IN THE RCT REQUIRED READING PROGRAM, AND WAS DISCUSSED IN THE WEEKLY RADIATION CHEMISTRY DEPARTMENT MEETING AND INCLUDED IN THE DEPARTMENT NEWS LETTER. THE PERIOD FOR WHICH THE SAMPLE WAS MISSED WAS FROM 0800 ON JULY 3, 1987 TO 0900 ON JULY 4, 1987. THERE HAVE BEEN NO PREVIOUS OCCURENCES.

[14]   BRAIDWOOD 1   DOCKET 50-456   LER 87-035
REACTOR TRIP FROM ROD CONTROL SYSTEM MOTOR GENERATOR SET TRIP DUE TO MISCOMMUNICATION.
EVENT DATE: 070587   REPORT DATE: 072887   NSSS: WE   TYPE: PWR
VENDOR: DALE ELECTRONICS
        GENERAL ELECTRIC CO.
        INTERNATIONAL RECTIFIER

(NSIC 205697) WHILE TROUBLESHOOTING SPURIOUS TRIPS OF THE 1B ROD CONTROL SYSTEM MOTOR GENERATOR (M/G) SET, REACTOR TRIP OCCURRED DUE TO MISCOMMUNICATION ERROR. A TECHNICAL STAFF ENGINEER REQUESTED AN OPERATOR (A-MAN) BE SENT TO CLOSE THE OUTPUT BREAKER OF THE M/G SET. CONTROL ROOM PERSONNEL INSTRUCTED THE A-MAN TO ASSIST THE OAD ENGINEER. UPON ARRIVAL AN OPERATIONAL ANALYSIS DEPARTMENT (OAD) ENGINEER INSTRUCTED THE A-MAN TO CLOSE THE BREAKER. THE A-MAN NOTICED NUMEROUS PIECES OF TESTING EQUIPMENT, AND NOT KNOWING WHAT BYPASSES HAD BEEN DISABLED, CLOSED THE BREAKER, WITHOUT USING PROCEDURE BWOP RD-5, CONTROL ROD DRIVE M/G SET STARTUP AND PARALLELING TO OPERATING M/G SET. THIS CAUSED BOTH 1A AND 1B M/G SET BREAKERS TO OPEN, DROPPING CONTROL RODS INTO THE CORE. CAUSE OF THIS EVENT WAS MISCOMMUNICATION BETWEEN THE PERSONNEL INVOLVED. IT WAS NOT SPECIFIED THAT PROCEDURE BWOP RD-5 SHOULD BE USED. CORRECTIVE ACTIONS WERE TO REPLACE 2 RESISTORS, A VARISTOR, AND A DIODE. OPERATING WILL ENSURE THAT PROCEDURE BWOP RD-5 IS IN HAND WHEN PERFORMING PARALLELING OPERATION OF M/G SETS; THIS EVENT WILL BE REVIEWED BY OPERATING, OAD, AND TECHNICAL STAFF PERSONNEL, AND THIS EVENT WILL BE INCLUDED IN AN UPCOMING ISSUE OF TOPICS FOR SUCCESS, AN INFORMATION DISSEMINATING PROGRAM.

[15] BROWNS FERRY 1 DOCKET 50-259 LER 84-008 REV 01
UPDATE ON INOPERABLE PRESSURE TRANSMITTERS.
EVENT DATE: 012984 REPORT DATE: 071087 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 205422) DURING NORMAL UNIT OPERATION, ONE OR MORE PRESSURE DIFFERENTIAL TRANSMITTERS (PDT) 1-25A THROUGH D FAILED IN THE DOWNSCALE (NONCONSERVATIVE) DIRECTION ON FIVE SEPARATE OCCASIONS, CAUSING "TROUBLE" ALARMS TO ANNUNCIATE IN THE REACTOR PROTECTION SYSTEM ANALOG TRIP UNITS. PDT-1-25A FAILED ON 1/31/84, AND PDT-1-25C AND D BOTH FAILED ON 2/3/84. THIS INSTRUMENTATION SENSES HIGH FLOW ON EACH MAIN STEAMLINE. HIGH FLOW COULD INDICATE A BREAK IN THE MAIN STEAMLINE, AND FLOWS ABOVE THE INSTRUMENT TRIP SETTING INITIATING MAIN STEAMLINE ISOLATION. FOLLOWING EXTENSIVE TESTING OF THE TRANSMITTERS BY BOTH TVA AND THE MANUFACTURER, IT WAS CONCLUDED THAT THE OFFSCALE CONDITIONS COULD BE ATTRIBUTED TO THE BEHAVIOR OF THE PULSE DAMPENING DEVICES (SNUBBERS) INSTALLED IN THE INSTRUMENT SENSING LINES. THE SNUBBERS WERE REMOVED, AND UNIT 1 WAS OPERATED FOR APPROXIMATELY SEVEN MONTHS WITH NO FURTHER PROBLEMS NOTED.

[16] BROWNS FERRY 1 DOCKET 50-259 LER 87-015
ENGINEERED SAFETY FEATURE ACTUATIONS DUE TO CIRCUIT PROTECTOR TRIP CAUSED BY JARRING OF PANEL.
EVENT DATE: 070987 REPORT DATE: 080787 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 205661) ON JULY 9, 1987, AT 0951, A REACTOR PROTECTION SYSTEM (RPS) CIRCUIT PROTECTOR TRIPPED DE-ENERGIZING RPS BUS A. AS A RESULT THE FOLLOWING ENGINEERED SAFETY FEATURES ACTUATED CAUSING A RPS HALF SCRAM, PRIMARY CONTAINMENT ISOLATION SYSTEM GROUPS 2, 3, 6 AND 8 ISOLATED, STANDBY GAS TREATMENT INITIATED, AND CONTROL ROOM EMERGENCY VENTILATION INITIATED. THE OPERATOR SECURED THE SYSTEMS TO STANDBY READINESS. THE EVENT LASTED 8 MINUTES. MODIFICATIONS CRAFTSMEN WORKING NEAR THE CIRCUIT PROTECTOR JARRED THE PANEL CAUSING THE TRIP OF THE CIRCUIT PROTECTOR. THE CRAFTSMEN INVOLVED HAVE BEEN COUNSELED ON PROPER WORK METHODS REGARDING TREATMENT OF PLANT EQUIPMENT.

[17] BROWNS FERRY 2 DOCKET 50-260 LER 87-004
PRIMARY CONTAINMENT ISOLATION DUE TO REACTOR WATER CLEANUP INSTRUMENT DRIFT.
EVENT DATE: 062487 REPORT DATE: 071787 NSSS: GE TYPE: BWR
VENDOR: FENWALL, INC.

(NSIC 205520) ON JUNE 24, 1987, AT 1235, THE UNIT 2 REACTOR WATER CLEANUP (RWCU) ISOLATED UNEXPECTEDLY. THE CAUSE OF THE ISOLATION WAS DETERMINED TO BE A SPURIOUS SIGNAL FROM A TEMPERATURE INDICATING SWITCH (TIS) MONITORING THE WATER TEMPERATURE AT THE OUTLET OF THE NON-REGENERATIVE HEAT EXCHANGER. THE ISOLATION SIGNAL WAS CLEARED AND RWCU RETURNED TO SERVICE BY 1350. THE TEMPERATURE INDICATING SWITCH WAS CHECKED AND FOUND TO BE OUT OF CALIBRATION. THE TIS WAS RECALIBRATED. NO FURTHER CORRECTIVE ACTION IS REQUIRED.

[18] BRUNSWICK 1 DOCKET 50-325 LER 87-020
FAILURES OF REACTOR SAFETY RELIEF VALVES B21-F013J AND L TO OPEN AS A RESULT OF EXCESS LOCTITE ON THE INTERIOR OF THE VALVES' SOLENOID BONNET TUBES.
EVENT DATE: 070387 REPORT DATE: 073187 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 205615) FOLLOWING A REACTOR SCRAM ON JULY 1, 1987 (SEE LER 1-87-19), AN ATTEMPT WAS MADE TO CONTROL PRESSURE WITH REACTOR VESSEL SAFETY RELIEF VALVE (SRV) J. THE VALVE FAILED TO OPENED WHEN GIVEN A MANUAL OPENED SIGNAL. SRVS A, B, AND E WERE SUCCESSFULLY USED TO CONTROL PRESSURE DURING THIS EVENT. DUE TO

THE FAILURE OF THE SRV J TO OPENED, THE REMAINING SRVS WHICH HAD NOT BEEN OPENED DURING THE RECOVERY WERE TESTED AT APPROXIMATELY 250 PSIG TO ENSURE OPERABILITY PRIOR TO STARTUP OF THE UNIT. DURING THIS TEST, SRV L FAILED TO OPENED. INITIAL INVESTIGATION ON SITE DETERMINED THAT THE SOLENOIDS FOR THESE VALVES HAD FAILED. FURTHER INVESTIGATION INTO THE FAILURE MODE WAS CONDUCTED AT WYLE LABORATORY. IT WAS DETERMINED THAT LOCTITE WAS IN THE CLEARANCE BETWEEN THE SOLENOID PLUNGER AND BONNET TUBE, AND HAD APPARENTLY BONDED THE PLUNGER TO THE BONNET TUBE ON THE FAILED VALVES, PREVENTING PROPER SOLENOID OPERATION. THE SOLENOIDS WERE REPLACED ON SRVS J AND L AS WELL AS THE REMAINING AUTOMATIC DEPRESSURIZATION SYSTEM VALVES (5).

[19] BRUNSWICK 2 DOCKET 50-324 LER 86-017 REV 01
UPDATE ON AUTOMATIC SCRAM ON LOW WATER LEVEL RESULTING FROM FAILURE OF REACTOR FEEDWATER PUMP 2B DISCHARGE CHECK VALVE TO CLOSE.
EVENT DATE: 061886 REPORT DATE: 071687 NSSS: GE TYPE: BWR
VENDOR: CRANE VALVE CO.

(NSIC 205435) AT 0811 ON 6/18/86, WITH UNIT 2 AT 55% POWER, A REACTOR SCRAM OCCURRED DUE TO LOW LEVEL (LL) NO. 1 WHILE PLACING REACTOR MAIN STEAM-DRIVEN FEED PUMP (RFP) 2B INTO SERVICE. IMMEDIATELY AFTER THE SCRAM, THE CONTROL OPERATOR CLOSED THE RFP 2B DISCHARGE VALVE ALLOWING RFP 2A TO RESTORE NORMAL LEVEL CONTROL. A SCRAM RECOVERY WAS CARRIED OUT. PRIMARY CONTAINMENT GROUPS 2, 6, AND 8 ISOLATIONS OCCURRED DUE TO THE LL NO. 1. REACTOR LEVEL WAS CONTROLLED WITH THE REACTOR FEEDWATER SYSTEM. REACTOR PRESSURE DECREASED TO 870 PSIG. THE LL NO. 1 RESULTED FROM REVERSE FLOW OF REACTOR FEEDWATER THROUGH RFP 2B DISCHARGE CHECK VALVE FW.V2 PRIOR TO STARTING THE PUMP. A WORN VALVE DISC PIVOT PIN LOCKING STUD IN V2 ALLOWED ONE OF THE TWO PIVOT PINS TO LOOSEN AND CAUSE THE DISC TO BECOME IMPROPERLY SEATED. V2, CRANE VALVE COMPANY MODEL LIST 973A, WAS DISASSEMBLED AND THE DISC PIVOT PINS, PIVOT PIN BUSHINGS, AND PIVOT PIN LOCKING STUDS WERE REPLACED. FW-VI, THE RFP 2A DISCHARGE CHECK VALVE, WAS OVERHAULED DURING THE RECENT UNIT REFUELING/MAINTENANCE OUTAGE. APPROPRIATE PROCEDURES HAVE BEEN REVISED TO MINIMIZE THE POTENTIAL FOR A DEFECTIVE CHECK VALVE TO CAUSE A LEVEL TRANSIENT WHILE PLACING A REACTOR FEED PUMP IN SERVICE.

[20] BYRON 1 DOCKET 50-454 LER 85-027 REV 03
UPDATE ON FAILURE OF MSIV TO CLOSE ON MAIN STEAM ISOLATION SIGNAL DUE TO INSTRUMENT AIR HEATER SLOWLY DEPRESSURIZING.
EVENT DATE: 031485 REPORT DATE: 072187 NSSS: WE TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 205575) DURING PLANT RESTORATION FOLLOWING THE LOSS OF OFFSITE POWER STARTUP TEST, A SAFETY INJECTION AND STEAMLINE ISOLATION WERE INITIATED. MAIN STEAM ISOLATION VALVES (MSIV) 1B AND 1C CLOSED IMMEDIATELY AS RESULT OF THE STEAMLINE ISOLATION. HOWEVER, THE 1A AND 1D MSIV'S FAILED TO FULLY CLOSE. SUBSEQUENT INVESTIGATIONS DETERMINED THAT SEVERAL OF THE AIR SUPPLY CHECK VALVES INSTALLED ON THE MSIV ACTUATORS FAILED TO SEAT WHEN THE INSTRUMENT AIR SUPPLY HEADER SLOWLY DEPRESSURIZED. THE ABILITY OF THE CHECK VALVES TO SEAT AND MAINTAIN THE AIR CHARGE STORED IN THE MSIV ACTUATOR IS REQUIRED TO PERMIT MSIV CLOSURE FOLLOWING A LOSS OF AIR SUPPLY PRESSURE, WHICH IS AN EXPECTED RESULT FOLLOWING A LOSS OF OFFSITE POWER. AS A CORRECTIVE ACTION, THE AIR SUPPLY CHECK VALVES ON THE UNIT ONE MSIV'S HAVE BEEN REPLACED. STROKE TESTING OF THE MSIV'S FOLLOWING A SLOW DEPRESSURIZATION OF THE AIR SUPPLY HAS DEMONSTRATED THE ADEQUACY OF EACH OF THE REPLACEMENT CHECK VALVES AS WELL AS THE ABILITY OF THE MSIV'S TO CLOSE FOLLOWING A LOSS OF SUPPLY AIR PRESSURE. THE ORIGINALLY DESIGNED CHECK VALVES WERE REPLACED WITH AN UPGRADED BRASS CHECK VALVE ON A TEMPORARY BASIS UNTIL STAINLESS STEEL CHECK VALVES CAN BE PROCURED. MSIV CHECK VALVE TESTING WAS PERFORMED BI-WEEKLY UNTIL JULY 8, 1985.

[21] BYRON 1 DOCKET 50-454 LER 87-011 REV 01
UPDATE ON FIRE PROTECTION CARBON DIOXIDE SYSTEM INOPERABLE BECAUSE OF A MISALIGNED VALVE DUE TO PERSONNEL ERROR.
EVENT DATE: 041587 REPORT DATE: 072087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 2 (PWR)

(NSIC 205574) ON APRIL 15, DURING THE PERFORMANCE OF THE 18 MONTH SURVEILLANCE OF THE CARBON DIOXIDE (CO(2) SYSTEM IN THE DIESEL DRIVEN AUXILIARY FEEDWATER (AF) PUMP ROOM, IT WAS FOUND THAT CO(2) WAS ISOLATED TO THE ROOM. THE CO(2) TO THE AF PUMP ROOM WAS DECLARED INOPERABLE AND TROUBLESHOOTING TO FIND THE CAUSE BEGAN. ON APRIL 16, THE SYSTEM ENGINEER FOUND THE VAPOR PILOT VALVE WAS MISPOSITIONED CLOSED. THIS VALVE, WHICH IS NORMALLY OPEN, PROVIDES CO(2) AS A MOTIVE FORCE TO OPEN MAIN CO(2) HEADER VALVES. WITH THIS MOTIVE FORCE ISOLATED THE CO(2) SYSTEM CANNOT DISCHARGE AND IS INOPERABLE. THE VAPOR PILOT VALVE WAS IMMEDIATELY OPENED. THE LAST TIME IT CAN BE DEMONSTRATED THE VALVE WAS OPENED WAS ON APRIL 4 WHEN AN INADVERTENT CO(2) DISCHARGE OCCURRED. THE ROOT CAUSE IS TO HOW THIS VALVE BECAME CLOSED IS UNKNOWN. IT IS SUSPECTED THE VALVE WAS INADVERTENTLY CLOSED DURING THE RECHARGING OF THE TANK AFTER THE DISCHARGE ON APRIL 4. IN ADDITION, THIS VALVE WAS NOT ON ANY DESIGN DOCUMENTS AND CONSEQUENTLY NOT UNDER STATION CONTROL. THERE WERE EITHER HOURLY OR CONTINUOUS FIRE WATCHES ESTABLISHED IN ALL AREAS AFFECTED, THUS SATISFYING THE TECH SPEC REQUIREMENTS THAT CORRESPOND TO THE CO(2) SYSTEM BEING INOPERABLE. BASED ON THE SUPPLEMENTAL INFORMATION THIS EVENT IS NOT CONSIDERED REPORTABLE.

[22] BYRON 2 DOCKET 50-455 LER 87-009
MANUAL REACTOR TRIP IN RESPONSE TO DECREASING STEAM GENERATOR LEVELS RESULTING FROM A FEEDWATER PUMP TRIP DUE TO DEFECTIVE SPEED CONTROL FEEDBACK LOOP.
EVENT DATE: 062987 REPORT DATE: 072987 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 205573) AT 0921 ON JUNE 29, 1987 A MANUAL UNIT 2 REACTOR TRIP WAS INITIATED DUE TO THE 2C (TURBINE DRIVEN) MAIN FEEDWATER PUMP (FW) TRIP. THE LEVEL IN ALL FOUR STEAM GENERATORS WAS BELOW THE LO LEVEL ALARM SETPOINT AND TRENDING TOWARD THE LO-LO REACTOR TRIP SETPOINT. THE SHIFT ENGINEER (LICENSED) ORDERED THE NUCLEAR STATION OPERATOR (LICENSED) TO MANUALLY TRIP THE REACTOR IN ACCORDANCE WITH GOOD OPERATING PRACTICES. AT THE LO-LO STEAM GENERATOR LEVEL SETPOINT THE AUXILIARY FEEDWATER PUMPS (AF) AUTOMATICALLY STARTED AS DESIGNED. THE UNIT WAS STABILIZED IN MODE 3 - HOT STANDBY. THE CAUSE OF THE 2C MAIN FEEDWATER PUMP TRIP WAS A COLD SOLDER JOINT ON A CAPACITOR IN THE SPEED CONTROL FEEDBACK LOOP. THE SOLDER JOINT WAS REPAIRED AND TESTED. IN ADDITION, SELECTED PARAMETERS WILL BE MONITORED DURING SUBSEQUENT OPERATIONS TO ENSURE PROPER FUNCTIONING. RESULTS OF THIS MONITORING WILL BE REPORTED IN A SUPPLEMENTAL REPORT. THERE HAVE BEEN NO PREVIOUS OCCURRENCES.

[23] BYRON 2 DOCKET 50-455 LER 87-010
MANUAL REACTOR TRIP IN ANTICIPATION OF A TURBINE TRIP DUE TO THE ELECTROHYDRAULIC PUMP MANUAL TRIP NECESSITATED BY AN EXCESSIVE FLUID LEAK.
EVENT DATE: 070187 REPORT DATE: 072987 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 205572) AT 0605, ON JULY 1, 1987, A LOW ELECTRO-HYDRAULIC (EH) (JJ) RESERVOIR LEVEL ALARM WAS RECEIVED. AN INVESTIGATION DETERMINED THAT THERE WAS A FLUID LEAK AT THE EH SKID DUE TO A LINE BREAK. THE LEAK WAS UNISOLATABLE, CONSEQUENTLY A TURBINE POWER RAMP DOWN WAS BEGUN IN AN ATTEMPT TO BRING THE TURBINE BELOW THE REACTOR TRIP SETPOINT. AT 0628 THE EH PUMPS BEGAN TO CAVITATE DUE TO LOW LEVEL. THE EH PUMPS WERE TRIPPED. THE REACTOR WAS ALSO TRIPPED IN ANTICIPATION OF THE IMMINENT TURBINE TRIP DUE TO LOSS OF EH PRESSURE. THE ROOT CAUSE OF THE LINE BREAK APPEARS TO BE EXCESSIVE VIBRATION. THE PRESSURE SUPPLY LINE TO THE FULLERS EARTH FILTER WHICH BROKE WAS REMOVED AND A VALVE AND NIPPLE

INSTALLED FOR ISOLATION UNTIL ADDITIONAL LINE SUPPORTS CAN BE DESIGNED AND INSTALLED. THERE HAVE BEEN NO PREVIOUS EVENTS.

[24] CALVERT CLIFFS 1 DOCKET 50-317 LER 87-009 REV 01
UPDATE ON USE OF FASTENERS IN ASME CLASS 1, 2, AND 3 SYSTEMS WITHOUT PROPER CERTIFICATION, SPECIAL NDE, OR SPECIAL MARKING.
EVENT DATE: 042387 REPORT DATE: 072387 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 2 (PWR)

(NSIC 205562) ON MARCH 29 AND 30, WITH THE UNIT MODE 5, COLD SHUTDOWN, MAINTENANCE PERSONNEL WERE TRAINED ON OUR NEW ASME SECTION XI REPAIR AND REPLACEMENT PROGRAM. DURING THE TRAINING DESIGN ENGINEERS LEARNED THAT FASTENERS (BOLTS, STUDS, THREADED ROD AND NUTS) LACKING PROPER CERTIFICATION MAY HAVE BEEN INSTALLED IN SAFETY SYSTEMS. ON APRIL 23 WE DETERMINED THERE WERE INSTANCES OF UNCERTIFIED NUTS INSTALLED IN SAFETY SYSTEMS AND INFORMED THE NRC. THE ORIGINAL CONSTRUCTION CODES OF CERTAIN SAFETY SYSTEMS REQUIRE FASTENERS TO HAVE CERITIFICATION SUCH AS A CERTIFIED MATERIAL TEST REPORT OR CERTIFICATE OF COMPLIANCE AND TO MEET NON-DESTRUCTIVE EXAMINATION AND MARKING REQUIREMENTS ABOVE THOSE REQUIRED BY THE ASTM SPECIFICATION. WE SCREENED 30,000 TO 40,000 MAINTENANCE REQUESTS AND FOUND 61 INSTANCES WHERE UNCERTIFIED FASTENERS WERE MISTAKENLY INSTALLED IN CERTAIN SAFETY SYSTEMS. THE EVENT WAS CAUSED BY INADEQUATE PRECAUTIONS PLACED ON REPAIR AND REPLACEMENT PLANNING ACTIVITIES REGARDING THE USE OF COMMERCIAL-QUALITY FASTENERS. ALL UNCERTIFIED FASTENERS THAT WERE FOUND TO BE MISTAKENLY INSTALLED IN SAFETY SYSTEMS WERE REPLACED WITH CERTIFIED FASTENERS. THOUGH UNCERTIFIED, WE ARE CONFIDENT THE FASTENERS ARE MADE OF THE SAME MATERIAL AS CERTIFIED FASTENERS.

[25] CATAWBA 1 DOCKET 50-413 LER 87-001 REV 01
UPDATE ON INADVERTENT WASTE GAS RELEASE DUE TO FAILURE OF A WASTE GAS DRAIN TRAP.
EVENT DATE: 010587 REPORT DATE: 071387 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 205478) ON JANUARY 5, 1987, AT APPROXIMATELY 1030 HOURS, AN INADVERTENT RADIOACTIVE GAS RELEASE OCCURRED. AT THE TIME OF THE RELEASE, PERSONNEL WORKING ON A NUCLEAR STATION MODIFICATION, WERE STROKING VALVES ON THE WASTE GAS (WG) SYSTEM. THE AUXILIARY BUILDING VENTILATION (VA) SYSTEM SWAPPED FROM THE BYPASS MODE TO THE FILTER MODE DUE TO A TRIP 2 CONDITION ON THE AUXILIARY BUILDING VENTILATION RADIATION MONITOR. UNIT 1 AND UNIT 2 WERE AT 100% POWER AT THE TIME OF THIS INCIDENT. CAUSE CODE X, OTHER, HAS BEEN ASSIGNED TO THIS INCIDENT DUE TO THE FAILURE OF WG DRAIN TRAP T-04. CAUSE CODE E, MANAGEMENT/QUALITY ASSURANCE DEFICIENCY, HAS BEEN ASSIGNED TO THIS INCIDENT DUE TO SUE OF AN INADEQUATE WORK CONTROL PROGRAM, THE USE OF WHICH HAS BEEN SUBSEQUENTLY DELETED. IN ADDITION, NO ESTABLISHED METHODS WERE PROVIDED FOR DETERMINING THE OPERATIONAL CONTROL GROUPS FOR SYSTEMS TO BE USED DURING THE WORK PLANNING PHASE. CAUSE CODE A, PERSONNEL ERROR, HAS BEEN ASSIGNED TO THIS INCIDENT DUE TO VIOLATION OF PROCEDURES AND A STATION DIRECTIVE AND DUE TO INSUFFICIENT INVESTIGATION OF THE CAUSE OF THE VENTILATION FILTER SWAP. CAUSE CODE D, PROCEDURE DEFICIENCY, HAS ALSO BEEN ASSIGNED TO THIS INCIDENT. THE NRC RESPONSE REQUIREMENT PROCEDURE INCORRECTLY IDENTIFIED THE AUTOMATIC VA SWAP DUE TO EMF-41.

[26] CATAWBA 1 DOCKET 50-413 LER 87-023
MISSED SURVEILLANCE OF FIRE DETECTION ZONES DUE TO LACK OF ADMINISTRATIVE CONTROLS FOR RETIRING TECH SPEC STANDING WORK REQUESTS.
EVENT DATE: 041187 REPORT DATE: 071687 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 205545) ON JUNE 16, 1987, AT 1630 HOURS, DUKE POWER PERSONNEL DETERMINED THAT THE SIX MONTH SURVEILLANCE OF THE FIRE DETECTION (EFA) SYSTEM SUPERVISORY

CIRCUITS HAD NOT BEEN PERFORMED SINCE AUGUST 27, 1986. INVESTIGATION REVEALED THAT THE STANDING WORK REQUEST (SWR) USED TO SATISFY THIS SURVEILLANCE HAD BEEN RETIRED AND NO REPLACEMENT SWRS WERE FORMALLY CREATED AND SCHEDULED. THIS RESULTED IN A TECHNICAL SPECIFICATION VIOLATION DUE TO THE EFA SYSTEM BEING UNKNOWINGLY INOPERABLE AND NO COMPENSATORY ACTIONS BEING TAKEN. THE TECHNICAL SPECIFICATION WAS VIOLATED ON APRIL 11, 1987. BOTH UNITS WERE AT 100% POWER AT THE TIME OF DISCOVERY. THIS INCIDENT IS CLASSIFIED AS EVENT CAUSE CODE E, MANAGEMENT/QUALITY ASSURANCE DEFICIENCY. NO ADMINISTRATIVE CONTROLS EXISTED FOR RETIRING TECHNICAL SPECIFICATION RELATED SWRS AND ENSURING THAT CORRECT FOLLOW-UP ACTION WAS TAKEN. THE EFA SYSTEM WAS DECLARED INOPERABLE AND THE SURVEILLANCES WERE PERFORMED. A LETTER WAS ISSUED TO APPROPRIATE PERSONNEL PROVIDING GUIDELINES FOR RETIRING AND TRACKING TECHNICAL SPECIFICATION RELATED SWRS. THE MAINTENANCE MANAGEMENT PROCEDURE FOR THE TECHNICAL SPECIFICATIONS PROGRAM WILL BE REVISED TO INCLUDE INFORMATION ON RETIRING TECHNICAL SPECIFICATION RELATED SWRS. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS INCIDENT.

[27] CATAWBA 1 DOCKET 50-413 LER 87-022
LEAK RATE TESTING NOT PERFORMED RESULTING IN TECHNICAL SPECIFICATION VIOLATION DUE TO INCORRECT PROCEDURE.
EVENT DATE: 061087 REPORT DATE: 071087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)
VENDOR: CONAX CORP.

(NSIC 205479) ON JUNE 10, 1987, AT 1500 HOURS, THE DUKE POWER DESIGN ENGINEERING DEPARTMENT NOTIFIED CATAWBA NUCLEAR STATION THAT CONAX ELECTRICAL PENETRATION ASSEMBLIES REQUIRE TYPE B LEAK RATE TESTING AS SPECIFIED IN APPENDIX J OF 10CFR50. PRIOR TO THIS TIME, THE PRESSURE HAD BEEN MONITORED PERIODICALLY TO VERIFY CONDUCTOR SEAL INTEGRITY AS DESCRIBED IN THE MANUFACTURER'S INSTRUCTION MANUAL. THE FAILURE TO ADEQUATELY TEST THE CONDUCTOR SEALS RESULTED IN A TECHNICAL SPECIFICATION VIOLATION. THE TEST PROCEDURE WAS CHANGED TO INCORPORATE TYPE B LEAK RATE TEST REQUIREMENTS AND ALL CONAX PENETRATIONS WERE RETESTED. BOTH UNITS HAD BEEN IN ALL MODES OF OPERATION DURING THE PERIOD WHEN THE PENETRATIONS WERE BEING IMPROPERLY TESTED. THIS INCIDENT IS CLASSIFIED AS EVENT CAUSE CODE D, DEFECTIVE PROCEDURE. THE CONAX INSTRUCTION MANUAL REFERS TO THE CONDUCTOR SEAL MATERIAL AS RESILIENT, WHICH REQUIRES THAT THE SEALS BE TYPE B LEAK RATE TESTED. THE DUKE POWER PROCEDURE DID NOT SPECIFY THAT A TYPE B TEST HAD TO BE PERFORMED WHICH REQUIRED A LEAK RATE CALCULATION TO BE MADE. THE TYPE B LEAK RATE TEST WAS CONDUCTED, AND THE TEST RESULTS WERE ACCEPTABLE. DUKE POWER WILL ISSUE A COMPLETE REWRITE OF THE APPROPRIATE TEST PROCEDURE. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS INCIDENT.

[28] CATAWBA 1 DOCKET 50-413 LER 87-025
UNIT 1 VENT FLOW RATE ESTIMATE NOT PERFORMED WITHIN REQUIRED TIME DUE TO A PERSONNEL ERROR.
EVENT DATE: 062987 REPORT DATE: 072987 NSSS: WE TYPE: PWR

(NSIC 205635) ON JUNE 29, 1987, AT 1100 HOURS, THE UNIT VENT FLOW RATE MONITORING CHANNEL WAS REMOVED FROM SERVICE IN ORDER TO PERFORM ITS CHANNEL CALIBRATION PER TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS. THE UNIT VENT FLOW RATE MANUAL CALCULATION WAS NOT PERFORMED WITHIN THE FOUR HOUR TIME TECHNICAL SPECIFICATION LIMIT. THE UNIT WAS AT 100% POWER AT THE TIME OF THIS INCIDENT. THIS INCIDENT IS CLASSIFIED AS EVENT CAUSE CODE A, PERSONNEL ERROR. THE DUKE POWER TECHNICIAN WHO REMOVED THE CHANNEL FROM SERVICE DID NOT NOTIFY THE OPERATOR AT THE CONTROLS (OATC) PRIOR TO DOING SO, AS REQUIRED BY STATION DIRECTIVES. THE UNIT VENT FLOW RATE MANUAL CALCULATION WAS IMMEDIATELY PERFORMED UPON DISCOVERY THAT THE TIME LIMIT HAD BEEN EXCEEDED. THE UNIT FLOW RATE MONITOR CHANNEL INSTRUMENTATION WAS SUBSEQUENTLY CALIBRATED, RETURNED TO SERVICE, AND DECLARED OPERABLE. THIS INCIDENT WAS REVIEWED WITH THE APPROPRIATE PERSONNEL. A PROCEDURE CHANGE WAS INCORPORATED IN THE CALIBRATION PROCEDURE TO CLARIFY THE

PROPER NOTIFICATION REQUIRED PRIOR TO PERFORMING THE PROCEDURE. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS EVENT.

[29] CATAWBA 2 DOCKET 50-414 LER 87-015 REV 01
UPDATE ON CONTAINMENT AIR RETURN ISOLATION DAMPERS ACTUATED DUE TO DEFECTIVE PROCEDURE.
EVENT DATE: 040687 REPORT DATE: 071387 NSSS: WE TYPE: PWR

(NSIC 205480) ON APRIL 6, 1987, AT 1439:41 HOURS, THE CONTAINMENT AIR RETURN FAN B ISOLATION DAMPER UNEXPECTEDLY OPENED, WHICH CONSTITUTED AN ENGINEERED SAFEGUARD FEATURE ACTUATION. DUKE POWER STATION PERSONNEL WAS IN THE PROCESS OF CONDUCTING THE CONTAINMENT AIR RETURN FAN 2B AND HYDROGEN SKIMMER FAN 2B PERFORMANCE TEST PROCEDURE. THE UNIT WAS IN MODE 3, HOT STANDBY, AT THE TIME OF THIS INCIDENT. THIS INCIDENT IS CLASSIFIED AS EVENT CAUSE CODE D, DEFECTIVE PROCEDURE. THE CONTAINMENT AIR RETURN FAN 2B AND HYDROGEN SKIMMER FAN 2B PERFORMANCE TEST PROCEDURE DID NOT CONTAIN INSTRUCTIONS TO RESET THE ACTUATION SIGNAL FOR THE ISOLATION DAMPER. THIS ALLOWED THE ISOLATION DAMPER TO OPEN WHEN THE REQUIRED LOGIC WAS COMPLETED DURING THE PERFORMANCE OF ANOTHER SECTION OF THE PROCEDURE. CONTROL ROOM PERSONNEL CLOSED THE ISOLATION DAMPER. A PROCEDURE CHANGE WAS INCORPORATED INTO THE CONTAINMENT AIR RETURN FANS AND HYDROGEN SKIMMER FANS PERFORMANCE TEST PROCEDURES FOR BOTH UNITS TO RESET THE ACTUATION SIGNAL FOR THE ISOLATION DAMPERS. THE PLANT WAS ALWAYS WITHIN ITS DESIGN BASIS AND THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED BY THIS INCIDENT.

[30] CLINTON 1 DOCKET 50-461 LER 87-031
AUTOMATIC ISOLATION OF THE REACTOR WATER CLEANUP SYSTEM DUE TO SPURIOUS TRIP OF THE HEAT EXCHANGER ROOM HIGH DIFFERENTIAL TEMPERATURE CHANNEL.
EVENT DATE: 061787 REPORT DATE: 071787 NSSS: GE TYPE: BWR
VENDOR: RILEY-BEAIRD, INC.

(NSIC 205490) ON JUNE 17, 1987, THE REACTOR WAS AT APPROXIMATELY 20% POWER IN MODE 1 (POWER OPERATION) AND THE REACTOR WATER CLEANUP SYSTEM WAS IN THE NORMAL OPERATING LINEUP. AT 1848 HOURS THE ANNUNCIATOR FOR THE DIVISION II WEST REACTOR WATER CLEANUP HEAT EXCHANGER ROOM HIGH TEMPERATURE/HIGH DIFFERENTIAL TEMPERATURE ALARMED. THE REACTOR WATER CLEANUP SYSTEM INBOARD ISOLATION VALVES CLOSED AND THE PUMPS TRIPPED. THE CONTROL ROOM OPERATOR CLEARED THE ANNUNCIATOR AND VERIFIED THAT THERE WERE NO OTHER INDICATIONS OF INCREASED TEMPERATURE. A CHANNEL FUNCTIONAL TEST WAS PERFORMED SATISFACTORILY ON THE HIGH TEMPERATURE AND HIGH DIFFERENTIAL TEMPERATURE DETECTORS. THE REACTOR WATER CLEANUP SYSTEM WAS UNISOLATED AND RETURNED TO SERVICE. ON JULY 5, 1987, AT 2032 HOURS WITH THE REACTOR AT APPROXIMATELY 54% POWER IN MODE 1, A SPURIOUS SIGNAL WAS RECEIVED AGAIN RESULTING IN A SECOND ISOLATION OF THE REACTOR WATER CLEANUP SYSTEM. AFTER THE ANNUNCIATOR WAS CLEARED AND ATTEMPTS TO RECREATE THE SPURIOUS SIGNAL WERE UNSUCCESSFUL, CONTROL ROOM OPERATORS BEGAN UNISOLATING THE SYSTEM AND A THIRD SPURIOUS TRIP OCCURRED AT 2100 HOURS ON JULY 5, 1987. TROUBLESHOOTING DETERMINED THAT THE CAUSE OF THESE EVENTS WAS A FAILED DIFFERENTIAL TEMPERATURE POINT MODULE IN THE LEAK DETECTION SYSTEM. THE MODULE WAS REPLACED, TESTED AND THE SYSTEM RETURNED TO SERVICE.

[31] CLINTON 1 DOCKET 50-461 LER 87-041
VIOLATION OF PLANT'S TECHNICAL SPECIFICATIONS DUE TO UTILITY PERSONNEL ERROR RESULTING FROM PROCEDURAL DEFICIENCIES.
EVENT DATE: 070187 REPORT DATE: 072887 NSSS: GE TYPE: BWR

(NSIC 205552) ON JULY 1, 1987, AT 1730 HOURS, WITH THE PLANT IN MODE 1 (POWER OPERATION), AT APPROXIMATELY 50% REACTOR POWER, IT WAS DETERMINED THAT CONTAINMENT PENETRATIONS, 1MC-31 (VALVES 1E12-F436 AND 1E12-F437), 1MC-33 (VALVE 1E22-F376) AND 1MC-34 (VALVES 1SF004 AND 1SF034), WERE LEAKED TESTED WITH AIR

INSTEAD OF WATER AS REQUIRED BY TECHNICAL SPECIFICATION 3.6.1.2(E). THE VALVES HAD SATISFACTORILY PASSED THE LEAK TESTS WITH AIR IN ACCORDANCE WITH CPS SURVEILLANCE PROCEDURE 9861.02, "LOCAL LEAK RATE TESTING REQUIREMENTS." IMMEDIATE CORRECTIVE ACTION VERIFIED THAT THE VALVES MET THE LEAK RATE REQUIREMENTS BY PERFORMING THE FOLLOWING ACTIVITIES: 1) THE LOCAL LEAK RATE TESTS WERE CHECKED TO ENSURE THAT THE REMAINING VALVES IN PROCEDURE 9861.02 WERE TESTED WITH THE PROPER MEDIUM. 2) IN ADDITION, ALL OF THE ABOVE VALVES WERE SHUT AGAINST TEST PRESSURE DURING THE INTEGRATED LEAK RATE TEST WHICH WAS COMPLETED SATISFACTORILY. 3) THE AIR LEAKAGE RATE FOR THE ABOVE VALVES WAS CONVERTED TO AN EQUIVALENT WATER LEAKAGE RATE. THE EQUIVALENT WATER LEAKAGE RATE FOR EACH VALVE MET THE LEAKAGE REQUIREMENTS OF TECHNICAL SPECIFICATIONS. 4) THE VALVES WERE SATISFACTORILY LEAK TESTED WITH WATER ON JULY 3, 1987. THIS EVENT WAS NOT SAFETY SIGNIFICANT FOR EXISTING PLANT CONDITIONS OR OTHER PLANT MODES IN THAT THE AFFECTED VALVES WOULD HAVE PERFORMED THEIR REQUIRED FUNCTION.

[32] CLINTON 1 DOCKET 50-461 LER 87-038
VIOLATION OF THE PLANT'S TECHNICAL SPECIFICATIONS DUE TO PERSONNEL ERROR RESULTING IN A MISWIRED CONTROL ROOM VENTILATION MAKEUP FAN.
EVENT DATE: 071487 REPORT DATE: 072387 NSSS: GE TYPE: BWR

(NSIC 205551) ON JULY 14, 1987, PERFORMANCE OF CPS SURVEILLANCE PROCEDURE 9070.01 DETERMINED THAT CONTROL ROOM VENTILATION MAKEUP FAN B WAS PRODUCING LESS THAN THE REQUIRED FLOW RATE. THIS PROBLEM WAS DETERMINED TO HAVE BEEN CAUSED BY A MAINTENANCE WORK REQUEST WORKED ON JUNE 23, 1987. THE MOTOR LEADS WERE DISCONNECTED AND RECONNECTED IN THE INCORRECT CONFIGURATION. PRIOR TO RELANDING THE MOTOR LEADS, A QUESTION AS TO WHETHER TWO OF THE WIRES SHOULD BE REVERSED BASED ON THE WAY IN WHICH THE LEADS ENTERED THE TERMINATION AREA WAS DISCUSSED BETWEEN THE ELECTRICAL TECHNICIAN AND QUALITY CONTROL INSPECTOR. IGNORING PREVIOUSLY RECORDED DATA AS TO THE "AS FOUND" CONDITION OF THE LEADS, AND WITHOUT CONSULTATION WITH SUPERVISION, THE ELECTRICIAN AND INSPECTOR REVERSED THE LEADS THUS CAUSING THEM TO BE LANDED INCORRECTLY. ON JULY 14, 1987, THE LEADS WERE PROPERLY INSTALLED AND THE FAN PASSED THE SURVEILLANCE TEST. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO UTILITY PERSONNEL ERROR. THE INDIVIDUALS INVOLVED HAVE BEEN COUNSELED AND TECHNICIANS WILL BE TRAINED ON THE LESSONS LEARNED FROM THIS EVENT. OTHER SAFETY RELATED FANS AND RELATED WORK ACTIVITIES WERE REVIEWED WITH NO ADDITIONAL DISCREPANCIES DISCOVERED. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B) SINCE ONE CONTROL ROOM VENTILATION TRAIN WAS INOPERABLE IN EXCESS OF TECHNICAL SPECIFICATION REQUIREMENTS.

[33] CONNECTICUT YANKEE DOCKET 50-213 LER 87-008
BLOWN CONTROL ROD FUSE CAUSES DROPPED ROD AND STEAM GENERATOR OVERFEED.
EVENT DATE: 062587 REPORT DATE: 072187 NSSS: WE TYPE: PWR

(NSIC 205438) AT 1318 HOURS ON JUNE 25, 1987, WITH THE PLANT OPERATING IN MODE 1 AT 98 PERCENT REACTOR POWER, CONTROL ROD NUMBER 31 DROPPED INTO THE CORE DUE TO A BLOWN FUSE IN THE STATIONARY GRIPPER COIL POWER SUPPLY. FEEDWATER FLOW CONTROL VALVE (FRV) NUMBER 2 HAD PREVIOUSLY BEEN BLOCKED OPEN FOR MAINTENANCE. THE DROPPED ROD INITIATED A TURBINE LOAD RUNBACK WHICH, SINCE FRV NUMBER 2 WAS BLOCKED OPEN, CAUSED AN OVERFEED TO STEAM GENERATOR NUMBER 2. OPERATORS PROPERLY RESPONDED TO THE EVENT AND TERMINATED THE OVERFEED IN 2 MINUTES AND AT 93 PERCENT WATER LEVEL (NARROW RANGE). ROD 31 WAS RECOVERED AND REALIGNED IN 1 HOUR AND 8 MINUTES. THIS IS A VIOLATION OF TECHNICAL SPECIFICATION 3.10.E.4 WHICH REQUIRES THESE ACTIONS TO BE COMPLETED WITHIN 1 HOUR, AND THUS IS REPORTABLE PER 10CFR50.73(A)(2)(I)(B). THIS EVENT IS ALSO REPORTABLE UNDER 10CFR50.73(A)(2)(IV) SINCE IT INVOLVED ACTUATION OF THE REACTOR PROTECTION SYSTEM. ROOT CAUSE OF THE DROPPED ROD WAS A FAULTY FUSE, WHICH WAS REPLACED. ROOT CAUSE OF THE OVERFEED INCIDENT WAS AN INADEQUATE MAINTENANCE PROCEDURE WHICH WAS REVISED TO PREVENT RECURRENCE OF THIS TYPE OF INCIDENT.

[34] COOK 1 DOCKET 50-315 LER 85-023 REV 01
UPDATE ON TRAVEL OF HEAVY LOAD OVER SPENT FUEL DUE TO NUREG 0612 MISINTERPRETATION.
EVENT DATE: 050285 REPORT DATE: 071487 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)

(NSIC 205433) ON MAY 2, 1985 WITH THE UNIT 1 REACTOR IN MODE 6 (REFUELING) AND UNIT 2 REACTOR IN MODE 1 (POWER OPERATION) AT 100 PERCENT THERMAL POWER, THE AUXILIARY BUILDING CRANE TRAVELED, UNLOADED, OVER THE SPENT FUEL POOL. THIS CONDITION WAS CONTRARY TO TECHNICAL SPECIFICATION 3.9.7 IN THAT THE CRANE'S LOAD BLOCK WAS AT THAT TIME DEFINED AS A HEAVY LOAD. THE CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE THE FAILURE OF PLANT PERSONNEL TO RECOGNIZE THAT THE MAIN LOAD BLOCK ON THE AUXILIARY BUILDING CRANE CONSTITUTED A HEAVY LOAD. IT WAS BELIEVED THAT HEAVY LOADS WERE APPLICABLE TO ITEMS BEING TRANSPORTED BY THE CRANE AND NOT COMPONENTS OF THE CRANE. THE CORRECTIVE ACTION TAKEN WAS TO CHANGE TECHNICAL SPECIFICATION 3.9.7 BY ADDING A FOOTNOTE WHICH DEFINES THE CONDITIONS NECESSARY IN ORDER TO PLACE THE MAIN LOAD BLOCK OVER THE SPENT FUEL POOL. THE PREVENTIVE ACTIONS WERE TO IMPOSE RESTRICTIONS ON THE CRANE AND PROVIDE TRAINING TO THE OPERATORS.

[35] COOK 1 DOCKET 50-315 LER 87-011
MAIN STEAM SAFETY VALVES OUT OF SPECIFICATION DUE TO APPARENT SETPOINT DRIFT.
EVENT DATE: 062487 REPORT DATE: 072387 NSSS: WE TYPE: PWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV

(NSIC 205561) BETWEEN JUNE 24 AT 1700 HOURS AND JUNE 25 AT 1400 HOURS, 1987, WITH THE UNIT 1 REACTOR IN MODE 1 (POWER OPERATION) AT 78 PERCENT THERMAL POWER, EIGHT OF THE TWENTY MAIN STEAM SAFETY VALVES (MSSV) LIFT SETPOINTS WERE FOUND TO BE OUT OF SPECIFICATION DURING SURVEILLANCE TESTING. THE MSSV LIFT SETPOINTS RANGED FROM 1 TO 9 PSI BELOW THE TECH SPEC REQUIRED RANGE. IN EACH CASE THE MSSV'S LIFT SETPOINTS WERE CORRECTED AND THE SAFETY VALVES LEFT OPERABLE PRIOR TO COMPLETION OF THE SURVEILLANCE TEST PROCEDURE (STP). THE APPARENT MSSV SETPOINT DRIFT COULD HAVE BEEN ATTRIBUTABLE TO TWO FACTORS, 1) TESTING METHOD, AND; 2) SETPOINT DRIFT DUE TO VALVE DESIGN/APPLICATION. THE INVESTIGATION CONCLUDED THAT THE OLD TESTING METHOD HAD A HIGH PROBABILITY OF CONTRIBUTING TO THE APPARENT MSSV SETPOINT DRIFT. THE IMMEDIATE CORRECTIVE ACTION, AS REQUIRED BY THE SURVEILLANCE TEST PROCEDURE, WAS TO RESET THE SAFETY VALVES SETPOINTS TO WITHIN THEIR SPECIFIED RANGES UTILIZING AN IMPROVED TESTING METHOD. TO PREVENT RECURRENCE FUTURE MSSV SETPOINTS WILL BE TESTED WITH THE IMPROVED TESTING METHOD. THIS WILL MORE ACCURATELY REFLECT THE MSSV SETPOINTS.

[36] COOK 1 DOCKET 50-315 LER 87-012
CONTAINMENT TYPE B&C LEAKAGE EXCEEDS LCO VALUE DUE TO DEGRADATION OF ISOLATION VALVE.
EVENT DATE: 070287 REPORT DATE: 072487 NSSS: WE TYPE: PWR

(NSIC 205529) ON JULY 2, 1987, AT 1400 HOURS, WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN), THE ACCUMULATED LEAKAGE FOUND WHILE PERFORMING THE TYPE B AND C LEAK RATE TESTS ON CONTAINMENT PENETRATIONS EXCEEDED THE L.C.O. VALUE (0.60 LA) OF TECHNICAL SPECIFICATION 3.6.1.2.B. THOSE CONTAINMENT ISOLATION VALVES THAT EXHIBIT EXCESSIVE LEAK RATES ARE BEING REPAIRED AND RETESTED TO ENSURE THE COMBINED LEAK RATES ARE WITHIN ALLOWABLE LIMITS. THE B AND C LEAK RATE TESTING IS STILL IN PROGRESS AND WILL NOT BE COMPLETED UNTIL THE END OF THE CURRENT REFUELING OUTAGE. THIS IS AN INTERIM REPORT. THE FINAL REPORT WILL BE SUBMITTED BY SEPTEMBER 30, 1987, FOLLOWING COMPLETION OF THE B AND C LEAK RATE TESTS. PREVIOUS OCCURRENCES OF A SIMILAR NATURE INCLUDE LERS: 50-315/83-072, 50-315/82-058, 50-315/81-025, 50-316/81-018, 50-315/81-011, 50-316/79-053, 50-315/79-020, 50-315/85-017, 50-316/86-009.

[37] COOK 2 DOCKET 50-316 LER 87-006
IMPROPERLY PERFORMED TECHNICAL SPECIFICATION SURVEILLANCE DUE TO A BREAKDOWN IN CONTROL OF WORK PROCESSES AND PLANT MODIFICATIONS.
EVENT DATE: 061587 REPORT DATE: 071487 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 1 (PWR)

(NSIC 205448) ON JUNE 15, 1987 AT 1800 HOURS, WITH UNIT 2 AT 90 PERCENT RATED THERMAL POWER, BOTH CONTAINMENT SPRAY (CTS) PUMPS WERE DECLARED INOPERABLE DUE TO MISSED TECHNICAL SPECIFICATION (T/S) SURVEILLANCES. IMPROPER LABELING OF THE TEST FLOW GAUGE METER FACE RESULTED IN SURVEILLANCE TESTS HAVING BEEN PERFORMED BELOW THE T/S 4.6.2.1.B MINIMUM FLOW REQUIREMENT SINCE THE INITIAL STARTUP OF THE UNIT (1978). A UNIT SHUTDOWN WAS INITIATED PER T/S 3.0.3 AND AN UNUSUAL EVENT WAS DECLARED AT 1855 HOURS. THE EAST CTS PUMP WAS SATISFACTORILY TESTED IN ACCORDANCE WITH T/S 4.6.2.1.B AT 2053 HOURS AT WHICH TIME THE UNIT SHUTDOWN AND UNUSUAL EVENT WERE TERMINATED. THE WEST CTS PUMP WAS SATISFACTORILY TESTED AND DECLARED OPERABLE AT 0038 HOURS ON JUNE 16, 1987. A REVIEW OF THE DATA DETERMINED THAT ALTHOUGH THE TEST FLOW RATE WAS BELOW THE T/S LIMITS, THE PUMPS HAD NOT DEGRADED BELOW THE ORIGINAL FACTORY TEST DATA AND THEIR PERFORMANCE WAS, AND STILL IS, WELL ABOVE THE T/S CRITERIA. BECAUSE NO PUMP DEGRADATION HAS OCCURRED, NO DECREASE IN CONTAINMENT SPRAY FLOW BELOW THAT ASSUMED IN THE ANALYSIS WOULD BE EXPECTED. BASED ON THE ABOVE INFORMATION, IT IS CONCLUDED THAT THE CONDITION DID NOT CONSTITUTE A SIGNIFICANT SAFETY PROBLEM AS DEFINED IN 10CFR50.59.

[38] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-010
INADEQUATE CHANNEL CHECK OF OTSG OPERATING LEVEL INSTRUMENTATION.
EVENT DATE: 061687 REPORT DATE: 072087 NSSS: BW TYPE: PWR

(NSIC 205558) ON JUNE 18, 1987, CRYSTAL RIVER UNIT 3 (CR-3) WAS OPERATING AT 88% OF RATED THERMAL POWER, GENERATING 761 MWE. DURING AN ENGINEERING REVIEW OF SURVEILLANCE PROCEDURE SP-300 "OPERATING DAILY SURVEILLANCE LOG" PRIOR TO A SPECIAL TEST, IT WAS DISCOVERED THAT THE REQUIREMENTS FOR OTSG OPERATING RANGE LEVEL RECORDERS DID NOT MEET THE TECH SPECS REQUIREMENTS FOR A MONTHLY CHANNEL CHECK. SP-300 ONLY SPECIFIES A MAXIMUM VALUE, IN ACCORDANCE WITH THE TECH SPECS REQUIRED MAXIMUM OTSG LEVEL. THE CAUSE IS ATTRIBUTED TO PERSONNEL ERROR. A PROCEDURE CHANGE TO SP-300 WAS MADE IN 1977 BY A SENIOR LICENSED UTILITY INDIVIDUAL, WHICH DELETED REQUIREMENTS FOR OTSG OPERATING RANGE LEVEL RECORDERS. THESE CHANGES DID NOT INCLUDE ADDITION OF THE PROPER TECH SPECS CHANNEL CHECK REQUIREMENTS. APPROPRIATE PLANT PROCEDURES WILL BE REVISED WITH THE APPROPRIATE ACCEPTANCE CRITERIA TO MEET THE REQUIREMENTS OF A TECH SPECS CHANNEL CHECK. THIS ACCEPTANCE CRITERIA HAS ALREADY BEEN DEVELOPED. THIS EVENT IS BEING REPORTED AS A CONDITION PROHIBITED BY PLANT TECH SPECS.

[39] DIABLO CANYON 2 DOCKET 50-323 LER 85-029
IMPROPER INSTALLATION OF REACTOR COOLANT PUMP SEAL INJECTION FLOW ORIFICES CAUSED CONTROLLED LEAKAGE IN EXCESS OF TECHNICAL SPECIFICATION LIMIT.
EVENT DATE: 082785 REPORT DATE: 072887 NSSS: WE TYPE: PWR

(NSIC 205509) ON AUGUST 27, 1985, UNIT 2 ENTERED MODE 2 (STARTUP), WITH CONTROLLED LEAKAGE EXCEEDING THE LIMIT OF TECHNICAL SPECIFICATION (TS) 3.4.6.2. PLANT PERSONNEL PERFORMING SURVEILLANCE TEST PROCEDURE (STP) M-54, "MEASUREMENT OF REACTOR COOLANT PUMP SEAL INJECTION FLOW," WERE UNAWARE THAT TWO OF THE FOUR FLOW MEASUREMENT ORIFICES (FE-144 AND FE-115) HAD PREVIOUSLY BEEN INSTALLED BACKWARDS IN THE REACTOR COOLANT PUMP (RCP) SEAL INJECTION LINES. ON JUNE 29, 1987, THE INCORRECT INSTALLATION WAS DISCOVERED BY INSTRUMENTATION AND CONTROLS TECHNICIANS. THE ENGINEERING DEPARTMENT DETERMINED THAT THE REVERSED ORIFICE PLATES HAD CAUSED AN ERRONEOUS RCP SEAL INJECTION FLOWRATE INDICATION APPROXIMATELY 20 PERCENT (1.5 GPM) LESS THAN THE ACTUAL FLOWRATE FOR EACH OF THE TWO SEAL INJECTION LINES. THE ORIFICES WERE INSTALLED CORRECTLY AND THE FLOW MEASUREMENT SURVEILLANCE TEST OF THE RCP SEAL INJECTION LINES WAS COMPLETED JULY

2, 1987. THE CAUSE OF THIS EVENT WAS LACK OF ADEQUATE DETAIL IN THE WRITTEN INSTRUCTIONS FOR ORIFICE INSTALLATION. ALL FLOW MEASUREMENT ORIFICES THAT PROVIDE DATA FOR TS-RELATED SURVEILLANCE TESTS WERE INSPECTED TO VERIFY PROPER ORIENTATION. FOUR OTHER ORIFICE PLATES WERE FOUND INSTALLED BACKWARDS. THESE INCORRECT INSTALLATIONS DID NOT ADVERSELY AFFECT ANY TS SURVEILLANCE TESTS OR SYSTEM PERFORMANCE.

[40] DIABLO CANYON 2 DOCKET 50-323 LER 86-014 REV 01
UPDATE ON DIESEL GENERATOR START AND LOADING DUE TO AN INCORRECTLY TERMINATED JUMPER.
EVENT DATE: 050686 REPORT DATE: 072787 NSSS: WE TYPE: PWR

(NSIC 205513) AT 1424 PDT, MAY 6, 1986, WHILE THE UNIT WAS IN MODE 1 (POWER OPERATION) AT 20 PERCENT POWER, DIESEL GENERATOR 2-1 STARTED ON BUS UNDERVOLTAGE AND ABNORMALLY LOADED ONTO 4 KV BUS G. DURING A MANUALLY INITIATED AUTOMATIC BUS TRANSFER, THE AUXILIARY POWER FEEDER BREAKER TRIPPED ON OVERCURRENT AND LOCKED OUT THE STARTUP POWER FEEDER BREAKER, RESULTING IN AN UNDERVOLTAGE CONDITION ON BUS G. ON MAY 7, 1986, PLANT PERSONNEL DETERMINED THAT A JUMPER WAS INCORRECTLY TERMINATED IN THE 4 KV BUS G AUTO TRANSFER SCHEME, WHICH ALLOWED THE DIESEL GENERATOR TO LOAD ONTO THE BUS WHEN THE AUXILIARY FEEDER BREAKER TRIPPED ON OVERCURRENT. ON MAY 7, 1986, THE JUMPER WAS CORRECTLY TERMINATED AND A FUNCTIONAL TEST WAS SATISFACTORILY PERFORMED. THE INCORRECT TERMINATION MOST LIKELY OCCURRED DURING FEBRUARY 1985, WHEN CONSTRUCTION TECHNICIANS WERE RETESTING CIRCUITRY IN THE 4 KV SWITCHGEAR THAT HAD BEEN PREVIOUSLY LIFTED TO REMOVE EXCESS PYROCRETE. SIMILAR TASKS PERFORMED BY THE TECHNICIANS WERE INSPECTED AND NO SIMILAR ERRORS FOUND. A TERMINAL-BY-TERMINAL INSPECTION OF ALL CONTROL CIRCUIT WIRING IN UNIT 2 VITAL 4 KV BUS SWITCHGEAR HAS BEEN COMPLETED WITH ONLY MINOR DISCREPANCIES FOUND. TO PREVENT RECURRENCE, PLANT PROCEDURES HAVE BEEN REVISED TO ENHANCE THE DETERMINATION/RETERMINATION PROCESS.

[41] DIABLO CANYON 2 DOCKET 50-323 LER 87-006
RHR PUMP 2-2 FAILED TO START DUE TO BENT SOLENOID MOUNTING BRACKET IN PUMP MOTOR BREAKER.
EVENT DATE: 041187 REPORT DATE: 072087 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 205450) THIS VOLUNTARY LER IS BEING SUBMITTED FOR INFORMATION PURPOSES ONLY AS DESCRIBED IN ITEM 19 OF SUPPLEMENT 1 TO NUREG-1022. WITH THE UNIT IN MODE 5 (COLD SHUTDOWN), RESIDUAL HEAT REMOVAL (RHR) PUMP 2-2 WAS INOPERABLE FROM 1308 PDT, APRIL 11, 1987, UNTIL 2134 PDT, APRIL 12, 1987. TECHNICAL SPECIFICATION 3.4.1.4.2 REQUIRES TWO OPERABLE RESIDUAL HEAT REMOVAL TRAINS, BUT ACTION A ALLOWS ONE TRAIN TO BE INOPERABLE IF CORRECTIVE ACTION IS INITIATED IMMEDIATELY TO RETURN THE REQUIRED TRAIN TO OPERABILITY. ON APRIL 12, 1987, AT 2039 PDT, RHR PUMP 2-2 FAILED TO START. AT 2134 PDT, RHR PUMP 2-2 WAS STARTED AFTER THE PROBLEM WITH THE BREAKER CLOSING SPRINGS WAS BELIEVED TO HAVE BEEN RESOLVED. AN ACTION REQUEST WAS INITIATED TO FURTHER INVESTIGATE THE BREAKER AFTER RCS MIDLOOP OPERATIONS, WHEN THE RHR SYSTEM IS NOT REQUIRED TO BE OPERABLE. ON APRIL 29, 1987, ELECTRICAL MAINTENANCE REMOVED THE RHR PUMP 2-2 MOTOR BREAKER IN RESPONSE TO THE ACTION REQUEST INITIATED ON APRIL 12, 1987. THE LOOSE SOLENOID AND BENT SOLENOID BRACKET WERE DISCOVERED ON APRIL 29, 1987, DURING A BENCH TEST AND PHYSICAL EXAMINATION OF THE BREAKER. THE LOOSE SOLENOID PROBLEM WAS CORRECTED. THE BREAKER TESTED SATISFACTORILY AND WAS RETURNED TO SERVICE. ON JUNE 2, 1987, RHR PUMP 2-2 AGAIN FAILED TO START.

[42] DIABLO CANYON 2 DOCKET 50-323 LER 87-011
INOPERABILITY OF BOTH RHR TRAINS DUE TO A FAULTED RELAY WHICH CAUSED A VOLTAGE TRANSIENT ON THE RHR AUTOCLOSURE INTERLOCK POWER SUPPLY.
EVENT DATE: 062987 REPORT DATE: 072987 NSSS: WE TYPE: PWR

VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 205609) AT 1829 PDT ON JUNE 29, 1987, WITH THE UNIT IN MODE 4 (HOT SHUTDOWN), A FAULTED COIL ON RELAY 2TC441HX INITIATED A VOLTAGE TRANSIENT ON INSTRUMENT POWER PANEL PY-24, WHICH RESULTED IN CLOSURE OF RESIDUAL HEAT REMOVAL (RHR) VALVE 8701. THE MOMENTARY VOLTAGE TRANSIENT RESULTED IN AN ACTUATION OF THE RHR AUTOCLOSURE INTERLOCK (ACI) FUNCTION SINCE THE ACI AND THE FAULTED RELAY SHARE A COMMON INSTRUMENT POWER SUPPLY. THE ACI ACTIVATION CAUSED RHR VALVE 8701 TO CLOSE. THE LICENSED OPERATORS NOTED THAT RHR PUMP 2-1 DISCHARGE TEMPERATURE WAS DECREASING RAPIDLY AND THEN OBSERVED RHR VALVE 8701 WAS SHUT. IN RESPONSE TO THE VALVE CLOSURE, RHR PUMP 2-1 WAS SECURED AND VALVE 8701 WAS REOPENED. RHR PUMP 2-1 WAS RESTARTED AT 1834 PDT, AND NO ABNORMAL PUMP SEAL LEAKAGE WAS OBSERVED. THE FOUR HOUR NON-EMERGENCY REPORT REQUIRED BY 10 CFR 50.72(B)(2)(III)(B) WAS MADE AT 1945 PDT. THE FAILURE OF THE RELAY WAS A RESULT OF THE COIL SHORTING. MELTED NYLON AROUND THE IRON CORE INTERFACE SURFACES AND SOME SLIGHT CORROSION OF THE SURFACES WAS NOTED. PGANDE IS PREPARING A LETTER FOR SUBMITTAL TO THE NRC REQUESTING AND JUSTIFYING REMOVAL OF THE RHR ACI FUNCTION. THE RHR ACI FUNCTION WILL BE REMOVED AFTER NRC STAFF REVIEW AND APPROVAL OF THE PROPOSED CHANGE.

[43] DIABLO CANYON 2 DOCKET 50-323 LER 87-013
AUTOSTART OF AFW PUMPS IN MODE 4 ON LOW-LOW STEAM GENERATOR LEVEL DUE TO PERSONNEL ERROR.
EVENT DATE: 070187 REPORT DATE: 073187 NSSS: WE TYPE: PWR

(NSIC 205610) AT 0052 PDT, JULY 1, 1987, WITH THE UNIT IN MODE 4 (HOT SHUTDOWN) AT 330 DEGREES FAHRENHEIT AND 360 PSIG AND THE REACTOR TRIP BREAKERS OPENED, STEAM GENERATOR (S/G) 2-2 WATER LEVEL DECREASED TO THE LOW-LOW S/G LEVEL SETPOINT, RESULTING IN AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION OF THE AUXILIARY FEEDWATER SYSTEM (AFW). THE OPERATORS OBSERVED THAT THE S/G LEVELS WERE SLOWLY DECREASING TO THEIR LOW-LOW SETPOINTS, BUT DID NOT RECOGNIZE THAT AN ESF ACTUATION WOULD RESULT WHEN THE REACTOR TRIP BREAKERS ARE OPENED. THIS EVENT WILL BE REVIEWED WITH OPERATIONS PERSONNEL. AN OPERATIONS INCIDENT REPORT WILL BE INITIATED STRESSING THE IMPORTANCE OF BEING AWARE OF POSSIBLE ESF ACTUATIONS EVEN WHEN THE REACTOR TRIP BREAKERS ARE OPENED. ALL STEAM AND FEEDWATER PIPING FOR STEAM GENERATORS 2-2 AND 2-3 WAS INSPECTED FOR LEAKAGE, AND NO ABNORMALITIES WERE FOUND.

[44] DRESDEN 2 DOCKET 50-237 LER 87-021
FAILURE TO PLACE CONDENSER PIT LEVEL SWITCH LS-2-4441-24B IN A TRIPPED CONDITION DUE TO PERSONNEL ERROR.
EVENT DATE: 061787 REPORT DATE: 071587 NSSS: GE TYPE: BWR
VENDOR: MAGNETROL, INC.

(NSIC 205439) AT 1330 HOURS ON JUNE 17, 1987, WHILE DRESDEN UNIT 2 WAS OPERATING AT 78% RATED THERMAL POWER, IT WAS DISCOVERED THAT MAIN CONDENSER HIGH-HIGH LEVEL SWITCH LS-2.4441.24B HAD NOT BEEN PLACED IN THE TRIPPED CONDITION AFTER BEING FOUND INOPERABLE DURING SURVEILLANCE TESTING. ALTHOUGH TECHNICAL SPECIFICATION (T.S.) 3.5.L.2 REQUIRES THIS ACTION, THE STATION CONTROL ROOM ENGINEER (SCRE) FAILED TO RECOGNIZE THIS WHEN INFORMED OF THE LEVEL SWITCH FAILURE ON JUNE 16, 1987. AS A RESULT, THE SWITCH WAS NOT PLACED IN A TRIPPED CONDITION UNTIL 26 HOURS AND 45 MINUTES AFTER IT WAS DISCOVERED INOPERATIVE. SAFETY SIGNIFICANCE WAS MINIMAL SINCE THE REDUNDANT PAIR OF MAIN CONDENSER PIT HIGH-HIGH LEVEL SWITCHES WERE OPERABLE, AND WOULD HAVE TRIPPED THE CIRCULATING WATER PUMPS AT THE FIVE FOOT LEVEL AS REQUIRED BY T.S. 4.5.L.1.C. ADDITIONAL SWITCHES WOULD HAVE PROVIDED CONTROL ROOM ALARMS AT THE ONE AND THREE FOOT LEVELS. CORRECTIVE ACTIONS INCLUDE REVIEW OF THIS EVENT WITH APPROPRIATE PERSONNEL AND PROCEDURE CHANGES TO CLARIFY T.S. REQUIREMENTS. A PREVIOUS EVENT INVOLVING FAILURE OF A

SIMILAR TYPE LEVEL SWITCH IS LISTED IN REPORTABLE OCCURRENCE 86-027 ON DOCKET 50-237.

[45] FARLEY 1 DOCKET 50-348 LER 87-011
INOPERABLE CONTROL ROOM FIRE DAMPERS DUE TO DESIGN DEFICIENCY.
EVENT DATE: 061187 REPORT DATE: 071387 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: FARLEY 2 (PWR)
VENDOR: RUSKIN MANUFACTURING COMPANY

(NSIC 205454) ON 6-5-87, FOUR FIRE DAMPERS IN THE CONTROL ROOM VENTILATION SYSTEM WERE DECLARED INOPERABLE DUE TO CONCERNS ABOUT THEIR ABILITY TO PERFORM UNDER ALL OPERATING CONDITIONS. A TESTING PROGRAM WAS ESTABLISHED. AS A RESULT OF THIS TESTING, ON 6-11-87 IT WAS DETERMINED THAT THE FIRE DAMPERS WOULD NOT FULLY CLOSE AND LATCH WITH AIR FLOW. THE DAMPERS CONTINUE TO NOT BE FULLY OPERABLE. TECHNICAL SPECIFICATION 3.7.12 REQUIRES THESE FIRE DAMPERS TO BE OPERABLE WITHIN SEVEN DAYS OR A SPECIAL REPORT MUST BE SUBMITTED WITHIN THE FOLLOWING 30 DAYS. THEREFORE, THIS SPECIAL REPORT IS BEING SUBMITTED. ALL TECHNICAL SPECIFICATION ACTION STATEMENT REQUIREMENTS FOR THE FIRE DAMPERS ARE BEING MET. A DESIGN CHANGE IS BEING DEVELOPED TO ENSURE THAT THESE DAMPERS FULLY CLOSE AND LATCH WITH AIR FLOW. THIS DESIGN CHANGE IS EXPECTED TO BE IMPLEMENTED WITHIN THE NEXT SIX MONTHS.

[46] FARLEY 1 DOCKET 50-348 LER 87-012
ENVIRONMENTAL QUALIFICATION OF WIRING SPLICES AND TERMINATIONS.
EVENT DATE: 072187 REPORT DATE: 073087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: FARLEY 2 (PWR)

(NSIC 205619) IN REVIEWING THE ENVIRONMENTAL QUALIFICATION (EQ) FINDINGS AT ANOTHER NUCLEAR PLANT, PROBLEMS WITH THE CONFIGURATION OF EQ SOLENOID VALVE SPLICES AND TERMINATIONS WERE NOTED. CONSEQUENTLY A REVIEW OF ENGINEERING SPECIFICATIONS AND INSTALLATION PRACTICES FOR THE FARLEY NUCLEAR PLANT WAS INITIATED. ON 7/21/87 IT WAS CONCLUDED THAT SUCH PROBLEMS MIGHT ALSO EXIST IN THE FARLEY NUCLEAR PLANT. IN AN EVALUATION OF POTENTIAL INOPERABILITY INVOLVING THE EQ SOLENOIDS, IT WAS DETERMINED THAT NO SYSTEM COVERED BY TECHNICAL SPECIFICATIONS WOULD BE RENDERED INOPERABLE AS A RESULT OF ADVERSE EFFECTS BY DESIGN BASIS ACCIDENTS ON THE SUBJECT SOLENOID VALVES. A JUSTIFICATION FOR CONTINUED OPERATION (JCO) WAS DEVELOPED, REVIEWED BY THE PLANT OPERATIONS REVIEW COMMITTEE, AND APPROVED ON 7/22/87. THIS EVENT WAS CAUSED BY THE INSTALLATION OF SPLICES AND TERMINATIONS UTILIZING METHODS NOT EVALUATED BY DESIGN FOR ENVIRONMENTAL QUALIFICATION. A JCO WAS APPROVED FOR THE EXISTING SOLENOID SPLICE OR TERMINATION CONFIGURATION. A SYSTEMATIC PLAN WAS DEVELOPED TO MODIFY SOLENOID VALVE SPLICES AND TERMINATIONS ACCESSIBLE DURING PLANT OPERATION. ALL OTHER EQ SOLENOID VALVE SPLICES AND TERMINATIONS WILL BE MODIFIED AS PLANT CONDITIONS PERMIT. FARLEY NUCLEAR PLANT IS CONTINUING ITS REVIEW OF EQ FILES, PROCEDURES, RECORDS, AND OTHER INSTALLATIONS RELATED TO EQ SPLICES AND TERMINATIONS.

[47] FERMI 2 DOCKET 50-341 LER 85-026 REV 01
UPDATE ON EMERGENCY EQUIPMENT COOLING WATER SYSTEM AUTOMATIC INITIATION DUE TO A PRESSURE TRANSIENT.
EVENT DATE: 061585 REPORT DATE: 072787 NSSS: GE TYPE: BWR

(NSIC 205511) ON JUNE 15, 1985 AT 0645 HOURS, WHILE IN OPERATIONAL CONDITION 4 AND PRIOR TO INITIAL CRITICALITY, THE EMERGENCY EQUIPMENT COOLING WATER (EEC) SYSTEM INITIATED AUTOMATICALLY. THIS OCCURRED WHEN A THIRD REACTOR BUILDING CLOSED COOLING WATER (RBCCL) SYSTEM PUMP WAS SHUT DOWN AS PART OF A NORMAL PUMP ROTATION SCHEME. THE RESULTING PRESSURE TRANSIENT CAUSED THE DIFFERENTIAL PRESSURE ACROSS THE EECW HEADERS TO DECREASE BELOW THE SETPOINT WHICH INITIATED THE EECW SYSTEM. TO PREVENT FURTHER EECW ACTUATION FROM THIS CAUSE, THE SYSTEM

OPERATING PROCEDURE WILL BE MODIFIED BASED ON THE RESULTS OF TESTING TO BE PERFORMED.

[48] FERMI 2 DOCKET 50-341 LER 87-029
INADEQUATE SURVEILLANCE COVERAGE OF A.C. POWER SOURCES DUE TO MISINTERPRETATION OF TECHNICAL SPECIFICATIONS.
EVENT DATE: 061587 REPORT DATE: 071587 NSSS: GE TYPE: BWR

(NSIC 205500) ON JUNE 15, 1987 A QUESTION AROSE AS TO HOW THE TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT FOR DETERMINING OPERABILITY OF THE CIRCUITS BETWEEN THE OFFSITE TRANSMISSION NETWORK AND THE ONSITE CLASS 1E DISTRIBUTION SYSTEM WAS BEING IMPLEMENTED. A SUBSEQUENT INVESTIGATION REVEALED THAT THE PROCEDURE IMPLEMENTING THIS REQUIREMENT WAS INADEQUATE. THIS RESULTED IN NOT FULLY COMPLYING WITH THE TECHNICAL SPECIFICATION ACTION STATEMENTS WHEN AN EMERGENCY DIESEL GENERATOR WAS INOPERABLE. THE CAUSE OF THIS EVENT WAS A MISINTERPRETATION OF THE TECHNICAL SPECIFICATION. THIS RESULTED IN AN INADEQUATE PROCEDURE. IMMEDIATE CORRECTIVE ACTION CONSISTED OF ISSUING A NIGHT ORDER INSTRUCTING OPERATORS ON HOW TO CORRECTLY PERFORM THE SURVEILLANCE. IN ADDITION. THE SURVEILLANCE PROCEDURE WILL BE REVISED TO CORRECTLY REFLECT THE INTENT OF THE SURVEILLANCE REQUIREMENT.

[49] FERMI 2 DOCKET 50-341 LER 87-024
PLANT SHUTDOWN DUE TO INOPERABILITY OF A REACTOR RECIRCULATION SYSTEM LOOP.
EVENT DATE: 062587 REPORT DATE: 072487 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 205532) ON JUNE 24, 1987 AT 1316 HOURS ARCING WAS OBSERVED BETWEEN A WORN BRUSH AND THE ANODE SLIP RINGS ON THE MOTOR-GENERATOR SET EXCITER OF THE B REACTOR RECIRCULATION PUMP (RCS). PUMP SPEED WAS PROMPTLY REDUCED AND THE EXCITER WAS TRIPPED AT 1321 HOURS. IN COMPLIANCE WITH TECHNICAL SPECIFICATION ACTION STATEMENT REQUIREMENTS, THE PLANT WAS PLACED IN HOT SHUTDOWN AT 0120 HOURS ON JUNE 25, 1987. INVESTIGATION DETERMINED THAT SOME EXICTER BRUSHES SHOWED EXCESSIVE WEAR. AS COPRECTIVE ACTION, ALL THE BRUSHES WERE INSPECTED ON THE A AND B RCS PUMPS MG SET EXCITERS AND REPLACED. THE BRUSHES AND BRUSH RIGGINGS WILL BE INSPECTED FOR ABNORMAL WEAR AGAIN AT AN INCREASED FREQUENCY. A DESIGN CHANGE IS BEING PREPARED TO INSTALL TRANSPARENT COVERS SO THAT THE BRUSHES CAN BE INSPECTED WITHOUT REQUIRING REMOVAL OF THE COVER PLATES ON THE BRUSH COMPARTMENT.

[50] FERMI 2 DOCKET 50-341 LER 87-025
ACTUATION OF THE REACTOR PROTECTION SYSTEM DUE TO A TRANSIENT IN COMMON REFERENCE LEG.
EVENT DATE: 062587 REPORT DATE: 072087 NSSS: GE TYPE: BWR

(NSIC 205533) ON JUNE 25, 1987 AT 1456 HOURS THE REACTOR PROTECTION SYSTEM ACTUATED AND SEVERAL ENGINEERED SAFETY FEATURES WERE CHALLENGED. THIS EVENT OCCURRED AS AN INSTRUMENT REPAIRMAN WAS OPENING THE ISOLATION VALVE TO A NEWLY REPLACED TRANSMITTER ON A COMMON REFERENCE LEG. DUE TO A COMBINATION OF TRAPPED AIR IN TRANSMITTER AND DIFFICULTY IN OPERATING THE ISOLATION VALVE SMOOTHLY, A PRESSURE TRANSIENT OCCURRED. THIS AFFECTED THE REACTOR WATER LEVEL TRANSMITTERS AND CAUSED A REACTOR WATER LEVEL 1 TRIP SIGNAL. WHILE THIS TYPE OF EVENT HAS OCCURRED PREVIOUSLY, THE INSTALLATION OF A NEW DRY TRANSMITTER MADE THIS TYPE OF OCCURRENCE MORE PROBABLE. THEREFORE MANAGEMENT DECIDED TO PERFORM THE REPLACEMENT DURING A SHUTDOWN TO MINIMIZE THE EFFECT ON THE PLANT. IN ORDER TO MINIMIZE THE POSSIBILITY OF RECURRENCE OF THIS EVENT, MULTI-TURN NEEDLE VALVES WILL BE INSTALLED TO IMPROVE THE REPAIRMAN'S CONTROL WHEN RETURNING INSTRUMENTS ON THE COMMON REFERENCE LEGS TO SERVICE.

[51] FERMI 2 DOCKET 50-341 LER 87-027
INADVERTENT INCREASE IN REACTOR COOLANT TEMPERATURE DURING OPERATIONAL CONDITION 4 (COLD SHUTDOWN) DUE TO PERSONNEL ERROR.
EVENT DATE: 062687 REPORT DATE: 072787 NSSS: GE TYPE: BWR

(NSIC 205556) ON JUNE 26, 1987, WHILE IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN), IT WAS NOTICED AT 1505 HOURS, THAT THE REACTOR COOLANT TEMPERATURE HAD EXCEEDED, 200 DEGREES FAHRENHEIT. THIS PLACED THE REACTOR INTO OPERATIONAL CONDITION 3 (HOT SHUTDOWN). CONTROL ROOM OPERATORS IMMEDIATELY TOOK CORRECTIVE ACTION TO DECREASE REACTOR COOLANT TEMPERATURE BELOW 200 DEGREES FAHRENHEIT BY UTILIZING DIVISION 1 OF RHR IN SHUTDOWN COOLING MODE. THE SURVEILLANCE TESTS REQUIRED TO BE PERFORMED BY TECHNICAL SPECIFICATIONS BEFORE ENTRY INTO OPERATIONAL CONDITION 3 WERE NOT COMPLETE. THIS EVENT WAS CAUSED BY A PERSONNEL ERROR. CONTRARY TO APPROVED PROCEDURES, THE OPERATOR (UNDER INSTRUCTION) ASSIGNED TO MONITOR AND CONTROL THE REACTOR TEMPERATURE DID NOT TAKE TIMELY ACTIONS TO REPORT OR CONTROL THE INCREASE IN THE REACTOR COOLANT TEMPERATURE. FOLLOWING AN INTERNAL INVESTIGATION, SEVERAL CORRECTIVE ACTIONS WERE TAKEN. THESE ACTIONS CONSISTED OF DISCIPLINARY ACTIONS AGAINST PERSONNEL INVOLVED, PROPOSED MODIFICATIONS TO CONTROL ROOM ANNUNCIATOR AND DISPLAYS, REVISION OF PROCEDURES, ISSUANCE OF GUIDANCE TO PLANT PERSONNEL, EXAMINATION OF SHIFT PERSONNEL ON THEIR DUTIES, AND COMMUNICATION WITH ALL SITE PERSONNEL ON THIS INCIDENT.

[52] FERMI 2 DOCKET 50-341 LER 87-028
EMERGENCY EQUIPMENT COOLING WATER SYSTEM ACTUATION DUE TO INADEQUATE PROCEDURE REVISION.
EVENT DATE: 062787 REPORT DATE: 072787 NSSS: GE TYPE: BWR

(NSIC 205534) ON JUNE 27, 1987 DURING THE PERFORMANCE OF A SURVEILLANCE OF THE OPERABILITY OF THE PUMPS AND VALVES IN THE EMERGENCY EQUIPMENT COOLING WATER SYSTEM, AN AUTOMATIC INITIATION OF THE SYSTEM OCCURRED. THE CONTROL ROOM OPERATOR MANUALLY OVERRODE THE LOGIC AND RESTORED THE COOLING TO THE NON-ESSENTIAL LOADS. THE ROOT CAUSE OF THIS EVENT WAS PROCEDURAL ERROR. THIS ACTUATION WAS AN UNANTICIPATED CONSEQUENCE OF A TEMPORARY CHANGE IN THE SURVEILLANCE PROCEDURE. THE PROCEDURE WILL BE REVISED TO CLARIFY THAT THE INITIATION IS AN EXPECTED RESULT OF THE PROCEDURE.

[53] FT. CALHOUN 1 DOCKET 50-285 LER 87-017 REV 01
UPDATE ON VIOLATION OF EMERGENCY DIESEL GENERATOR TEST INTERVAL REQUIREMENTS.
EVENT DATE: 050687 REPORT DATE: 072087 NSSS: CE TYPE: PWR
VENDOR: GENERAL MOTORS

(NSIC 205524) THE ANNUAL INSPECTION OF EMERGENCY DIESEL GENERATOR D-2 WAS SCHEDULED FOR JANUARY, 1987. DUE TO THE REFUELING SHUTDOWN SCHEDULED TO COMMENCE IN EARLY MARCH, 1987, OPPD PERSONNEL BEGAN INVESTIGATING THE ALLOWABLE DELAY FOR PERFORMANCE OF THE SURVEILLANCE. DURING THIS REVIEW IT WAS DISCOVERED THAT DELAY UNTIL REFUELING WAS APPROPRIATE BASED UPON COMPLETION DATE OF THE TEST IN 1986. THE 1986 TEST WAS CONDUCTED DURING THE WEEK OF FEBRUARY 24, 1986, BEING RETURNED TO SERVICE MARCH 3, 1986. HOWEVER, IT WAS ALSO DISCOVERED THAT THE ANNUAL INSPECTION OF EMERGENCY DIESEL GENERATOR D-2, SCHEDULED FOR JANUARY, 1986, COINCIDED WITH THE PLANT START UP FOF THE TENTH OPERATING CYCLE. DUE TO AN ASSUMPTION THAT THE TEST CARRIED A TIME ALLOWANCE OF PLUS OR MINUS TWENTY-FIVE PERCENT, THE TEST WAS POSTPONED UNTIL THE LAST WEEK IN FEBRUARY. DURING CONVERSATIONS WITH THE NUCLEAR REGULATORY COMMISSION ON FEBRUARY 28, 1987, CONCERNING DELAY, IF NECESSARY, OF THE 1987 TEST, IT WAS DETERMINED THAT THE TEST INTERVAL REQUIREMENTS MAY NOT BE INTERPRETED IN THAT MANNER. USING THE REVISED INTERPRETATION, TECHNICAL SPECIFICATION 3.7(1D) WAS VIOLATED. TECHNICAL SPECIFICATION 3.7(ID) REQUIRES THE EMERGENCY DIESEL GENERATORS TO, BE GIVEN A THOROUGH INSPECTION AT LEAST ANNUALLY FOLLOWING THE MANUFACTURERS RECOMMENDATIONS FOR THIS CLASS OF STANDBY SERVICE.

[54] FT. CALHOUN 1 DOCKET 50-285 LER 87-015 REV 01
UPDATE ON MOMENTARY VOLTAGE LOSS FOLLOWING AUTO TRANSFER OF INVERTER TO ALTERNATE AC SOURCES.
EVENT DATE: 052087 REPORT DATE: 061987 NSSS: CE TYPE: PWR

(NSIC 205523) WITH THE PLANT IN MODE 5 (REFUELING SHUTDOWN), AT 1311 CDT ON MAY 20, 1987, OPERATORS CLOSED THE CIRCUIT BREAKER WHICH SUPPLIES POWER TO THE CONTROL ROD CLUTCH POWER SUPPLIES. THE CLOSING OF THE CIRCUIT BREAKER INITIATED A TRANSIENT ON INSTRUMENT BUS "D" WHICH CAUSED INVERTER "D" TO AUTOMATICALLY TRANSFER TO ITS ALTERNATE AC SOURCE. DURING THE AUTOMATIC TRANSFER A MOMENTARY VOLTAGE LOSS ON INSTRUMENT BUS "D" OCCURRED. INSTRUMENT BUS "D" SUPPLIES POWER TO THE PPLS-B/BLOCK RELAY WHICH IS IN OPERATION DURING REFUELING SHUTDOWN. THE MOMENTARY VOLTAGE LOSS ON THE "D" INSTRUMENT BUS RESULTED IN A POWER LOSS TO THE "B" BLOCKING RELAY. REMOVAL OF POWER FROM EITHER THE "A" OR "B" BLOCKING RELAYS PERMITS INITIATION OF PRESSURIZER PRESSURE LOW SIGNAL (PPLS). INITIATION OF PPLS ACTUATES SEVERAL ENGINEERED SAFETY FEATURES. HAD THIS EVENT OCCURRED DURING POWER OPERATION, NO ESF ACTUATION WOULD HAVE RESULTED. SINCE THE PLANT WAS IN MODE 5, ONLY EQUIPMENT ASSOCIATED WITH THE VENTILATION ISOLATION ACTUATION SIGNAL (VIAS) ACTUALLY OPERATED. PLANT SYSTEMS INVOLVED IN THIS INCIDENT OPERATED WITHIN THEIR DESIGN BASIS WITH NO EQUIPMENT DAMAGE OR FAILURE. NO TECHNICAL SPECIFICATION VIOLATIONS OR OPERATOR ERRORS OCCURRED. IN AN ATTEMPT TO DETERMINE THE ROOT CAUSE OF THE EVENT, TESTING PROCEDURES WERE PREPARED AND CONDUCTED IN AN ATTEMPT TO DUPLICATE THE EVENT.

[55] FT. CALHOUN 1 DOCKET 50-285 LER 87-023
RADIATION MONITOR OUT OF SERVICE.
EVENT DATE: 062587 REPORT DATE: 072787 NSSS: CE TYPE: PWR

(NSIC 205525) DURING A ROUTINE MONTHLY NRC INSPECTION ON JUNE 25, 1987, THE NRC INSPECTOR NOTED THAT THE WIDE RANGE NOBLE GAS STACK RADIATION MONITORS (RM-063L, -063M AND -063H) WERE NOT IN SERVICE. THE INOPERABLE MONITORS WERE IN VIOLATION OF THE TECHNICAL SPECIFICATION 2.21 REQUIREMENT THAT WITH ONE OR MORE MONITORS INOPERABLE FOR GREATER THAN SEVEN DAYS WHILE IN MODES 1, 2 OR 3, A REPORT BE SENT TO THE NRC WITHIN 14 DAYS OF THE MONITOR(S) BECOMING INOPERABLE. THE MONITORS HAD BEEN REMOVED FROM SERVICE DURING MAY, 1987, WHILE THE PLANT WAS IN A REFUELING AND MAINTENANCE OUTAGE. ON JUNE 2, 1987, WITH THE MONITORS STILL INOPERABLE, THE PLANT ENTERED MODE 3. THIS PLACED THE PLANT IN A LIMITING CONDITION FOR OPERATION. AFTER SEVEN DAYS OF INOPERABILITY, A SPECIAL REPORT TO THE NRC IS REQUIRED WITHIN SEVEN ADDITIONAL DAYS. THIS TIME FRAME WAS NOT MET. TO PREVENT FUTURE OCCURRENCES, THE OPERATING INSTRUCTION FOR REACTOR COOLANT SYSTEM STARTUP HAS BEEN REVISED TO INCLUDE A PREREQUISITE THAT THE REQUIREMENTS OF TECHNICAL SPECIFICATION 2.21 ARE MET. ADDITIONALLY, A MEMORANDUM WAS SENT FROM THE SUPERVISOR-OPERATIONS TO ALL OPERATIONS PERSONNEL TO STRESS THE NEED FOR COMPLYING WITH THE TECHNICAL SPECIFICATIONS REQUIRING SPECIAL REPORTING.

[56] GRAND GULF 1 DOCKET 50-416 LER 87-009
REACTOR SCRAM DUE TO RELAY FAILURE.
EVENT DATE: 062987 REPORT DATE: 072987 NSSS: GE TYPE: BWR
VENDOR: AGASTAT RELAY CO.

(NSIC 205546) ON JUNE 29, 1987 AGASTAT RELAY N62-R33 SUSTAINED AN INTERMITTENT FAILURE WHICH CAUSED THE MAIN STEAM INLET VALVE (N62-F001B) TO STEAM JET AIR EJECTOR (SJAE) "B" TO CLOSE. THE CLOSURE OF THIS VALVE CAUSED A LOSS OF CONDENSER VACUUM RESULTING IN A MAIN TURBINE TRIP AND REACTOR SCRAM. THE CLOSURE OF VALVE N62-F001B PREVENTED THE SJAE FROM REMOVING NON-CONDENSABLE GASES FROM THE MAIN CONDENSER. IN ADDITION, MOTOR OPERATED VALVE N62-F003B FAILED TO CLOSE ALLOWING REVERSE FLOW THROUGH THE OFF-GAS SYSTEM BACK INTO THE CONDENSER. THESE TWO FAILURES COMBINED TO DECREASE MAIN CONDENSER VACUUM RESULTING IN A MAIN TURBINE TRIP AND REACTOR SCRAM ON THE TURBINE STOP VALVE FAST CLOSURE SIGNAL.

FOLLOWING THE SCRAM THE SJAES WERE SECURED WHICH CLOSED THE N62-F003B VALVE AND TERMINATED THE LOSS OF VACUUM EVENT. VACUUM STABILIZED AT APPROXIMATELY 21 INCHES MERCURY. DURING THE SCRAM RECOVERY, A DIVISION II GROUP 8 AUTOMATIC ISOLATION OCCURRED WHEN OPERATORS PREPARED TO PLACE THE REACTOR WATER CLEANUP (RWCU) SYSTEM INTO THE BLOWDOWN MODE OF OPERATION. THE RELAY WAS REPLACED. A MAINTENANCE WORK ORDER WAS INITIATED TO INVESTIGATE THE POTENTIAL PROBLEM WITH FLOW SWITCHES ASSOCIATED WITH LOW STEAM FLOW FROM THE SECOND STAGE AIR EJECTORS AND IS SCHEDULED TO BE WORKED IN AN UPCOMING PLANT OUTAGE.

[57]   HATCH 1   DOCKET 50-321   LER 87-010
PERSONNEL ERROR RESULTS IN REDUCTION OF VESSEL WATER LEVEL AND ESF ACTUATIONS.
EVENT DATE: 061587   REPORT DATE: 071587   NSSS: GE   TYPE: BWR

(NSIC 205449) ON 6/15/87 AT APPROXIMATELY 1456 CDT, WHILE IN THE REFUELING MODE, OPERATIONS PERSONNEL WERE PERFORMING LOGIC TESTING AND VALVE OPERABILITY TESTING ON ONE TRAIN OF THE RESIDUAL HEAT REMOVAL (RHR EIIS CODE BO) SYSTEM. THE OTHER TRAIN OF THE RHR SYSTEM WAS IN THE SHUTDOWN COOLING (SDC) MODE OF OPERATION. DURING THE TESTING, OPERATIONS PERSONNEL ATTEMPTED TO EQUALIZE PRESSURE AROUND A CHECK VALVE BY OPENING OTHER SYSTEM VALVES. A FLOW PATH FROM THE REACTOR VESSEL TO THE TORUS WAS INADVERTENTLY ESTABLISHED AND REACTOR WATER LEVEL DECREASED. A REACTOR SCRAM AND A PARTIAL PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS EIIS CODE JM) VALVE GROUP 2 ISOLATION OCCURRED. THE ROOT CAUSE OF THIS EVENT WAS COGNITIVE PERSONNEL ERROR BY OPERATIONS PERSONNEL. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED: 1) RESTORING REACTOR WATER LEVEL, 2) COUNSELING OF INVOLVED PERSONNEL, 3) DISCUSSING THE EVENT WITH OPERATIONS PERSONNEL, 4) INVESTIGATING VALVE ISOLATIONS, 5) EVALUATING THE NEED FOR VALVE INTERLOCKS, AND 6) ASSIGNING A SHIFT TEST COORDINATOR TO THE CONTROL ROOM.

[58]   HATCH 2   DOCKET 50-366   LER 87-004
CAUSES OF HIGH PRESSURE COOLANT INJECTION INOPERABILITY AND ESF ACTUATIONS UNDER REVIEW.
EVENT DATE: 061687   REPORT DATE: 071687   NSSS: GE   TYPE: BWR
VENDOR: WOODWARD GOVERNOR COMPANY

(NSIC 205466) ON 6/16/87 AT APPROXIMATELY 1724 CDT, PLANT OPERATIONS PERSONNEL WERE PERFORMING A SURVEILLANCE ON THE HIGH PRESSURE COOLANT INJECTION (HPCI EIIS CODE BJ) AND DETERMINED THE SYSTEM WAS BEHAVING ERRATICALLY. THIS ERRATIC BEHAVIOR WAS SUCH THAT THE HPCI WAS DETERMINED TO BE INCAPABLE OF PERFORMING ITS INTENDED SAFETY FUNCTION. CORRECTIVE MAINTENANCE WAS INITIATED ON THE HPCI SYSTEM. DURING SYSTEM TESTING ON 6/18/87, TWO PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS EIIS CODE JM) HPCI STEAM SUPPLY VALVE ISOLATIONS OCCURRED. THESE ISOLATIONS WERE UNPLANNED ACTUATIONS OF AN ENGINEERED SAFETY FEATURE (ESF). THE CAUSE FOR THE HPCI SYSTEM BEING INCAPABLE OF PERFORMING ITS INTENDED SAFETY FUNCTION AND THE TWO ESF ACTUATIONS ARE CURRENTLY UNDER INVESTIGATION. LONG TERM CORRECTIVE ACTIONS TO PREVENT RECURRENCE OF THE EVENTS DESCRIBED IN THIS LER WILL BE DEVELOPED WHEN ALL ON SITE INVESTIGATIVE ACTIVITIES HAVE CONCLUDED. THE RESULTS OF THE INVESTIGATIONS AND THE CORRECTIVE ACTIONS TAKEN WILL BE DOCUMENTED IN AN UPDATE TO THIS REPORT. IT IS ANTICIPATED THAT THIS UPDATE WILL BE DEVELOPED BY APPROXIMATELY 10/30/87.

[59]   HOPE CREEK 1   DOCKET 50-354   LER 87-025
NON-CONSERVATIVE LIQUID EFFLUENT SAMPLING FREQUENCY DUE TO INCONSISTENCY BETWEEN TECHNICAL SPECIFICATION REQUIREMENTS AND PROCEDURAL REQUIREMENTS.
EVENT DATE: 061187   REPORT DATE: 071387   NSSS: GE   TYPE: BWR

(NSIC 205462) A PROCEDURAL DEFICIENCY RESULTED IN THE STATION OPERATING IN VIOLATION OF TECHNICAL SPECIFICATION 3.4.11.1. THIS INCIDENT WAS THE RESULT OF AN INCONSISTENCY WHICH EXISTED BETWEEN A CHEMISTRY DEPARTMENT PROCEDURE AND

TECHNICAL SPECIFICATIONS WITH REGARD TO LIQUID EFFLUENT SAMPLING FREQUENCY. THE ROOT CAUSE OF THIS INCONSISTENCY WAS AN INADEQUATE REVIEW OF FINAL DRAFT TECHNICAL SPECIFICATION 3.4.11.1 BY THE TECH SPEC COORDINATOR AND CHEMISTRY DEPARTMENT PERSONNEL RESPONSIBLE FOR REVIEWING TECHNICAL SPECIFICATION CHANGES AGAINST PROCEDURAL AND ADMINISTRATIVE REQUIREMENTS. IMMEDIATE CORRECTIVE ACTIONS INCLUDED REVISING THE AFFECTED CHEMISTRY DEPARTMENT PROCEDURE AND REVIEWING ALL OTHER CHEMISTRY DEPARTMENT PROCEDURES AGAINST TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS. LONG TERM CORRECTIVE ACTIONS CONSIST OF REVIEWING CURRENT STATION ADMINISTRATIVE PROGRAMS TO ENSURE ADEQUATE DIRECTION IS PROVIDED IN IDENTIFYING TECHNICAL SPECIFICATION CHANGES TO AFFECTED DEPARTMENTS.

[60]   HOPE CREEK 1   DOCKET 50-354   LER 87-026
TECHNICAL SPECIFICATION VIOLATION - MOV THERMAL OVERLOADS INSTALLED WITHOUT BYPASS CAPABILITY DUE TO INADEQUATE TECH SPEC AND DESIGN REVIEWS.
EVENT DATE: 061587   REPORT DATE: 071587   NSSS: GE   TYPE: BWR

(NSIC 205463) WHILE PREPARING TO PERFORM A ROUTINE 18 MONTH FUNCTIONAL TEST OF SERVICE WATER SCREEN WASH VALVE OPERATORS (MOVS), IT WAS DETERMINED THAT THE MOVS DID NOT HAVE THERMAL OVERLOAD BYPASS CIRCUITRY INSTALLED AS REQUIRED BY TECHNICAL SPECIFICATION 3.8.4.2. SUBSEQUENT INVESTIGATION DETERMINED THAT THE MOVS SHOULD HAVE BEEN DESIGNED WITH BYPASS CIRCUITRY, BUT WERE NOT. WHILE INVESTIGATING THIS INCIDENT FOUR ADDITIONAL SAFETY AUXILIARY COOLING SYSTEM (SACS) MOVS WERE FOUND WITH BYPASS CIRCUITS DEFEATED DUE TO A PREVIOUSLY IMPLEMENTED DESIGN CHANGE. THE ROOT CAUSE OF THIS OCCURRENCE WAS AN INADEQUATE DESIGN REVIEW AND IMPLEMENTATION OF TECHNICAL SPECIFICATION REQUIREMENTS DURING THE FINAL DRAFT PHASE OF TECHNICAL SPECIFICATION PREPARATION. IMMEDIATE CORRECTIVE ACTIONS INCLUDED INSTALLING A TEMPORARY MODIFICATION IN THE MOV BYPASS CIRCUITS AND INITIATING A DESIGN CHANGE TO ADD PERMANENT BYPASS CIRCUITRY TO ALL SIMILAR MOVS IN TECHNICAL SPECIFICATION 3.8.4.2. IN ADDITION, ALL DEPARTMENTS ARE REVIEWING THE TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS AGAINST PROCEDURES AND SCHEDULING PROGRAMS. NUCLEAR SYSTEMS ENGINEERING IS REVIEWING THE DESIGN CHANGE PROCESS TO IDENTIFY APPROPRIATE CORRECTIVE ACTIONS.

[61]   HOPE CREEK 1   DOCKET 50-354   LER 87-027
SPURIOUS ISOLATION OF HIGH PRESSURE COOLANT INJECTION (HPCI) INBOARD STEAM ISOLATION VALVE DUE TO FAILED TEMPERATURE MODULE.
EVENT DATE: 062687   REPORT DATE: 072787   NSSS: GE   TYPE: BWR
VENDOR: RILEY COMPANY, THE - PANALARM DIVISION

(NSIC 205535) A CHANNEL 'C' HPCI ISOLATION OCCURRED WHEN A RILEY TEMPMATIC 86 TEMPERATURE MODULE IN THE STEAM LEAK DETECTION SYSTEM FAILED, CAUSING SPURIOUS UPSCALE TEMPERATURE SPIKES. THE TEMPERATURE MODULE IN QUESTION (E41-N602C) PROVIDES HPCI CHANNEL 'C' HIGH ROOM TEMPERATURE ISOLATION. HPCI INBOARD STEAM SUPPLY VALVE HV-F002 ISOLATED UPON RECEIPT OF THE SPURIOUS HIGH TEMPERATURE SIGNAL. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE AN ISOLATED FAILURE OF THE SUBJECT TEMPERATURE MODULE. SHORT TERM CORRECTIVE ACTIONS CONSISTED OF REPLACING THE FAILED TEMPERATURE MODULE. LONG TERM CORRECTIVE ACTIONS IDENTIFIED FROM A PREVIOUS EVENT INCLUDE REPLACING ALL CURRENTLY INSTALLED RILEY TEMPMATIC 86 TEMPERATURE MODULES WITH A NEWER MODEL.

[62]   HOPE CREEK 1   DOCKET 50-354   LER 87-028
ISOLATIONS OF THE RWCU SYSTEM ON HIGH DIFFERENTIAL FLOW DUE TO DESIGN DEFICIENCIES.
EVENT DATE: 062987   REPORT DATE: 072987   NSSS: GE   TYPE: BWR

(NSIC 205536) ON JUNE 29, 1987 AT 2135 HOURS, THE PLANT WAS IN OPERATIONAL CONDITION 1 (POWER OPERATION) AT 100% POWER - 1060 MWE. A RWCU SYSTEM ISOLATION OCCURRED WHILE ATTEMPTING TO PLACE THE "A" RWCU FILTER DEMINERALIZER (F/D) IN

SERVICE. THE RWCU SYSTEM ISOLATED ON HIGH SYSTEM DIFFERENTIAL FLOW DURING AN ATTEMPT TO BACKWASH THE RESIN TRAP. THE ROOT CAUSE OF THIS EVENT WAS DESIGN DEFICIENCIES WHICH REQUIRED COMPENSATORY PROCEDURAL CHANGES. ON JULY 1, 1987 AT 1611 HOURS, THE PLANT WAS IN OPERATIONAL CONDITION 1 (POWER OPERATION) AT 100% POWER - 1060 MWE. A RWCU SYSTEM ISOLATION OCCURRED WHILE ATTEMPTING TO PLACE THE "A" RWCU F/D IN SERVICE. THE RWCU SYSTEM ISOLATED ON HIGH DIFFERENTIAL FLOW. THE ROOT CAUSE OF THIS EVENT WAS LEAK-BY THROUGH THE RWCU PRECOAT INLET VALVE. DESIGN CHANGES TO MODIFY THE RWCU SYSTEM LOGIC TO REDUCE THE NUMBER OF F/D BACKWASH OPERATIONS AND TO PROVIDE DOUBLE VALVE ISOLATION BETWEEN THE HIGH AND LOW PRESSURE SECTIONS OF THE SYSTEM ARE UNDER MANAGEMENT REVIEW. PROCEDURAL AND OPERATIONAL CHANGES HAVE BEEN MADE TO IMPROVE THE PERFORMANCE OF RESIN TRAP BACKWASHING.

[63] INDIAN POINT 2 DOCKET 50-247 LER 87-008
CENTRAL CONTROL ROOM CHARCOAL FILTERS - LOW METHYL-IODIDE ADSORPTION EFFICIENCY.
EVENT DATE: 062387 REPORT DATE: 072387 NSSS: WE TYPE: PWR
VENDOR: MINE SAFETY APPLIANCES COMPANY

(NSIC 205503) ON JUNE 3, 1987, AS PART OF A ROUTINE SURVEILLANCE, CHARCOAL IN BOTH CHARCOAL FILTERS (FLT) IN THE CENTRAL CONTROL ROOM (CCR) FILTRATION SYSTEM WAS REPLACED, AND A REPRESENTATIVE SAMPLE OF USED CHARCOAL FROM EACH OF TWO FILTERS WAS SENT TO CALLERY CHEMICAL CORPORATION, EVANS CITY, PA, FOR ANALYSIS. ON JUNE 23, 1987, THE LABORATORY ANALYSIS RESULTS WERE RECEIVED, REVEALING THAT THE "AS FOUND" ADSORPTION EFFICIENCY FOR METHYL-IODIDE WAS 93.66% FOR FILTER NO. 1 AND 79.07% FOR FILTER NO. 2. SINCE THE CHARCOAL IN BOTH CHARCOAL FILTERS HAD BEEN REPLACED ON JUNE 3, 1987, THERE WAS NO NEED FOR FURTHER CORRECTIVE ACTION. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED.

[64] INDIAN POINT 2 DOCKET 50-247 LER 87-009
REACTOR TRIP DUE TO STEAM GENERATOR LEVEL RELAY MALFUNCTION.
EVENT DATE: 062787 REPORT DATE: 072687 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 205519) ON JUNE 27, 1987, WHILE ADJUSTING STEAM GENERATOR (SG) LOW-LOW LEVEL SETPOINTS, A REACTOR TRIP OCCURRED WHEN REACTOR (RCT) TRIP BREAKER "A" OPENED. THE TRIP OCCURRED IMMEDIATELY AFTER STEAM GENERATOR NO. 24 CHANNEL III BISTABLE WAS PLACED IN THE TEST (TRIP) CONDITION IN ORDER TO ADJUST THE TRIP SETPOINT. THE STEAM GENERATOR (SG) LOW-LOW LEVEL TRIP CIRCUIT IS CONFIGURED IN A TWO-OUT-OF-THREE LOGIC. A CONTACT IN THE TRIP CIRCUIT FROM ONE OF THE OTHER TWO CHANNEL RELAYS (RLY) ON STEAM GENERATOR NO. 24 WAS NOT MADE UP PROPERLY, ALLOWING THE TRIP LOGIC TO BE MADE UP. THE RELAYS (RLY) ARE WESTINGHOUSE BF-57F RELAYS. RELAYS (RLY) LC-447BX AND LC-447DX WERE REPLACED WITH NEW RELAYS PRIOR TO RETURNING THE PLANT TO SERVICE. ALL OTHER SAFETY-RELATED EQUIPMENT OPERATED AS REQUIRED. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED.

[65] INDIAN POINT 3 DOCKET 50-286 LER 87-008
FUEL ASSEMBLY IN INCORRECT LOCATION IN SPENT FUEL PIT.
EVENT DATE: 062787 REPORT DATE: 072787 NSSS: WE TYPE: PWR

(NSIC 205526) ON JUNE 27, 1987, DURING A REFUELING OUTAGE, UNIRRADIATED FUEL ASSEMBLY U50 WAS FOUND TO HAVE BEEN INADVERTENTLY LOADED INTO SPENT FUEL PIT (SFP) LOCATION S-21 INSTEAD OF SFP LOCATION R-21. THE PRESENCE OF THIS HIGH-ENRICHED (3.6 WEIGHT PERCENT) U-235 IN THE WRONG LOCATION DISRUPTED THE "CHECKERBOARD" PATTERN REQUIRED BY TECHNICAL SPECIFICATIONS FOR HIGHLY-ENRICHED (I.E., GREATER THAN 3.5 WEIGHT PERCENT) ASSEMBLIES IN THE SFP. WORST-CASE ANALYSES PERFORMED PRIOR TO THIS EVENT HAD SHOWED THAT THERE WAS NO RISK OF INADVERTENTLY CREATING A CRITICALITY IN THE SFP FOR THIS TYPE OF EVENT. ANALYSES PERFORMED SUBSEQUENT TO THIS EVENT, SPECIFICALLY ADDRESSING THE MISLOCATION OF AN

ASSEMBLY ENRICHED TO 3.6 WEIGHT PERCENT, CONFIRMED A NEUTRON MULTIPLICATION FACTOR OF LESS THAN 0.90. IN ORDER TO PREVENT ITS RECURRENCE, SFP LOADING PROCEDURES WILL BE CHANGED TO MINIMIZE THE RISK OF MISLOADING THE SFP BY LOADING LOW- ENRICHED ASSEMBLIES FIRST IN INTERVENING LOCATIONS WHENEVER POSSIBLE. ASSEMBLY U50 WAS RETRIEVED FROM SFP LOCATION S-21, IDENTIFIED, AND INSERTED INTO THE CORE. NO SIMILAR EVENT HAS BEEN REPORTED TO DATE.

[66] KEWAUNEE DOCKET 50-305 LER 87-006
POTENTIAL VIOLATION OF COMPONENT COOLING WATER CONTAINMENT ISOLATION FUNCTION IDENTIFIED BY WESTINGHOUSE IN A 10 CFR 21 REPORT.
EVENT DATE: 061887 REPORT DATE: 072087 NSSS: WE TYPE: PWR

(NSIC 205446) ON JUNE 18, 1987, WHILE THE PLANT WAS OPERATING AT 100% POWER, WESTINGHOUSE ELECTRIC CORPORATION NOTIFIED WISCONSIN PUBLIC SERVICE CORPORATION (WPSC) OF A 10 CFR 21 ITEM WITH REGARD TO WESTINGHOUSE SUPPLIED COMPONENT COOLING WATER SYSTEM (CCWS). DUE TO A RECOMMENDATION MADE BY WESTINGHOUSE, WPSC REMOVED THE INTERNALS FROM THE RELIEF VALVE ON THE CCW SURGE TANK, IN JULY OF 1984. WESTINGHOUSE MADE THE RECOMMENDATION TO PREVENT THE POSSIBLE OVERPRESSURIZATION OF THE CCWS DURING A LEAK IN A REACTOR COOLANT PUMP THERMAL BARRIER. UPON REEVALUATION OF THEIR RECOMMENDATION, WESTINGHOUSE DETERMINED THAT THE REMOVAL OF THE VALVE'S INTERNALS COULD RESULT IN A VIOLATION OF THE CONTAINMENT INTEGRITY DESIGN BASIS FOR SOME PLANTS, INCLUDING KEWAUNEE. WPSC IMMEDIATELY BEGAN EVALUATING THE APPLICABILITY OF THE JUNE 18, 1987 10 CFR 21 NOTIFICATION WITH THE ASSISTANCE OF WESTINGHOUSE AND FLUOR ENGINEERS, INC., THE KNPP ARCHITECTURAL ENGINEERING FIRM. THIS EVALUATION DETERMINED THAT THE PART 21 NOTIFICATION DID NOT APPLY TO THE KEWAUNEE PLANT. THEREFORE, NO FURTHER CORRECTIVE ACTION WAS REQUIRED. THIS EVENT IS BEING REPORTED BY WPSC IN RESPONSE TO THE 10 CFR 21 REPORT SENT TO THE NRC BY WESTINGHOUSE.

[67] KEWAUNEE DOCKET 50-305 LER 87-007
STEAM GENERATOR BLOWDOWN AND SAMPLING ISOLATION DUE TO STICKY CONTACT.
EVENT DATE: 062287 REPORT DATE: 072287 NSSS: WE TYPE: PWR
VENDOR: POTTER & BRUMFIELD

(NSIC 205527) AT 1148 ON JUNE 22, 1987, STEAM GENERATOR BLOWDOWN AND BLOWDOWN SAMPLING WERE AUTOMATICALLY ISOLATED DURING PERFORMANCE OF A SURVEILLANCE PROCEDURE TO CALIBRATE R-19, THE STEAM GENERATOR BLOWDOWN LINE RADIATION MONITOR. THE EVENT OCCURRED AS A RESULT OF A STICKY CONTACT ON A RELAY, WHICH FAILED TO CLOSE, ERRONEOUSLY INDICATING A HIGH RADIATION SIGNAL FROM R-19 AND ISOLATING STEAM GENERATOR BLOWDOWN AND BLOWDOWN SAMPLING. AFTER VERIFYING THAT HIGH RADIATION DID NOT EXIST IN THE STEAM GENERATOR BLOWDOWN LINES, THE OPERATORS REESTABLISHED STEAM GENERATOR BLOWDOWN. IN ADDITION TO THE ISOLATION SIGNAL, R-15, THE CONDENSER AIR EJECTOR RADIATION MONITOR, AND R-16, THE CONTAINMENT FAN COIL UNIT RADIATION MONITOR, ALARMED IN THE CONTROL ROOM . SINCE NEITHER ALARM WAS REINITIATED AFTER CONTROL ROOM OPERATORS RESET THE ALARMS, IT WAS CONCLUDED THAT HIGH RADIATION IN THE LINES DID NOT EXIST. THE ROOT CAUSE OF THE ISOLATION WAS A STICKY CONTACT IN THE RELAY THAT CONTROLS THE STEAM GENERATOR BLOWDOWN AND BLOWDOWN SAMPLING ISOLATION VALVES. ACTUATION OF R-15 AND R-16 WAS A RESULT OF A VOLTAGE SPIKE THAT OCCURRED DURING THE EVENT IN THE CIRCUIT WHICH POWERS R-15, R-16 AND R-19. IMMEDIATELY FOLLOWING THE EVENT, A JUMPER WAS INSTALLED, BYPASSING THE FAULTY RELAY, TO ALLOW STEAM GENERATOR BLOWDOWN TO BE REESTABLISHED.

[68] KEWAUNEE DOCKET 50-305 LER 87-008
UNIT TRIP DUE TO PERSONNEL MISUNDERSTANDING OF WORK INSTRUCTION.
EVENT DATE: 062687 REPORT DATE: 072787 NSSS: WE TYPE: PWR

(NSIC 205504) ON JUNE 26, 1987, AT 1115, WITH THE PLANT AT 100% POWER, A PLANT ELECTRICIAN PULLED OUT THE DRAWER OF THE UPPER PHASE B MAIN GENERATOR POTENTIAL

TRANSFORMER CABINET. THIS CAUSED A LOSS-OF-EXCITATION SIGNAL TO THE MAIN GENERATOR RESULTING IN A GENERATOR TRIP, FOLLOWED BY A TURBINE/REACTOR TRIP. INFORMATION WAS NEEDED ON THE ELECTRICAL SIZE OF THE POTENTIAL TRANSFORMERS LOCATED IN THE EMERGENCY DIESEL GENERATOR CONTROL AND EXCITATION CABINET. A PLANT ELECTRICIAN WAS ASKED TO COLLECT THIS INFORMATION BY OPENING THE CABINET DOOR AND READING THE NAMEPLATE DATA. THE ELECTRICIAN MISUNDERSTOOD THE INSTRUCTIONS AND PULLED OUT THE MAIN GENERATOR POTENTIAL TRANSFORMER CABINET DRAWER. WHEN THE POTENTIAL TRANSFORMER WAS PULLED OUT AN OPEN CIRCUIT RESULTED, AND THE VOLTAGE SIGNAL TO THE LOSS-OF-EXCITATION RELAY WAS DISCONNECTED. THIS LOSS OF INPUT CAUSED THE LOSS-OF-EXCITATION RELAY TO BE ENERGIZED, RESULTING IN A LOSS-OF-EXCITATION GENERATOR TRIP. THE ROOT CAUSE OF THE EVENT WAS PERSONNEL ERROR. THE ELECTRICIAN MISUNDERSTOOD HIS JOB TASK. TO PREVENT RECURRENCE, THE MAIN GENERATOR POTENTIAL TRANSFORMER CABINET WILL BE LOCKED AND A CAUTION SIGN WILL BE PLACED ON THE FRONT OF THE DRAWER. THE PLANT SYSTEMS RESPONDED AS DESIGNED AND THE OPERATORS FOLLOWED APPROPRIATE. PROCEDURES FOR PLANT STABILIZATION.

[69] LA SALLE 2 DOCKET 50-374 LER 87-015
REACTOR WATER CLEAN-UP SUCTION ISOLATION VALVE CLOSURE ON HI FILTER/DEMINERALIZER INLET TEMPERATURE DUE TO FAILED TEMPERATURE SWITCH.
EVENT DATE: 063087 REPORT DATE: 073087 NSSS: GE TYPE: BWR
VENDOR: FENWALL, INC.

(NSIC 205692) AT 0944 HOURS ON JUNE 30, 1987, WITH UNIT 2 IN OPERATIONAL CONDITION 1 (RUN) AT 79% POWER, THE UNIT 2 REACTOR WATER CLEANUP (RWCU) FILTER/DEMINERALIZER INLET TEMPERATURE SWITCH (TIS-2G33-N008) FAILED IN THE TRIPPED (HIGH TEMPERATURE) CONDITION, CAUSING THE RWCU OUTBOARD CONTAINMENT ISOLATION VALVE (2G33-F004) TO CLOSE THROUGH THE PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) GROUP V OUTBOARD ISOLATION LOGIC. UPON VERIFICATION THAT A HIGH TEMPERATURE CONDITION NEVER EXISTED, A JUMPER WAS INSTALLED TO BYPASS THE HIGH TEMPERATURE TRIP AND THE RWCU SYSTEM WAS RESTARTED AT 1400 HOURS THE SAME DAY. THE CAUSE OF THIS EVENT WAS A FAILED CAPACITOR IN THE TEMPERATURE SWITCH (TIS-2G33-N008). THE SAFETY CONSEQUENCES OF THIS EVENT WERE MINIMAL SINCE THE OUTBOARD ISOLATION VALVE OF THE RWCU SYSTEM CLOSED AS DESIGNED. WORK REQUEST L70132 WAS INITIATED TO REPAIR THE FAILED TEMPERATURE SWITCH. THE SWITCH WAS REPAIRED BY JULY 6, 1987, AND NO FURTHER PROBLEMS HAVE BEEN EXPERIENCED. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV) DUE TO AN ENGINEERED S AFETY FEATURE (ESF) ACTUATION.

[70] LIMERICK 1 DOCKET 50-352 LER 87-030
FAILURE TO PERFORM RADIATION MONITOR SURVEILLANCE TEST WHILE MOVING IRRADIATED FUEL.
EVENT DATE: 060487 REPORT DATE: 071787 NSSS: GE TYPE: BWR

(NSIC 205461) ON JUNE 4, 1987, AND JUNE 5, 1987, THE 'D' AND 'B' CHANNELS OF THE REFUELING AREA VENTILATION EXHAUST RADIATION MONITORS WERE NOT IN COMPLIANCE WITH THE SURVEILLANCE REQUIREMENTS DUE TO THE FAILURE TO PERFORM THE MONTHLY FUNCTIONAL TESTS. THE SITUATION WAS RECOGNIZED BY THE SURVEILLANCE TEST COORDINATOR ON JUNE 16 DURING A REVIEW OF THE SURVEILLANCE REQUIREMENTS NEEDED TO REFUEL THE REACTOR. THE 'D' AND 'B' CHANNELS WERE TESTED AND FOUND TO BE OPERATING PROPERLY WITH THEIR TRIP SETPOINTS WITHIN THE ACCEPTABLE LIMITS. THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL RESULTING FROM THIS EVENT AND NO ADVERSE CONSEQUENCES ASSOCIATED WITH THIS EVENT. THE SURVEILLANCE TEST SCHEDULE WILL BE REVISED TO GROUP AND UNIQUELY IDENTIFY CERTAIN SPECIAL OPERATING CONDITIONS DURING WHICH SURVEILLANCE TESTS ARE REQUIRED.

[71] LIMERICK 1 DOCKET 50-352 LER 87-023
ENGINEERED SAFETY FEATURE ACTUATION DUE TO STATION BATTERY CHARGER FAILURE.
EVENT DATE: 061187 REPORT DATE: 071587 NSSS: GE TYPE: BWR
VENDOR: C & D BATTERIES, DIV OF ELTRA CORP.

(NSIC 205455) ON JUNE 11, 1987, THE STANDBY GAS TREATMENT AND REACTOR ENCLOSURE RECIRCULATION SYSTEMS (ENGINEERED SAFETY FEATURES) INITIATED AS A CONSEQUENCE TO ACTIONS TAKEN DUE TO FAILURE OF THE 1A1D103 STATION BATTERY CHARGER. THE 125 VDC STATION BATTERIES (1A1) WERE DISCONNECTED FROM THE BUS AT THE TIME OF THE EVENT TO ACCOMMODATE MAINTENANCE WORK. THE REACTOR ENCLOSURE AND REFUEL FLOOR HVAC SYSTEMS ISOLATED AS A RESULT OF DE-ENERGIZATION OF THEIR LOGIC CIRCUITS AND THE ESF SYSTEMS INITIATED AS DESIGNED. A TEMPORARY CIRCUIT ALTERATION (TCA) WAS INSTALLED TO PROVIDE AN ALTERNATE POWER SUPPLY TO THE DE-ENERGIZED BUS. DURING RE ENERGIZATION OF THE BUS, A REACTOR PROTECTION SYSTEM SERIES BREAKER SHUNT TRIPPED AND CAUSED THE INBOARD INSTRUMENT GAS VALVE TO CLOSE. THE BATTERY CHARGER FAILED AS A RESULT OF AN INTEGRATED CIRCUIT CONTROLLER CARD FAILURE. THE CARD HAS BEEN SENT TO THE MANUFACTURER FOR FAILURE ANALYSIS. THE CAUSE FOR THE RPS BREAKER TRIP IS UNKNOWN AND WILL BE FURTHER EVALUATED. THE CONSEQUENCES OF THIS EVENT WERE MINIMAL BECAUSE THE ENGINEERED SAFETY FEATURES INITIATED AS DESIGNED AND THE UNIT WAS SHUTDOWN WITH THE CORE OFFLOADED AT THE TIME OF THE EVENT. A SUPPLEMENTAL REPORT WILL PROVIDE INFORMATION ON THE CONTROLLER CARD FAILURE AND ANY APPROPRIATE ACTIONS TO PREVENT RECURRENCE OF THIS EVENT.

[72] LIMERICK 1 DOCKET 50-352 LER 87-024
REFUELING FLOOR VENTILATION ISOLATION DUE TO LOW NEGATIVE DIFFERENTIAL PRESSURE.
EVENT DATE: 061187 REPORT DATE: 071387 NSSS: GE TYPE: BWR

(NSIC 205456) ON JUNE 11, 1987, AT 1130 HOURS, A NUCLEAR STEAM SUPPLY SHUTDOWN SYSTEM ISOLATION OF THE REFUEL FLOOR VENTILATION OCCURRED DUE TO LOW NEGATIVE DIFFERENTIAL PRESSURE BETWEEN THE REFUEL FLOOR AND THE OUTSIDE ENVIRONMENT. THE ISOLATION OCCURRED AFTER A WORKER LIFTED AN INCORRECT WIRE FROM A TERMINAL STRIP WHILE WORKING UNDER A "TROUBLESHOOTING CONTROL FORM". LIFTING OF THIS WIRE CAUSED THE 'A' CHANNEL REFUEL FLOOR ISOLATION VALVES TO CLOSE, CAUSING THE SUPPLY AND EXHAUST FANS TO TRIP, RESULTING IN A LOW NEGATIVE DIFFERENTIAL PRESSURE. THE STANDBY GAS TREATMENT SYSTEM, AN ENGINEERED SAFETY FEATURE, RESPONDED AS DESIGNED FOLLOWING THE ISOLATION AND AT 1230 HOURS NORMAL VENTILATION WAS RESTORED AND THE ISOLATION WAS RESET. THERE WERE NO ADVERSE CONSEQUENCES RESULTING FROM THIS EVENT AND NO RADIOACTIVE MATERIAL WAS RELEASED FROM THIS EVENT. THE WORKER WAS COUNSELED ON THE IMPORTANCE OF LIFTING ONLY THE WIRE WHICH IS DESIGNATED TO BE LIFTED AND THE IMPORTANCE OF INVESTIGATING THE FUNCTION OF ANY OTHER WIRE WHICH MAY BE AFFECTED BY HIS WORK.

[73] LIMERICK 1 DOCKET 50-352 LER 87-026
NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM ISOLATION SIGNALS AND REACTOR SCRAM SIGNALS WHILE PERFORMING RADIOGRAPHIC EXAMINATION.
EVENT DATE: 061287 REPORT DATE: 071387 NSSS: GE TYPE: BWR

(NSIC 205458) ON JUNE 12, 1987, AT 0331 HOURS, WITH THE UNIT SHUTDOWN FOR REFUELING AND THE CORE OFFLOADED, A NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) ISOLATION SIGNAL (AN ENGINEERED SAFETY FEATURE) AND FULL REACTOR SCRAM SIGNAL (REACTOR PROTECTION SYSTEM ACTUATION) WERE CAUSED BY NONDESTRUCTIVE RADIOGRAPHIC EXAMINATIONS BEING PERFORMED ON THE 'A' MAIN STEAM LINE. THIS RESULTED IN THE ISOLATION OF THE INBOARD MAIN STEAM ISOLATION VALVES (MSIVS) (WHICH HAD BEEN BLOCKED FOR MAINTENANCE), MAIN STEAM LINE DRAIN VALVES, MAIN STEAM LINE SAMPLE VALVES, REACTOR RECIRCULATION SAMPLE VALVES, AND THE PARTIAL INSERTION OF 28 CONTROL ROD BLADES. THE ISOLATION AND SCRAM SIGNALS WERE RESET. THE ISI PERSONNEL WERE TOLD TO DISCONTINUE THE EXAMINATIONS UPON COMPLETION OF THE JOB IN PROGRESS. AT 0355 HOURS, WHILE RETURNING THE SOURCE TO ITS SHIELD CHAMBER, A SECOND NSSSS ISOLATION SIGNAL AND FULL REACTOR SCRAM SIGNAL OCCURRED WITH NO

CONTROL ROD MOVEMENT RESULTING. THE ISOLATIONS AND SCRAM SIGNAL WERE RESET AND THE SYSTEMS WERE RETURNED TO OPERATION BY 0430 HOURS. A TEMPORARY CIRCUIT ALTERATION (TCA) WAS IMPLEMENTED TO BYPASS THE TRIP RELAYS ON THE MAIN STEAM LINE RADIATION MONITORS TO PREVENT ADDITIONAL ISOLATION SIGNALS DURING TESTING. THERE WAS NO RADIOLOGICAL RELEASE TO THE ENVIRONMENT AS A RESULT OF THIS EVENT.

[74] LIMERICK 1 DOCKET 50-352 LER 87-037
NON-COMPLIANCE WITH TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS AS RESULT OF MISSING FLUENCE WIRE DOSIMETRY.
EVENT DATE: 061287 REPORT DATE: 072487 NSSS: GE TYPE: BWR

(NSIC 205506) ON JUNE 12, 1987, AT 0300 HOURS, MAINTENANCE PERSONNEL ATTEMPTED TO REMOVE A NEUTRON FLUENCE WIRE SPECIMEN DOSIMETER FROM ITS SAMPLE HOLDER WHICH IS ATTACHED TO A MATERIAL SURVEILLANCE CAPSULE INSIDE THE REACTOR PRESSURE VESSEL. THE DEVICE WAS NOT FOUND IN THE SAMPLE HOLDER NOR COULD IT BE LOCATED IN THE VESSEL BY UNDERWATER CAMERA INSPECTION. ON JUNE 24, 1987, PLANT PERSONNEL DETERMINED THAT THE FAILURE TO LOCATE THE DOSIMETER LEADS TO A VIOLATION OF THE TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS 4.4.6.1.4. REQUIRING ITS REMOVAL FOR FLUENCE ANALYSIS AT THE FIRST REFUELING OUTAGE. AN INVESTIGATION DETERMINED THAT REACTOR PRESSURE VESSEL INSTALLATION SPECIFICATIONS DID NOT INCLUDE THE DOSIMETRY DEVICE AS PART OF THE REACTOR MATERIAL SURVEILLANCE PROGRAM. THEREFORE, IT IS BELIEVED THAT THE DOSIMETER WAS NEVER INSTALLED DURING REACTOR PRESSURE VESSEL ASSEMBLY PRIOR TO INITIAL STARTUP. HOWEVER, A LOST PART ANALYSIS CONCLUDED THAT THERE ARE NO ADVERSE SAFETY CONSEQUENCES SHOULD THE DOSIMETER NOT BE FOUND. THE CONSEQUENCES OF THE LOST NEUTRON FLUENCE DATA ARE MINIMAL BECAUSE THE MATERIAL SURVEILLANCE PROGRAM PROVIDES ADDITIONAL FLUENCE WIRE SPECIMENS WHICH WILL BE REMOVED PERIODICALLY DURING PLANT LIFE.

[75] LIMERICK 1 DOCKET 50-352 LER 87-025
PERSONNEL ERROR RESULTS IN REFUELING PLATFORM OPERATION NOT IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS.
EVENT DATE: 061387 REPORT DATE: 071387 NSSS: GE TYPE: BWR

(NSIC 205457) ON JUNE 11, 1987, AT 1230 HOURS, IT WAS DETERMINED THAT THE CONTROL ROD BLOCK LOGIC REQUIRED BY TECHNICAL SPECIFICATION 3.9.6 WAS NOT OPERABLE. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR DUE TO A UTILITY EMPLOYED TECHNICAL ASSISTANT (TA) DEFEATING THE REFUELING PLATFORM TRAVEL LIMIT SWITCHES TO COMPENSATE FOR A MALFUNCTIONING REACTOR MANUAL CONTROL RELAY CONTACT WHICH SIMULATED THE REACTOR MODE SWITCH IN STARTUP TO THE REFUELING PLATFORM. THE ROOT CAUSE OF THE EVENT WAS THE FACT THAT THE TA DID NOT BELIEVE THAT THE TRAVEL LIMIT SWITCHES AND ASSOCIATED CONTROL ROD BLOCKS WERE REQUIRED WITH THE CORE OFFLOADED. THERE WERE NO ADVERSE CONSEQUENCES TO THE EVENT. UPON RECOGNITION OF THE SITUATION, THE CONTROL ROD SHUFFLE WAS IMMEDIATELY SUSPENDED. THE LIMIT SWITCHES WERE RETURNED TO NORMAL AND THE RELAY CONTACT CLEANED AND TESTED. THE TA INVOLVED WAS COUNSELED AS TO THE REQUIREMENTS OF THE TECHNICAL SPECIFICATIONS AND THE IMPORTANCE OF FOLLOWING ADMINISTRATIVE PROCEDURES. ADDITIONALLY, THE REACTOR ENGINEER ISSUED A MEMORANDUM TO PERSONNEL INVOLVED WITH THE REFUELING OPERATIONS STRESSING THE IMPORTANCE OF FOLLOWING ADMINISTRATIVE CONTROLS WHEN TROUBLESHOOTING AND/OR REPAIRS ARE REQUIRED WITH THE REFUELING PLATFORM AND ASSOCIATED EQUIPMENT.

[76] LIMERICK 1 DOCKET 50-352 LER 87-031
HOT MAINTENANCE SHOP VENTILATION OPERATION NOT IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS DUE TO PERSONNEL ERROR.
EVENT DATE: 061387 REPORT DATE: 071687 NSSS: GE TYPE: BWR

(NSIC 205501) ON JUNE 14, 1987 AT 0305 HOURS, THE HOT MAINTENANCE SHOP VENTILATION WAS DISCOVERED OPERATING WITHOUT THE ASSOCIATED EXHAUST RADIATION

MONITOR FUNCTIONING AND WITHOUT COMPENSATORY SAMPLING TAKING PLACE AS REQUIRED BY TECHNICAL SPECIFICATION 3.3.7.12. ALL WORK IN THE HOT MAINTENANCE SHOP WAS DISCONTINUED AND THE VENTILATION WAS IMMEDIATELY SHUT DOWN. THE VENTILATION HAD BEEN IN SERVICE FOR APPROXIMATELY 18 HOURS WITHOUT SAMPLE FLOW THROUGH THE RADIATION MONITOR PRIOR TO DISCOVERY. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THE EVENT. NO WORK WAS BEING PERFORMED IN THE SHOP WHICH INVOLVED COMPONENTS WITH HIGH CONTAMINATION LEVELS DURING THE PERIOD IN WHICH THE RADIATION MONITOR WAS NOT FUNCTIONING. OTHER MONITORING DEVICES WERE IN SERVICE IN THE AREA AND DID NOT INDICATE ANY AIRBORNE CONTAMINATION CONCERN. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR IN THAT A UTILITY EMPLOYED PERSON DID NOT TAKE THE PROPER ACTION AS PROVIDED BY THE STATION PROCEDURES FOLLOWING THE ACKNOWLEDGEMENT OF A LOSS OF RADIATION MONITOR SAMPLE FLOW ALARM. A MEMO FROM THE OPERATIONS ENGINEER HAS BEEN SENT TO ALL OPERATIONS PERSONNEL DISCUSSING THE IMPORTANCE OF FOLLOWING PROCEDURES WHEN RESPONDING TO ALARMS GENERATED BY THE RM-11 TERMINAL LOCATED IN THE MAIN CONTROL ROOM.

[77]   LIMERICK 1   DOCKET 50-352   LER 87-027
ACTUATION OF ESF DUE TO RPS/UPS BREAKER TRIP OF UNKNOWN CAUSE.
EVENT DATE: 061587   REPORT DATE: 071587   NSSS: GE   TYPE: BWR

(NSIC 205459) ON JUNE 15, 1987, WHILE PREPARING TO TEST A MODIFICATION WHICH UPGRADED THE 'B' REACTOR PROTECTION SYSTEM (RPS) ELECTRIC POWER MONITORING UNDERVOLTAGE AND OVERVOLTAGE RELAYS, THE 'B' AND 'D' NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) CHANNELS GENERATED ISOLATION SIGNALS FOR SEVERAL SYSTEMS. THE MODIFICATION ACCEPTANCE TEST REQUIRED THE RELAYS' POWER SOURCE TO BE SUPPLIED FROM THE PREFERRED SOURCE RATHER THAN THE ALTERNATE SOURCE. WHEN THE POWER SUPPLY WAS TRANSFERRED, BOTH THE RCS SERIES BREAKERS SHUNT TRIPPED ON A TRANSIENT OVERVOLTAGE CONDITION, RESULTING IN DE-ENERGIZATION OF THE ASSOCIATED NSSSS LOGIC. THE CAUSE FOR THE OVERVOLTAGE CONDITION IS UNKNOWN AND EFFORTS TO RECREATE THE EVENT COULD NOT DETERMINE A PROBABLE CAUSE. THE CONSEQUENCES OF THE EVENT WERE MINIMAL BECAUSE THE SYSTEMS WHICH WERE AFFECTED BY THE ISOLATIONS WERE NOT REQUIRED BY TECHNICAL SPECIFICATIONS BUT WERE ONLY IN SERVICE FOR HABITABILITY CONCERNS. THE RPS POWER SUPPLY WAS RESTORED WITHIN TEN MINUTES OF THE EVENT AND THE NSSSS ISOLATIONS WERE RESET SHORTLY AFTERWARD.

[78]   LIMERICK 1   DOCKET 50-352   LER 87-029
NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM ISOLATIONS DUE TO AN INADEQUATE PROCEDURE.
EVENT DATE: 061687   REPORT DATE: 071687   NSSS: GE   TYPE: BWR

(NSIC 205460) ON JUNE 16, 1987, AT 1323 HOURS, VARIOUS NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) ISOLATIONS (AN ENGINEERED SAFETY FEATURE) OCCURRED. THE ISOLATED SYSTEMS INCLUDED THE REFUEL FLOOR HEATING, VENTILATING AND AIR CONDITIONING (HVAC) AND THE PRIMARY CONTAINMENT INSTRUMENT GAS (PCIG) SYSTEMS. THE STANDBY GAS TREATMENT SYSTEM (SGTS) STARTED AS DESIGNED IN CONJUNCTION WITH THE REFUEL FLOOR HVAC ISOLATION. THE IMMEDIATE CAUSE WAS THE LIFTING OF A COMMON NEUTRAL WIRE DURING THE INSTALLATION OF A RELAY FOR A PLANT MODIFICATION. THE ROOT CAUSE OF THE EVENT WAS AN INCOMPLETE REVIEW OF THE IMPLEMENTATION METHOD OF A MODIFICATION TO ASSESS THE POTENTIAL IMPACT ON PLANT OPERATIONS. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THIS EVENT. THE UNIT WAS IN A REFUELING OUTAGE WITH THE CORE COMPLETELY OFF-LOADED AT THE TIME OF THE EVENT. A MEETING WAS HELD ON JUNE 7, 1987, BETWEEN THE PLANT MANAGER AND VARIOUS PLANT PERSONNEL TO DISCUSS THIS EVENT AND ACTIONS TO PREVENT RECURRENCE. A REVIEW OF THE IMPLEMENTATION METHOD OF ALL ONGOING MODIFICATIONS AS PERFORMED TO DETERMINE THE IMPACT ON PLANT OPERATIONS. A REVIEW IS UNDERWAY TO EVALUATE THE NEED TO UPGRADE THE EXISTING MODIFICATION IMPLEMENTATION METHODS. THIS REVIEW WILL BE COMPLETED BY FEBRUARY 26, 1988.

[79] LIMERICK 1 DOCKET 50-352 LER 87-033
CONTROL ROOM EMERGENCY FRESH AIR SUPPLY DUE TO A FALSE HIGH CHLORINE CONCENTRATION SIGNAL.
EVENT DATE: 070187 REPORT DATE: 073187 NSSS: GE TYPE: BWR

(NSIC 205622) ON JULY 1, 1987, AT 2300 HOURS, THE CONTROL ROOM VENTILATION SYSTEM ISOLATED AND THE 'A' TRAIN OF THE CONTROL ROOM FRESH AIR SUPPLY (CREFAS) SYSTEM, AN ENGINEERED SAFETY FEATURE (ESF), AUTOMATICALLY INITIATED AS DESIGNED. THE CAUSE OF THE CONTROL ROOM VENTILATION SYSTEM ISOLATION WAS A SPURIOUS SPIKE OF THE MAIN CONTROL ROOM VENTILATION CHLORINE DETECTOR. THE SPIKE WAS DUE TO A DEFECTIVE MANUFACTURING SOLDER CONNECTION WITHIN THE CABLE WHICH CONNECTS THE LOCAL PROCESSING UNIT TO THE ANALYZER PROBE. THE BAD CONNECTION CAUSED A SHORT IN THE WIRE WHICH WAS INTERPRETED BY THE LOCAL PROCESSING UNIT OF THE ANALYZER AS A HIGH CHLORINE SIGNAL. THIS FALSE CHLORINE SIGNAL CAUSED AN ISOLATION OF THE CONTROL ROOM VENTILATION AND INITIATION OF THE CREFAS SYSTEM. AT 1700 HOURS ON JULY 17, 1987 THE NORMAL VENTILATION WAS RESTORED IN THE CONTROL ROOM AFTER THE SUCCESSFUL COMPLETION OF SURVEILLANCE TEST, ST-2- 078-606-0 FOR THE 'C' DETECTOR. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THIS EVENT. THE EIIS CODE FOR THE AFFECTED SYSTEM, CONTROL ROOM VENTILATION IS VI AND THE CODE FOR THE ANACON MODEL M-17 CHLORINE ANALYZER IS AE.

[80] MAINE YANKEE DOCKET 50-309 LER 87-006
MANUAL REACTOR TRIP AFTER LOSS OF LEVEL CONTROL IN NO. 3 STEAM GENERATOR.
EVENT DATE: 062787 REPORT DATE: 072787 NSSS: CE TYPE: PWR
VENDOR: ROCHESTER INSTRUMENT SYSTEMS, INC.

(NSIC 205577) ON JUNE 24, 1987, A MANUAL REACTOR TRIP WAS INITIATED DUE TO AN UNCONTROLLABLE INCREASE IN STEAM GENERATOR NO. 3 LEVEL. THE INCREASE IN STEAM GENERATOR LEVEL OCCURRED WHEN AN INSTRUMENT AND CONTROLS TECHNICIAN WAS REPLACING A COMPONENT IN THE FEEDWATER CONTROL CABINET. THE TECHNICIAN REMOVED OUTPUT LEADS FROM THE FAILED TIME LAG UNIT AND INTERRUPTED THE CONTROL SIGNAL TO THE NO. 3 MAIN FEEDWATER REGULATING VALVE. THE VALVE FAILED OPEN CAUSING A RAPID RISE IN STEAM GENERATOR LEVEL. THE REACTOR WAS MANUALLY TRIPPED AS STEAM GENERATOR LEVEL APPROACHED THE AUTOMATIC TURBINE TRIP SETPOINT. ALL SYSTEMS RESPONDED PROPERLY FOLLOWING THE PLANT TRIP. THE ROOT CAUSE OF THIS EVENT IS STILL BEING EVALUATED.

[81] MCGUIRE 1 DOCKET 50-369 LER 87-011
IMPROPER SURVEILLANCES PERFORMED ON UNIT 1 AND 2 CONTAINMENT ELECTRICAL PENETRATIONS AS DEFINED BY 10CFR50 APPENDIX J DUE TO INADEQUATE ADMINISTRATIVE CONTROLS.
EVENT DATE: 052987 REPORT DATE: 071387 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)
VENDOR: CONAX CORP.

(NSIC 205467) ON JUNE 10, 1987, IT WAS DETERMINED THAT THE REQUIRED LEAK RATE TESTING OF CONTAINMENT ELECTRICAL PENETRATIONS E121 AND E247, UNITS 1 AND 2, HAD NOT BEEN PERFORMED AS DESCRIBED IN 10CFR 50, APPENDIX J. THE NITROGEN GAS PRESSURE INSIDE THE PENETRATIONS WAS NOT MONITORED PROPERLY IN LIEU OF PERFORMING LEAK RATE TESTS. PRESSURES WERE TO BE INITIALIZED AT OR ABOVE THE PEAK ACCIDENT PRESSURE AS SPECIFIED IN TECHNICAL SPECIFICATION 3.6.1.2. DUKE PERSONNEL DETERMINED THE PENETRATIONS WERE OPERABLE BY USING THE AS FOUND" PRESSURE DATA OBTAINED DURING PAST SURVEILLANCE TO CALCULATE A LEAK RATE. PERFORMANCE CHANGED THE LEAK RATE PROCEDURES FOR TESTING THE PENETRATIONS TO PERFORM SURVEILLANCES ON ELECTRICAL PENETRATIONS AS SPECIFIED IN 10CFR50 APPENDIX J. PERFORMANCE PERFORMED THE CORRECTED LEAK RATE PROCEDURE FOR THE UNIT 2 PENETRATIONS ON JUNE 9, AND ON JUNE 10, 1987 FOR THE UNIT 1 PENETRATIONS. UNIT 1 WAS IN MODE 1, POWER OPERATION, AT 100% POWER AND UNIT 2 WAS IN MODE 6, REFUELING, AT THE TIME THE SURVEILLANCES WERE DETERMINED TO BE IMPROPER. THIS INCIDENT HAS BEEN ATTRIBUTED TO MANAGEMENT DEFICIENCY DUE TO INADEQUATE ADMINISTRATIVE CONTROLS TO ENSURE THAT

REQUIRED SURVEILLANCE TESTING WAS PERFORMED. THIS INCIDENT IS CONSIDERED TO BE OF NO SIGNIFICANCE WITH RESPECT TO THE HEALTH AND SAFETY OF THE PUBLIC.

[82] MCGUIRE 2 DOCKET 50-370 LER 87-007
INOPERABLE AUXILIARY BUILDING FIRE BARRIER DUE TO A WALL SECTION CONSTRUCTED WITHOUT PROPER JOINT TREATMENT.
EVENT DATE: 060987 REPORT DATE: 071387 NSSS: WE TYPE: PWR

(NSIC 205468) AT APPROXIMATELY 1600 ON JUNE 9, 1987 WHILE PERFORMING AN UNRELATED INSPECTION, A QUALITY ASSURANCE INSPECTOR DISCOVERED A SMALL GAP BETWEEN THE DRYWELL AND CONCRETE ABOVE A WALL SECTION BETWEEN THE 750 FT ELEVATION ELECTRICAL PENETRATION AND SWITCHGEAR ROOMS ON THE UNIT 2 SIDE OF THE AUXILIARY BUILDING. THE NEXT DAY QA REVIEWED ARCHITECTURAL DRAWINGS AND FOUND THAT THE WALL WAS A FIRE BARRIER THAT SHOULD HAVE BEEN SEALED WITH CAULK DURING CONSTRUCTION. MAINTENANCE INVESTIGATED AND DETERMINED THAT WITHOUT THE PROPER JOINT TREATMENT THE FIRE BARRIER WAS INOPERABLE. OPERATIONS WAS INFORMED OF THE INOPERABLE FIRE BARRIER AND A WORK REQUEST WAS WRITTEN TO REPAIR THE WALL. OPERATIONS IMMEDIATELY DECLARED THE FIRE BARRIER INOPERABLE AT 2200 AND INITIATED AN HOURLY FIRE WATCH. THE WALL SECTION WAS REPAIRED ON JUNE 18, AND AT 1335 OPERATIONS DECLARED THE FIRE BARRIER OPERABLE. UNIT 2 WAS IN MODE 6, REFUELING OPERATIONS, AT THE TIME OF DISCOVERY; HOWEVER, UNIT 2 HAD OPERATED IN ALL MODES DURING THE PERIOD OF TIME THE DEFICIENCY EXISTED. THIS INCIDENT HAS BEEN ATTRIBUTED TO A CONSTRUCTION/INSTALLATION DEFICIENCY BECAUSE THE FIRE BARRIER SECTION WAS CONSTRUCTED WITHOUT PROPER JOINT TREATMENT. A MANAGEMENT/QUALITY ASSURANCE DEFICIENCY ALSO CONTRIBUTED TO THIS INCIDENT.

[83] MCGUIRE 2 DOCKET 50-370 LER 87-008
FOUR SNUBBERS WERE NOT INCLUDED ON THE PERIODIC INSPECTION LIST DUE TO A DEFECTIVE PROCEDURE.
EVENT DATE: 062487 REPORT DATE: 072487 NSSS: WE TYPE: PWR

(NSIC 205538) ON JUNE 24, 1987, MAINTENANCE DISCOVERED THAT TWO SNUBBERS ON REACTOR COOLANT SYSTEM PIPING HAD BEEN LEFT OFF OF THE VISUAL INSPECTION LIST FOR UNIT 2 DURING A WALKDOWN INSPECTION OF SNUBBERS IN HIGH MAINTENANCE AREAS. THESE TWO SNUBBERS WERE FUNCTIONALLY TESTED AND VERIFIED OPERABLE ON JUNE 25, 1987. A COMPLETE REVIEW OF THE SNUBBER INSPECTION LIST WAS PERFORMED ON JUNE 26, 1987. TWO ADDITIONAL SNUBBERS WERE FOUND TO BE MISSING FROM THE LIST, AND WERE THEN ADDED TO CORRECT THE LIST. THE LAST DOCUMENTED INSPECTION OF THESE FOUR SNUBBERS WAS COMPLETED ON JANUARY 19, 1984. ALTHOUGH THE SNUBBER INSPECTION PROGRAM WAS CARRIED OUT WITHIN THE ALLOWED TIME, FOUR SNUBBERS WERE NOT INSPECTED BECAUSE THEY WERE NOT ON THE INSPECTION LIST. UNIT 2 WAS IN MODE 5, COLD SHUTDOWN, WHEN THE SNUBBER INSPECTION LIST DISCREPANCIES WERE FOUND; HOWEVER, THE UNIT HAS OPERATED IN ALL MODES SINCE JANUARY 1984. THIS EVENT HAS BEEN ATTRIBUTED TO A DEFECTIVE PROCEDURE. THE PROCEDURE FOR VISUAL INSPECTIONS DOES NOT INCLUDE, NOR REFERENCE, A CONTROLLED LIST OF SNUBBERS TO BE INSPECTED. THE INSPECTION LIST MUST BE CREATED FOR EACH INSPECTION MANUALLY. THE REMAINING TWO SNUBBERS WERE SUBSEQUENTLY INSPECTED AND FOUND TO BE IN AN OPERABLE CONDITION. THIS EVENT IS CONSIDERED TO BE OF NO SIGNIFICANCE WITH RESPECT TO THE HEALTH AND SAFETY OF THE PUBLIC.

[84] MILLSTONE 1 DOCKET 50-245 LER 87-020
RECIRCULATION SYSTEM PIPE WELD CRACK.
EVENT DATE: 062687 REPORT DATE: 072487 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 205518) ON JUNE 26, 1987 AT 1010 HOURS, WHILE THE UNIT WAS SHUT DOWN FOR REFUELING, AN INDICATION WAS DISCOVERED ON CLASS I STAINLESS STEEL PIPING INSIDE THE DRYWELL. THE INDICATION WAS FOUND ON THE 22 INCH RECIRCULATION SYSTEM (AD)

PIPE WELD RMBJ-1 (PIPE TO CAP WELD) DURING INSERVICE INSPECTION (ISI) OF THE RECIRCULATION SYSTEM. THE INDICATION WAS REVIEWED BY GENERATION MECHANICAL ENGINEERING AGAINST THE ACCEPTANCE CRITERIA OF A.S.M.E. SECTION XI, AND ON JULY 1, 1987 WAS DECLARED AS BEING AN UNACCEPTABLE FLAW AND THUS REPORTABLE. THE WELD IN QUESTION WILL BE REPAIRED USING THE WELD OVERLAY TECHNIQUE AND A PRESSURE TEST OF THE SYSTEM WILL BE PERFORMED.

[85] MILLSTONE 1 DOCKET 50-245 LER 87-023
PUMP ANCHOR BOLT NONCONFORMANCE.
EVENT DATE: 070887 REPORT DATE: 080787 NSSS: GE TYPE: BWR

(NSIC 205591) ON JULY 8, 1987 AT 1300 HOURS, DURING A SCHEDULED REFUELING OUTAGE, INSPECTION OF CORE SPRAY SYSTEM PUMP ANCHORS REVEALED NONCONFORMANCE TO THE ORIGINAL DESIGN. THE SAME CONDITION WAS SUSPECTED TO EXIST FOR THE LOW PRESSURE COOLANT INJECTION SYSTEM PUMP ANCHORS. THE SUPPORT SYSTEMS WERE MODIFIED TO MEET ALL REQUIRED LOADINGS. THERE WERE NO CONSEQUENCES.

[86] MILLSTONE 3 DOCKET 50-423 LER 87-031
REACTOR TRIP DUE TO TURBINE TRIP ON LOW LUBE OIL HEADER PRESSURE.
EVENT DATE: 061487 REPORT DATE: 071387 NSSS: WE TYPE: PWR

(NSIC 205481) ON JUNE 14, 1987 AT 0320 WITH THE PLANT AT 100% POWER (MODE 1) THE REACTOR TRIPPED AS A RESULT OF A TURBINE TRIP. THE TURBINE TRIPPED ON LOW BEARING LUBE OIL HEADER PRESSURE IMMEDIATELY FOLLOWING A TRIP OF THE TURNING GEAR OIL PUMP (TGOP). THE TGOP HAD BEEN AUTOSTARTED APPROXIMATELY 9 SECONDS EARLIER AS PART OF WEEKLY TURBINE/GENERATOR PREVENTIVE MAINTENANCE TESTING, AND WAS NOT THE PRIMARY SOURCE OF LUBE OIL HEADER PRESSURE. PRESSURE WAS PRIMARILY SUPPLIED BY THE (TURBINE) SHAFT DRIVEN OIL PUMP, WHICH WAS OPERATING CORRECTLY. ALL EQUIPMENT OPERATED AS EXPECTED IN RESPONSE TO THE TRIP AND THE PLANT RETURNED TO POWER OPERATION (MODE 1) AT APPROXIMATELY 2012 HOURS. THE CAUSE FOR THIS EVENT HAS NOT BEEN DETERMINED. A SUPPLEMENTAL LICENSEE EVENT REPORT WILL BE SUBMITTED ON OR ABOUT FEBRUARY 28, 1988, FOLLOWING REFUELING OUTAGE EQUIPMENT INSPECTIONS.

[87] MILLSTONE 3 DOCKET 50-423 LER 87-032
INADVERTENT DISCHARGE OF CO(2) DUE TO PROCEDURAL DEFECT.
EVENT DATE: 070687 REPORT DATE: 080587 NSSS: WE TYPE: PWR

(NSIC 205638) ON JULY 6, 1987 AT 1232 PM, WHILE OPERATING AT 100% POWER (MODE 1), TECHNICAL SPECIFICATIONS WERE VIOLATED WHEN FIRE WATCHES WERE NOT POSTED DUE TO AN INADVERTENT DISCHARGE OF THE CARBON DIOXIDE (CO2) FIRE SUPPRESSION SYSTEM IN THE EAST MCC/ROD CONTROL AREA. THE DISCHARGE OCCURRED WHILE NON-LICENSED OPERATORS WERE PERFORMING A SURVEILLANCE ON THE FIRE DETECTION SYSTEM, WHICH PROVIDES THE INPUT TO ACTUATE THE CARBON DIOXIDE SYSTEM. THE DISCHARGE WAS DUE TO A PROCEDURAL DEFICIENCY. THE OPERATION OF THE PLANT WAS NOT AFFECTED BY THE DISCHARGE. THERE WERE NO INJURIES. INITIAL OPERATOR ACTIONS WERE TO EVACUATE THE AFFECTED AREA, EVACUATE THE ADJACENT AREAS, AND ALLOW THE CO2 DISCHARGE TO COMPLETE A FULL CYCLE. THE ROOT CAUSE OF THE EVENT WAS A DEFICIENT PROCEDURE IN THAT THE TEST OF THE ZONE MODULES IN THE FIRE DETECTION SYSTEM PANELS WAS ALSO CAPABLE OF ACTUATING THE CO2 SYSTEM, BUT NO PRECAUTIONS OR RESETS WERE INCLUDED IN THE PROCEDURE. IMMEDIATE CORRECTIVE ACTION CONSISTED OF SUSPENDING TESTING UNTIL THE PROCEDURAL DEFICIENCY COULD BE RECTIFIED AND REMOVING THE CO2 SYSTEM FROM SERVICE UNTIL IT COULD BE RESTORED DELIBERATELY AND SAFELY. FIRE SUPPRESSION WAS RESTORED TO MOST AREAS IN APPROXIMATELY 2 HOURS, AND TO ALL AREAS IN APPROXIMATELY 8 HOURS. ACTION TO PREVENT RECURRENCE CONSISTED OF REVISING THE SURVEILLANCE PROCEDURE TO LOCK-OUT.

[88] NINE MILE POINT 2 DOCKET 50-410 LER 87-042
REACTOR CLEANUP SYSTEM ISOLATION DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 061487 REPORT DATE: 071487 NSSS: GE TYPE: BWR

(NSIC 205477) ON JUNE 14, 1987 AT 1657 HOURS, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED THE ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF), SPECIFICALLY, ISOLATION OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM. AT THE TIME OF THE EVENT, THE PLANT WAS AT APPROXIMATELY 2% POWER WITH THE MODE SWITCH IN THE "STARTUP" POSITION. REACTOR PRESSURE AND TEMPERATURE WERE APPROXIMATELY 409 POUNDS PER SQUARE INCH GAUGE (PSIG) AND 445F, RESPECTIVELY. THE ROOT CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE A PROCEDURAL DEFICIENCY. AS A RESULT OF THIS DEFICIENCY, A NIAGARA MOHAWK OPERATOR REPOSITIONED THE RWCU REJECT FLOW CONTROL VALVE SUBSEQUENT TO HAVING ESTABLISHED REJECT FLOW TO THE MAIN CONDENSER. REPOSITIONING THE REJECT FLOW CONTROL VALVE INDUCED ERRATIC FLOW OSCILLATIONS ON THE REJECT FLOW TRANSMITTERS AND INITIATED THE RWCU ISOLATION. CORRECTIVE ACTION HAS BEEN TO REVISE THE DEFICIENT PROCEDURE. THE REVISED PROCEDURE INSTRUCTS OPERATORS TO USE THE FEEDWATER LEVEL CONTROL VALVES TO MAINTAIN REACTOR WATER LEVEL, RATHER THAN USE OF THE RWCU SYSTEM. A CAUTION HAS BEEN INCLUDED TO INSTRUCT OPERATORS THAT THE REJECT FLOW CONTROL VALVE SHALL NOT BE REPOSITIONED WITH A FILTER/DEMINERALIZER INSERVICE AND TOTAL REJECT FLOW TO THE MAIN CONDENSER.

[89] NINE MILE POINT 2 DOCKET 50-410 LER 87-034
REACTOR RECIRCULATION PUMP TRIP DUE TO TROUBLESHOOTING ON RRCS CIRCUITRY.
EVENT DATE: 061787 REPORT DATE: 071487 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 205476) WHILE IN COLD SHUTDOWN ON JUNE 17, 1987, BOTH REACTOR RECIRCULATION PUMPS TRIPPED DUE TO TROUBLESHOOTING ACTIVITIES ON REDUNDANT REACTIVITY CONTROL SYSTEM (RRCS) CIRCUITRY. THE UNIT HAD BEEN SHUTDOWN SINCE JUNE 15, 1987 WHEN AN ALTERNATE ROD INSERTION (ARI) AND REACTOR SCRAM WERE RECEIVED DURING PERFORMANCE OF AN RRCS SURVEILLANCE TEST. THE CAUSES OF THE EVENT WERE THE RAPID POWERING UP AND DOWN OF RRCS POWER SUPPLIES DURING TROUBLESHOOTING AND A FAILED HIGH POWER OUTPUT ISOLATOR (HPOI) CARD. THIS EVENT IS REPORTED VOLUNTARILY AS AN AUTOMATIC ACTUATION OF A SAFETY SYSTEM NOT REQUIRED FOR SAFE SHUTDOWN. IMMEDIATE CORRECTIVE ACTIONS WERE TO RESTART THE RECIRCULATION PUMPS IN SLOW SPEED AND TO RESTORE REACTOR WATER LEVEL TO NORMAL WITHIN ONE HOUR. THE FAILED HPOI CARD WAS REPLACED, POST-MAINTENANCE TESTS PERFORMED, AND THE RRCS WAS RETURNED TO SERVICE ON JULY 1, 1987.

[90] NINE MILE POINT 2 DOCKET 50-410 LER 87-037
MAIN STEAM TUNNEL DIFFERENTIAL TEMPERATURE INSTRUMENTATION INOPERABLE DUE TO A DESIGN DEFICIENCY.
EVENT DATE: 062287 REPORT DATE: 072287 NSSS: GE TYPE: BWR

(NSIC 205542) ON JUNE 24, 1987 A LIMITING CONDITION FOR OPERATION (LCO) AS DEFINED BY TECHNICAL SPECIFICATION 3.3.2 WAS FOUND TO HAVE BEEN EXCEEDED AT NINE MILE POINT UNIT 2. THE LCO WAS EXCEEDED AS THE RESULT OF THE UTILIZATION OF INOPERABLE INSTRUMENTATION IN THE LEAK DETECTION SYSTEM. NINE MILE POINT UNIT 2 WAS AT LESS THAN 2% POWER WITH THE MODE SWITCH IN THE "STARTUP" POSITION. THE CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE A DESIGN DEFICIENCY. TEMPERATURE ELEMENTS 2MSS*TE48A, B, C, D MEASURE THE SUPPLY AIR TEMPERATURE TO THE MAIN STEAM TUNNEL. THESE ELEMENTS PROVIDE FEEDBACK TO THE MAIN STEAM TUNNEL DIFFERENTIAL TEMPERATURE INSTRUMENTATION AND ARE REQUIRED TO BE OPERABLE AS DEFINED IN TECHNICAL SPECIFICATION 3.3.2. 2MSS*TE48B, D WERE INSTALLED AS DESIGNED AWAY FROM THE INLET AIR STREAM AND WERE SENSING AMBIENT AIR TEMPERATURE. THESE TEMPERATURE ELEMENTS INDICATED IN A NON-CONSERVATIVE DIRECTION. CORRECTIVE ACTION HAS BEEN TO MODIFY THE SUPPLY DUCT TO DIRECT SUPPLY AIR TO THE SUBJECT TEMPERATURE ELEMENTS.

[91] NINE MILE POINT 2 DOCKET 50-410 LER 87-041
TECHNICAL SPECIFICATION VIOLATION RESULTS FROM FAILURE TO INCREASE SURVEILLANCE MONITORING OF THE SERVICE WATER SUPPLY TEMPERATURE - PROCEDURAL DEFICIENCY.
EVENT DATE: 071087 REPORT DATE: 073187 NSSS: GE TYPE: BWR

(NSIC 205633) ON JULY 11, 1987 AT 2036 WITH THE REACTOR IN HOT SHUTDOWN (OPERATIONAL CONDITION 3) AND AT A TEMPERATURE AND PRESSURE OF 317 DEGREES FAHRENHEIT AND 100 POUNDS PER SQUARE INCH GAUGE RESPECTIVELY, IT WAS DETERMINED THAT NINE MILE POINT UNIT 2 WAS IN VIOLATION OF TECHNICAL SPECIFICATION (TS) SECTION 4.7.1.1.1A-3. THIS VIOLATION OCCURRED DUE TO THE FAILURE TO PERFORM INCREASED SURVEILLANCE MONITORING OF THE SERVICE WATER SUPPLY HEADER WATER TEMPERATURE. SEVEN SURVEILLANCES REQUIRED BY TS SECTION 4.7.1.1.1A-3 WERE NOT PERFORMED OVER A 28 HOUR PERIOD. THE ROOT CAUSE OF THIS EVENT IS A PROCEDURAL DEFICIENCY. THE CORRECTIVE ACTIONS TAKEN SUBSEQUENT TO THIS EVENT ARE: 1) THE SURVEILLANCE MONITORING WAS INCREASED TO TWO HOURS. 2) THE SURVEILLANCE PROCEDURE HAS BEEN REVISED TO INCLUDE TS OPERABILITY LIMITS FOR THE SERVICE WATER SUPPLY HEADER TEMPERATURE. 3) OTHER SURVEILLANCE LOGS WILL BE REVIEWED TO ENSURE APPLICABLE TS OPERABILITY LIMITS ARE INCORPORATED AS REQUIRED. 4) THIS EVENT WILL BE INCLUDED IN THE OPERATION'S DEPARTMENT LESSONS LEARNED BOOK. 5) A TRAINING MODIFICATION RECOMMENDATION HAS BEEN SUBMITTED REQUESTING OPERATOR TRAINING CONCERNING THIS EVENT.

[92] NORTH ANNA 1 DOCKET 50-338 LER 87-015
REACTOR TRIP DUE TO 5A FEEDWATER HEATER HIGH-HIGH LEVEL.
EVENT DATE: 062987 REPORT DATE: 072887 NSSS: WE TYPE: PWR

(NSIC 205617) ON JUNE 29, 1987, AT 2240 HOURS, UNIT 1 TRIPPED FROM 18 PERCENT POWER. UNIT 2 WAS STABLE IN MODE 1 AT 92 PERCENT POWER. THE INITIATING SIGNAL FOR THIS REACTOR TRIP WAS A TURBINE SOLENOID TRIP WHICH RESULTED FROM A 5A FEEDWATER HEATER HIGH-HIGH LEVEL SIGNAL. THE HIGH-HIGH LEVEL IN THE 5A FEEDWATER HEATER WAS CAUSED BY AN IMPROPER VALVE LINE-UP FOLLOWING A REFUELING OUTAGE. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV). THE ROOT CAUSE FOR THE IMPROPER VALVE LINE-UP IS FAILURE TO FOLLOW ADMINISTRATIVE CONTROLS FOR REMOVING DANGER TAGS AND RETURNING VALVES TO SERVICE.

[93] NORTH ANNA 2 DOCKET 50-339 LER 87-007
SURVEILLANCE TEST OF LOW PRESSURE CO2 SYSTEM MISSED DUE TO INADEQUATE SCHEDULING CONTROLS.
EVENT DATE: 070887 REPORT DATE: 080687 NSSS: WE TYPE: PWR

(NSIC 205618) ON JULY 8, 1987, WITH UNIT 1 IN MODE 1 AT 50 PERCENT POWER AND UNIT 2 1 IN MODE 1 AT 85 PERCENT POWER, IT WAS DISCOVERED THAT THE LOW PRESSURE CO2-TOTAL FLOODING ZONES SURVEILLANCE TEST (2-PT-102.1), REQUIRED BY TECHNICAL SPECIFICATION 3.4.14.2, TECHNICAL SPECIFICATION 4.0.3, AND THE NATIONAL FIRE PROTECTION ASSOCIATION CODE 12, HAD NOT BEEN PERFORMED SATISFACTORY ON UNIT 2 SINCE JANUARY 28, 1983. THIS SURVEILLANCE IS PERFORMED TO ENSURE PROPER OPERATION OF THE ALARMS, LIGHTS, VALVES AND NOZZLES ASSOCIATED WITH THE FIRE PROTECTION SYSTEM FOR THE UNIT 2 EMERGENCY DIESEL ROOMS, CABLE VAULTS, AND CABLE TUNNELS. THE MISSED SURVEILLANCE WAS DUE TO INADEQUATE SCHEDULING CONTROLS. AS A CORRECTIVE ACTION, 2-PT-102.1 HAS BEEN PLACED ON THE PERIODIC TEST SCHEDULING SYSTEM (PTSS). ADDITIONALLY, DURING 1985, ADMINISTRATIVE PROCEDURES WERE WRITTEN WHICH PROVIDE INSTRUCTIONS AND SIGN-OFF SHEETS FOR SCHEDULING NEW AND REVISED PERIODIC TESTS. THESE ADMINISTRATIVE PROCEDURES ENSURE THAT ANY SUBSEQUENTLY REVISED SURVEILLANCE PROCEDURES ARE ENTERED INTO PTSS. THE PERIODIC TEST SCHEDULING SYSTEM WILL BE REVIEWED TO REVERIFY THAT ALL CURRENTLY APPROVED SURVEILLANCE PROCEDURES ARE PROPERLY SCHEDULED. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B).

[94]   OCONEE 1   DOCKET 50-269   LER 87-006
MISSED INSERVICE INSPECTION RESULTING IN TECHNICAL SPECIFICATION VIOLATIONS.
EVENT DATE: 060287   REPORT DATE: 071787   NSSS: BW   TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
                      OCONEE 3 (PWR)

(NSIC 205521) ON MAY 21, 1987 DURING A QUALITY ASSURANCE REVIEW OF COMPLETED WORK REQUESTS, IT WAS DETERMINED THAT A REQUIRED VALVE BODY PRESSURE TEST WAS NOT PERFORMED ON VALVES 3HP-43 AND 3LPSW-18 WHICH HAD RECENTLY BEEN REPAIRED OR REPLACED. WHEN THE PRESSURE TEST WAS NOT PERFORMED A VIOLATION OF TECHNICAL SPECIFICATION 4.0.4 AND 4.2.1 OCCURRED. THE ROOT CAUSE OF THIS INCIDENT WAS DETERMINED TO BE A PERSONNEL ERROR. THE INDIVIDUAL RESPONSIBLE FOR REVIEWING THE WORK REQUEST IN THIS INCIDENT AND INITIATING THE REQUIRED TESTS FAILED TO DO SO. THE INDIVIDUAL WAS KNOWLEDGEABLE OF HIS RESPONSIBILITIES AND REMEMBERS REVIEWING THE WORK REQUEST, BUT FAILED TO FOLLOW-UP. THE IMMEDIATE CORRECTIVE ACTION WAS TO PERFORM THE REQUIRED PRESSURE TEST ON THE TWO VALVES. THIS WAS COMPLETED AND DOCUMENTED ON APRIL 22, 1987. THERE WERE NO RELEASES OF RADIOACTIVITY ASSOCIATED WITH THIS EVENT, AS SUCH THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED.

[95]   PALISADES   DOCKET 50-255   LER 87-018
IMPROPER VALVE OPERATIONS RESULTS IN REACTOR CRITICAL UNDER 525F.
EVENT DATE: 062087   REPORT DATE: 072087   NSSS: CE   TYPE: PWR
VENDOR: FISHER CONTROLS CO.
        WESTINGHOUSE ELECTRIC CORP.

(NSIC 205441) ON JUNE 20, 1987 AT APPROXIMATELY 2338, WITH THE REACTOR CRITICAL AT APPROXIMATELY 10 PERCENT OF RATED POWER, THE PRIMARY COOLANT SYSTEM TEMPERATURE DROPPED BELOW 525 DEGREES F. THIS OCCURRENCE, WHICH IS CONTRARY TO PALISADES TECHNICAL SPECIFICATION 3.1.3(C) RESULTED FROM IMPROPER VALVE OPERATION. DURING A POWER REDUCTION INITIATED BY AN ELECTRO-HYDRAULIC SYSTEM FLUID LEAK, THE "A" TRAIN MAIN FEEDWATER REGULATING VALVE CV-0701 FAILED TO CLOSE. THIS RESULTED IN CONTINUED FLOW TO STEAM GENERATOR E-50A. SHORTLY AFTER, WHEN THE TURBINE GENERATOR WAS REMOVED FROM SERVICE, A MOISTURE SEPARATOR AND REHEATER CONTROL VALVE FAILED TO CLOSE. THIS RESULTED IN CONTINUED STEAM DRAW FROM THE STEAM GENERATORS. THE COMBINED EFFECTS OF THESE FAILURES RESULTED IN THE PRIMARY COOLANT SYSTEM DROPPING BELOW 525 DEGREES FOR APPROXIMATELY 30 SECONDS. OPERATIONS DEPARTMENT PERSONNEL INITIATED ACTION TO ISOLATE STEAM SUPPLY TO THE MOISTURE SEPARATOR REHEATERS AND TO HAVE INSTRUMENT AND CONTROL PERSONNEL AFFECT REPAIRS ON THE FEEDWATER REGULATION VALVE. SYSTEM MODIFICATIONS AND REPAIRS HAVE BEEN COMPLETED TO PRECLUDE RECURRENCE. THIS EVENT IS BEING REPORTED PER 10CFR50.73(A)(2)(I) AS AN OPERATIONAL CONDITION PROHIBITED BY PLANT TECHNICAL SPECIFICATIONS.

[96]   PALISADES   DOCKET 50-255   LER 87-019
FAILURE OF AUTOSTOP OIL PRESSURE SWITCH RESULTS IN INADVERTENT DIESEL GENERATOR ACTUATION.
EVENT DATE: 062187   REPORT DATE: 072187   NSSS: CE   TYPE: PWR
VENDOR: MESKER, GEORGE L. CO.

(NSIC 205497) ON JUNE 21, 1987 AT 0820, DIESEL GENERATORS 1-1 AND 1-2 (EK;DG) WERE INADVERTENTLY ACTUATED WHILE OPERATIONS DEPARTMENT PERSONNEL ATTEMPTED TO "LATCH" (BRING ON-LINE) THE TURBINE GENERATOR (SB;TG). THE PLANT WAS IN HOT STANDBY CONDITION (PRIMARY COOLANT SYSTEM: 533 DEGREES F, 2016 PSIA) AT THE TIME OF THE EVENT. A RAPID INCREASE IN AUTOSTOP OIL PRESSURE, INHERENT TO SYSTEM DESIGN, CAUSED AUTOSTOP OIL PRESSURE SWITCHES 63 AST/1&2 (TD;PS) TO ENERGIZE AND THEN DEENERGIZE. THE DEENERGIZATION OF THE PRESSURE SWITCH WAS THE RESULT OF MERCURY (THE CONDUCTIVE MECHANISM) BOUNCING WITHIN THE SWITCH. UPON PRESSURE SWITCH DEENERGIZATION, TURBINE TRIP AUXILIARY RELAYS 305-L&R/AST ENERGIZED AND ACTUATED THE DIESEL GENERATORS.

[97]   PALO VERDE 1   DOCKET 50-528   LER 87-017
AUXILIARY OPERATOR ENTERS HIGH RADIATION AREA WITHOUT PROPER RADIATION MONITORING DEVICE.
EVENT DATE: 063087   REPORT DATE: 072987   NSSS: CE   TYPE: PWR

(NSIC 205671) AT APPROXIMATELY 1635 ON JUNE 30, 1987, PALO VERDE UNIT 1 WAS IN MODE 4 (HOT SHUTDOWN) WHEN IT WAS DISCOVERED THAT AN AUXILIARY OPERATOR (UTILITY NON-LICENSED) HAD ENTERED A HIGH RADIATION AREA WITHOUT THE RADIATION MONITORING DEVICES THAT ARE REQUIRED BY TECH SPECS. THE OPERATOR RECEIVED AN INTAKE OF 6.6 MPC (MAXIMUM PERMISSIBLE CONCENTRATION HOURS. HIS EXPOSURE WAS DETERMINED TO BE 101 MILLIRAD TO THE THYROID AND LESS THAN 1 MILLIRAD WHOLE BODY DOSE, WHICH ARE WITHIN THE 10CFR 20 EXPOSURE LIMITS. THE HIGH RADIATION AREA WAS PROPERLY POSTED, HOWEVER THE OPERATOR DID NOT REALIZE THAT HE WAS ENTERING A HIGH RADIATION AREA. THIS WAS A COGNITIVE PERSONNEL ERROR THAT WAS CONTRARY TO APPROVED PROCEDURES. TO PREVENT RECURRENCE APPROPRIATE DISCIPLINARY ACTION HAS BEEN TAKEN. NO SIMILAR EVENTS HAVE BEEN REPORTED.

[98]   PALO VERDE 3   DOCKET 50-530   LER 87-002
CPIS AND CREFAS ACTUATIONS DUE TO UNKNOWN CAUSE.
EVENT DATE: 061587   REPORT DATE: 071387   NSSS: CE   TYPE: PWR
VENDOR: ELGAR, CORP.

(NSIC 205430) AT APPROXIMATELY 0611 MST ON JUNE 15, 1987, PALO VERDE UNIT 3 WAS IN MODE 5 (COLD SHUTDOWN) WHEN CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNALS (CREFAS) AND CONTAINMENT RGE ISOLATION ACTUATION SIGNALS (CPIAS) WERE RECEIVED ON BOTH CHANNELS OF THE LANCE OF PLANT ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (BOP ESFAS). THESE BOP ESFAS ACTUATIONS WERE ANNUNCIATED IN THE CONTROL ROOM ON THE MAIN CONTROL BOARD. ALL ASSOCIATED EQUIPMENT RESPONDED AS DESIGNED. CONTROL ROOM PERSONNEL SECURED ALL ACTUATED EQUIPMENT IN ACCORDANCE WITH APPROVED PROCEDURES BY 0657 MST ON JUNE 15, 1987. THE DURATION OF THIS EVENT WAS APPROXIMATELY 46 MINUTES. AT APPROXIMATELY 0611 MST, THE DISTRIBUTION PANEL D26 MOMENTARILY LOST POWER AND A "120VAC INV B AC/DC STATUS" TROUBLE ALARM WAS RECEIVED FOR INVERTER "B". AS DESIGNED, THE STATIC TRANSFER SWITCH AUTOMATICALLY TRANSFERRED FROM THE "B" INVERTER TO THE 120 VAC VOLTAGE REGULATOR. IMMEDIATELY FOLLOWING THE STATIC TRANSFER SWITCH OPERATION, BOTH CHANNELS OF CREFAS AND CPIAS WERE ACTUATED ON THE BOP ESFAS SYSTEM. THE ROOT CAUSE OF THIS EVENT IS CURRENTLY UNDER INVESTIGATION. A SUPPLEMENT TO THIS LER WILL BE SUBMITTED FOLLOWING THE COMPLETION OF THE INVESTIGATION AND WILL ADDRESS THE ROOT CAUSE AND CORRECTIVE ACTION.

[99]   PEACH BOTTOM 2   DOCKET 50-277   LER 87-007
REACTOR SCRAM DURING INTERMEDIATE RANGE MONITOR TESTING CAUSED BY PROCEDURAL DEFICIENCIES AND PERSONNEL ERRORS.
EVENT DATE: 061987   REPORT DATE: 071787   NSSS: GE   TYPE: BWR

(NSIC 205444) ON APRIL 19, 1987, WITH UNIT 2 IN THE REFUELING MODE WITH THE CORE OFF LOADED, A FULL SCRAM SIGNAL WAS GENERATED BY THE REACTOR PROTECTION SYSTEM (RPS) LOGIC. THE SCRAM OCCURRED DURING THE PERFORMANCE OF THE SURVEILLANCE TEST PROCEDURE FOR VERIFICATION OF INTERMEDIATE RANGE MONITOR (IRM) OPERABILITY. THE TEST INVOLVES THE VERIFICATION OF PROPER IRM INPUT TO THE RPS, AND PROPER OPERATION OF IRM ALARMS AND INDICATORS. THE SCRAM WAS CAUSED BY PROCEDURAL DEFICIENCIES, COMBINED WITH PERSONNEL ERROR. PROCEDURAL REVISIONS ARE BEING MADE AND DISCIPLINARY GUIDELINES HAVE BEEN EXERCISED AS PART OF THE EFFORTS TO PREVENT RECURRENCE. THE UNPLANNED RPS ACTUATION MAKES THIS EVENT REPORTABLE.

[100] PEACH BOTTOM 3 DOCKET 50-278 LER 87-001 REV 01
UPDATE ON SCRAM DUE TO HIGH FLUX RESULTING FROM TURBINE CONTROL VALVE FLUCTUATIONS.
EVENT DATE: 031787 REPORT DATE: 071787 NSSS: GE TYPE: BWR
VENDOR: SPARTON SOUTHWEST INC.

(NSIC 205445) ON MARCH 17, 1987 AT 0254 HOURS, UNIT 3 SCRAMMED FROM 85% POWER ON HIGH NEUTRON FLUX SIGNALS FROM THE NEUTRON MONITORING SYSTEM. THE UNPLANNED ACTIVATION OF AN ENGINEERED SAFETY FEATURE MAKES THIS EVENT REPORTABLE. THE HIGH FLUX RESULTED FROM AN INCREASE IN REACTOR COOLANT SYSTEM PRESSURE CAUSED BY TURBINE CONTROL VALVE FLUCTUATIONS. INVESTIGATION INDICATED THAT CIRCUITRY IN THE ELECTRO-HYDRAULIC CONTROL SYSTEM (EHC) WAS AFFECTED BY SPURIOUS ELECTRICAL NOISE BELIEVED TO HAVE BEEN CAUSED BY MALFUNCTIONING CABINET COOLING FANS. PHILADELPHIA ELECTRIC COMPANY INVESTIGATIONS HAVE DETERMINED THAT OPERATORS WERE INATTENTIVE TO INSTRUMENTATION INDICATIONS DURING THE HOURS PRECEDING THE SCRAM. FIVE FANS WERE ELECTRICALLY DISCONNECTED, 4 FANS WERE REPLACED, AND 4 EHC CARDS WERE REPLACED. TO PREVENT RECURRENCE, THE COOLING FANS IN ALL 6 EHC ELECTRONICS BAYS WILL BE REPLACED, AS PART OF THE PREVENTIVE MAINTENANCE PROGRAM, DURING EVERY OTHER REFUELING OUTAGE. THE OPERATORS ON DUTY DURING THIS EVENT RECEIVED COUNSELING REGARDING THE IMPORTANCE OF ATTENTIVENESS. THE OPERATIONS STAFF RECEIVED A LETTER FROM PLANT MANAGEMENT, EMPHASIZING THE NEED TO MONITOR VARIOUS PLANT PARAMETERS AND RESPOND TO DEVIATIONS FROM NORMAL CONDITIONS. OTHER THAN THE UNIT SHUTDOWN TRANSIENT, THERE WERE NO ADVERSE CONSEQUENCES OF THIS EVENT.

[101] PERRY 1 DOCKET 50-440 LER 87-041
PLANT OPERATOR UNINTENTIONALLY DEPRESSED A RADIATION MONITOR TRIP TEST BUTTON RESULTING IN THE BACKUP HYDROGEN PURGE SYSTEM ISOLATING.
EVENT DATE: 061187 REPORT DATE: 070387 NSSS: GE TYPE: BWR

(NSIC 205464) ON JUNE 11, 1987 AT 1405 THE BACKUP HYDROGEN PURGE SYSTEM ISOLATED DUE TO AN INADVERTENT TRIP SIGNAL ON THE DRYWELL ATMOSPHERIC GASEOUS RADIATION MONITOR. THE BACKUP HYDROGEN PURGE SYSTEM WAS IN OPERATION TO VENT THE DRYWELL DUE TO HIGH PRESSURE FOLLOWING PLANT HEATUP. AN "ALERT" ALARM WAS RECEIVED ON THE DRYWELL ATMOSPHERIC GASEOUS RADIATION MONITOR DUE TO THE LEVELS OF BROMINE 82 AS A RESULT OF PAINT CURING IN THE DRYWELL. WHILE ACKNOWLEDGING THE ALARM A SUPERVISING OPERATOR PRESSED THE "ALARM TRIP TEST" BUTTON INSTEAD OF THE "ALARM ACKNOWLEDGE" BUTTON. THIS INITIATED AN ISOLATION SIGNAL TO THE BACKUP HYDROGEN PURGE SYSTEM AND CAUSED A DRYWELL EVACUATION ALARM. THE TRIP WAS RESET AND DRYWELL VENTING RECOMMENCED AT 1410. THE CAUSE OF THE EVENT WAS OPERATOR ERROR. IN AN ATTEMPT TO ACKNOWLEDGE AN ALARM THE OPERATOR UNINTENTIONALLY PRESSED THE ALARM TRIP TEST BUTTON ON THE DRYWELL ATMOSPHERIC GASEOUS RADIATION MONITOR. THE OPERATOR INVOLVED WITH THE EVENT HAS BEEN COUNSELED ON THE NEED FOR ATTENTION TO DETAIL.

[102] PERRY 1 DOCKET 50-440 LER 87-042
DEENERGIZATION OF RPS BUS DUE TO FAILED LOGIC CARD RESULTS IN CONTAINMENT ISOLATION, PLUS UNEXPECTED MAIN STEAM ISOLATION VALVE CLOSURE AND REACTOR SCRAM.
EVENT DATE: 061787 REPORT DATE: 071787 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 205486) ON JUNE 17, 1987 AT 1139, A REACTOR PROTECTION SYSTEM (RPS) ELECTRICAL PROTECTION ASSEMBLY (EPA) BREAKER TRIPPED, RESULTING IN DEENERGIZATION OF RPS BUS A AND CLOSURE OF THE OUTBOARD MAIN STEAM ISOLATION VALVES (MSIV). CLOSURE OF THE MSIVS RESULTED IN A REACTOR SCRAM. LOSS OF THE RPS BUS ALSO RESULTED IN A NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) ISOLATION SIGNAL AND CLOSURE OF THE ASSOCIATED CONTAINMENT ISOLATION VALVES. THE ROOT CAUSE OF THE CONTAINMENT ISOLATION WAS A FAILED COMPONENT OF AN EPA LOGIC CARD WHICH PRODUCED SPURIOUS OUTPUT VOLTAGE PULSES AND RESULTANT TRIPPING OF THE EPA BREAKER. THE CAUSE OF THE MSIV CLOSURE AND RESULTANT SCRAM WAS THE DESIGN OF THE POWER

SUPPLIES TO THE MSIV PILOT VALVE SOLENOIDS. BOTH OF THE THE SOLENOIDS FOR EACH OUTBOARD MSIV WERE POWERED FROM THE SAME RPS BUS, THEREFORE, MSIV CLOSURE RESULTED WHEN NORMAL RPS POWER WAS LOST TO BOTH SOLENOIDS. THE LOGIC CARDS FOR THE RPS BUS A EPAS WERE REPLACED, CHANGES WERE MADE TO THE APPROPRIATE PLANT OPERATING INSTRUCTIONS, AND SURVEILLANCE TESTING OF THE NORMAL RPS POWER SUPPLY OUTPUT BREAKERS WAS COMPLETED ON JUNE 20. THE EPA CARDS FOR THE NORMAL RPS B POWER SUPPLY WERE REPLACED AND THE POWER SUPPLY CONFIGURATION TO THE INBOARD AND OUTBOARD MSIV SOLENOIDS WAS MODIFIED DURING THE CURRENT PLANT OUTAGE.

[103] PERRY 1 DOCKET 50-440 LER 87-043
INSTRUMENT DRIFT OF AEGTS EXHAUST FAN B DIFFERENTIAL PRESSURE SWITCH CAUSES AUTOMATIC START OF AEGTS EXHAUST FAN A.
EVENT DATE: 062287 REPORT DATE: 071787 NSSS: GE TYPE: BWR

(NSIC 205487) ON JUNE 22, 1987 AT 0115, THE STANDBY TRAIN OF THE ANNULUS EXHAUST GAS TREATMENT SYSTEM (AEGTS) AUTOMATICALLY STARTED. THE CAUSE OF THIS EVENT WAS INSTRUMENT SETPOINT DRIFT OF THE 1M15-N061B LOW DIFFERENTIAL PRESSURE (D/P) SWITCH. DUE TO DRIFT OF THE SETPOINT, AEGTS EXHAUST FAN A AUTOSTARTED ON THE D/P SENSED BY 1M15-N061B. SETPOINT CHANGE REQUESTS HAD BEEN INITIATED PRIOR TO THIS EVENT AS A RESULT OF AN EVALUATION OF PREVIOUS CALIBRATION PROBLEMS WITH 1M15-N061A & B. THESE CHANGES RESULT IN INCREASING THE ALLOWABLE BAND BASED UPON THE ACCURACY OF THE INSTRUMENT, AND REDUCING THE AMOUNT OF CONSERVATISM FOR THE LOW D/P SETPOINT. HOWEVER, THESE CHANGES HAD NOT BEEN IMPLEMENTED AT THE TIME OF THIS EVENT. THE SETPOINT CHANGES FOR D/P INSTRUMENTS 1M15-N061A & B HAVE BEEN IMPLEMENTED. NO ADDITIONAL CORRECTIVE ACTIONS ARE PLANNED AT THIS TIME. HOWEVER, THE ROUTINE SYSTEM PERFORMANCE MONITORING WILL CONTINUE AS REQUIRED.

[104] PERRY 1 DOCKET 50-440 LER 87-044
TRANSMITTER SENSING LINE PRESSURE ANOMALIES RESULT IN REACTOR CORE ISOLATION COOLING SYSTEM ISOLATIONS.
EVENT DATE: 062487 REPORT DATE: 072187 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 205488) ON JUNE 24, AT 1328, AND JUNE 28, AT 2204 AND 2337 THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM AUTOMATICALLY ISOLATED. AT THE TIME OF THESE EVENTS THE RCIC SYSTEM WAS IN STANDBY READINESS. ON JUNE 24 AT 1328 THE RCIC SYSTEM ISOLATED DURING TROUBLESHOOTING OF SIGNAL OSCILLATIONS GENERATED ON RCIC LEAK DETECTION (LD) HIGH FLOW TRANSMITTER 1E31-N084B. ON JUNE 28 AT 2204 AND 2337 RCIC AUTOMATICALLY ISOLATED DUE TO CONTINUED SIGNAL OSCILLATIONS ON RCIC/LD HIGH FLOW TRANSMITTER 1E31-N084A. THE INITIATING CAUSE OF THESE EVENTS IS PRESSURE OSCILLATIONS IN THE RCIC/LD HIGH FLOW INSTRUMENT PROCESS LINES. DURING TROUBLESHOOTING OF THESE OSCILLATIONS ON JUNE 24 A LEAK DEVELOPED ON THE B TRANSMITTER TEST CONNECTION WHICH CAUSED THE OUTPUT SIGNAL TO GO OFFSCALE LOW, RESULTING IN A RCIC ISOLATION. THE AMPLITUDE OF THE OSCILLATIONS INCREASED WITH INCREASING REACTOR POWER AND ON JUNE 28 CAUSED THE A TRANSMITTER TO REACH THE HIGH FLOW TRIP SETPOINT, RESULTING IN A RCIC ISOLATION. ON JUNE 30 THE LEAKING TRANSMITTER TEST CONNECTION WAS REPAIRED. A 25 MICRON SINTERED DISC (SNUBBER) HAS BEEN ADDED TO THE INSTRUMENT PROCESS LINES FOR THE 1E31-N084A AND B TRANSMITTERS IN ORDER TO DAMPEN THE EFFECT OF SYSTEM PRESSURE OSCILLATIONS. THE EFFECTIVENESS OF THESE SNUBBERS WILL BE EVALUATED DURING THE NEXT PLANT STARTUP.

[105] PRAIRIE ISLAND 1 DOCKET 50-282 LER 87-009
ONE SAFETY INJECTION PUMP DID NOT START ON DEMAND WHEN ITS MOTOR BREAKER WAS NOT RACKED IN PROPERLY.
EVENT DATE: 061887 REPORT DATE: 071787 NSSS: WE TYPE: PWR
VENDOR: I-T-E CIRCUIT BREAKER

(NSIC 205498) ON JUNE 18, 1987, UNIT 1 WAS AT A STEADY STATE 94% POWER. THE

MONTHLY OPERABILITY EST OF THE SAFETY INJECTION PUMPS WAS IN PROGRESS. WHEN NO. 11 SAFETY INJECTION (SI) PUMP WAS GIVEN A START SIGNAL, THE PUMP DID NOT START. OPERATORS IMMEDIATELY EXAMINED THE CONDITION OF THE MOTOR BREAKER (MO)(BKR) FOR NO. 11 SI PUMP AND FOUND IT NOT COMPLETELY RACKED IN, EVEN THOUGH CONTROL ROOM INDICATIONS SHOWED THE BREAKER CONDITION TO BE PROPER. THE LOCKING LEVER TANG WAS NOT ENGAGED IN THE RACKING SCREW LOT; IN THIS CONDITION A MECHANICAL TRIP SIGNAL IS CONTINUOUSLY APPLIED, PREVENTING REAKER CLOSURE. ONE-HALF TURN ON THE RACKING SCREW RESULTED IN PROPER ENGAGEMENT. THEN THE MOTOR BREAKER WAS SUCCESSFULLY CLOSED AND THE OPERABILITY TEST COMPLETED (THE NO. 12 SI PUMP OPERABILITY TEST WAS UNEVENTFUL). INVESTIGATION SHOWED THAT THE MOTOR BREAKER FOR NO. 11 PUMP WAS LAST RACKED IN ON MAY 22 AND UNIT HEATUP OCCURRED N MAY 24, SO THE PUMP MAY HAVE BEEN INOPERABLE FOR 25 DAYS WHILE THE UNIT WAS ABOVE OLD SHUTDOWN. ROOT CAUSE OF THE EVENT WAS FAILURE TO LEAVE THE ITE TYPE 5HK250 4160V BREAKER COMPLETELY RACKED IN. PROCEDURAL DEFICIENCIES CONTRIBUTED TO THE EVENT. THIS EVENT IS REPORTABLE UNDER 10CFR50.73.(A)(2)(I)(B) SINCE NO. 11 PUMP WAS INOPERABLE FOR MORE THAN THE 24 HOURS ALLOWED BY TECHNICAL SPECIFICATIONS.

[106]   PRAIRIE ISLAND 1   DOCKET 50-282   LER 87-010
PERSONNEL ERROR CAUSED AUTO-START OF ONE DIESEL GENERATOR.
EVENT DATE: 061987   REPORT DATE: 071787   NSSS: WE   TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)

(NSIC 205499) ON JUNE 19, 1987, UNIT 1 WAS AT 94.3% POWER AND UNIT 2 WAS AT 99.7% POWER. THE MONTHLY BUS 26 UNDERVOLTAGE RELAY TEST WAS IN PROGRESS. ACTUATION SETTINGS FOR ONE UNDERVOLTAGE RELAY WERE FOUND TO BE OUT OF SPECIFICATION. SINCE THE SETTINGS HAD CHANGED SIGNIFICANTLY FROM THE LAST MONTHLY TEST, THE ELECTRICAL MAINTENANCE SUPERVISOR DECIDED THE RELAY SHOULD BE REPLACED RATHER THAN RECALIBRATED. WHILE THE SYSTEM ENGINEER WAS PREPARING A WORK REQUEST, THE SUPERVISOR WITHDREW A REPLACEMENT RELAY FROM THE WAREHOUSE, AND, SINCE THE DEFECTIVE RELAY APPEARED TO BE ISOLATED PER THE TEST PROCEDURE, THE SUPERVISOR DIRECTED THE ELECTRICIANS TO PREPARE FOR THE REPLACEMENT BY DISCONNECTING WIRING AT THE RELAY. TWO OF THE WIRES THAT WERE DISCONNECTED FORMED A COMMON NEUTRAL FOR ALL OF THE VOLTAGE SENSING SERVICES, SO WHEN THIS NEUTRAL CIRCUIT WAS BROKEN, AT 1252, OTHER UV RELAYS WERE DEENERGIZED AND THE UV SCHEME WAS ACTUATED, WHICH INCLUDES AUTO-START OF DL DIESEL GENERATOR. OPERATOR RESPONSE WAS PROPER, TAKING MANUAL CONTROL AND RESTORING BUS 26 TO THE NORMAL SOURCE WITHIN 7.5 MINUTES. RELAY REPLACEMENT WAS COMPLETED AND THE BUS RESTORED TO AUTOMATIC OPERABILITY BY 1555 HOURS. CAUSE OF THE EVENT WAS PERSONNEL ERROR ON THE PART OF THE SUPERVISOR IN DECIDING TO PROCEED WITH WORK WHICH WAS OUTSIDE THE SCOPE OF THE PROCEDURE IN USE.

[107]   PRAIRIE ISLAND 1   DOCKET 50-282   LER 87-012
THREE SURVEILLANCE TESTS DONE LATE DUE TO SCHEDULING ERROR.
EVENT DATE: 070287   REPORT DATE: 073187   NSSS: WE   TYPE: PWR

(NSIC 205674) ON JULY 2, 1987, UNIT 1 WAS AT 94% POWER. REVIEW OF THE SURVEILLANCE TEST PROGRAM BY THE SURVEILLANCE COORDINATOR REVEALED THAT THREE TESTS WERE DONE OUTSIDE OF THEIR ALLOWABLE TIME SPAN. TWO OF THE TESTS WERE DUE BEFORE JUNE 24TH AND ONE TEST DUE BEFORE JUNE 26TH; ALL WERE DONE ON JUNE 29TH. CAUSE OF THE EVENT WAS PERSONNEL ERROR ON THE PART OF THE SURVEILLANCE COORDINATOR. THE THREE SURVEILLANCE TESTS WERE RESCHEDULED BECAUSE OF THE UNIT 1 REFUELING OUTAGE. IN SETTING THE NEW DATE, THE COORDINATOR INADVERTENTLY TRANSPOSED THE WEEK/DAY CODE FOR THE NEW SCHEDULE, RESULTING IN THE TESTS BEING SCHEDULED LATER THAN THEY SHOULD HAVE BEEN SCHEDULED. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B). THE TESTS, THOUGH DONE LATE, SHOWED THAT THE EQUIPMENT WAS OPERABLE THROUGHOUT THE PERIOD. HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED. THE SYSTEM FOR RESCHEDULING SURVEILLANCE PROCEDURES HAS WORKED QUITE WELL AND NO MAJOR CHANGES ARE FORESEEN. TO HELP PREVENT RECURRENCE OF THIS TYPE OF EVENT, A COPY OF THIS REPORT WILL BE CIRCULATED AMONG RESPONSIBLE

SUPERVISORS TO ENCOURAGE DOUBLE CHECKING DATE CHANGES. SURVEILLANCE TESTS HAVE BEEN MISSED IN THE PAST, BUT NOT DUE TO THIS TYPE OF RESCHEDULING ERROR.

[108] RANCHO SECO DOCKET 50-312 LER 86-021 REV 03
UPDATE ON FAILURE TO IMPLEMENT IN-SERVICE TESTING OF SAFETY-RELATED VALVES.
EVENT DATE: 092486 REPORT DATE: 071787 NSSS: BW TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 205434) DURING COLD SHUTDOWN CONDITIONS, IT WAS REPORTED, AS PART OF NRC INSPECTION REPORT 87-16, THAT SAFETY RELATED VALVES WERE NOT APPROPRIATELY INCLUDED IN THE TECHNICAL SPECIFICATION 4.2.2.1 REQUIRED VALVE IN-SERVICE TESTING PROGRAM. AS A RESULTS THESE VALVES WERE NOT PROPERLY SURVEILLANCE TESTED. THIS RESULTED IN THE PLANT BEING OPERATED WITH SYSTEMS WHICH WERE NOT FORMALLY "OPERABLE" AS DEFINED IN TECHNICAL SPECIFICATION 1.3, ITEM 2. THIS IS REPORTABLE AS A CONDITION PROHIBITED BY THE PLANT TECHNICAL SPECIFICATIONS AS INDICATED IN 10 CFR PART 50.73(A)(2)(I)(B). THE SPECIFIC VALVES WERE TESTED OR ACTUALLY USED ON A PERIODIC BASIS DURING PLANT LIFE, SO THERE IS NO GENUINE CONCERN THAT THE AFFECTED SYSTEMS WOULD NOT HAVE PERFORMED THEIR REQUIRED SAFETY FUNCTIONS. A LIST OF THE PARTICULAR VALVES OF INTERESTS AND A DESCRIPTION OF THEIR FUNCTIONS AND HOW THEY ARE VERIFIED TO FUNCTION PROPERLY IS PROVIDED IN THE TEXT. THE PUBLIC HEALTH AND SAFETY WERE UNAFFECTED BY THIS CONDITION.

[109] RANCHO SECO DOCKET 50-312 LER 87-035
ABANDONMENT OF CONTINUOUS FIRE WATCH FROM FIRE ALARM PANEL.
EVENT DATE: 062587 REPORT DATE: 072587 NSSS: BW TYPE: PWR

(NSIC 205559) DURING COLD SHUTDOWN CONDITIONS ON JUNE 25, 1987, STARTUP TEST PROCEDURE (STP) 199 WAS PERFORMED TO VERIFY A POST DISCHARGE CARBON DIOXIDE CONCENTRATION IN A FIRE ZONE (ZONE 76) LOCATED IN THE NUCLEAR SERVICE ELECTRICAL BUILDING (NSEB). DURING THE PERFORMANCE OF THIS PROCEDURE, THE CONTINUOUS FIRE WATCH WAS ABANDONED FROM MONITORING ALARM PANEL H4FCP5 FOR APPROXIMATELY THREE HOURS. THE CONTINUOUS FIRE WATCH WAS ABANDONED BECAUSE SPURIOUS UNCONTROLLED DISCHARGES OF CARBON DIOXIDE IN THE NSEB EXPOSED PERSONNEL TO THE RISK OF ASPHYXIATION. THESE UNCONTROLLED DISCHARGES OCCURRED FOLLOWING THE PLANNED DISCHARGE PORTION OF THE STP-199, AND DURING THE POST-TEST RECOVERY PORTION OF THE PROCEDURE. PRIOR TO, AND THROUGHOUT THE EVENT, HOURLY FIRE WATCHES CONTINUED TO BE CONDUCTED WITHIN THE NSEB, AND AT THIS PANEL, BY PERSONNEL WEARING PORTABLE BREATHING APPARATUS. THE CONTINUOUS FIRE WATCH WAS REINSTATED AFTER THE SAFETY DEPARTMENT AND THE STARTUP TEST GROUP HAD DETERMINED THE ATMOSPHERE TO BE SAFE FOR BREATHING.

[110] RIVERBEND 1 DOCKET 50-458 LER 86-052 REV 01
UPDATE ON SPURIOUS CONTROL ROOM VENTILATION ISOLATION DUE TO NOISE RESULTING FROM MAINTENANCE WORK ON MONITOR.
EVENT DATE: 082286 REPORT DATE: 072287 NSSS: GE TYPE: BWR

(NSIC 205516) AT 0220 ON 8/22/86 WITH THE UNIT IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) A MAIN CONTROL ROOM VENTILATION SYSTEM ISOLATION OCCURRED. THIS WAS CAUSED BY AN INDUCED SIGNAL ON A RADIATION MONITOR RESULTING FROM MAINTENANCE WORK BEING PERFORMED ON THE MONITOR. THE INVESTIGATION REVEALED THAT ELECTRICAL NOISE COULD BE REDUCED BY INSTALLING ADDITIONAL NOISE SUPPRESSION ON THE PREAMPLIFIER POWER SUPPLY LINES. FURTHER INVESTIGATION WITH THE RADIATION MONITOR VENDOR IS BEING CONDUCTED TO PROVIDE ADDITIONAL REDUCTION IN SENSITIVITY TO ELECTRICAL NOISE. THE SAFE OPERATION OF THE PLANT AND HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED AS A RESULT OF THIS EVENT SINCE THE ACTUATION OF THE ESF WOULD HAVE PREVENTED ANY RADIOACTIVE OUTSIDE AIR FROM ENTERING THE CONTROL ROOM.

[111] RIVERBEND 1 DOCKET 50-458 LER 86-062 REV 01
UPDATE ON AUTOMATIC INITIATION OF SGTS DUE TO A RADIATION MONITOR SPIKE.
EVENT DATE: 102886 REPORT DATE: 071787 NSSS: GE TYPE: BWR

(NSIC 205423) AT 0100 ON 10/28/86 WITH THE UNIT IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN), AN AUTOMATIC INITIATION OF THE DIVISION I STANDBY GAS TREATMENT SYSTEM (SGTS) AND ANNULUS MIXING FAN, BOTH ENGINEERED SAFETY FEATURES, OCCURRED. THE INITIATION WAS CAUSED BY SPURIOUS SIGNALS FROM RADIATION MONITOR 1RMS*RE11A. RADIATION MONITOR 1RMS*RE11A HAS SHOWN PREVIOUS SUSCEPTABILITY TO ELECTRICAL NOISE AND MODIFICATION REQUEST 86-1471 WAS INITIATED TO INSTALL A NOISE FILTER AS A RESULT OF AN EARLIER OCCURRENCE (SEE LER 86-020). HOWEVER, THIS MODIFICATION HAD NOT BEEN IMPLEMENTED AT THE TIME OF THE EVENT BECAUSE IT WAS DETERMINED THAT THE MODIFICATION WOULD NOT COMPLETELY ELIMINATE THE SENSITIVITY OF THE MONITOR TO ELECTRICAL NOISE. FURTHER INVESTIGATION IS BEING CONDUCTED TO PROVIDE ADDITIONAL REDUCTION IN SENSITIVITY TO ELECTRICAL NOISE. THE SAFE OPERATION OF THE PLANT AND HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED AS A RESULT OF THIS EVENT SINCE SYSTEMS WHICH WERE ACTUATED PLACED THE PLANT IN A MORE CONSERVATIVE CONDITION BY FILTERING THE AIR PRIOR TO RELEASING IT TO THE ENVIRONMENT.

[112] RIVERBEND 1 DOCKET 50-458 LER 87-012
REACTOR SCRAM ON HIGH LEVEL SETPOINT DUE TO FEEDWATER REGULATING VALVE LOCKUP.
EVENT DATE: 061887 REPORT DATE: 072087 NSSS: GE TYPE: BWR
VENDOR: ELGAR, CORP.

(NSIC 205489) ON 6/18/87 AT 0322, WITH THE UNIT AT APPROXIMATELY 70 PERCENT POWER, A REACTOR TRIP OCCURRED. INITIATION OF THE REACTOR PROTECTION SIGNAL WAS CAUSED BY A REACTOR VESSEL WATER LEVEL - HIGH LEVEL 8 (51 INCHES) CONDITION. A LOSS OF CONTROL POWER TO PANEL 1VBN-PNL01B1 OCCURRED INADVERTENTLY DURING THE TROUBLE SHOOTING OF BATTERY INVERTER 1BYS- INV01B *INVT*. LOSS OF CONTROL POWER TO THE FEEDWATER REGULATING VALVES CAUSED THEM TO LOCKUP IN A POSITION CONSISTENT WITH 70 PERCENT REACTOR POWER. ALSO, AS A RESULT OF THE LOSS OF THE INVERTER, THE RECIRCULATION SYSTEM FLOW CONTROL VALVES RAN BACK. SIMULTANEOUSLY, THE RECIRCULATION PUMPS RECEIVED A SIGNAL TO TRANSFER TO THE LOW FREQUENCY MOTOR GENERATORS. THIS CAUSED SUFFICIENT FEEDWATER FLOW/STEAM FLOW MISMATCH TO INCREASE THE VESSEL LEVEL TO HIGH LEVEL 8. OPERATIONS PERSONNEL RESPONDED BY SATISFACTORILY IMPLEMENTING THE IMMEDIATE AND SUBSEQUENT ACTIONS REQUIRED BY "REACTOR SCRAM" PROCEDURES. PROCEDURAL REVISIONS HAVE BEEN COMPLETED THAT WILL PRECLUDE RECURRENCE BY REQUIRING THE PLACEMENT OF THE BATTERY INVERTER IN MANUAL BYPASS MODE PRIOR TO TROUBLE SHOOTING. THERE WAS NO ADVERSE IMPACT ON THE SAFE OPERATION OF THE PLANT OR TO THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT. THE PLANT'S RESPONSE WAS IN A CONSERVATIVE DIRECTION WITH NO IMPACT ON SAFETY SYSTEMS.

[113] RIVERBEND 1 DOCKET 50-458 LER 87-013
MISSED GAS SAMPLE DUE TO INADEQUATE COMMUNICATION.
EVENT DATE: 062887 REPORT DATE: 072487 NSSS: GE TYPE: BWR

(NSIC 205507) AT 0930 ON 6/29/87 WITH THE UNIT AT APPROXIMATELY 67 PERCENT POWER, IT WAS DISCOVERED THAT THE SURVEILLANCE FOR NOBLE GAS AND TRITIUM SAMPLES FROM THE MAIN PLANT EXHAUST HAD NOT BEEN TAKEN WITHIN 1 HOUR FOLLOWING UNIT SHUTDOWN. THE REQUIRED SURVEILLANCE WAS COMPLETED UPON UNIT RESTART AT APPROXIMATELY 1435 ON 6/28/87 UPON UNIT RESTART BY THE CHEMISTRY TECHNICIANS ON THE NEXT SHIFT (14.5 HOURS AFTER THE UNIT SHUTDOWN). INVESTIGATION REVEALED THAT THE CHEMISTRY TECHNICIANS WERE UNAWARE THAT A PLANT SHUTDOWN HAD OCCURRED. THE TECHNICIANS HAD INFORMED THEIR RELIEF AT 0600 THE NEXT MORNING THAT THE PLANT WAS STILL OPERATING IN MODE 2. ANNOUNCEMENTS ARE MADE OVER THE GAITRONICS COMMUNICATIONS SYSTEM WHEN PLANT SHUTDOWNS ARE INITIATED. HOWEVER, THESE ANNOUNCEMENTS WERE INAUDIBLE OVER THE NOISE CREATED BY THE INSTRUMENTATION AND FUME HOODS IN THE CHEMISTRY LABORATORY. AS INTERIM CORRECTIVE ACTION, THE VOLUME ON THE GAITRONICS SYSTEM

WITHIN THE CHEMISTRY LABORATORY HAS BEEN INCREASED. THE GROSS GAMMA AND BETA ACTIVITIES IN THE REACTOR COOLANT TAKEN AFTER THE UNIT SHUTDOWN INDICATED NORMAL LEVELS. THERE WAS NO IMPACT ON THE HEALTH AND SAFETY AS A RESULT OF THIS MISSED SURVEILLANCE. ANY ABNORMAL RELEASE ACTIVITIES WOULD HAVE BEEN DETECTED BY THE MAIN PLANT EXHAUST RADIATION MONITOR AND ALARMED IN THE MAIN CONTROL ROOM.

[114] ROBINSON 2 DOCKET 50-261 LER 87-018
AFW PUMP AUTOSTART MALFUNCTION.
EVENT DATE: 061587 REPORT DATE: 071587 NSSS: WE TYPE: PWR

(NSIC 205442) ON JUNE 15, 1987, WITH THE UNIT 2 REACTOR CRITICAL, A CONDENSATE PUMP TRIPPED CAUSING A TRIP OF THE MAIN FEEDWATER PUMP IN OPERATION. THE TRIP OF THE FEEDWATER PUMP INITIATED AN AUTOSTART SIGNAL TO THE TWO MOTOR DRIVEN AUXILIARY FEEDWATER (AFW) PUMPS BUT ONLY THE "A" AFW PUMP STARTED. DURING A RECENT PLANT MODIFICATION, THE AUTOSTART CIRCUIT FOR THE "B" AFW PUMP WAS MISWIRED BUT ACCEPTANCE TESTING FAILED TO INDENTIFY THE WIRING PROBLEM. THE MISWIRING WAS CORRECTED AND THE WIRING FOR BOTH AFW PUMPS WAS VERIFIED PROPERLY INSTALLED. FOLLOWING THIS, BOTH AFW PUMPS WERE TESTED TO ASSURE AUTOMATIC STARTING ON LOSS OF FEEDWATER AS WELL AS ON LOW LOW STEAM GENERATOR LEVEL. IN ADDITION, A PLANT NONCONFORMANCE REPORT WAS WRITTEN TO CAPTURE THE EVENT FOR MANAGEMENT EVALUATION AND DETERMINATION OF CORRECTIVE ACTION TO PRECLUDE RECURRENCE.

[115] ROBINSON 2 DOCKET 50-261 LER 87-019
INOPERABLE LOOP 1 DELTA T.
EVENT DATE: 061687 REPORT DATE: 071487 NSSS: WE TYPE: PWR

(NSIC 205443) ON JUNE 16, 1987, DURING NORMAL STARTUP OF UNIT 2, THE LOAD ON THE UNIT HAD INCREASED TO ABOUT TWENTY-THREE PERCENT WHEN IT WAS DISCOVERED THAT THE THOT AND TCOLD WERE REVERSED REACTOR COOLANT SYSTEM (RCS) LOOP 1. TO COMPENSATE, THE ASSOCIATED BISTABLE WERE PLACED IN THE TRIPPED POSITION AND THE LOAD ON THE UNIT WAS MAINTAINED. RECENT REPAIR WORK ON THE THOT AND TCOLD CONNECTIONS INSIDE CONTAINMENT AT THE ELECTRIC PENETRATION ASSEMBLIES HAD INADVERTENTLY REVERSED THE CIRCUITS. THE CIRCUITS WERE CORRECTED BY SWITCHING TERMINATIONS ON THE HAGAN RACKS TO SPARE RCS 1 RESISTANCE THERMAL DETECTORS (RTDS). THE CONNECTIONS AT THE PENETRATIONS INSIDE CONTAINMENT WILL BE INSPECTED AT THE FIRST OPPORTUNITY WHEN PERSONNEL ARE ALLOWED ACCESS IN LIGHT OF THE ENVIRONMENT IN-CONTAINMENT DURING POWER OPERATION.

[116] SALEM 2 DOCKET 50-311 LER 87-009
APPENDIX R CRITERIA NON-CONFORMANCE.
EVENT DATE: 061987 REPORT DATE: 072087 NSSS: WE TYPE: PWR

(NSIC 205447) ON JUNE 19, 1987 AT 1515 HOURS, IT WAS DISCOVERED THAT THE CABLING FOR THE THREE (3) ELECTRICAL TRAINS OF THE UNIT 2 SERVICE WATER (SW) SYSTEM (BI) DID NOT MEET THE SEPARATION REQUIREMENTS OF THE CODE OF FEDERAL REGULATIONS, 10CFR 50 APPENDIX R. UPON DISCOVERY, A CONTINUOUS FIRE WATCH WAS ESTABLISHED AT THE ENTRANCE TO THE SW PIPE TUNNEL. THIS CABLE CONFIGURATION WAS IDENTIFIED BY A PSE&G TASK FORCE ESTABLISHED TO REVIEW AND EVALUATE SALEM STATION'S COMPLIANCE WITH 10CFR 50 APPENDIX R. THE ROOT CAUSE OF THIS DEFICIENCY IS INADEQUATE DESIGN REVIEW. THE CURRENT DESIGN MEETS THE ORIGINAL ELECTRICAL SEPARATION REQUIREMENTS FOR SALEM STATION UNIT 2, HOWEVER, IT DOES NOT MEET THE 10CFR 50 APPENDIX R REQUIREMENTS AS PUBLISHED IN THE FEDERAL REGISTER ON SEPTEMBER 8, 1981. THE APPENDIX R CRITERIA WAS NOT APPLIED TO THE SW PIPING TUNNEL BECAUSE OF ITS RESTRICTED ACCESS AND CONFINED SPACE. DUE TO THIS OVERSIGHT, THE SW PUMP CABLING CONFIGURATION WAS NOT MODIFIED. PSE&G IS REVIEWING DESIGN MODIFICATION OPTIONS TO CORRECT THIS DEFICIENCY. THE FIRE PROTECTION TASK FORCE IS CONTINUING ITS REVIEW.

[117] SALEM 2 DOCKET 50-311 LER 87-010
TECHNICAL SPECIFICATION 3.7.11 - FIRE BARRIER IMPAIRMENT NON-COMPLIANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 062387 REPORT DATE: 072387 NSSS: WE TYPE: PWR

(NSIC 205528) ON JUNE 23, 1987 AT 1645 HOURS, A FIRE PROTECTION OPERATOR DISCOVERED THE PENETRATION ABOVE FIRE DOOR 135-2 HAD BEEN IMPAIRED TO ALLOW PASSAGE OF WELDING LEADS. AFTER RUNNING THE LEADS, THROUGH THE PENETRATION THE MAINTENANCE WORKER SEALED THE PENETRATION, HOWEVER, THE SEAL WAS NOT ADEQUATE TO ASSURE THE REQUIRED THREE (3) HOUR FIRE BARRIER RATING. THE PENETRATION WAS IN THIS CONDITION FOR APPROXIMATELY 24 HOURS. FURTHER INVESTIGATION REVEALED AN IMPAIRMENT PERMIT HAD NOT BEEN COMPLETED (PER ADMINISTRATIVE PROCEDURE AP-25, "FIRE PROTECTION PROGRAM") NOR HAD A FIRE WATCH BEEN ASSIGNED IN STATEMENT 3.7.11.A. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. CORRECTIVE ACTION INCLUDED INITIATING AN IMPAIRMENT PERMIT AND AN HOURLY FIRE WATCH UNTIL THE PENETRATION WAS ADEQUATELY SEALED. THE MAINTENANCE WORKER WHO HAD IMPAIRED THE PENETRATION WAS COUNSELED AND RE-INDOCTRINATED IN THE REQUIREMENT OF AP-25.

[118] SAN ONOFRE 2 DOCKET 50-361 LER 87-008
ENTRY INTO TECHNICAL SPECIFICATION 3.0.3 - CONTROL ROD POSITION INDICATION INOPERABLE.
EVENT DATE: 061387 REPORT DATE: 071387 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 205465) ON JUNE 13, 1987, AT 0500, WITH THE UNIT AT 100% POWER, IT WAS DETERMINED, AS A RESULT OF PERFORMING A SURVEILLANCE, THAT ONE OF THE TWO ROD POSITION INDICATION CHANNELS REQUIRED BY TECHNICAL SPECIFICATION 3.1.3.2 HAD NOT CORRECTLY TRACKED PART LENGTH CONTROL ELEMENT ASSEMBLY (PLCEA) MOVEMENT PERFORMED EARLIER, AND WAS THEREBY DECLARED INOPERABLE. SINCE THE LCO ACTION STATEMENT REQUIRES OPERATION OF AT LEAST TWO POSITION INDICATING CHANNELS, THIS CONSTITUTED ENTRY INTO TECHNICAL SPECIFICATION 3.0.3. AT 537, THE CORRECT ROD POSITIONS WERE ENTERED INTO THE ROD POSITION INDICATION PULSE COUNTER OF THE BACK-UP CORE OPERATING LIMIT SUPERVISORY SYSTEM (COLSS) COMPUTER, AND TECHNICAL SPECIFICATION 3.0.3 WAS EXITED. TESTING REVEALED THE PROGRAMMED TIME DELAY FOR THE COMPUTER TO RECEIVE ROD CONTROL PULSE DATA WAS INSUFFICIENT TO RESET SPURIOUS INTERRUPTS WHICH INHIBIT PULSE COUNTING. IT WAS DETERMINED AN INTERRUPT MAY HAVE PREVENTED THE BACK-UP COLSS COMPUTER FROM RECORDING MOVEMENT OF ANY CEA. THEREFORE, DURING THE PERIOD FROM 0537 TO 0637 (WHEN ADDITIONAL REDUNDANT INDICATION WAS PLACED IN OPERATION), HIGH RELIABILITY OF THE BACK-UP COLSS POSITION INDICATION WAS NOT ASSURED. CORRECTIVE ACTION WAS IMPLEMENTED ON 6/25/87 TO REPROGRAM THE BACK-UP COLSS COMPUTERS ON UNITS 2 AND 3, TO INCREASE THE DATA ACQUISITION TIME DELAY.

[119] SAN ONOFRE 2 DOCKET 50-361 LER 87-006
MISPOSITIONED EMERGENCY CORE COOLING SYSTEM VALVE CIRCUIT BREAKERS.
EVENT DATE: 062687 REPORT DATE: 072787 NSSS: CE TYPE: PWR

(NSIC 205565) ON 6/26/87, AT 0325, WITH UNIT 2 AT 91% POWER, WHILE PERFORMING THE SHIFTLY SURVEILLANCE OF EMERGENCY CORE COOLING SYSTEM (ECCS) CIRCUIT BREAKER (CB) ALIGNMENTS REQUIRED BY TECH SPEC 4.5.2.A, CBS FOR TWO ECCS VALVES WERE FOUND CLOSED CONTRARY TO THE TECH SPEC SURVEILLANCE REQUIREMENT. THE TWO CBS WERE PROMPTLY OPENED. THE ASSOCIATED VALVES, HOWEVER, WERE IN THE REQUIRED POSITION AND THERE WAS NO FUNCTIONAL IMPAIRMENT OF THE ECCS. THEREFORE, THERE WAS NO IMPACT ON THE HEALTH AND SAFETY OF PLANT PERSONNEL OR THE PUBLIC AS A RESULT OF THIS EVENT. INVESTIGATION REVEALED THAT THESE VALVES WERE PROPERLY REMOVED FROM SERVICE FOR MAINTENANCE AT 0625, ON 6/25/87. UPON RESTORATION TO SERVICE, THE TWO CBS WERE CLOSED IN ACCORDANCE WITH AN INCORRECTLY SPECIFIED "RETURN TO SERVICE" (RTS) ALIGNMENT. AT 1200, THE ECCS COMPONENTS WERE DECLARED OPERABLE BASED ON COMPLETION OF THE RTS ALIGNMENT. PERFORMANCE OF THE TECH SPEC SURVEILLANCE ON THE FOLLOWING SHIFT FAILED TO DETERMINE THAT THE CBS WERE CLOSED.

AS NOTED ABOVE, THE CIRCUIT BREAKERS WERE OPENED AT 0325 ON 6/26/87. THE CONDITION RESULTED FROM ERRONEOUSLY SPECIFYING THE RTS ALIGNMENT, AS WELL AS, FAILURE TO IDENTIFY THE ERROR DURING THE REVIEW AND APPROVAL OF THE RTS ALIGNMENT.

[120] SAN ONOFRE 2 DOCKET 50-361 LER 87-009
FAILURE TO ESTABLISH CONTINUOUS FIRE WATCHES DUE TO PERSONNEL ERROR.
EVENT DATE: 062787 REPORT DATE: 072987 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 205623) ON 6/27/87, AT 0130, WITH UNIT 3 AT 100% POWER AND AT 0745, WITH UNIT 2 AT 90% POWER, THE FIRE SUPPRESSION SYSTEMS IN THE UNIT 3 AND UNIT 2 AUXILIARY FEEDWATER PUMP (AFWP) ROOMS, RESPECTIVELY, WERE REMOVED FROM SERVICE FOR PLANNED FIRE PROTECTION OUTAGES. ON 27, AT 1830, AND 6/28, AT 0300 THE UNIT 3 AND THE UNIT 2 AFWP ROOM FIRE SUPPRESSION STEMS, RESPECTIVELY, WERE RETURNED TO SERVICE. ON 6/29, AT APPROXIMATELY 1300, DURING THE PROCESS OF REMOVING THE COMPENSATORY MEASURES, IT WAS IDENTIFIED THAT CONTINUOUS FIRE WATCHES HAD NOT BEEN ESTABLISHED, CONTRARY TO TECHNICAL SPECIFICATION 3.7.8.2, ACTION STATEMENT 'A'. THERE WAS NO SAFETY SIGNIFICANCE SINCE THE FIRE DETECTION EQUIPMENT IN BOTH ROOMS REMAINED OPERABLE. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR. MISCOMMUNICATION BETWEEN EMERGENCY SERVICES OFFICER (ESO) PERSONNEL AND THE ESO SHIFT CAPTAIN RESULTED IN THE FAILURE TO ESTABLISH THE REQUIRED COMPENSATORY FIRE WATCHES. IN ADDITION, CONTRARY TO PROCEDURAL REQUIREMENTS AND TRAINING, THE ESO SHIFT CAPTAIN DID NOT VERIFY THAT ALL APPROPRIATE COMPENSATORY MEASURES WERE IN PLACE. SIMILAR FAILURES TO ESTABLISH APPROPRIATE FIRE WATCHES DUE TO PERSONNEL ERROR HAVE BEEN REPORTED PREVIOUSLY IN LERS 84-034 AND 85-038 (DOCKET NO. 50-361), AND 85-022 (DOCKET NO. 50.362). CORRECTIVE ACTION TAKEN FOR THOSE OCCURRENCES INVOLVED DISCIPLINARY ACTION AND TRAINING, AS APPROPRIATE.

[121] SAN ONOFRE 3 DOCKET 50-362 LER 86-015
APPARENT EXTREMITY EXPOSURE IN EXCESS OF REGULATORY LIMITS.
EVENT DATE: 121286 REPORT DATE: 011287 NSSS: CE TYPE: PWR

(NSIC 205578) ON DECEMBER 12, 1986, SCE DETERMINED THAT SUFFICIENT PRELIMINARY INFORMATION EXISTED TO BELIEVE THAT AN INDIVIDUAL EXTREMITY EXPOSURE IN EXCESS OF REGULATORY LIMITS MAY HAVE OCCURRED IN OCTOBER 1986. DURING OCTOBER 1986, AN SCE MAINTENANCE INDIVIDUAL WORE TWO RING THERMOLUMINESCENT DOSIMETERS (TLD) PROVIDED BY A VENDOR TLD SERVICE. THE RINGS WERE WORN ON FIVE SEPARATE OCCASIONS DURING ROUTINE MAINTENANCE WORK. ALTHOUGH THE INDIVIDUAL'S WHOLE BODY DOSIMETRY INDICATED 114 MREM GAMMA, ONE OF THE TWO RING TLDS INDICATED AN APPARENT RIGHT HAND EXTREMITY EXPOSURE IN EXCESS OF THE 18 3/4 REM/QUARTER REGULATORY LIMIT. THE TLD WORN ON THE LEFT HAND DID NOT SHOW ANY SIGNIFICANT EXPOSURE. ONGOING INVESTIGATIONS THUS FAR HAVE BEEN UNABLE TO ESTABLISH WHETHER THE EXTREMITY EXPOSURE ACTUALLY OCCURRED. HOWEVER, IF IT DID ACTUALLY OCCUR, THE ONLY PLAUSIBLE MECHANISM WOULD BE FROM A SMALL FISSION FUEL FRAGMENT (FFF). AT PRESENT, IT APPEARS EQUALLY LIKELY THAT THE TLD READING WAS ANOMALOUS OR WAS CAUSED BY TAMPERING. THE ONGOING INVESTIGATIONS SHOULD BE COMPLETED IN FEBRUARY 1987, AND A REVISED LER WILL BE SUBMITTED SUBSEQUENT TO ITS COMPLETION. THE INDIVIDUAL WAS RESTRICTED FROM FURTHER EXPOSURE FOR THE FALL 1986 QUARTER. HE RESUMED HIS NORMAL ACTIVITIES ON JANUARY 1, 1987.

[122] SAN ONOFRE 3 DOCKET 50-362 LER 87-011
REACTOR TRIP ON LOW STEAM GENERATOR WATER LEVEL DUE TO INTERMITTENT LOSS OF POWER IN INSTRUMENT BUS.
EVENT DATE: 062187 REPORT DATE: 072187 NSSS: CE TYPE: PWR
VENDOR: FOXBORO CO., THE
PACIFIC SCIENTIFIC COMPANY

(NSIC 205566) ON JUNE 21, 1987 AT 0258, WITH UNIT 3 IN MODE 1 AT 100% POWER, THE

REACTOR AUTOMATICALLY TRIPPED ON LOW STEAM GENERATOR (SG) WATER LEVEL. THE LOW SG WATER LEVEL WAS CAUSED BY AN INTERMITTENT LOSS OF POWER IN ONE PHASE OF A 120 VAC NON-1E INSTRUMENT BUS WHICH RESULTED IN THE INABILITY TO CONTROL MAIN FEEDWATER AND THE CONSEQUENT REDUCTION IN SG FEEDWATER ACTUATION SETPOINT. FOLLOWING THE REACTOR TRIP, 120 VAC NON-1E POWER RETURNED AND MAIN FEEDWATER FLOW RESUMED. THE WATER LEVEL IN SG E088 INCREASED FROM THE LOW LEVEL TRIP SETPOINT TO ABOVE THE HIGH LEVEL ALARM SETPOINT AS OPERATORS WERE IMPLEMENTING THEIR IMMEDIATE POST-TRIP ACTIONS IN ACCORDANCE WITH EMERGENCY OPERATING INSTRUCTIONS (EOIS). THIS RESULTED IN COOLING DOWN OF THE REACTOR COOLANT SYSTEM (RCS) TO BELOW THE SAFETY INJECTION ACTUATION SIGNAL (SIAS) SETPOINT. THE 120 VAC POWER MALFUNCTION WAS DETERMINED TO BE DUE TO A LOOSE BOLT CONNECTING THE "B" PHASE OF INSTRUMENT BUS #1 TO THE MAIN BUS BARS OF THE NON-1E UNINTERRUPTABLE POWER SUPPLY (UPS) MAIN DISTRIBUTION SWITCHBOARD, WHICH RESULTED IN INTERMITTENT LOSS OF CIRCUIT CONTINUITY. THIS WAS EVIDENCED BY ARCING AND PITTING AT THE CONNECTION, AND CONFIRMED BY SUBSEQUENT DUPLICATION OF POWER INTERRUPTIONS WHEN THE ASSEMBLY WAS MANUALLY MOVED.

[123] SAN ONOFRE 3 DOCKET 50-362 LER 87-012
TWO SAFETY INJECTION TANKS (SIT) INOPERABLE DURING FILLING.
EVENT DATE: 062287 REPORT DATE: 072287 NSSS: CE TYPE: PWR

(NSIC 205567) AT 1843 ON 6/22/87, WITH UNIT 3 IN HOT STANDBY, DURING THE PREPARATION OF ADDING WATER TO SAFETY INJECTION TANK (SIT) T-010, THE NITROGEN COVER PRESSURE IN T-010 DECREASED TO 580 PSIG. SIT T-010 BECAME INOPERABLE SINCE TECH SPEC 3.5.1 SPECIFIES A NITROGEN COVER PRESSURE RANGE OF 600 TO 625 PSIG. SIT T-008 WAS ALSO INOPERABLE SINCE ITS VENT VALVE FUSES HAD BEEN INSTALLED IN PREPARATION FOR VENTING. SINCE TECH SPEC 3.5.1 ALLOWS ONLY ONE INOPERABLE SIT, TECH SPEC 3.0.3 WAS ENTERED. AT 1846, T-008 VENT VALVE FUSES WERE REMOVED AND TECH SPEC 3.0.3 WAS EXITED. AT 1855, THE NITROGEN COVER PRESSURE FOR T-010 WAS RESTORED TO WITHIN LIMITS. THE CAUSE OF THE EVENT WAS FAILURE OF THE CONTROL OPERATOR (CO) TO PROPERLY IMPLEMENT THE SIT FILL PROCEDURE. CONTRARY TO PROCEDURE, A DRAIN VALVE WAS NOT OPENED PRIOR TO COMMENCING RECIRCULATION OF THE SIT COMMON FILL AND DRAIN HEADER, RESULTING IN EXCESSIVE IN-LEAKAGE TO T-008 WHICH CAUSED PRESSURE IN T-008 TO INCREASE. WHILE INITIATING T-008 VENTING, THE POSITION OF THE DRAIN VALVE WAS NOTED AND WAS OPENED IMMEDIATELY, CAUSING THE COMMON HEADER TO DEPRESSURIZE WHICH RESULTED IN DRAINING OF T-010. AS CORRECTIVE ACTION, THE CO INVOLVED RECEIVED APPROPRIATE DISCIPLINARY ACTION. THE IMPORTANCE OF PROPERLY FOLLOWING PROCEDURES WAS RE-EMPHASIZED TO ALL UNIT 2 AND 3 OPERATORS.

[124] SAN ONOFRE 3 DOCKET 50-362 LER 87-013
HIGH PRESSURE SAFETY INJECTION (HPSI) PUMP BYPASS VALVE FOUND NOT FULLY CLOSED.
EVENT DATE: 062487 REPORT DATE: 072487 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 2 (PWR)

(NSIC 205537) ON 6/24/87, WITH THE UNIT IN MODE 3, AN INSERVICE TEST (IST) OF HIGH PRESSURE SAFETY INJECTION (HPSI) PUMP 3P018 INDICATED A HIGH BYPASS FLOW TO THE REFUELING WATER STORAGE TANK. A VISUAL INSPECTION OF THE MANUALLY OPERATED BYPASS VALVE REVEALED THAT THE PASS VALVE WAS PARTIALLY OPEN DUE TO INTERFERENCE WITH THE LIMIT SWITCH ACTUATION ASSEMBLY, WHICH HAD LOOSENED. AN INSPECTION OF ALL REMAINING UNIT 2 AND 3 HPSI BYPASS VALVES REVEALED THAT THE VALVE ASSOCIATED WITH 3P019 WAS ALSO FOUND PARTIALLY OPEN, ALTHOUGH TO A LESSER DEGREE. THE LACK OF A LIMIT SWITCH ACTUATING ASSEMBLY COLLAR RETAINING DEVICE IS CONSIDERED TO BE THE ROOT CAUSE OF THIS EVENT. 3P018 HAD BEEN ALIGNED TO HPSI TRAIN "B" DURING THE PERIOD ITS BYPASS VALVE WAS NOT IN ITS FULLY CLOSED POSITION. SINCE THIS CONDITION IS CONTRARY TO SURVEILLANCE REQUIREMENT 4.5.2.B, THE UNIT WAS CONSIDERED TO HAVE BEEN OPERATING IN A CONDITION NOT PERMITTED BY TECHNICAL SPECIFICATIONS. AN EVALUATION OF HPSI FLOW WITH THE BYPASS VALVE PARTIALLY OPEN CONFIRMED THAT 3P018 WOULD HAVE BEEN ABLE TO PROVIDE SUFFICIENT FLOW TO THE REACTOR COOLANT SYSTEM IN THE EVENT OF THE MOST LIMITING SMALL BREAK LOSS OF

COOLANT ACCIDENT. THEREFORE, THERE WAS NO LOSS OF SAFETY FUNCTION AS A RESULT OF THIS EVENT. THE LIMIT SWITCH ASSEMBLIES WERE RETURNED TO THEIR ORIGINAL CONFIGUATION AND THE VALVES WERE PROPERLY CLOSED AND LOCKED.

[125] SEQUOYAH 1 DOCKET 50-327 LER 87-013 REV 01
UPDATE ON REACTOR COOLANT SYSTEM SPILLAGE WHILE IN MODE 5 DUE TO A PROCEDURAL INADEQUACY.
EVENT DATE: 020187 REPORT DATE: 071487 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 205424) THIS LER IS BEING REVISED TO INCLUDE ADDITIONAL INFORMATION FROM THE REPORT GENERATED BY THE TASK FORCE ASSIGNED TO INVESTIGATE THIS EVENT. ON FEBRUARY 1, 1987, WITH BOTH UNITS 1 AND 2 IN MODE 5 AT 0 PERCENT POWER, A REACTOR COOLANT SYSTEM (RCS) SPILLAGE OCCURRED ON UNIT 1 AS A RESULT OF A PROCEDURAL INADEQUACY. THE RCS HAD BEEN DRAINED DOWN TO ALLOW U-BEND HEAT TREATMENT ON THE STEAM GENERATOR (SG) U-TUBES WITH THE PRIMARY SIDE MANWAYS REMOVED. OPERATIONS RECEIVED A WORK PACKAGE TO STROKE TIME TEST VALVE 1-FCV-63-1 (REFUELING WATER STORAGE TANK (RWST) SUPPLY TO THE RESIDUAL HEAT REMOVAL SYSTEM). OPERATIONS RECOGNIZED THAT THE SURVEILLANCE INSTRUCTION (SI) WAS INADEQUATE FOR THE GIVEN PLANT CONFIGURATION AND ATTEMPTED TO SUPPLEMENT THE SI BY ISOLATING THE RWST FROM THE RCS. THIS ISOLATION WAS INCOMPLETE, AND WHEN THE VALVE WAS STROKED, APPROXIMATELY 6000 GALLONS OF RWST WATER FLOWED BY GRAVITY INTO THE RCS AND APPROXIMATELY 3000 GALLONS FLOWED FROM THE OPEN SG MANWAYS ONTO THE FLOOR OF THE LOWER COMPARTMENT. NO PERSONNEL WERE INJURED DUE TO THE TIMING OF THE EVENT. AIR SAMPLES TAKEN AFTER THE EVENT SHOWED NO INCREASE IN ACTIVITY. THERE WAS NO IDENTIFIED DILUTION OF THE RCS BORON CONCENTRATION. A TASK FORCE WAS ESTABLISHED TO INVESTIGATE THE EVENT. ADDITIONAL ADMINSTRATIVE CONTROLS WERE PLACED ON ACTIVITIES WHICH COULD AFFECT SG WORK.

[126] SEQUOYAH 1 DOCKET 50-327 LER 87-028
POTENTIAL FOR LOSS OF ALL CONTROL AIR RESULTING FROM A LOSS OF OFFSITE POWER EVENT DUE TO A DESIGN OVERSIGHT.
EVENT DATE: 061187 REPORT DATE: 071087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 205451) ON JUNE 11, 1987, WITH UNITS 1 AND 2 IN MODE 5 (0 PERCENT POWER, 4 PSIG, 103 DEGREES F AND 0 PERCENT POWER, 260 PSIG, 133 DEGREES F, RESPECTIVELY), IT WAS DETERMINED THAT BOTH TRAINS OF CONTROL AIR COULD BE LOST IN THE EVENT OF A LOSS OF OFFSITE POWER. A DESIGN FLAW IN THE ELECTRICAL CONTROL LOGIC WILL PREVENT THE AUXILIARY CONTROL AIR COMPRESSORS FROM STARTING ON A LOSS OF OFFSITE POWER SIGNAL. IT IS BELIEVED A PERSONNEL OVERSIGHT BY THE SYSTEM DESIGNER CAUSED THIS CONDITION. ABNORMAL OPERATING INSTRUCTION-35, "LOSS OF OFFSITE POWER," REQUIRES THE OPERATORS TO LOCALLY START THE STATION CONTROL AIR COMPRESSORS A & B. THIS IS THE FIRST MANUAL ACTION REQUIRED OF THE OPERATOR. HENCE, CONTROL AIR COULD NOT BE LOST FOR MORE THAN A FEW MINUTES. IN ORDER TO CORRECT THIS PROBLEMS JUMPERS WILL BE INSTALLED AROUND THE SEAL-IN CONTACTS TO PROVIDE AN AUTOMATIC RESET OF THE COMPRESSORS AFTER A LOSS OF POWER. DIVISION OF NUCLEAR ENGINEERING WILL FURTHER INVESTIGATE A LONG-TERM CORRECTIVE ACTION.

[127] SEQUOYAH 1 DOCKET 50-327 LER 87-029
YEARLY REPORTING AND A DOCUMENTED EVALUATION OF ENVIRONMENTAL IMPACT FOR PLANT DESIGN AND OPERATING CHANGES NOT MADE DUE TO A LACK OF PROPER PROCEDURES.
EVENT DATE: 061587 REPORT DATE: 071587 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 205425) ON JUNE 15, 1987, WITH BOTH UNITS IN MODE 5 (COLD SHUTDOWN) AT ZERO PERCENT POWER, PROCEDURE AND DOCUMENT REVIEW DETERMINED THAT SEQUOYAH HAD NOT BEEN MAINTAINING EVALUATION RECORDS OR SUBMITTING PERIODIC ANNUAL REPORTS OF

ONSITE CHANGES OF THE ENVIRONMENTAL PORTION OF THE TECH SPECS. THIS IS IN VIOLATION OF TECH SPEC 5.3.C (APPENDIX B) WHICH SAYS THAT THE LICENSEE MUST MAINTAIN RECORDS OF CHANGES IN FACILITY DESIGN OR OPERATION THAT COULD AFFECT AN ENVIRONMENTAL IMPACT. FUTTHER, THE LICENSEE IS TO FURNISH TO NRC AN ANNUAL REPORT CONTAINING ANY DESCRIPTIONS, ANALYSES, INTERPRETATIONS, AND EVALUATIONS OF SUCH CHANGES, TESTS AND EXPERIMENTS. THE CAUSE OF THE EVENT WAS A RESULT OF (1) LACK OF CLEAR LINES OF RESPONSIBILITY FOR THESE REQUIREMENTS AND (2) LACK OF ADEQUATE PROCEDURES TO ADDRESS THESE REPORTING REQUIREMENTS. THIS EVENT IS REPORTABLE IN ACCORDANCE WITH 10 CFR 50.73, PARAGRAPH A.2.I.B. THE PLANT MANAGER HAS THE RESPONSIBILITY TO ENSURE THAT THE REQUIRED ANNUAL REPORTING REQUIREMENTS ARE MET AND THAT ADEQUATE PROCEDURES EXIST TO ADDRESS THE DOCUMENTATION OF CHANGES WHICH MAY AFFECT THE ENVIRONS. PLANT ADMINISTRATIVE INSTRUCTIONS WILL BE REVISED TO REQUIRE A DOCUMENTED EVALUATION OF CHANGES, TESTS, AND EXPERIMENTS TO THE FACILITY WHICH COULD AFFECT THE ENVIRONMENT AS REQUIRED BY TECH SPEC 5.3.C.

[128] SEQUOYAH 1 DOCKET 50-327 LER 87-030
BLOWN FUSE IN EMERGENCY START CIRCUITS RESULT IN SPURIOUS EMERGENCY DIESEL GENERATOR STARTS ON TWO OCCASIONS.
EVENT DATE: 062087 REPORT DATE: 072087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
VENDOR: LITTLEFUSE INC

(NSIC 205426) ON JUNE 20, 1987, AT 0055 EST, ALL FOUR STANDBY DIESEL GENERATORS (D/GS) STARTED DUE TO A BLOWN FUSE IN A REMOTE EMERGENCY START CIRCUITRY FOR 2A-A D/G. THE BLOWN FUSE (FLAS-5, LOT NO. 3) CAUSED LOSS OF VOLTAGE TO THE 2A-A REMOTE EMERGENCY START CIRCUITRY. STANDBY D/GS 1A-A, 1B-B, 2A-A, AND 2B-B STARTED AS REQUIRED. ON JULY 4, 1987, AT 1212 EST, ALL FOUR STANDBY D/GS STARTED DUE TO A BLOWN FUSE IN 125V DC VITAL BATTERY BOARD II, CIRCUIT C-16. THE BLOWN FUSE (FLAS-5, LOT NO. 2) CAUSED LOSS OF POWER TO THE LOGIC RELAY PANEL FOR 1B-B D/G. UPON LOSS OF VOLTAGE TO THE 1B-B D/G REMOTE EMERGENCY START CIRCUITRY, STANDBY D/GS 1A-A, 1B-B, 2A-A, AND 2B-B STARTED AS REQUIRED. FOR BOTH EVENTS DESCRIBED ABOVE, NO DAMAGE OCCURRED TO THE D/GS, AND NONE OF THE D/GS LOADED BECAUSE A DEGRADED VOLTAGE CONDITION DID NOT EXIST ON THE 6.9 KV SHUTDOWN BOARDS. TVA HAS CONTRACTED WITH LITTLEFUSE INC. TO SUPPLY 15,000 FLAS-5 FUSES. APPROXIMATELY 3,200 OUT OF 3,702 (DELIVERED) FUSES WERE INSTALLED IN MARCH AND APRIL 1987, 1,683 OF WHICH WERE FROM LOTS 2 AND 3. AS OF JULY 13, 1987, 69 FLAS-5 FUSE FAILURES HAVE OCCURRED. OF THESE, 67 FAILED FUSES WERE SUPPLIED IN LOTS 2 AND 3. THE LOT NUMBER OF THE REMAINING TWO FAILED FUSES COULD NOT BE DETERMINED. A TELEPHONE CONVERSATION REVEALED THAT CHANGES HAD BEEN MADE IN BOTH SOLDER MATERIAL AND THE SOLDERING PROCESS.

[129] SHEARON HARRIS 1 DOCKET 50-400 LER 87-011 REV 01
UPDATE ON AUXILIARY OPERATOR TRIPS AUXILIARY BUS BREAKER DE-ENERGIZING AUXILIARY AND SAFETY BUSES.
EVENT DATE: 030787 REPORT DATE: 072287 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 205541) ON MARCH 7, 1987, AN AUXILIARY OPERATOR (AO) WAS PERFORMING A WEEKLY SURVEILLANCE TEST (OST-1023) WHICH IS FOR OFFSITE POWER VERIFICATION. ONE OF THE REQUIREMENTS OF THE OST IS TO ENSURE THE BREAKER RELEASE LEVER IS IN THE NEUTRAL POSITION WITH THE BREAKER CLOSED. WHEN THE AO CHECKED AUXILIARY BUS IE FEEDER BREAKER 121, THE BREAKER WAS INADVERTENTLY TRIPPED OPEN CAUSING THE DE-ENERGIZATION OF AUXILIARY BUS IE AND SAFETY BUS IB-SB. THIS OCCURRED AT 2230 HOURS ON MARCH 7, 1987. THE DE-ENERGIZATION OF BUS 1B-SB CAUSED THE 1B-SB DIESEL GENERATOR TO START AND THE ACTUATION OF SEQUENCER 1B-SB ON BUS UNDERVOLTAGE. THE AO IMMEDIATELY CLOSED THE BREAKER CABINET AND NOTIFIED THE CONTROL ROOM OF THE INCIDENT. THE SENIOR CONTROL OPERATOR (SCO) THEN INITIATED AOP-025 FOR THE LOSS OF ONE EMERGENCY BUS AND ALL PLANT SYSTEMS WERE THEN RETURNED TO NORMAL. AT THE TIME OF THE INCIDENT, THE PLANT WAS IN MODE 4 AT 345F AND 350 PSIG. THERE WERE

NO ADVERSE CONSEQUENCES DUE TO THIS EVENT AND SAFETY SYSTEMS PERFORMED AS REQUIRED. TO PREVENT RECURRENCE, THE APPLICABLE PROCEDURE HAS BEEN REVISED.

[130] SHEARON HARRIS 1 DOCKET 50-400 LER 87-039
INTERMEDIATE RANGE HIGH NEUTRON FLUX TRIP SETPOINT OUT OF CALIBRATION.
EVENT DATE: 061987 REPORT DATE: 072087 NSSS: WE TYPE: PWR

(NSIC 205475) ON JUNE 19, 1987 A PROBLEM WITH THE SCALING AND SETPOINTS FOR THE INTERMEDIATE RANGE (IR) NEUTRON FLUX MONITORS (N35 AND N36) WAS IDENTIFIED. AT THE TIME, THE PLANT WAS OPERATING AT 100 PERCENT REACTOR POWER AND TECHNICAL SPECIFICATIONS DID NOT REQUIRE THE IR MONITORS TO BE OPERABLE. THE IR CHANNELS WERE READING OFFSCALE HIGH ABOVE 70% REACTOR POWER AND IT WAS DETERMINED THAT A NEW CORRELATION BETWEEN POWER AND IR CURRENT WAS NECESSARY. DATA FOR A NEW CORRELATION WAS OBTAINED DURING A STARTUP ON JUNE 18, 1987, BETWEEN THE HOURS OF 2200 TILL 2300. UPON REDUCTION OF DATA, ON JUNE 19, 1987, AT APPROXIMATELY 1045 HOURS IT WAS DETERMINED THAT THE INTERMEDIATE RANGE HIGH FLUX TRIP SETPOINTS REQUIRED AT 25% REACTOR POWER FOR CHANNELS N35 AND N35 WERE AT ACTUALLY 31% REACTOR POWER, WHICH IS IN EXCESS OF THE TECHNICAL SPECIFICATION (TABLE 2.2-1) ALLOWABLE OF 30.9%. THE SHIFT FOREMAN WAS NOTIFIED AND A PRIORITY 1 WORK REQUEST WAS ISSUED TO RECALIBRATE CHANNELS N35 AND N36 IN ACCORDANCE WITH THE NEW CORRELATION BETWEEN POWER RANGE AND INTERMEDIATE RANGE INSTRUMENTS. THE CAUSE WAS DETERMINED TO BE DUE TO PREVIOUS ADJUSTMENTS MADE ON JANUARY 21, 1987. NEW SCALING CURRENTS WERE INCORPORATED INTO APPROPRIATE SURVEILLANCE TEST PROCEDURES. THESE TESTS WERE COMPLETED ON JUNE 19, 1987.

[131] SHEARON HARRIS 1 DOCKET 50-400 LER 87-036
MISSED DIESEL GENERATOR AIR ROLL FOLLOWING ENGINE OPERATION.
EVENT DATE: 062187 REPORT DATE: 072087 NSSS: WE TYPE: PWR
VENDOR: TOUSLEY-IBER COMPANY

(NSIC 205472) ON JUNE 21, 1987, WITH THE PLANT IN MODE 3, IT WAS DISCOVERED THAT AN AIR ROLL OF THE 1A-SA DIESEL ENGINE HAD NOT BEEN PERFORMED WITHIN FOUR TO EIGHT HOURS AFTER ENGINE SHUTDOWN, AS REQUIRED BY THE SHEARON HARRIS OPERATING LICENSE (NPF-63). THIS AIR ROLL IS PERFORMED TO DETECT ANY WATER INGRESS INTO THE ENGINE'S CYLINDERS, AND MAY ONLY BE OMITTED WHEN THE PLANT IS ALREADY IN AN ACTION STATEMENT OF TECHNICAL SPECIFICATION 3/4.8.1.1, "ELECTRICAL POWER SYSTEMS, AC SOURCES." WHEN THE FOUR HOUR INTERVAL BEGAN, THIS ACTION STATEMENT WAS IN EFFECT, AND THE PROCEDURE STEP FOR THE AIR ROLL WAS MARKED AS "NOT APPLICABLE". HOWEVER, THIS ACTION STATEMENT WAS EXITED PRIOR TO EXPIRATION OF THE FOUR HOUR INTERVAL, AND SO THE AIR ROLL WAS REQUIRED TO BE PERFORMED. THIS SITUATION WAS NOT DISCOVERED UNTIL AFTER THE FOUR HOUR INTERVAL HAD ELAPSED. UPON DISCOVERY, THE ENGINE WAS AIR ROLLED AND NO ACCUMULATION OF WATER WAS DETECTED. THIS EVENT IS REPORTED UNDER CONDITION 2.G OF THE SHEARON HARRIS OPERATING LICENSE.

[132] SHEARON HARRIS 1 DOCKET 50-400 LER 87-037
TRIP OF OPERATING MAIN FEEDWATER PUMP IN 30% POWER.
EVENT DATE: 062187 REPORT DATE: 071787 NSSS: WE TYPE: PWR

(NSIC 205473) ON JUNE 21, 1987 AT 2208, A MANUAL REACTOR TRIP WAS INITIATED FROM 30% POWER AFTER THE OPERATING MAIN FEEDWATER TRAIN TRIPPED, RESULTING IN A LOSS OF MAIN FEEDWATER. THE MOTOR-DRIVEN AUXILIARY FEEDWATER PUMPS STARTED AS REQUIRED. PLANT RESPONSE TO THE REACTOR TRIP WAS AS EXPECTED, AND THE PLANT WAS STABILIZED IN MODE 3. THE TRIP OF THE FEEDWATER PUMPS WAS CAUSED BY A FAILED OPEN FEEDWATER PUMP RECIRCULATION VALVE, RESULTING FROM OPEN CIRCUIT IN THE VALVE CONTROL WIRING. THE CIRCUIT WAS REPAIRED, AND THE PLANT WAS RETURNED TO SERVICE ON JUNE 22, 1987.

[133] SHEARON HARRIS 1 DOCKET 50-400 LER 87-038
REACTOR TRIP ON TURBINE TRIP DURING SYNCHRONIZATION.
EVENT DATE: 062287 REPORT DATE: 072287 NSSS: WE TYPE: PWR

(NSIC 205474) ON JUNE 22, 1987, AT 0307, WITH THE PLANT AT 6% THERMAL POWER LEVEL, THE TURBINE GENERATOR WAS SYNCHRONIZED TO THE POWER GRID DURING A PLANT STARTUP. THE TURBINE CONTROL SYSTEM PICKED UP EXCESSIVE ELECTRICAL LOAD IMMEDIATELY UPON TIE IN TO THE GRID. THE RESULTING TRANSIENT CAUSED A HIGH STEAM GENERATOR WATER LEVEL TURBINE TRIP AND MAIN FEEDWATER ISOLATION, AND THE TURBINE TRIP ACTUATED AN AUTOMATIC REACTOR TRIP ON INTERLOCK. THE PLANT RESPONSE TO THE REACTOR TRIP WAS AS EXPECTED. AUXILIARY FEEDWATER STARTED AUTOMATICALLY WHEN THE MAIN FEEDWATER PUMPS TRIPPED ON THE HIGH STEAM GENERATOR WATER LEVEL. THE PLANT WAS STABILIZED IN MODE 3. THE CAUSE OF THE SUDDEN INCREASE IN ELECTRICAL LOAD COULD NOT BE PRECISELY DETERMINED. SPECIFIC PRECAUTIONARY MEASURES WERE TAKEN TO REDUCE THE EFFECTS OF ANY FUTURE SIMILAR TRANSIENT, AND THE PLANT WAS RETURNED TO SERVICE WITHOUT REPEAT OF THIS EVENT.

[134] SURRY 1 DOCKET 50-280 LER 87-013
HIGH REACTOR COOLANT SYSTEM LEAK RATE DUE TO FAILED PACKING ON LOOP STOP VALVES.
EVENT DATE: 062387 REPORT DATE: 072387 NSSS: WE TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 205522) ON JUNE 23, 1987 AT 1532 HOURS, WITH UNIT 1 AT 100% POWER, A UNIT SHUTDOWN WAS COMMENCED DUE TO AN UNIDENTIFIED REACTOR COOLANT SYSTEM (RCS) (EIIS-AB) LEAK RATE OF 47 GPM. THIS EVENT IS CONTRARY TO TECHNICAL SPECIFICATION 3.1.C.3 WHICH LIMITS UNIDENTIFIED RCS LEAKAGE TO LESS THAN 10 GPM, AND AN UNUSUAL EVENT WAS DECLARED. AT 1628 HOURS, A CONTAINMENT WALKDOWN WAS CONDUCTED AND OPERATORS SUBSEQUENTLY PLACED THE LOOP STOP VALVES ONTO THEIR BACKSEATS. THIS REDUCED THE UNIDENTIFIED RCS LEAK RATE TO .88 GPM, AND THE UNUSUAL EVENT WAS TERMINATED AT 1924 HOURS. THE CAUSE OF THE HIGH RCS LEAK RATE WAS FAILED PACKING ON LOOP STOP VALVE (EIIS-ISV) MOV-1593. ADDITIONALLY, DUE TO THE INCREASED TEMPERATURE AND PRESSURE IN THE PACKING DRAIN HEADER, THE DIAPHRAGM RUPTURED ON THE DRAIN ISOLATION VALVE (EIIS-LOV) 1-DG-14. SUBSEQUENTLY, THE UNIT WAS PLACED IN COLD SHUTDOWN AND THE LOOP STOP VALVE WAS REPACKED. THE DIAPHRAGM ON THE DRAIN HEADER ISOLATION VALVE WAS REPLACED.

[135] SURRY 2 DOCKET 50-281 LER 86-018 REV 01
UPDATE ON IMPROPERLY INSTALLED RAYCHEM SPLICES ON EQ EQUIPMENT DUE TO INADEQUATE PROCEDURES.
EVENT DATE: 111286 REPORT DATE: 072387 NSSS: WE TYPE: PWR
VENDOR: RAYCHEM CORP.

(NSIC 205512) ON NOVEMBER 12, 1986, IN RESPONSE TO NRC IEIN 86-53 CONCERNING THE IMPROPER INSTALLATION OF RAYCHEM SPLICE MATERIAL, AN INSPECTION OF THE ENVIRONMENTALLY QUALIFIED (EQ) EQUIPMENT WAS PERFORMED. AT THE TIME OF THE INSPECTION, UNIT 2 WAS IN COLD SHUTDOWN. THE INSPECTION DISCOVERED A NUMBER OF EQ COMPONENTS THAT WERE NOT INSTALLED PER APPROVED RAYCHEM GUIDELINES. PROCEDURES USED DURING THE INITIAL INSTALLATION OF THE RAYCHEM HEAT SHRINK TUBING DID NOT CONFORM TO THE RAYCHEM APPLICATION GUIDE WHICH RESULTED IN IMPROPER INSTALLATION. A CONTRIBUTING FACTOR WAS INADEQUATE TRAINING OF PERSONNEL INSTALLING THE HEAT SHRINK MATERIAL. THE DISCREPANCIES IDENTIFIED WERE CORRECTED. THE PROCEDURES HAVE BEEN REVISED TO REFLECT THE CURRENT APPLICATION GUIDE AND ADDITIONAL TRAINING HAS BEEN CONDUCTED.

[136] SUSQUEHANNA 1 DOCKET 50-387 LER 86-024 REV 01
UPDATE ON AUTOMATIC START RELAYS FOR THE EMERGENCY SERVICE WATER PUMPS NOT SEISMICALLY QUALIFIED.
EVENT DATE: 070386 REPORT DATE: 072087 NSSS: GE TYPE: BWR

OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)
VENDOR: WESTINGHOUSE ELECTRIC SUPPLY COMPANY

(NSIC 205496) ON JUNE 2, 1986, THE SEISMIC QUALIFICATION OF THE EMERGENCY SERVICE WATER (ESW) AUTOMATIC PUMP START RELAYS WAS QUESTIONED BY A DESIGN ENGINEER. AN ENGINEERING WORK REQUEST (EWR) WAS INITIATED ON JUNE 9, 1986 TO DETERMINE IF THE RELAYS WERE SEISMICALLY QUALIFIED AS REQUIRED BY THE FINAL SAFETY ANALYSIS REPORT. THE EWR WAS COMPLETED JULY 3, 1986 AND FOUND THAT THE RELAYS WERE NOT SEISMICALLY QUALIFIED. AT THE TIME WHEN THE SEISMIC QUALIFICATION WAS QUESTIONED, UNITS ONE AND TWO WERE SHUTDOWN (CONDITION 4). THE RELAYS WERE REPLACED BY JUNE 12, 1986 WITH SEISMICALLY QUALIFIED COMPONENTS PRIOR TO RESTARTING THE UNITS.

[137] SUSQUEHANNA 1 DOCKET 50-387 LER 87-022
CONTROL STRUCTURE CHILLER REPAIRS.
EVENT DATE: 061987 REPORT DATE: 072087 NSSS: GE TYPE: BWR
VENDOR: CARRIER AIR CONDITIONING CO.

(NSIC 205428) ON JUNE 19, 1987 WITH UNIT 1 OPERATING AT 100% POWER, LIMITING CONDITION FOR OPERATION (LCO) 3.0.3 WAS ENTERED AT 0509 AND CLEARED AT 0530 TO REPAIR THE "A" CONTROL STRUCTURE CHILLER. WHILE THE "B" CHILLER WAS OUT OF SERVICE FOR MAINTENANCE, THE RUNNING "A" CHILLER WAS EXPERIENCING SPURIOUS TRIPS DUE TO PROBLEMS WITH THE CYCLE TIMER AND LOW REFRIGERANT TRIP SWITCH. MAINTENANCE PERSONNEL REPLACED THE TIMER AND SWITCH. FOLLOWING REPLACEMENT THE CHILLER OPERATED SATISFACTORILY. THE CONTROL STRUCTURE CHILLERS PROVIDE COOLING WATER TO THE EMERGENCY SWITCHGEAR ROOM COOLERS. THE EQUIPMENT INSIDE THE SWITCHGEAR ROOM IS ENVIRONMENTALLY QUALIFIED TO OPERATE POST ACCIDENT PROVIDED THE ROOM TEMPERATURE DOES NOT EXCEED 100F. WITHOUT ONE CONTROL STRUCTURE CHILLER IN SERVICE THERE IS A POSSIBILITY THAT THE ROOM TEMPERATURE COULD EXCEED 104F IN THE EVENT OF A LOSS OF COOLANT ACCIDENT (LOCA). ALTHOUGH THE SWITCHGEAR ARE NOT QUALIFIED TO OPERATE ABOVE 104F, THERE IS SOME PROBABILITY THAT THEY WOULD CONTINUE TO OPERATE ABOVE 104F FOR SOME PERIOD OF TIME FULFILLING THEIR SAFETY FUNCTION; HOWEVER, PP&L CONSIDERS THE SWITCHGEAR INOPERABLE WITHOUT AT LEAST ONE CONTROL STRUCTURE CHILLER IN-SERVICE. AS SUCH, ENTRANCE INTO LCO 3.0.3 WAS NECESSITATED.

[138] SUSQUEHANNA 1 DOCKET 50-387 LER 87-021
ENTRY INTO LCO 3.0.3 TO PERFORM 4KV ESS BUS DEGRADED VOLTAGE RELAY SURVEILLANCES.
EVENT DATE: 062287 REPORT DATE: 071487 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)

(NSIC 205469) ON JUNE 22, 1987 WITH UNITS 1 AND 2 OPERATING AT 100% POWER, LIMITING CONDITION FOR OPERATION (LCO) 3.0.3 WAS ENTERED AND CLEARED FOUR (4) TIMES ON EACH UNIT TO PERFORM SURVEILLANCES ON THE 4.16 KV ENGINEERED SAFEGUARD SYSTEM (ESS) BUSSES (EIIS CODE: EB). TO PERFORM THE MONTHLY DEGRADED VOLTAGE CHANNEL FUNCTIONAL TESTS ON THE ESS BUSES, ALL DEGRADED VOLTAGE PROTECTION ON THE BUS IS TAKEN OUT OF SERVICE ALTHOUGH THE BUS REMAINS ENERGIZED. TECHNICAL SPECIFICATIONS REQUIRE 2 CHANNELS OF DEGRADED VOLTAGE PROTECTION PER BUS, AND BOTH CHANNELS MUST BE OPERABLE. THE LOSS OF BOTH CHANNELS OF DEGRADED VOLTAGE PROTECTION IS NOT ADDRESSED BY THE ACTION STATEMENT, THEREFORE ENTERING TECH SPEC. 3.0.3 IS REQUIRED. AN APPROVED AMENDMENT TO EACH UNIT'S LICENSE HAS BEEN RECEIVED FROM THE NRC SINCE THE OCCURRENCE WHICH CLARIFIES THE ACTION STATEMENT OF TABLE 3.3.3.1 SECTION 5 TO ADDRESS THE SITUATION WHERE BOTH CHANNELS OF DEGRADED VOLTAGE PROTECTION ARE INOPERABLE AT THE SAME TIME. THIS WILL PREVENT THE NECESSITY OF ENTERING TECH SPEC 3.0.3 TO PERFORM THIS TESTING.

[139] SUSQUEHANNA 1 DOCKET 50-387 LER 87-023
INOPERABLE PRIMARY CONTAINMENT VACUUM BREAKER SOLENOID VALVE.
EVENT DATE: [illegible]87 REPORT DATE: 081087 NSSS: GE TYPE: BWR
VENDOR: CIRCLE SEAL

(NSIC 205677) ON JULY 9, 1987 AT 0103 HOURS UNIT 1 WAS MANUALLY SHUT DOWN FROM 29% POWER AS REQUIRED BY TECHNICAL SPECIFICATION 3.6.4 ACTION STATEMENT. THIS WAS CAUSED WHEN A SOLENOID VALVE IN THE TEST CIRCUITRY OF VACUUM BREAKER PSV15704EL FAILED DURING MONTHLY SURVEILLANCE TESTING. THE VACUUM BREAKER ITSELF WAS CONFIRMED CLOSED AND SUBSEQUENT INVESTIGATION DETERMINED THE VACUUM BREAKER REMAINED OPERABLE, EXCEPT IN THE TEST NODE, AND WOULD HAVE PERFORMED ITS DESIGN FUNCTION UNDER ACCIDENT CONDITIONS. THE SHUTDOWN WAS COMMENCED AT 2000 HOURS ON 7-8-87 AND COMPLETED AT 0103 HOURS ON 7-9-87. IT WAS DETERMINED THAT A SOLENOID VALVE COIL WHICH IS UTILIZED IN THE TEST CIRCUIT OF THE VACUUM BREAKER WAS DEFECTIVE PREVENTING IT FROM STROKING UNDER TEST. THE SOLENOID COIL WAS REPLACED ON 7-10-87 AND THE UNIT SUBSEQUENTLY RETURNED TO SERVICE.

[140] THREE MILE ISLAND 2 DOCKET 50-320 LER 86-001 REV 01
UPDATE ON INOPERABILITY OF EMERGENCY DIESEL GENERATOR DUE TO DEFECTIVE GOVERNORS.
EVENT DATE: 122086 REPORT DATE: 071487 NSSS: BW TYPE: PWR
VENDOR: FAIRBANKS MORSE
WILMAR ELECTRONICS INC.

(NSIC 205495) AT 2050 HOURS ON DECEMBER 13, 1985, THE ANNUAL PREVENTIVE MAINTENANCE FOR EMERGENCY DIESEL GENERATOR DF-X-1B COMMENCED. ON DECEMBER 20, 1985, WHILE PERFORMING THE REQUIRED DEMONSTRATION OF ENGINE OPERABILITY, IT WAS DETERMINED THAT THERE WAS NO REMOTE SPEED CONTROL. TROUBLESHOOTING EVENTUALLY IDENTIFIED THE GOVERNOR AS BEING DEFECTIVE. A REPLACEMENT GOVERNOR WAS OBTAINED AND INSTALLED. RETESTING INDICATED THAT THE REMOTE SPEED CONTROL PROBLEM WAS CORRECTED, BUT THERE WAS NO SHUTDOWN CAPABILITY. FURTHER TROUBLESHOOTING DISCOVERED A DEFECTIVE SOLENOID IN THE REPLACEMENT GOVERNOR. A NEW SOLENOID WAS OBTAINED AND INSTALLED. ENGINE OPERABILITY WAS SUCCESSFULLY DEMONSTRATED AND THE ENGINE WAS RETURNED TO SERVICE AT 1709 HOURS ON DECEMBER 21, 1985. THUS, EMERGENCY DIESEL GENERATOR DF-X-1B WAS OUT-OF-SERVICE FOR MORE THAN SEVEN (7) DAYS, EXCEEDING THE TIMECLOCK OF THE ACTION 10 CFR 50.73(A)(2)(I)(B). THE GOVERNOR AND SOLENOID WERE RETURNED TO THE MANUFACTURER FOR ANALYSIS. THE GOVERNOR WAS LEAKING OIL INTERNALLY AS A RESULT OF A DEFECTIVE PILOT VALVE PUSHING AND PILOT VALVE PLUNGER. THE SUBSEQUENT SHUTDOWN SOLENOID FAILURE WAS ATTRIBUTABLE TO LOOSE FITTINGS ON THE SHUTDOWN ASSEMBLY WHICH RESULTED IN THE SHUTDOWN ASSEMBLY BEING UNABLE TO CONTROL THE GOVERNOR. BASED ON THE ANALYSIS RESULTS, NO LONG-TERM CORRECTIVE ACTIONS ARE CONSIDERED NECESSARY.

[141] TROJAN DOCKET 50-344 LER 87-015
STEAM GENERATOR LEVEL TRANSMITTERS IMPROPERLY CALIBRATED.
EVENT DATE: 041387 REPORT DATE: 072487 NSSS: WE TYPE: PWR

(NSIC 205564) DURING APRIL 13 THROUGH APRIL 15, 1987 ANNUAL CALIBRATION OF STEAM GENERATOR LEVEL TRANSMITTERS WAS PERFORMED. TEN STEAM GENERATOR LEVEL TRANSMITTERS WERE FOUND APPARENTLY OUT-OF-CALIBRATION SUCH THAT THEY WOULD NOT HAVE ACTUATED A TURBINE TRIP AND FEEDWATER ISOLATION ON HIGH-HIGH STEAM GENERATOR WATER LEVEL WITHIN THE TECH SPEC ALLOWED VALUE OF LESS THAN OR EQUAL TO 76%. FURTHER INVESTIGATION REVEALED THAT THE CALIBRATION WAS PERFORMED IMPROPERLY. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR AND PROCEDURE DEFICIENCY. THE TECHNICIANS PERFORMING THE CALIBRATION FAILED TO DRAIN THE WATER FROM THE TRANSMITTERS PRIOR TO BEGINNING CALIBRATION. THE PROCEDURE DID NOT SPECIFY THAT WATER SHOULD BE DRAINED FROM THE TRANSMITTERS. THE LEVEL TRANSMITTERS WERE PROPERLY RE-CALIBRATED TO WITHIN THE ALLOWED TOLERANCE. I&C TECHNICIANS WERE COUNSELED ON THE NEED TO DRAIN THE WATER FROM THE LEVEL TRANSMITTERS PRIOR TO PERFORMING CALIBRATION. THE CALIBRATION PROCEDURE WILL BE REVISED PRIOR TO THE

NEXT CALIBRATION TO REQUIRE DRAINING WATER FROM THE TRANSMITTERS. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[142] TROJAN DOCKET 50-344 LER 87-017
LEAKAGE THROUGH CONTROL ROOM FLOOR CABLE PENETRATIONS - INSUFFICIENT CONTROL ROOM POSITIVE PRESSURE.
EVENT DATE: 041787 REPORT DATE: 073087 NSSS: WE TYPE: PWR

(NSIC 205667) ON APRIL 17, 1987 DURING TESTING OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM, A POSITIVE PRESSURE OF 0.125 INCHES OF WATER COULD NOT BE MAINTAINED. INVESTIGATION REVEALED THAT OPERATION OF THE CABLE SPREADING ROOM SMOKE EXHAUST FAN (CB-13) WAS AFFECTING CONTROL ROOM PRESSURIZATION. THE CABLE SPREADING ROOM IS LOCATED IMMEDIATELY BELOW THE CONTROL ROOM. THE CAUSE OF THIS EVENT WAS AIR LEAKAGE THROUGH CABLE PENETRATIONS IN THE CONTROL ROOM FLOOR. WITH CB-13 OPERATING, AIR WAS DRAWN FROM THE CONTROL ROOM INTO THE CABLE SPREADING ROOM. OPERATING PROCEDURES ARE BEING REVISED TO PREVENT OPERATION OF CB-13 WHEN CONTROL ROOM PRESSURIZATION IS REQUIRED. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY AND IS NOT REPORTABLE UNDER 10 CFR 50.73. THIS EVENT IS BEING REPORTED VOLUNTARILY AS AN ITEM OF INTEREST.

[143] TROJAN DOCKET 50-344 LER 87-018
VALVE PACKING LEAKAGE EXCEEDED FSAR ASSUMED LEAKAGE DUE TO NORMAL PACKING DEGRADATION.
EVENT DATE: 050987 REPORT DATE: 073187 NSSS: WE TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 205668) DURING LOCAL LEAK RATE TESTING (LLRT), ON MAY 9, 1987, THE CONTAINMENT SPRAY AND RESIDUAL HEAT REMOVAL RECIRCULATION SUCTION VALVES OUTSIDE CONTAINMENT (MO2052B AND MO-8811B) EXHIBITED PACKING LEAKS. THE LEAKAGE EXCEEDED THE 1580 CUBIC CENTIMETERS PER HOUR ASSUMED IN THE FINAL SAFETY ANALYSIS REPORT FOR POST-ACCIDENT RECIRCULATION LEAKAGE. THE CAUSE OF THE VALVE PACKING LEAKS WAS ATTRIBUTED TO NORMAL PACKING DEGRADATION. THE VALVE PACKINGS WERE TIGHTENED AND TESTED SATISFACTORILY. THE VALVE PACKING PROGRAM FOR VALVES IN THE RECIRCULATION FLOW PATH WILL BE REVIEWED FOR CHANGES NECESSARY TO PREVENT RECURRENCE. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY. THE LLRT OF THIS PENETRATION WAS PERFORMED WITH AIR AT 60 PSIG. ACTUAL LEAKAGE IN THE EVENT OF OPERATION OF THE RECIRCULATION SUMP FLOW PATH WOULD HAVE BEEN LIQUID INSTEAD OF GAS, AND (2) RECIRCULATION DOES NOT COMMENCE UNTIL POST-ACCIDENT CONTAINMENT PRESSURES HAVE DECREASED SIGNIFICANTLY BELOW 60 PSIG.

[144] TROJAN DOCKET 50-344 LER 87-016
REACTOR TRIP BREAKERS INADVERTENTLY OPENED DUE TO PROCEDURE DEFICIENCY.
EVENT DATE: 062987 REPORT DATE: 072987 NSSS: WE TYPE: PWR

(NSIC 205666) ON JUNE 29, 1987, THE REACTOR TRIP BREAKERS WERE INADVERTENTLY ACTUATED DURING TESTING. THE "A" AND "B" REACTOR TRIP BREAKERS HAD BEEN CLOSED DURING A SIMULATED LOSS OF OFFSITE POWER AND EMERGENCY DIESEL GENERATOR START TEST. UPON CLOSURE OF THE "B" BYPASS BREAKER AS REQUIRED BY THE TEST, A REACTOR TRIP SIGNAL WAS GENERATED AND ALL THREE BREAKERS OPENED. THE CAUSE OF THIS EVENT WAS A PROCEDURE DEFICIENCY. THE PROCEDURE DID NOT REQUIRE VERIFICATION OF THE STATUS OF PROTECTION SYSTEM TROUBLE ANNUNCIATORS. AN "A" TRAIN GENERAL WARNING CONDITION EXISTED DUE TO AN OPEN 48 VOLT DC "A" TRAIN POWER SUPPLY BREAKER. CLOSURE OF THE "B" BYPASS BREAKER ACTUATED THE "B" GENERAL WARNING LOGIC WHICH RESULTED IN THE REACTOR TRIP SIGNAL. PROCEDURES WILL BE REVISED TO REQUIRE VERIFICATION OF PROTECTION SYSTEM TROUBLE ANNUNCIATORS STATUS AND TO NOTE THAT CLOSURE OF THE BYPASS BREAKERS ACTUATES THE GENERAL WARNING LOGIC. CONTROL ROOM INDICATION OF GENERAL WARNING CONDITIONS WILL BE REVIEWED FOR IMPROVEMENT. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[145] TURKEY POINT 3 DOCKET 50-250 LER 87-020
INTAKE COOLING WATER TEMPERATURE EXCEEDED LIMITS BASED ON ENGINEERING EVALUATION FOR COMPONENT COOLING WATER HEAT EXCHANGERS.
EVENT DATE: 121186 REPORT DATE: 071687 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)
VENDOR: ENGINEERS AND FABRICATORS, INC.

(NSIC 205440) ON JUNE 16, 1987, A REVIEW OF COMPONENT COOLING WATER (CCW) HEAT EX CHANGER PERFORMANCE TEST DATA FROM DECEMBER, 1986 REVEALED THAT UNIT 3 HAD OPERATED OUTSIDE OF A ENGINEERING EVALUATION FOR THE CCW HEAT EXCHANGERS FOR APPROXIMATELY 17 HOURS ON DECEMBER 11, 1986. IN FEBRUARY OF 1986, FPL ENGINEERING NOTIFIED TURKEY POINT OF A POTENTIAL CONCERN ON THE ABILITY OF THE INTAKE COOLING WATER (ICW) SYSTEM TO MEET FLOW REQUIREMENTS FOR A DESIGN BASIS ACCIDENT. ENGINEERING CONTINUED TO REVIEW THIS CONCERN AND DEVELOPED AN EVALUATION OF THIS CONDITION FOR UNITS 3 AND 4. THE EVALUATION PROVIDED GUIDANCE TO THE PLANT TO DEVELOP A PROGRAM TO ROUTINELY EVALUATE THE CCW HEAT EXCHANGER PERFORMANCE. ON DECEMBER 1, 1986, PERFORMANCE TESTS WERE PERFORMED ON THE 3A, 3B, AND 3C CCW HEAT EXCHANGERS. THE PRELIMINARY ANALYSIS OF THE DATA FROM THESE TESTS INDICATED THAT THE LEVEL OF CLEANLINESS OF THE CCW HEAT EXCHANGERS AFTER CLEANING WAS NOT AS EXPECTED. AFTER ADDITIONAL REVIEW OF THE DATA IT WAS DETERMINED ON JUNE 16, 1987 THAT DURING THE CLEANING OF THE 3B CCW HEAT EXCHANGER ON DECEMBER 11, 1986 THE REQUIREMENTS OF THE EVALUATION WERE EXCEEDED. A CONTINUOUS TUBE CLEANING SYSTEM IS BEING INSTALLED ON UNIT 3 DURING THE CURRENT REFUELING OUTAGE. SIMILAR MODIFICATIONS WILL BE DONE ON UNIT 4 DURING THE NEXT REFUELING OUTAGE.

[146] VOGTLE 1 DOCKET 50-424 LER 87-035
FAULTY MAIN FEEDWATER PUMP TURBINE HYDRAULIC TUBING CONNECTION LEADS TO REACTOR TRIP.
EVENT DATE: 061487 REPORT DATE: 071487 NSSS: SS TYPE: PWR
VENDOR: GEN ELEC CO (STEAM TURB/ENGRD PROD)

(NSIC 205482) ON JUNE 14, 1987, AT 1724 CDT WITH UNIT 1 IN MODE 1 AT 97% OF RATED THERMAL POWER, THE REACTOR TRIPPED ON STEAM GENERATOR NO. 4 LOW-LOW WATER LEVEL. THE REACTOR PROTECTION SYSTEM INITIATED A REACTOR TRIP, AND STARTED THE MOTOR-DRIVEN AUXILIARY FEEDWATER PUMPS. A TURBINE TRIP AND FEEDWATER ISOLATION (FWI) OCCURRED. THE FWI CAUSED A TRIP OF BOTH MAIN FEEDWATER (MFW) PUMPS. PLANT OPERATORS RESPONDED PROPERLY TO ENSURE PLANT SAFETY. THE EVENT WAS CAUSED BY THE FAILURE OF A HYDRAULIC TUBING CONNECTION. FAILURE TO INSTALL A FERRULE IN THE CONNECTION EVENTUALLY LED TO ITS FAILURE. THE RESULTANT LOW HYDRAULIC PRESSURE CAUSED A MFW PUMP TURBINE CONTROL VALVE TO PARTIALLY CLOSE, LEADING TO A SLOWDOWN IN PUMP SPEED AND A DECREASE IN FEEDWATER TO THE STEAM GENERATORS. THE FAULTY HYDRAULIC TUBING CONNECTION WAS REPLACED AND SIMILAR CONNECTIONS WERE INSPECTED TO ENSURE THEIR INTEGRITY.

[147] VOGTLE 1 DOCKET 50-424 LER 87-036
AUXILIARY FEEDWATER ACTUATION CIRCUITRY INOPERABLE DUE TO PERSONNEL ERROR.
EVENT DATE: 061587 REPORT DATE: 071587 NSSS: SS TYPE: PWR

(NSIC 205483) ON JUNE 15, 1987 AT APPROXIMATELY 0600 CDT WITH THE UNIT IN MODE 3, AN OPERATOR DISCOVERED THAT THE AUXILIARY FEEDWATER (AFW) ACTUATION CIRCUITRY HAD BEEN DISABLED FOR APPROXIMATELY 11 HOURS. THIS WAS CONTRARY TO TECHNICAL SPECIFICATION REQUIREMENTS FOR MODE 3. A CLEARANCE, WRITTEN TO SUPPORT MAINTENANCE ACTIVITIES ON THE MAIN FEEDWATER (MFW PUMPS, ALLOWED THE CIRCUITRY TO BE IMPROPERLY DISABLED AT 1904 CDT ON JUNE 14, 1987. THE CLEARANCE WAS MEANT TO ALLOW BLOCKING AN AFW SYSTEM ACTUATION UPON RECEIPT OF A MFW PUMP TRIP SIGNAL. ON JUNE 22, 1987, DURING EVENT INVESTIGATION, A REVIEW OF PREVIOUS AFW CLEARANCES REVEALED A SIMILAR EVENT HAD OCCURRED ON JUNE 3, 1987. THIS SIMILAR EVENT WILL BE DISCUSSED IN A SUPPLEMENTAL REPORT. PLANT PERSONNEL DISABLED THE AFW VALVE ACTUATION CIRCUITRY BY PULLING INCORRECT FUSES DUE TO INCORRECTLY USING

AND INTERPRETING A PROCEDURE. THIS WAS CAUSED BY A PERSONNEL ERROR COMPOUNDED BY PROCEDURAL INADEQUACY. THE CLEARANCE WAS RELEASED AND THE IMPROPERLY PULLED FUSES WERE REINSTALLED. PROCEDURAL CHANGES HAVE BEEN MADE TO CLEARLY DESIGNATE THE PROPER FUSES TO BE PULLED FOR BLOCKING SPECIFIC FEATURES OF THE AFW ACTUATION CIRCUITRY.

[148] VOGTLE 1 DOCKET 50-424 LER 87-037
FAILURE TO MEET TECHNICAL SPECIFICATION ACTION STATEMENT DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 061687 REPORT DATE: 071687 NSSS: SS TYPE: PWR

(NSIC 205547) ON JUNE 16, 1987, PLANT PERSONNEL DISCOVERED THAT TECHNICAL SPECIFICATIONS HAD POTENTIALLY BEEN VIOLATED BECAUSE A LEAKING CONTAINMENT ISOLATION VALVE (CIV) MAY HAVE BEEN INOPERABLE AND THE APPROPRIATE TECHNICAL SPECIFICATION ACTION STATEMENT WAS NOT BEING MET. ON JUNE 18, 1987, AT 1440 CDT, THE CIV WAS DECLARED INOPERABLE AND THE APPROPRIATE LCO ACTION WAS MET. THE VALVE WAS FOUND LEAKING ON FEBRUARY 22, 1987, AND AN MWO WAS WRITTEN TO INVESTIGATE. THE MWO STATED THAT THE VALVE WAS SUBJECT TO LLRT REQUIREMENTS. THIS EVENT OCCURRED BECAUSE A DEFICIENCY CARD WAS NOT WRITTEN WHICH WOULD HAVE PROVIDED FOR AN ENGINEERING EVALUATION. ADDITIONALLY, THE DEFICIENCY CONTROL PROCEDURE FAILED TO PROVIDE ADEQUATE INSTRUCTIONS OR GUIDANCE FOR INITIATING A DEFICIENCY CARD. THE ADMINISTRATIVE PROCEDURE 00150-C, "DEFICIENCY CONTROL", WILL BE REVISED TO PROVIDE ADEQUATE DIRECTIONS FOR INITIATING A DEFICIENCY CARD. THIS LER WAS NOT SUBMITTED WITHIN THE THIRTY (30) DAY TIME PERIOD DUE TO EXTENDED EVALUATION. THE DELAY WAS DISCUSSED WITH APPROPRIATE REGION II PERSONNEL.

[149] VOGTLE 1 DOCKET 50-424 LER 87-043
IMPROPER PERFORMANCE OF CONTAINMENT PRESSURE SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 061687 REPORT DATE: 072887 NSSS: SS TYPE: PWR

(NSIC 205639) ON JUNE 28, 1987, WITH UNIT 1 OPERATING IN MODE 1, A REACTOR OPERATOR DID NOT NOTICE A MARKED-UP TEMPORARY CHANGE ON A DATA SHEET IN PROCEDURE 14000-1 "OPERATIONS SHIFT AND DAILY SURVEILLANCE LOGS". CONSEQUENTLY, THE REACTOR OPERATOR DID NOT OBTAIN A CONTAINMENT PRESSURE READING NEEDED TO VERIFY THAT THE PLANT WAS IN COMPLIANCE WITH TECHNICAL SPECIFICATION (T.S.) 3.6.1.4, "INTERNAL PRESSURE". THIS OMISSION WAS DISCOVERED BY THE NIGHT SHIFT, ALBEIT BEYOND THE ALLOWED GRACE PERIOD. SUBSEQUENTLY, CONTAINMENT PRESSURE WAS VERIFIED TO BE WITHIN T.S. LIMITS. PROCEDURE 14000-1 DATA SHEETS HAVE BEEN PERMANENTLY REVISED AND SHOULD PRECLUDE THE INADVERTENT OMISSION OF THIS MEASUREMENT OF CONTAINMENT PRESSURE. THE REACTOR OPERATOR AND SHIFT SUPERVISOR WERE COUNSELED ON PROPER OBSERVANCE OF TEMPORARY CHANGES AND COMPLETION OF TECHNICAL SPECIFICATION SURVEILLANCES.

[150] VOGTLE 1 DOCKET 50-424 LER 87-038
MANUAL REACTOR TRIPS DUE TO OVERLY CONSERVATIVE ANNUNCIATOR RESPONSE PROCEDURE.
EVENT DATE: 062087 REPORT DATE: 072087 NSSS: SS TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 205484) ON JUNE 20, 1987 AND ON JUNE 25, 1987, THE UNIT WAS IN MODE 3 AND THE SHUTDOWN ROD BANKS WERE PARTIALLY WITHDRAWN WHEN THE REACTOR WAS MANUALLY TRIPPED AFTER RECEIPT OF AN "RPI URGENT ALARM" ANNUNCIATOR ON INDICATED FAILURE OF THE DIGITAL ROD POSITION INDICATOR (DRPI) FOR SHUTDOWN BANK "B", ROD G-3. AFTER THE INITIAL TRIP, A THOROUGH INSPECTION OF THE DRPI SYSTEM DID NOT IDENTIFY OR LOCATE A DEFECTIVE COMPONENT. AFTER THE SECOND EVENT, AN INSPECTION OF THE DRPI SYSTEM WAS AGAIN PERFORMED WHICH REPRODUCED THE FAILURE MODE. THE DETECTOR/ENCODER CARD FOR ROD G-3 IN THE "B" DRPI DATA CABINET WAS DISCOVERED TO HAVE FAILED. THIS CARD WAS REPLACED AND SUBSEQUENT TESTING DEMONSTRATED THAT PROPER INDICATION HAD BEEN RESTORED FOR ROD G-3. FURTHER REVIEW DETERMINED THAT

ANNUNCIATOR RESPONSE PROCEDURE 17010-1 WAS INCONSISTENT WITH THE SAFETY SIGNIFICANCE OF A FAILED DRPI INDICATOR CHANNEL AND TECHNICAL SPECIFICATION 3.1.3.3. PROCEDURE 17010-1 HAS BEEN REVISED TO REQUIRE THE OPERATOR, BEFORE OPENING THE TRIP BREAKERS, TO IMMEDIATELY PLACE THE DRPI SYSTEM IN DATA A ONLY AND DATA B ONLY TO VERIFY THAT THE REQUIRED ACTION OF TECHNICAL SPECIFICATIONS 3.1.3.3 APPLIES.

[151] VOGTLE 1 DOCKET 50-424 LER 87-039
PRESSURE TRANSMITTER FAILURE CAUSES ESF ACTUATION ON STEAM GENERATOR HI-HI WATER LEVEL.
EVENT DATE: 062087 REPORT DATE: 072087 NSSS: SS TYPE: PWR
VENDOR: ROSEMOUNT, INC.

(NSIC 205485) ON JUNE 20, 1987, AT 1552 CDT WITH UNIT 1 IN MODE 3, A MAIN FEEDWATER (MFW) ISOLATION, TRIP OF BOTH MFW PUMPS AND SUBSEQUENT MOTOR-DRIVEN AUXILIARY FEEDWATER (AFW) PUMP START SIGNAL OCCURRED DUE TO STEAM GENERATOR (SG) NO. 2 REACHING ITS HIGH-HIGH WATER LEVEL SETPOINT. THE SG HIGH-HIGH LEVEL WAS CAUSED BY AN ATMOSPHERIC RELIEF VALVE (ARV) THAT HAD LIFTED AND REMAINED OPEN. THE RESULTANT PRESSURE DROP CAUSED THE SG WATER VOLUME TO SWELL. THE MOTOR DRIVEN AFW PUMPS WERE BEING USED FOR SG LEVEL CONTROL; THE "A" MFW PUMP WAS OPERATING IN THE RECIRCULATION MODE AND THE "B" MFW PUMP WAS IN THE TRIPPED CONDITION. PLANT OPERATORS MANUALLY CLOSED THE ARV AND MANUALLY CONTROLLED THE MOTOR DRIVEN AFW PUMPS TO STABILIZE PLANT CONDITIONS. A FAILED PRESSURE TRANSMITTER (PT) CAUSED THE OPENING OF THE ARV. THE FAILED PRESSURE TRANSMITTER WAS REPLACED.

[152] VOGTLE 1 DOCKET 50-424 LER 87-040
CONTAINMENT VENTILATION ISOLATION DUE TO PERSONNEL ERROR.
EVENT DATE: 062287 REPORT DATE: 072287 NSSS: SS TYPE: PWR

(NSIC 205548) ON JUNE 22, 1987, WHILE IN MODE 1 AT 15% REACTOR POWER, A CONTAINMENT VENTILATION ISOLATION (CVI) OCCURRED ON A SIGNAL FROM CONTAINMENT VENTILATION RADIATION MONITOR 1RE-2565C. PLANT PERSONNEL WERE MODIFYING THE WIRING CONFIGURATION FOR THE RADIATION MONITOR ACTUATION BLOCK MECHANISM. THE CVI OCCURRED AFTER LIFTING A WIRE ON TERMINAL TB1-3, WHICH POWERED THE BLOCKING RELAY. THIS ALLOWED THE ALARM RELAYS, WHICH HAD POWER, TO INITIATE A CVI SIGNAL. TO CONFIRM THE CAUSE OF THE CVI, RADIATION MONITOR 1RE-2565C WAS PLACED IN THE OFF POSITION AND THE WIRE TO TERMINAL TB1-3 WAS AGAIN LIFTED. WHEN THE WIRE WAS LIFTED, VOLTAGE DROPPED TO THE BLOCKING RELAY WHICH ALLOWED THE HIGH RADIATION ALARM RELAYS TO INITIATE THE CVI. THE CVI WAS CAUSED BY PERSONNEL FAILING TO PERFORM ADEQUATE REVIEW OF CLEARANCES PRIOR TO MAKING EQUIPMENT MODIFICATIONS. THUS, INADEQUATE MEASURES WERE EMPLOYED TO PREVENT THE CVI SIGNAL FROM ENTERING THE SOLID STATE PROTECTION SYSTEM. ENGINEERING AND MAINTENANCE PERSONNEL WERE COUNSELED CONCERNING THE CAUSE OF THIS INCIDENT AND WERE PROVIDED GUIDANCE TO AVOID REPETITION.

[153] VOGTLE 1 DOCKET 50-424 LER 87-041
REACTOR TRIP DUE TO IMPROPERLY CALIBRATED FIELD CURRENT TRANSDUCERS.
EVENT DATE: 062387 REPORT DATE: 072387 NSSS: SS TYPE: PWR

(NSIC 205549) ON JUNE 23, 1987, WHILE IN MODE 1 A REACTOR TRIP OCCURRED DUE TO THE AUTOMATIC TRIP OF THE MAIN TURBINE. THE REACTOR TRIP AND SUBSEQUENT TRANSIENT RESULTED IN A FEEDWATER ISOLATION AND ACTUATION OF THE AUXILIARY FEEDWATER SYSTEM. THE OPERATORS RECOVERED SG LEVELS AND RESTORED THE PLANT TO STABLE CONDITIONS. THE MAIN TURBINE TRIP WAS INITIATED BY A GENERATOR TRIP FROM A GENERREX VOLTAGE REGULATOR TROUBLE TRIP. THE HIGH VOLTAGE SWITCHYARD 500 KV INDUCTIVE REACTORS HAD BEEN CLOSED TO REDUCE GRID VOLTAGE. AN EXCESSIVE FIELD CURRENT SIGNAL RESULTING FROM AN IMPROPERLY CALIBRATED FIELD EXCITATION

TRANSDUCER CAUSED THE GENERATOR TRIP. THE EVENT WAS CAUSED BY THE FAILURE TO USE THE FIELD CURRENT TRANSDUCER CALIBRATION PROCEDURE. CORRECTIVE ACTIONS INCLUDED THE RECALIBRATION OF FIELD CURRENT TRANSDUCERS, THE PERFORMANCE OF A 'NO-LOAD' FIELD CURRENT CHECK OF THE GENERATOR TO VERIFY DESIGN, AND RESTRICTIONS TO ENSURE THAT TRANSMATION DEVICES ARE USED ONLY WITH CALIBRATED MEASURING AND TEST EQUIPMENT. INSTRUMENT AND CONTROLS SUPERVISION WERE COUNSELLED REGARDING IN-PROGRESS AND POST-WORK REVIEWS, AND THE CONTROL OF ENGINEERS AND CONTRACTOR PERSONNEL.

[154] VOGTLE 1 DOCKET 50-424 LER 87-042
BORON CONCENTRATION EXCEEDS TECH SPEC LIMITING CONDITION OF OPERATION TIME LIMIT.
EVENT DATE: 062687 REPORT DATE: 072687 NSSS: SS TYPE: PWR

(NSIC 205550) ON JUNE 20, 1987, AT 1400 CDT, THE PLANT WAS IN MODE 3 (HOT STANDBY), WHEN A TECHNICAL SPECIFICATION (T.S.) SURVEILLANCE WAS CONDUCTED BY CHEMISTRY PERSONNEL FOR THE BORON CONCENTRATION IN THE REFUELING WATER STORAGE TANK (RWST). THE MEASURED CONCENTRATION OF 2194 PPM EXCEEDED THE REQUIRED CONCENTRATION OF 2000-2100 PPM. THIS SURVEILLANCE RESULT WAS NOT COMMUNICATED TO THE PLANT OPERATORS. THUS, THE APPROPRIATE LIMITING CONDITION FOR OPERATION (LCO) WAS NOT ENTERED, AND THE T.S. WAS VIOLATED. A REVIEW OF SURVEILLANCE LOGS ON JUNE 26, 1987, DISCOVERED THIS EVENT. CORRECTIVE ACTION INCLUDED COUNSELING THE PERSONNEL INVOLVED AND ISSUING A MEMORANDUM TO ALL DEPARTMENT PERSONNEL EMPHASIZING THE IMPORTANCE OF REPORTING SURVEILLANCE TEST RESULTS. AN ADDITIONAL SURVEILLANCE, AFTER MIXING OF THE RWST CONTENTS, CONFIRMED THE BORON CONCENTRATION WAS WITHIN ALLOWED VALUES.

[155] WOLF CREEK 1 DOCKET 50-482 LER 87-015 REV 01
UPDATE ON ENGINEERED SAFETY FEATURES ACTUATION - CONTROL ROOM VENTILATION ISOLATION SIGNAL CAUSED BY PAPER TAPE BUNCHING UP ON CHLORINE MONITOR.
EVENT DATE: 041287 REPORT DATE: 071787 NSSS: WE TYPE: PWR
VENDOR: M D A SCIENTIFIC, INC.

(NSIC 205491) ON APRIL 12, 1987, AT 0044 CDT, A CONTROL ROOM VENTILATION ISOLATION SIGNAL (CRVIS) OCCURRED DUE TO CHLORINE MONITOR GK-AITS-2 INDICATING HIGH CHLORINE LEVEL IN THE OUTSIDE AIR MAKEUP TO THE CONTROL BUILDING HEATING, VENTILATING AND AIR CONDITIONING SYSTEM. UPON RECEIPT OF THE CRVIS, ALL ENGINEERED SAFETY FEATURES EQUIPMENT REQUIRED TO OPERATE RESPONDED PROPERLY. DURING THIS EVENT, THE PLANT WAS IN MODE 1, POWER OPERATION, WITH THE REACTOR AT 100 PERCENT POWER. NO CHLORINE WAS PRESENT AS EVIDENCED BY NORMAL READINGS ON THE REDUNDANT CHLORINE MONITOR. THE CONTROL BUILDING HEATING, VENTILATING AND AIR CONDITIONING SYSTEM WAS RETURNED TO A NORMAL CONFIGURATION AT 0057 CDT, APRIL 12, 1987, AND THE AFFECTED MONITOR WAS PLACED IN BYPASS FOR TROUBLESHOOTING. EXAMINATION OF THE MONITOR AFTER THE EVENT REVEALED THAT THE CHEMICALLY SENSITIVE PAPER TAPE USED TO DETECT CHLORINE HAD STOPPED WINDING ONTO THE TAKE-UP SPOOL AND WAS, INSTEAD, BUNCHING UP. THIS EVENTUALLY CAUSED LESS LIGHT TO BE ABLE TO PASS THROUGH THE PAPER, INITIATING THE CRVIS. THE PAPER TAPE WAS REPLACED. NO FURTHER PROBLEMS WITH THE MONITOR WERE FOUND AND IT WAS RETURNED TO OPERATION AT 1246 CDT ON APRIL 13, 1987.

[156] WOLF CREEK 1 DOCKET 50-482 LER 87-020 REV 01
UPDATE ON ENGINEERED SAFETY FEATURES ACTUATION - CONTROL ROOM VENTILATION ISOLATION SIGNAL CAUSED BY PAPER TAPE BREAKING ON CHLORINE MONITOR.
EVENT DATE: 050687 REPORT DATE: 071787 NSSS: WE TYPE: PWR
VENDOR: M D A SCIENTIFIC, INC.

(NSIC 205492) ON MAY 6, 1987, AT 2319 CDT, A CONTROL ROOM VENTILATION ISOLATION SIGNAL (CRVIS) OCCURRED DUE TO CHLORINE MONITOR GK-AITS-3 INDICATING HIGH CHLORINE LEVEL IN THE OUTSIDE AIR MAKEUP TO THE CONTROL BUILDING HEATING,

VENTILATING AND AIR CONDITIONING SYSTEM. UPON RECEIPT OF THE CRVIS, ALL ENGINEERED SAFETY FEATURES EQUIPMENT REQUIRED TO OPERATE RESPONDED PROPERLY. DURING THIS EVENT, THE PLANT WAS IN MODE 1, POWER OPERATION, WITH THE REACTOR AT 100 PERCENT POWER. NO CHLORINE WAS PRESENT AS EVIDENCED BY NORMAL READINGS ON THE REDUNDANT CHLORINE MONITOR. THE CONTROL BUILDING HEATING, VENTILATING AND AIR CONDITIONING SYSTEM WAS RETURNED TO A NORMAL CONFIGURATION AT 0007 CDT, MAY 7, 1987. EXAMINATION OF THE MONITOR AFTER THE EVENT REVEALED THAT THE CHEMICALLY SENSITIVE PAPER TAPE USED TO DETECT CHLORINE HAD BROKEN. THIS CAUSED MORE LIGHT TO BE ABLE TO PASS THROUGH, RESULTING IN A CHANGE IN OPACITY READING SUFFICIENT FOR THE MONITOR TO INITIATE A CRVIS. THE PAPER TAPE WAS REPLACED. NO FURTHER PROBLEMS WITH THE MONITOR WERE FOUND AND IT WAS RETURNED TO OPERATION AT 0007 CDT ON MAY 7, 1987.

[157] WOLF CREEK 1 DOCKET 50-482 LER 87-024
ENGINEERED SAFETY FEATURES ACTUATION - CONTROL ROOM VENTILATION ISOLATION SIGNAL CAUSED BY LOSS OF POWER TO CHLORINE MONITOR BECAUSE OF FAULTY SAMPLE PUMP.
EVENT DATE: 061587 REPORT DATE: 071487 NSSS: WE TYPE: PWR
VENDOR: M D A SCIENTIFIC, INC.

(NSIC 205493) ON JUNE 15, 1987, AT 0802 CDT, A CONTROL ROOM VENTILATION ISOLATION SIGNAL (CRVIS) OCCURRED DUE TO CHLORINE MONITOR GK-AITS-3 INDICATING HIGH CHLORINE LEVEL IN THE OUTSIDE AIR MAKEUP TO THE CONTROL BUILDING HEATING, VENTILATING AND AIR CONDITIONING (HVAC) SYSTEM. UPON RECEIPT OF THE CRVIS, ALL ENGINEERED SAFETY FEATURES EQUIPMENT REQUIRED TO OPERATE RESPONDED PROPERLY. NO CHLORINE WAS PRESENT AS EVIDENCED BY NORMAL READINGS ON THE REDUNDANT CHLORINE MONITOR. THE HVAC SYSTEM WAS RETURNED TO A NORMAL CONFIGURATION AT 0830 CDT, JUNE 15, 1987. INVESTIGATIONS REVEALED THAT A FUSE HAD BLOWN IN THE POWER SUPPLY FOR GK-AITS-3. THE RESULTANT LOSS OF POWER TO THE MONITOR INITIATED THE CRVIS. THE BLOWN FUSE WAS CAUSED BY A SHORT WITHIN THE EXTERNALLY MOUNTED SAMPLE PUMP. THE PUMP WAS REPLACED AND THE MONITOR WAS RETURNED TO SERVICE AT 0155 CDT ON JUNE 16, 1987. BECAUSE OF NUMEROUS PROBLEMS EXPERIENCED WITH THIS TYPE OF MONITOR, A DESIGN CHANGE IS BEING CONSIDERED TO REDUCE THE POTENTIAL FOR UNNECESSARY CRVIS'S INITIATED BY MALFUNCTIONS OF THESE MONITOR.

[158] WOLF CREEK 1 DOCKET 50-482 LER 87-028
TECHNICAL SPECIFICATION VIOLATION - INOPERABLE CLASS IE BATTERIES DUE TO INADEQUATE POST-TEST REVIEW OF SURVEILLANCE TEST.
EVENT DATE: 061987 REPORT DATE: 072087 NSSS: WE TYPE: PWR
VENDOR: GOULD-NATIONAL BAIT

(NSIC 205494) ON TWO OCCASIONS, A QUARTERLY SURVEILLANCE TEST INDICATED THE CLASS IE 125 VOLT BATTERIES WERE OUTSIDE THE CATEGORY B LIMITS FOR THE PARAMETERS SHOWN ON TECHNICAL SPECIFICATIONS TABLE 4.8-2. THE FIRST OCCURRENCE WAS ON JANUARY 28, 1985, PRIOR TO RECEIPT OF THE LOW POWER OPERATING LICENSE. THE UNIT LOADED FUEL STARTING MARCH 12, 1985 AND OPERATED UP TO MODEL 4, HOT SHUTDOWN, BEFORE THE NEXT QUARTERLY SURVEILLANCE TEST WHICH VERIFIED COMPLIANCE WITH TECHNICAL SPECIFICATIONS TABLE 4.8-2 WAS CONDUCTED ON APRIL 25, 1985. THE SECOND OCCURRENCE WAS ON NOVEMBER 7, 1986, WITH THE PLANT IN MODE 6, REFUELING; AFTER WHICH THE UNIT OPERATED IN ALL MODES UP TO 100 PERCENT POWER BEFORE THE NEXT QUARTERLY SURVEILLANCE TEST, ON JANUARY 23, 1987. THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE POST-TEST REVIEW OF PROCEDURES BY THE ELECTRICAL GROUP SUPERVISOR. THE POST-TEST REVIEW PROCESS WAS STRENGTHENED BY COUNSELLING OF MAINTENANCE DEPARTMENT PERSONNEL AND BY REQUIRING AN ADDITIONAL POST-TEST REVIEW BE CONDUCTED BY A MAINTENANCE ENGINEER, AS REPORTED IN OUR RESPONSE TO VIOLATION 482/8632-01, DATED MARCH 13, 1987. CONTRIBUTING CAUSES INCLUDE THE QUARTERLY SURVEILLANCE TEST ACCEPTANCE CRITERIA WORDING, WHICH WAS CONFUSING, AND AN INADEQUATE EQUALIZING CHARGE PROCEDURE. THE PROCEDURES ARE BEING REVISED TO CORRECT THESE DISCREPANCIES.

[159] WOLF CREEK 1 DOCKET 50-482 LER 87-025
ENGINEERED SAFETY ACTUATION - CONTAINMENT PURGE ISOLATION AND CONTROL ROOM VENTILATION ISOLATION DUE TO PERSONNEL ERROR DURING RADIATION MONITOR SURVEILLANCE TESTING.
EVENT DATE: 062387 REPORT DATE: 072287 NSSS: WE TYPE: PWR

(NSIC 205553) ON JUNE 23, 1987, AT 0959 CDT, AN ENGINEERED SAFETY FEATURES ACTUATION SIGNAL (ESFAS) WAS INITIATED DURING THE PERFORMANCE OF A SURVEILLANCE TEST PROCEDURE WHEN THE INCORRECT ESFAS CABINET BISTABLE TEST SWITCH WAS ERRONEOUSLY REPOSITIONED. THIS RESULTED IN A CONTAINMENT PURGE ISOLATION SIGNAL AND A CONTROL ROOM VENTILATION ISOLATION SIGNAL BEING INITIATED. ALL REQUIRED ENGINEERED SAFETY FEATURES EQUIPMENT RESPONDED PROPERLY. THE HEATING, VENTILATING AND AIR CONDITIONING SYSTEMS WERE RESTORED TO NORMAL CONFIGURATION AT 1040 CDT. TO PREVENT RECURRENCE, A REVISION TO EIGHT RADIATION MONITOR SURVEILLANCE TEST PROCEDURES TO INCLUDE A PROVISION FOR PERFORMING THIS TEST WITHOUT NECESSITATING OPERATION OF THESE PARTICULAR ESFAS CABINET BISTABLE TEST SWITCHES HAS BEEN INITIATED. THERE WAS NO DAMAGE TO PLANT EQUIPMENT OR RELEASE OF RADIOACTIVITY AS A RESULT OF THIS EVENT, AND IT POSED NO THREAT TO THE PUBLIC HEALTH AND SAFETY.

[160] WOLF CREEK 1 DOCKET 50-482 LER 87-026
TECHNICAL SPECIFICATION VIOLATION - LATE PERFORMANCE OF SPENT FUEL BUILDING VENT TRITIUM ANALYSIS CAUSED BY PERSONNEL ERROR.
EVENT DATE: 062587 REPORT DATE: 072787 NSSS: WE TYPE: PWR

(NSIC 205508) ON JUNE 25, 1987, AT APPROXIMATELY 123? CDT, IT WAS DISCOVERED BY A CHEMISTRY SUPERVISOR THAT THE SPENT FUEL BUILDING VENT TRITIUM SAMPLING AND ANALYSIS, REQUIRED WEEKLY BY TECHNICAL SPECIFICATIONS (T/S), HAD NOT BEEN PERFORMED WITHIN THE T/S SPECIFIED TIME FRAME. PERFORMANCE OF THIS SURVEILLANCE WAS SCHEDULED FOR JUNE 23, WITH A "LATE DATE" OF JUNE 25 AT 0725 CDT. THE SURVEILLANCE WAS ALREADY IN PROGRESS AT THE TIME OF THIS DISCOVERY, HAVING BEEN INITIATED AT 1220 CDT, AND WAS COMPLETED SATISFACTORILY AT APPROXIMATELY 1345 CDT. THIS EVENT OCCURRED AS A RESULT OF PERSONNEL AND PROCEDURAL ERRORS. OVER THE COURSE OF SEVERAL SHIFTS, THE SHIFT CHEMISTRY TECHNICIANS WERE UNAWARE OF THE WEEKLY T/S REQUIREMENT AND FINALLY DETERMINED FROM A SAMPLING PROCEDURE THAT THE SURVEILLANCE WAS REQUIRED MONTHLY. THE SAMPLING PROCEDURE WAS IN ERROR AND HAS BEEN REVISED TO DELETE THE FREQUENCY STATEMENT. THE TECHNICIANS HAVE BEEN REMINDED THAT FREQUENCY REQUIREMENTS ARE SPECIFIED IN THE CHEMISTRY SURVEILLANCE PROGRAM. THIS SURVEILLANCE HAS BEEN ADDED TO THE CONTROL ROOMS LIST STATUS CHART TO PROVIDE AN ADDITIONAL CHECK TO ENSURE IT IS PERFORMED IN A TIMELY MANNER.

[161] WOLF CREEK 1 DOCKET 50-482 LER 87-027
REACTOR TRIP RESULTING FROM PERSONNEL ERROR IN NOT CORRECTLY TIGHTENING INSTRUMENT SENSING LINE WHICH CAUSED LOSS OF MAIN FEEDWATER PUMP.
EVENT DATE: 062987 REPORT DATE: 072987 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 205699) AT APPROXIMATELY 0647 CDT ON JUNE 29, 1987, A PARTIAL LOSS OF FEEDWATER FLOW OCCURRED. A FITTING ON AN INSTRUMENT LINE, INTERNAL TO THE LUBE OIL RESERVOIR, SENSING MAIN FEEDWATER PUMP (MFP) TRIP HEADER PRESSURE SEPARATED. THIS CAUSED THE PRESSURE SWITCH, WHICH CLOSES CONTROL VALVES WHEN STOP VALVES ARE TRIPPED, TO INCORRECTLY SENSE AN MFP TRIP SIGNAL (STOP VALVES TRIPPED), WHICH THEN CAUSED THE MFP TURBINE STEAM CONTROL VALVES TO CLOSE ON AN ELECTRIC SIGNAL. THIS CAUSED THE 'A' MFP TURBINE TO COAST DOWN. THE ENSUING SHRINK IN STEAM GENERATOR (S/G) LEVEL CAUSED A REACTOR TRIP DUE TO S/G LO-LO LEVEL AT APPROXIMATELY 0648 CDT. AN AUXILIARY FEEDWATER ACTUATION, A S/G BLOWDOWN AND SAMPLE ISOLATION, A MAIN FEEDWATER ISOLATION, AND A MAIN TURBINE TRIP ALSO OCCURRED DURING THIS EVENT. THE UNIT WAS STABILIZED IN ACCORDANCE WITH CONTROL PROCEDURES. ALL REQUIRED SAFETY EQUIPMENT PERFORMED AS DESIGNED. THE ROOT CAUSE

OF THIS EVENT WAS DETERMINED TO BE COGNITIVE PERSONNEL ERROR BY CONTRACTOR OR VENDOR PERSONNEL DURING INITIAL INSTALLATION OF THE MFP. EXAMINATION OF THE FITTING INDICATED THAT THE FITTING WAS NOT PROPERLY TIGHTENED DURING INITIAL INSTALLATION.

[162] WOLF CREEK 1 DOCKET 50-482 LER 87-029
REQUIRED TESTING DELETED FROM SURVEILLANCE PROCEDURES.
EVENT DATE: 070187 REPORT DATE: 073187 NSSS: WE TYPE: PWR

(NSIC 205700) ON JULY 1, 1987, AT APPROXIMATELY 1500 CDT, DURING A REVIEW OF THE ANALOG CHANNEL OPERATIONAL TEST (ACOT) PROCEDURES FOR THE CONTAINMENT PURGE RADIATION MONITORS, GT RE-22 AND GT RE-33, IT WAS DISCOVERED THAT VERIFICATION OF AUTOMATIC PATHWAY ISOLATION WAS NOT BEING PERFORMED ON A QUARTERLY BASIS AS REQUIRED BY TECH SPEC 3.3.3.11. FOLLOWING THIS DISCOVERY, THE RADIATION MONITORS WERE DECLARED INOPERABLE AND THE APPROPRIATE TECH SPEC ACTION STATEMENT WAS ENTERED. THIS SURVEILLANCE WAS COMPLETED SATISFACTORILY AT APPROXIMATELY 1336 CDT ON JULY 2, 1987, THEREBY RESTORING THE RADIATION MONITORS TO OPERABLE STATUS. IT WAS DETERMINED THAT THIS VERIFICATION OF PATHWAY ISOLATION HAD BEEN DELETED FROM THE ACOT PROCEDURES IN OCTOBER OF 1985, AT LEAST PARTIALLY BECAUSE OF DIFFICULTIES EXPERIENCED IN SYSTEM LINEUP. A NEW PROCEDURE IS BEING DEVELOPED TO IMPLEMENT THIS SURVEILLANCE REQUIREMENT. IN ORDER TO PREVENT RECURRENCE, AN ENHANCEMENT IS BEING MADE TO THE SURVEILLANCE PROCEDURE CHANGE REVIEW PROCESS.

[163] WPPSS 2 DOCKET 50-397 LER 87-004 REV 01
UPDATE ON MISSED FIRE DOOR SURVEILLANCE AND IMPROPER IDENTIFICATION OF FIRE SEAL PENETRATIONS.
EVENT DATE: 040787 REPORT DATE: 071787 NSSS: GE TYPE: BWR

(NSIC 205470) AS THE RESULT OF PERFORMING A TECHNICAL SPECIFICATION SURVEILLANCE AND SUBSEQUENT REVIEW, DURING THE PERIOD OF APRIL 7-14, 1987, A PLANT SYSTEM ENGINEER DISCOVERED THAT 1) TWO FIRE SEAL PENETRATIONS WERE NOT SEALED AS REQUIRED BY THE PLANT TECHNICAL SPECIFICATIONS, AND 2) ONE FIRE DOOR HAD NOT BEEN INCLUDED IN TECHNICAL SPECIFICATION SURVEILLANCE PROCEDURES. THE TWO UNSEALED PENETRATIONS WERE IMMEDIATELY PLACED ON THE HOURLY FIRE TOUR AND MAINTENANCE WORK REQUESTS (MWRS) WERE WRITTEN TO HAVE THE PENETRATIONS SEALED. THE FIRE DOOR WAS INCORPORATED INTO PLANT SURVEILLANCE PROCEDURES AND VERIFIED TO BE LOCKED AND OPERABLE. THE CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE INADEQUATE COMMUNICATION BETWEEN TWO ARCHITECT ENGINEER (BURNS AND ROE, INC.) GROUPS DURING THE DESIGN AND CONSTRUCTION PHASE OF THE PLANT. THIS LED TO INCORRECT DESIGN BASIS INFORMATION BEING USED IN PROCEDURE PREPARATION. ACCORDINGLY, AN INDEPENDENT ENGINEERING REVIEW BY THE SUPPLY SYSTEM IS CURRENTLY BEING COMPLETED TO ENSURE THAT ALL TECHNICAL SPECIFICATION FIRE DOORS, BARRIERS AND PENETRATIONS ARE PROPERLY IDENTIFIED. AT THE COMPLETION OF THIS REVIEW, A SUPPLEMENTAL REPORT WILL BE SUBMITTED.

[164] WPPSS 2 DOCKET 50-397 LER 87-015
PLANT OPERATING MODE CHANGED WHILE DIVISION 2 DRYWELL PRESSURE MONITORING INSTRUMENTATION WAS INOPERABLE DUE TO INADEQUATE VALVE TAGGING.
EVENT DATE: 062087 REPORT DATE: 072087 NSSS: GE TYPE: BWR

(NSIC 205471) ON JUNE 20, 1987 DURING THE INITIAL WNP-2 PLANT STARTUP AFTER REFUELING OUTAGE NUMBER 2 (R2), AT 0130 HOURS, PLANT OPERATORS NOTED THAT DIVISION 2 DRYWELL PRESSURE MONITORING INSTRUMENTATION WAS NOT RESPONDING TO PLANT CONDITIONS. PLANT OPERATORS DISCOVERED THE INSTRUMENT RACK ISOLATION VALVES WERE CLOSED. THE PLANT SHIFT SUPPORT SUPERVISOR INITIATED A NONCONFORMANCE REPORT BUT FAILED TO HAVE PLANT INSTRUMENT AND CONTROL TECHNICIANS OPEN THE VALVES. PLANT OPERATORS BEGAN TO PERIODICALLY VENT THE DRYWELL AND THEY MAINTAINED DRYWELL PRESSURE NEAR ZERO PSIG WHICH MASKED THE CONTINUED

INOPERABILITY OF THE DIVISION 2 INSTRUMENTS. AT 0555 HOURS ON JUNE 21, 1987 PLANT OPERATORS MANUALLY SCRAMED THE PLANT. THIS WAS A PLANNED SCRAM FOR CONTROL ROD SCRAM TIMING. AT APPROXIMATELY 0700 ON JUNE 22, 1987 THE PLANT I&C SUPERVISOR WHILE INVESTIGATING THE EVENT OPENED THE ISOLATION VALVES AND RESTORED THESE INSTRUMENTS TO OPERATIONAL STATUS. THE ROOT CAUSE OF THE EVENT WAS INADEQUATE PLANT LABELING OF INSTRUMENTATION VALVES WHICH MET PLANT PROCEDURES DID NOT SPECIFICALLY IDENTIFY THE ISOLATION VALVES BY A TAG NUMBER. AN INSTRUMENT VALVE TAGGING PROGRAM WAS BEGUN IN THE FALL OF 1986 AND IS NOW COMPLETE. PLANT PROCEDURES ARE BEING REVISED TO INCLUDE SPECIFIC VALVE TAG NUMBERS. THERE IS NO SAFETY SIGNIFICANCE ASSOCIATED WITH THIS EVENT.

[165] WPPSS 2 DOCKET 50-397 LER 87-016
CONTROL ROOM EMERGENCY FILTRATION SYSTEM ACTUATION DUE TO FAILURE OF RELAY TO RESET DURING SURVEILLANCE - CAUSE UNKNOWN.
EVENT DATE: 062287 REPORT DATE: 072287 NSSS: GE TYPE: BWR

(NSIC 205539) ON JUNE 22, 1987 AT 1020 HOURS, THE CONTROL ROOM EMERGENCY FILTRATION SYSTEM AUTOMATICALLY INITIATED. THE INITIATION WAS THE RESULT OF THE FAILURE OF THE TRIP LOGIC TO RESET DURING THE PERFORMANCE OF A REACTOR BUILDING EXHAUST RADIATION MONITOR SURVEILLANCE PROCEDURE. THE TRIP LOGIC FAILED TO RESET WHEN PLANT PERSONNEL DEPRESSED THE RESET PUSHBUTTONS DURING THE PROCEDURE. THIS CAUSED A HALF-TRIP CONDITION IN EXISTENCE AND, WHEN PLANT PERSONNEL CONTINUED WITH THE PROCEDURE, ANOTHER HALF-TRIP OCCURRED WHICH RESULTED IN THE AUTOMATIC START OF THE EMERGENCY FILTRATION SYSTEM. AN INVESTIGATION DID NOT REVEAL THE CAUSE OF THE FAILURE OF THE TRIP LOGIC TO RESET. PLANT PERSONNEL ATTEMPTED TO REPRODUCE THE FAILURE BY RE-PERFORMING APPLICABLE SECTIONS OF THE PROCEDURE, BUT THE CONDITION COULD NOT BE DUPLICATED. THERE IS NO SAFETY SIGNIFICANCE ASSOCIATED WITH THIS EVENT AS THERE WAS NO ACTUAL INITIATING CONDITION AND ALL EQUIPMENT OPERATED CORRECTLY TO PLACE THE CONTROL ROOM VENTILATION SYSTEM IN AN ISOLATION CONDITION.

[166] WPPSS 2 DOCKET 50-397 LER 87-017
REACTOR WATER CLEANUP SYSTEM ISOLATION DUE TO DEMINERALIZER INFLUENT ISOLATION VALVE LEAKAGE.
EVENT DATE: 062387 REPORT DATE: 072387 NSSS: GE TYPE: BWR

(NSIC 205540) ON JUNE 23, 1987 AT 1342 HOURS, THE REACTOR WATER CLEANUP (RWCU) SYSTEM AUTOMATICALLY ISOLATED DUE TO HIGH DIFFERENTIAL FLOW. THE ISOLATION OCCURRED DURING THE PERFORMANCE OF AN RWCU DEMINERALIZER BACKWASH AND PRECOAT PROCEDURE. WHEN THE RADWASTE CONTROL ROOM OPERATOR DE-ISOLATED AN RWCU FILTER DEMINERALIZER AFTER PRECOATING, A FLOW UPSET CAUSED THE DIFFERENTIAL FLOW INDICATION TO GO OFF-SCALE HIGH CAUSING THE SYSTEM TO ISOLATE AND BOTH RWCU PUMPS TO TRIP. THE CAUSE OF THIS EVENT WAS LEAKAGE THROUGH AN INFLUENT ISOLATION VALVE FOR A FILTER DEMINERALIZER, WHICH SUBJECTED THE PRECOAT SYSTEM TO AN ABNORMALLY HIGH PRESSURE. THE CAUSE OF THE LEAKAGE WAS DETERMINED TO BE INCOMPLETE WORK INSTRUCTIONS ASSOCIATED WITH THE REWORK OF THE VALVE DURING THE LAST REFUELING OUTAGE. THIS VALVE IS ADEQUATELY UNIQUE AND THE GENERIC VALVE PROCEDURE USED DID NOT COMPLETELY ADDRESS THE MAINTENANCE OF SUCH. THERE IS NO SAFETY SIGNIFICANCE ASSOCIATED WITH THIS EVENT IN THAT THE RWCU SYSTEM ISOLATED AS DESIGNED. THIS EVENT POSED NO THREAT TO THE HEALTH AND SAFETY OF EITHER THE PUBLIC OR PLANT PERSONNEL.

[167] WPPSS 2 DOCKET 50-397 LER 87-018
REACTOR SCRAMS RESULTING FROM REACTOR PROTECTION SYSTEM ACTUATION DUE TO TURBINE CONTROL VALVE FAST CLOSURE.
EVENT DATE: 062687 REPORT DATE: 072787 NSSS: GE TYPE: BWR

(NSIC 205678) ON JUNE 26, 1987 AT 1541 HOURS, THE REACTOR AUTOMATICALLY SCRAMMED

BY REACTOR PROTECTION SYSTEM (RPS) ACTUATION DUE TO TURBINE CONTROL VALVE (TCV) FAST CLOSURE. THE TCV FAST CLOSURE WAS IN RESPONSE TO A UNIT LOCKOUT SIGNAL RESULTING FROM THE ACTUATION OF A SUDDEN PRESSURE RELAY ON NORMAL AUXILIARY POWER TRANSFORMER TR-N1. ON JUNE 27, 1987 AT 1813 HOURS, A SIMILAR EVENT OCCURRED INVOLVING THE SUDDEN PRESSURE RELAY ON NORMAL AUXILIARY POWER TRANSFORMER TR-N2. THE CAUSE OF BOTH SUDDEN PRESSURE RELAY ACTUATIONS WAS DETERMINED TO BE THE UNEXPECTED OPENING OF A TEST (POPPET) VALVE ASSOCIATED WITH THE RELAYS (THE VALVES ARE USED TO TEST THE RELAYS). WHEN THE VALVE OPENED, A DIFFERENTIAL PRESSURE DEVELOPED ACROSS THE BELLOWS OF THE SUDDEN PRESSURE RELAYS. THE RELAYS RESPONDED AS DESIGNED BY CAUSING A GENERATOR TRIP, A TURBINE TRIP AND, BECAUSE REACTOR POWER WAS ABOVE 30%, A REACTOR SCRAM. IN BOTH INSTANCES, PLANT POST-TRIP RESPONSE WAS NORMAL IN ALL ASPECTS. THERE IS NO SAFETY SIGNIFICANCE ASSOCIATED WITH EITHER EVENT IN THAT PLANT PROTECTION SYSTEMS AND ELECTRICAL DISTRIBUTION REALIGNMENTS FUNCTIONED AS DESIGNED . THESE EVENTS POSED NO THREAT TO THE HEALTH AND SAFETY OF EITHER THE PUBLIC OR PLANT PERSONNEL.

[168] WPPSS 2 DOCKET 50-397 LER 87-021
INADVERTENT NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM ISOLATION DURING PERFORMANCE OF SURVEILLANCE PROCEDURE-PERSONNEL ERROR.
EVENT DATE: 070387 REPORT DATE: 080387 NSSS: GE TYPE: BWR

(NSIC 205628) ON JULY 3, 1987 AT 2207 HOURS, AN INADVERTENT ISOLATION OF THE NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS - GROUP 1) OCCURRED DURING THE PERFORMANCE OF A MAIN STEAM LINE (MSL) LOW PRESSURE SURVEILLANCE PROCEDURE. THE ISOLATION WAS THE RESULT OF THE FAILURE OF PLANT INSTRUMENT AND CONTROL (I&C) TECHNICIANS TO HAVE PLANT OPERATORS RESET THE HALF-TRIP LOGIC PRIOR TO CONTINUING WITH THE PROCEDURE. WITH A HALF-TRIP CONDITION IN EXISTENCE, THE PROCEDURE WAS CONTINUED AND ANOTHER HALF-TRIP SIGNAL OCCURRED WHICH COMPLETED THE NSSSS, GROUP 1, FULL-TRIP LOGIC. THE RESULT WAS THE AUTOMATIC CLOSURE OF MAIN STEAM ISOLATION VALVES (MSIVS) AND MAIN STEAM LINE (MSL) DRAIN VALVES. THE ISOLATION LOGIC WAS RESET, THE SYSTEM RETURNED TO NORMAL LINEUP AND THE PROCEDURE SUCCESSFULLY COMPLETED. THE CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE PERSONNEL ERROR IN THAT THE I&C TECHNICIANS FAILED TO HAVE OPERATIONS RESET THE HALF-TRIP LOGIC AS REQUIRED BY THE PROCEDURE. THERE IS NO SAFETY SIGNIFICANCE ASSOCIATED WITH THIS EVENT IN THAT THERE WAS NO ACTUAL INITIATING CONDITION AND ALL EQUIPMENT OPERATED CORRECTLY TO PLACE THE PRIMARY CONTAINMENT (NSSSS - GROUP 1) IN AN ISOLATION CONDITION. THIS EVENT POSED NO THREAT TO THE HEALTH AND SAFETY OF EITHER THE PUBLIC OR PLANT PERSONNEL.

[169] YANKEE ROWE DOCKET 50-029 LER 87-013
HIGH RADIATION EXCLUSION AREA DOOR LEFT OPEN AND UNATTENDED.
EVENT DATE: 071087 REPORT DATE: 080587 NSSS: WE TYPE: PWR

(NSIC 205685) ON JULY 10, 1987, WITH THE PLANT IN MODE 1 AT 79% POWER, AN ACCESS DOOR TO A HIGH RADIATION EXCLUSION AREA WAS FOUND OPEN AND UNATTENDED (TECH SPEC 6.12). THE CONDITION EXISTED FROM APPROXIMATELY 1145 HOURS ON 7/10 TO APPROXIMATELY 1242 HOURS ON 7/10, AT WHICH TIME THE DOOR WAS CLOSED AND LOCKED. THE CONDITION WAS DISCOVERED BY PLANT MANAGEMENT DURING A ROUTINE TOUR OF THE PLANT. RADIATION LEVELS IN THE EXCLUSION AREA DURING THE INCIDENT WERE FROM 2 MR/HR TO 800 MR/HR GENERAL AREA AND 1 R/HR AT 1 FOOT FROM ONE COMPONENT. THE ROOT CAUSE OF THE EVENT WAS DETERMINED TO BE PERSONNEL ERROR. TWO INSTRUMENTATION AND CONTROL (I&C) TECHNICIANS LEFT THE AREA WITHOUT SECURING THE DOOR. THIS WAS CONTRARY TO APPROVED RADIATION PROTECTION (RP) PROCEDURES. THE TWO I&C TECHNICIANS RESPONSIBLE FOR THE INCIDENT WERE RESTRICTED FROM THE PLANT RADIATION CONTROL AREA UNTIL THEY COMPLETED AN RP RETRAINING CLASS. THE TRAINING WAS COMPLETED ON JULY 17. THIS IS THE FIRST REPORT OF THIS NATURE. THERE WAS NO ADVERSE EFFECT ON HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS OCCURRENCE.

[170] ZION 1 DOCKET 50-295 LER 87-013
CONTAINMENT HATCH INNER DOOR OPENED WITH OUTER DOOR INOPERABLE DUE TO COMPONENT FAILURE.
EVENT DATE: 062187 REPORT DATE: 072187 NSSS: WE TYPE: PWR
VENDOR: CHICAGO BRIDGE AND IRON COMPANY
MCGILL MANUFACTURING CO., INC.

(NSIC 205431) ON 6/21/87 AT 0015 HOURS, WITH THE UNIT AT 92% POWER, THE OUTER DOOR ON THE CONTAINMENT PERSONNEL HATCH MALFUNCTIONED, WITH PERSONNEL IN THE HATCH. THE CAUSE WAS AN OUT OF ADJUSTMENT CAM AND FOLLOWER MECHANISM. THE DOOR COULD BE CLOSED BUT NOT LATCHED SHUT, AND COULD BE OPENED ABOUT SIX INCHES. THE LICENSED SHIFT SUPERVISOR WAS NOTIFIED, BUT COULD NOT LOCATE MECHANICS TO ASSIST THE PERSONNEL. OUT OF CONCERN FOR THEIR SAFETY, THE CREW WAS PERMITTED TO ENTER CONTAINMENT AND EXIT VIA THE EMERGENCY HATCH, CONTRARY TO PLANT TECH SPECS WHICH REQUIRE ONE DOOR KEPT CLOSED WITH THE OTHER DOOR INOPERABLE. THE HATCH INNER DOOR WAS OPEN LESS THAN ONE MINUTE AND WAS VERIFIED CLOSED AND LATCHED AFTER THE EVENT. WHILE THE DOOR WAS OPEN, A MAN OUTSIDE CONTAINMENT HELD THE OUTER DOOR SHUT. THE DOORS ARE DESIGNED SO THAT THE PRESSURE DIFFERENCE BETWEEN CONTAINMENT AND THE AUXILIARY BUILDING HOLDS THE HATCH DOORS SHUT. SAFETY EFFECT WAS MINIMIZED BY THIS DESIGN AND BY THE SHORT TIME THE DOOR WAS OPEN. SELECTED PERSONNEL ON ALL SHIFTS WILL BE TRAINED ON HOW TO OPEN THE HATCH DOOR WITH THE MECHANISM JAMMED, AND TOOLS WILL BE PLACED IN THE HATCH.

[171] ZION 2 DOCKET 50-304 LER 87-003
AUTOSTART OF REACTOR CONTAINMENT FAN COOLER DUE TO PERSONNEL ERROR.
EVENT DATE: 070787 REPORT DATE: 080687 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 205687) ON JULY 7, 1987 AT 1612 HOURS, WITH THE UNIT IN COLD SHUTDOWN, THE 2A REACTOR CONTAINMENT FAN COOLER (RCFC) AUTOSTARTED IN LOW SPEED DURING A TEST OF THE SECONDARY UNDERVOLTAGE PROTECTION RELAYS. THE CAUSE WAS PERSONNEL ERROR. PER THE TEST PROCEDURE, THE CIRCUIT SHOULD HAVE BEEN DEENERGIZED TO PREVENT EQUIPMENT ACTUATION. THE BREAKERS, WHICH ARE NORMALLY ON, HAD BEEN LEFT IN THE OFF POSITION EARLIER THAT DAY DUE TO MAINTENANCE WORK. THE PERSONNEL PERFORMING THE TEST ASSUMED THE BREAKERS WERE IN THEIR NORMAL "ON" POSITION AND TOGGLED THEM WITHOUT FIRST VERIFYING INITIAL POSITION. A SECOND CAUSE WAS THE FACT THAT THE ON/OFF LABELS FOR THE SAFE SHUTDOWN AND SAFETY INJECTION SEQUENCE TIMERS HAD BEEN PAINTED OVER IN RED. THERE WAS NO SAFETY SIGNIFICANCE.

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.

SYSTEM INDEX

SYSTEM INDEX

KEYWORD INDEX

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

KEYWORD INDEX

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VENDOR CODE INDEX

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11a. TYPE OF REPORT
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12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one month period identified on the cover of the document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting for revisions to those events occurring prior to 1984 are described in NRC Regulatory Guide 1.16 and NUREG-1061, 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.

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