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For month of October 1989

Oak Ridge National Laboratory

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Oak Ridge National Laboratory
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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-0161, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports. For those events occurring on and after January 1, 1984, LERs are being submitted in accordance with the revised rule contained in Title 10 Part 50.73 of the Code of Federal Regulations (10 CFR 50.73 - Licensee Event Report System) which was published in the Federal Register (Vol. 48, No. 144) on July 26, 1983. NUREG-1022, Licensee Event Report System - Description of Systems and Guidelines for Reporting, provides supporting guidance and information on the revised LER rule.

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

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[1] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 89-009 REV 01
UPDATE ON INADVERTENT SIGNALS FROM RADIATION AND CHLORINE DETECTORS DUE TO VARIOUS CAUSES RESULT IN AUTOMATIC ACTUATIONS OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM.
EVENT DATE: 022489 REPORT DATE: 083189 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 2 (PWR)

(NSIC 215160) BETWEEN FEBRUARY AND SEPTEMBER 1989, THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) HAS UNEXPECTEDLY AUTOMATICALLY STARTED THIRTY-FIVE TIMES AS A RESULT OF INVALID INITIATION SIGNALS FROM RADIATION MONITORS OR CHLORINE CONCENTRATION DETECTORS. FOLLOWING EACH START OF THE CREVS, THE RADIATION MONITOR OR CHLORINE DETECTOR WAS RESET AND THE CREVS WAS RETURNED TO STANDBY. DURING THESE EVENTS, THE CREVS ACTUATED AS DESIGNED AND NO ACTUAL HIGH RADIATION OR HIGH CHLORINE CONCENTRATION CONDITIONS EXISTED. THE CREVS FOR THE COMBINED ANO-1 AND ANO-2 CONTROL ROOMS CONSISTS OF TWO REDUNDANT FILTER TRAINS. THE ROOT CAUSE FOR THESE ACTUATIONS HAS BEEN DETERMINED TO BE DIRECTLY RELATED TO DEFICIENCIES ASSOCIATED WITH THE SYSTEM DESIGN. AS A RESULT OF THE EVENTS, SEVERAL CORRECTIVE ACTIONS HAVE BEEN TAKEN AND HAVE REDUCED THE FREQUENCY OF SYSTEM ACTUATIONS. THESE ACTIONS INCLUDE PROHIBITING THE USE OF HAND HELD RADIOS IN THE VICINITY OF THE CHLORINE MONITORS, INSTALLATION OF SUPPRESSION DIODES FOR THE OPERATION OF THE NORMAL CONTROL ROOM DAMPERS, AND REPLACEMENT OF THE DETECTOR AND TUBE HOLDER FOR A RADIATION MONITOR. AN ENGINEERING EVALUATION OF THE SYSTEM DESIGN HAS BEEN INITIATED TO DETERMINE IF ADDITIONAL ACTIONS ARE WARRANTED.

[2] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 89-027
FAILURE DUE TO A PERSONNEL ERROR TO MEET A TECH SPEC REQUIREMENT TO DEMONSTRATE OPERABILITY OF THE REDUNDANT VALVE PRIOR TO INITIATING MAINTENANCE ON A LOW PRESSURE INJECTION SYSTEM VALVE.
EVENT DATE: 082189 REPORT DATE: 092089 NSSS: BW TYPE: PWR

(NSIC 215255) ON 8/21/89, AT 0845 HOURS, THE DECAY HEAT REMOVAL/LOW PRESSURE INJECTION (DHR/LPI) 'B' TRAIN INJECTION VALVE CV-1400 WAS REMOVED FROM SERVICE TO PERFORM PREVENTIVE MAINTENANCE. SINCE CV-1400 IS AN LPI COMPONENT, TECH SPEC 3.3.5 REQUIRES THAT OPERABILITY OF THE REDUNDANT 'A', TRAIN VALVE CV-1401 MUST BE DEMONSTRATED PRIOR TO INITIATING MAINTENANCE ON CV-1400. CONTRARY TO TECH SPEC 3.3.5, THE SHIFT OPERATIONS SUPERVISOR (SOS) AUTHORIZED REMOVING CV-1400 FROM SERVICE WITHOUT PERFORMING A VALVE STROKE TEST ON CV-1401. AT 1400 HOURS, CV-1400 WAS RETURNED TO SERVICE AND THE SOS SUBSEQUENTLY REALIZED THAT CV-1401 HAD NOT BEEN TESTED. THE ROOT CAUSE OF THE EVENT WAS PERSONNEL ERROR. A CONTRIBUTING FACTOR WAS AN INOPERABLE EMERGENCY DIESEL GENERATOR (EDG), WHICH CAUSED THE SOS TO FOCUS ON ENSURING THAT REMOVING CV-1400 DID NOT VIOLATE THE TECH SPECS' LIMITATIONS ON OPERATING WITH AN INOPERABLE EDG. AS CV-1401 HAD BEEN DEMONSTRATED TO BE OPERABLE ON 8/8/89, AND ITS STATUS HAD BEEN VERIFIED APPROXIMATELY ONE HOUR PRIOR TO INITIATING MAINTENANCE ON CV 1400, THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL. THE SOS HAS BEEN COUNSELLED REGARDING THIS EVENT, AND A MEMORANDUM WAS SENT TO THE UNIT ONE LICENSED OPERATORS DESCRIBING THE EVENT AND EMPHASIZING COMPLIANCE WITH TECH SPEC 3.3.5.

[3] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 89-030
INADEQUATE REVIEW OF THE CODE OF FIRE BARRIERS WHERE THE POTENTIAL EXISTED FOR THE SPREAD OF COMBUSTIBLE LIQUIDS RESULTED IN NONFUNCTIONAL FIRE BARRIERS BETWEEN THE EMERGENCY DIESEL GENERATOR ROOMS.
EVENT DATE: 082489 REPORT DATE: 092289 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 2 (PWR)

(NSIC 215306) ON 8/25/89, DURING A SPECIAL NRC EVALUATION, IT WAS IDENTIFIED THAT THE FIRE BARRIER BETWEEN THE ADJACENT EMERGENCY DIESEL GENERATOR (EDG) ROOMS IN ARKANSAS NUCLEAR ONE, UNIT ONE (ANO-1) DID NOT MEET THE NATIONAL FIRE PROTECTION ASSOCIATION (NFPA) CODE REQUIREMENTS RELATED TO DESIGN PROVISIONS FOR PROTECTION

AGAINST THE SPREAD OF FLAMMABLE AND COMBUSTIBLE LIQUIDS BETWEEN PLANT AREAS. BASED UPON THIS, THE FIRE BARRIER BETWEEN THE EDG ROOMS IN ANO, UNIT 2 (ANO-2) WAS INSPECTED AND ALSO FOUND TO BE DEFICIENT. AFTER ENGINEERING EVALUATION OF THE CONDITIONS, THE BARRIERS WERE DECLARED NONFUNCTIONAL. IN BOTH ANO-1 AND ANO-2 EDG ROOMS, ALTHOUGH THE POTENTIAL EXISTED FOR THE SPREAD OF COMBUSTIBLE LIQUIDS TO THE ADJACENT ROOM, FIRE DETECTION INSTRUMENTATION, SUPPRESSION SYSTEMS AND FIRE BRIGADE PERSONNEL ARE AVAILABLE TO MITIGATE THE SPREAD OF A FIRE IF IT WERE TO OCCUR. THE DESIGN OF THE PENETRATION FIRE BARRIERS, WHICH HAD EXISTED SINCE INITIAL PLANT CONSTRUCTION ON EACH UNIT, HAD NOT BEEN ADEQUATELY EVALUATED INITIALLY TO ENSURE COMPLIANCE WITH THE NFPA CODE. AS A RESULT OF THESE FINDINGS FIRE PROTECTION PERSONNEL EVALUATED OTHER AREAS OF THE FACILITY WHERE COMBUSTIBLE LIQUID SYSTEMS ARE LOCATED TO ENSURE THE NFPA CODE WAS MET.

[4] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 89-031
UNEXPECTED AUTOMATIC ACTUATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM DUE TO AN UNKNOWN CAUSE.
EVENT DATE: 082989 REPORT DATE: 092889 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 2 (PWR)

(NSIC 215362) ON 8/29/89, AT APPROX. 1131, AN UNEXPECTED CONTROL ROOM ISOLATION AND ACTUATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) OCCURRED. THE INITIATING LOGIC FOR THE CREVS IS DESIGNED SUCH THAT A SIGNAL FROM ANY OF FOUR CHLORINE DETECTORS OR TWO RADIATION MONITORS WILL ISOLATE NORMAL VENTILATION AND ACTUATE THE CREVS FOR BOTH UNITS. HOWEVER, NONE OF THE INSTRUMENTATION WHICH INITIATES ACTUATION OF THE CREVS WAS TRIPPED, INDICATING THAT THEY HAD NOT INITIATED THE EVENT. ALSO, ONLY THE ANO-1 PORTION OF THE SYSTEM ACTUATED. AFTER AN INVESTIGATION, WHICH DETERMINED THAT THE ACTUATION WAS SPURIOUS, THE CONTROL ROOM VENTILATION LINEUP WAS RETURNED TO NORMAL. SINCE NONE OF THE INSTRUMENTS THAT ACTUATE THE CREVS WERE TRIPPED, THE CAUSE OF THIS EVENT COULD NOT BE IMMEDIATELY DETERMINED. HOWEVER, FURTHER INVESTIGATION INTO ITS CAUSE WILL BE CONDUCTED. ADDITIONALLY, A MONTHLY FUNCTIONAL TEST DEMONSTRATING PROPER SYSTEM OPERATION, WAS COMPLETED SUBSEQUENT TO THIS EVENT. ALSO, AS A RESULT OF NUMEROUS PREVIOUS INADVERTENT ACTUATIONS OF THE CREVS, AN ENGINEERING EVALUATION OF THE SYSTEM DESIGN IS IN PROGRESS TO DETERMINE THE APPROPRIATE CORRECTIVE ACTIONS REQUIRED TO REDUCE THE OCCURRENCE OF FUTURE INADVERTENT ACTUATIONS. SINCE NO ACTUAL HIGH CHLORINE CONCENTRATION OR HIGH RADIATION CONDITION EXISTED, THERE WAS NO ACTUAL SAFETY SIGNIFICANCE RELATED TO THIS EVENT.

[5] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 89-014
FAILURE TO RECOGNIZE MAXIMUM EXPECTED TEMPERATURES RESULTED IN OPERATIONS OUTSIDE THE DESIGN BASIS OF THE PLANT.
EVENT DATE: 071087 REPORT DATE: 090789 NSSS: CE TYPE: PWR

(NSIC 215215) ON 7/10/87, WHILE IN AN UNSCHEDULED MAINTENANCE OUTAGE, IT WAS DISCOVERED DURING AN ENGINEERING REVIEW ASSOCIATED WITH A DESIGN CHANGE, THAT THE ORIGINAL PIPING ANALYSIS PERFORMED BY THE ARCHITECT ENGINEERS QUALIFIED PORTIONS OF THE REACTOR COOLANT MAKEUP SYSTEM PIPING AND PORTIONS OF THE AUXILIARY SPRAY PIPING TO DESIGN TEMPERATURES OF 400F AND 120 DEGREES, RESPECTIVELY. UNDER CERTAIN PLANT CONDITIONS THE HIGHEST EXPECTED TEMPERATURE TO WHICH EITHER OF THESE SECTIONS OF PIPING MIGHT BE EXPOSED IS 479 DEGREES. UPON DISCOVERY OF THE DESIGN DEFICIENCY, A VISUAL INSPECTION OF THE PIPING AND SUPPORTS WAS PERFORMED AND NO STRUCTURAL DAMAGE TO EITHER WAS IDENTIFIED. ADDITIONALLY, ADMINISTRATIVE CONTROLS WERE ESTABLISHED WHICH DIRECTED OPERATIONS PERSONNEL TO CONTACT ENGINEERING WHEN THE SPRAY PIPING WAS EXPOSED TO TEMPERATURES GREATER THAN DESIGN. THESE CONTROLS, HOWEVER, DID NOT PROHIBIT THE USE OF THE SPRAY LINE AT TEMPERATURES GREATER THAN 120 DEGREES, NOR, WAS AN ENGINEERING EVALUATION PERFORMED OF THE CONSEQUENCES ASSOCIATED WITH THE USE OF THE LINE AT TEMPERATURES GREATER THAN DESIGN. DURING REFUELING OUTAGE, 2R6 (FEBRUARY THROUGH MAY 1988)

MODIFICATIONS WERE MADE TO PIPING SUPPORTS ON BOTH SYSTEMS TO ENSURE THE SUPPORTS WERE QUALIFIED FOR THE HIGHER EXPECTED TEMPERATURES.

[6] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 89-015
SINGLE FAILURE CRITERION OF THE HYDROGEN MONITORING SYSTEM COMPROMISED BY A DESIGN CHANGE THAT RESULTED FROM AN INADEQUATE DESIGN REVIEW PROCESS.
EVENT DATE: 080989 REPORT DATE: 090889 NSSS: CE TYPE: PWR

(NSIC 215216) ON 8/9/89, DESIGN ENGINEERING PERSONNEL IDENTIFIED A DISCREPANCY RELATED TO THE SINGLE FAILURE DESIGN CRITERION OF THE CONTAINMENT HYDROGEN ANALYZERS. THE TWO REDUNDANT HYDROGEN ANALYZER CHANNELS CONTAIN INDEPENDENT SUCTION AND SAMPLE RETURN LINES TO CONTAINMENT. THE MOTOR OPERATED ISOLATION VALVES INSIDE CONTAINMENT ON THE SAMPLE RETURN LINES FOR BOTH CHANNELS ARE POWERED FROM THE SAME (RED) ENGINEERED SAFETY FEATURE (ESF) BUS. IF A LOSS OF COOLANT ACCIDENT WERE TO OCCUR, THESE CONTAINMENT ISOLATION VALVES WOULD CLOSE AUTOMATICALLY. A LOSS OF THE "RED" ESF BUS SUBSEQUENT TO THE CONTAINMENT ISOLATION, BUT PRIOR TO THE INITIATION OF SAMPLING WOULD RESULT IN BOTH HYDROGEN ANALYZERS BEING RENDERED INOPERABLE SINCE THE VALVES WOULD BE CLOSED AND WITHOUT POWER. ORIGINALLY, THE SUBJECT VALVES WERE CHECK VALVES. THIS CONDITION RESULTED FROM THE INADEQUATE DESIGN REVIEW PROCESS WHICH WAS IN PLACE AT THE TIME OF THE MODIFICATION. PROVISIONS WERE MADE TO ENSURE THAT ALTERNATE POWER CAN BE ESTABLISHED TO THE APPROPRIATE ISOLATION VALVE WITHIN THREE HOURS OF THE OCCURRENCE OF THE POSTULATED EVENT. ADDITIONALLY, EVALUATIONS ARE BEING CONDUCTED IN ORDER TO DEVELOP PLANS TO ALLEVIATE THE SINGLE FAILURE DESIGN DEVIATION. RECENT ENHANCEMENTS TO THE DESIGN CHANGE REVIEW PROCESS SHOULD AID IN PREVENTING OCCURRENCE OF SIMILAR EVENTS.

[7] ARNOLD DOCKET 50-331 LER 89-011
TURBINE CONTROL VALVE FAST CLOSURE TRIP RESULTS IN REACTOR SCRAM WHILE PERFORMING TESTING.
EVENT DATE: 082689 REPORT DATE: 092589 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215366) ON 8/26/89, AT 1642 HOURS, WITH THE PLANT OPERATING AT 100% POWER, OPERATIONS PROCEDURE, "POWER/LOAD UNBALANCE AND RELAY CIRCUITS TEST" WAS IN PROGRESS. THIS TEST IS PERFORMED FOR CONTINUED RELIABLE OPERATION OF THE MAIN TURBINE. CONTRARY TO WHAT WAS EXPECTED, A TRIP OF THE MAIN TURBINE CONTROL VALVES AND SUBSEQUENT REACTOR SCRAM OCCURRED AT 1643 HOURS. A PRELIMINARY ROOT CAUSE INVESTIGATION INDICATES THAT AN INACCURATE EVALUATION OF A PREVIOUSLY IDENTIFIED FAILED GENERATOR CURRENT TRANSFORMER CONTRIBUTED TO THE POWER/LOAD UNBALANCE CIRCUITRY PERCEIVING AN UNBALANCE CONDITION AND TRIPPING THE TURBINE CONTROL VALVES. A FINAL ROOT CAUSE DETERMINATION WILL BE SUBMITTED IN A SUBSEQUENT REPORT FOLLOWING FURTHER INVESTIGATION. APPROXIMATELY FIVE MINUTES FOLLOWING THE SCRAM, PROBLEMS WERE ENCOUNTERED ON THE "B" ESSENTIAL AND NON-ESSENTIAL BUSSES. SUBSEQUENT INVESTIGATION REVEALED A FAILED TRIP COIL ON AN ASSOCIATED BREAKER. THE PLANT WAS BROUGHT TO A NORMAL, SAFE SHUTDOWN CONDITION AND THE APPROPRIATE NOTIFICATIONS WERE MADE. THERE WAS NO EFFECT ON THE SAFE OPERATION OF THE PLANT.

[8] BEAVER VALLEY 2 DOCKET 50-412 LER 89-024
STEAM GENERATOR BLOWDOWN CONTAINMENT ISOLATION VALVES CLOSURE DUE TO ESF ACTUATION.
EVENT DATE: 081589 REPORT DATE: 091489 NSSS: WE TYPE: PWR

(NSIC 215280) ON 8/15/89 AT 0930 HOURS, WITH THE UNIT IN POWER OPERATION AT 100% REACTOR POWER, A SHIFT CHEMIST WAS MAKING AN ADJUSTMENT TO THE "A" STEAM GENERATOR (SG) BLOWDOWN SAMPLE FLOW TO THE SG BLOWDOWN SAMPLE RADIATION MONITOR (2SSR-RQI100). SAMPLE FLOW FROM THE "A" SG WAS OBSERVED TO BE LOWER THAN NORMAL

AND THE CHEMIST WAS INCREASING FLOW. DURING THIS SAMPLE FLOW INCREASE, CRUD PARTICLES IN THE SAMPLE FLOWPATH WERE SWEPT INTO THE SG BLOWDOWN RADIATION MONITOR CAUSING A MOMENTARY HIGH RADIATION ALARM. THE HIGH RADIATION ALARM CAUSED THE SG BLOWDOWN CONTAINMENT ISOLATION VALVES (2BDG-AOV100A1, B1, C1) TO AUTOMATICALLY CLOSE. THE CLOSURE OF THESE VALVES IS REPORTABLE AS AN ENGINEERED SAFETY FEATURES ACTUATION UNDER 10CFR50.72.B.2.II. THE CAUSE OF THIS EVENT WAS ATTRIBUTED TO THE FLOW ADJUSTMENT DISLODGING THE RESIDUAL ACTIVITY IN THE SAMPLE PATHWAY COMBINED WITH A LOW SETPOINT (LESS THAN 10% OF THE TECH SPEC LIMIT) ON THE SG BLOWDOWN RADIATION MONITOR. 2SSR-RQI100 WAS FLUSHED AND THE SG BLOWDOWN SYSTEM WAS RESTORED TO NORMAL SYSTEM ARRANGEMENT AT 0943 HOURS AFTER SAMPLING VERIFIED BLOWDOWN ACTIVITY TO BE NORMAL. THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC AS A RESULT OF THIS EVENT BECAUSE THE RADIATION MONITOR ACTUATED AS DESIGNED TO ISOLATE THE BLOWDOWN SYSTEM IN THE EVENT OF ANY STEAM GENERATOR TUBE LEAKAGE.

[9] BIG ROCK POINT DOCKET 50-155 LER 89-007
OMISSION OF FIRE DETECTOR TESTING DUE TO LACK OF UNDERSTANDING OF PROCEDURAL REQUIREMENTS.
EVENT DATE: 081689 REPORT DATE: 091589 NSSS: GE TYPE: BWR

(NSIC 215230) BIG ROCK POINT TECH SPEC 4.3.3.8.1A REQUIRES THAT THE FIRE DETECTION INSTRUMENTS LISTED IN TABLE 3.3-8 OF THE BIG ROCK POINT TECH SPECS ARE TO BE DEMONSTRATED OPERABLE ONCE PER SIX (6) MONTHS. CONTRARY TO THIS REQUIREMENT, DURING PERFORMANCE OF A QA FIRE PROTECTION AUDIT ON 8/16/89, IT WAS DISCOVERED THAT DETECTORS IN THE EMERGENCY DIESEL GENERATOR ROOM AND THE SCREENWELL AND PUMP HOUSE AREA WERE NOT TESTED WITHIN THE SIX (6) MONTH INTERVAL. UPON DISCOVERY, THE DETECTORS WERE DECLARED INOPERABLE AND A FIRE WATCH/PATROL WAS ESTABLISHED PER THE REQUIREMENTS OF TECH SPEC 3.3.3.8 ACTIONS A. AND B. THE DETECTORS WERE TESTED AND DECLARED OPERABLE LATER ON 8/16/89 AND THE FIRE WATCH/PATROL WAS TERMINATED. CAUSE OF THE DEFICIENCIES WAS ATTRIBUTED TO THE LACK OF UNDERSTANDING OF PROCEDURAL REQUIREMENTS INVOLVING FIRE DETECTOR TESTING.

[10] BIG ROCK POINT DOCKET 50-155 LER 89-008
REACTOR TRIP RESULTING FROM TURBINE CONTROL FAILURE.
EVENT DATE: 082289 REPORT DATE: 092189 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215298) ON 8/22/89 THE UNIT WAS OPERATING AT APPROXIMATELY 74% POWER FOLLOWING START-UP FROM THE REFUELING OUTAGE A WEEK EARLIER. AT 0635 HOURS, OPERATORS NOTICED A SLOW REACTOR PRESSURE INCREASE AND ADJUSTED THE INITIAL PRESSURE REGULATOR TO RESTORE PRESSURE TO NORMAL. AT 0645 HOURS, REACTOR PRESSURE INCREASED RAPIDLY CAUSING A REACTOR TRIP ON HIGH NEUTRON FLUX. ALL CONTROL RODS INSERTED AND PLANT COOLING WAS MAINTAINED WITH THE MAIN CONDENSER. NO ENGINEERED SAFETY SYSTEMS OTHER THAN THE REACTOR PROTECTION SYSTEM WERE ACTUATED DURING THE EVENT. CAUSE OF THE TRIP WAS A RAPID CLOSURE OF THE TURBINE ADMISSION VALVES DUE TO GROSS LEAKAGE IN THE TURBINE INITIAL PRESSURE REGULATOR BELLOWS ASSEMBLY. AFTER REPAIR OF THE BELLOWS ASSEMBLY AND SUCCESSFUL TESTING, THE PLANT WAS RESTARTED ON 8/22/89 AT 2211 HOURS.

[11] BRAIDWOOD 1 DOCKET 50-456 LER 89-007
CONTAINMENT VENTILATION ISOLATION ACTUATION SIGNAL DUE TO FAILED HIGH VOLTAGE POWER SUPPLY IN CONTAINMENT BUILDING FUEL HANDLING INCIDENT RADIATION MONITOR.
EVENT DATE: 081089 REPORT DATE: 090189 NSSS: WE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 215219) AT 0253 ON 9/10/89, THE CONTAINMENT BLDG. FUEL HANDLING INCIDENT AREA RADIATION MONITOR 1RT-AR011 WENT INTO ALERT ALARM AND INTERLOCK ACTUATION DUE TO A LOSS OF PULSES. THIS INITIATED A TRAIN A CONTAINMENT VENTILATION

ISOLATION SIGNAL. NO COMPONENTS REPOSITIONED AS A RESULT OF THIS SIGNAL. THE ASSOCIATED CONTAINMENT ISOLATION VALVES WERE ALREADY CLOSED. INVESTIGATION REVEALED THAT THE HIGH VOLTAGE POWER SUPPLY IN 1RT-AR011 HAD NO OUTPUT. THE ROOT CAUSE OF THIS EVENT WAS A FAILED HIGH VOLTAGE POWER SUPPLY IN 1RT-AR011. THE IMMEDIATE CORRECTIVE ACTIONS WERE TO REPLACE THE HIGH VOLTAGE POWER SUPPLY IN 1RT-AR011 AND VERIFY OPERABILITY. POWER SUPPLIES FOR RADIATION DETECTORS 1RE-AR011, 1RE-AR012, 2RE-AR011, AND 2RE-AR012 ARE CHECKED DURING NORMALLY SCHEDULED DETECTOR REPLACEMENT. THERE HAVE BEEN TWO PREVIOUS OCCURRENCES OF CONTAINMENT VENTILATION ISOLATION SIGNAL DUE TO A LOSS OF PULSES FROM A RADIATION MONITOR. ONE EVENT WAS DUE TO CONSTRUCTION ACTIVITY WHICH DAMAGED THE DETECTOR, THE OTHER EVENT WAS DUE TO A VERY LOW BACKGROUND ACTIVITY PRIOR TO UNIT 2 INITIAL CRITICALITY. PREVIOUS CORRECTIVE ACTIONS ARE NOT APPLICABLE TO THIS EVENT.

[12]   BRAIDWOOD 1   DOCKET 50-456   LER 89-009
FAILURE OF MAIN STEAMLINE SAFETY VALVE TO RESEAT DUE TO A VALVE DESIGN DEFICIENCY.
EVENT DATE: 090289   REPORT DATE: 100289   NSSS: WE   TYPE: PWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV

(NSIC 215389) AT 0014 ON 9/2/89 THE UNIT 1 TURBINE WAS TAKEN OFF LINE AS PART OF A SCHEDULED PLANT SHUTDOWN. A 'LIFT' TEST OF MAIN STEAMLINE SAFETY VALVE, 1MS017C, WAS IN PROGRESS. AT 0100 A FOURTH LIFT WAS PERFORMED. THE 1MS017C DID NOT COMPLETELY RESEAT. THE 1C MAIN STEAMLINE WAS SAMPLED TO AID IN DETERMINING THE STATUS OF THE 1C STEAM GENERATOR (SG) TUBES. AT 0142 1MS017C FULLY RESEATED AT A MAIN STEAMLINE HEADER PRESSURE OF APPROXIMATELY 1030 PSIG. THIS WAS 28 PSIG BELOW THE MINIMUM RESET PRESSURE OF 1058 PSIG. RCS AVERAGE TEMPERATURE WAS 5 DEGREES BELOW NORMAL AT 552F. STABLE PLANT CONDITIONS WERE IMMEDIATELY ESTABLISHED. AT 0353 SAMPLE RESULTS CONFIRMED THERE WAS NO DETECTABLE RADIOACTIVITY IN THE 1C MAIN STEAMLINE. THE ROOT CAUSE OF THIS EVENT WAS A PRESERVICE ERROR IN THE DESIGN OF THE VALVE. THIS DEFICIENCY WAS THE ABILITY OF THE GUIDE TO ROTATE, WHICH CAN OCCUR AS A RESULT OF VIBRATION OR FLOW DURING NORMAL SERVICE. THE SETTING OF THE BLOWDOWN CONTROL UPPER ADJUSTING RING WAS FOUND APPROXIMATELY .392 INCHES LOWER THAN THE RECOMMENDED SETTING. AN EVALUATION HAS CONCLUDED THAT THE RING DRIFTED OUT OF ADJUSTMENT DURING NORMAL OPERATION. THE 1MS017C WAS VERIFIED TO BE FULLY CLOSED AT 0142. A VENDOR RECOMMENDED UPGRADE WILL BE INSTALLED ON 1MS017C AND TWO OTHER SAFETIES. THE REMAINING MAIN STEAMLINE SAFETY VALVES WILL BE CHECKED. THERE HAVE BEEN NO PREVIOUS OCCURRENCES.

[13]   BROWNS FERRY 1   DOCKET 50-259   LER 89-015 REV 01
UPDATE ON MOMENTARY LOSS OF SECONDARY CONTAINMENT CAUSED BY FAILURE OF WELDS ON THE DOOR LOCK MECHANISM.
EVENT DATE: 062789   REPORT DATE: 092289   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 215234) AT 0845 ON 6/27/89, WHILE A GROUP OF SIX MODIFICATIONS PERSONNEL WERE LEAVING THE REACTOR BUILDING FROM THE REFUEL FLOOR TO THE CONTROL BUILDING ROOF, BOTH DOORS OF THE UNIT 1 AIRLOCK WERE OPENED SIMULTANEOUSLY FOR APPROXIMATELY FIVE (5) SECONDS. THIS CONDITION CONSTITUTED A BREACH OF SECONDARY CONTAINMENT. THIS EVENT WAS CAUSED BY THE FAILURE OF TWO WELDS WHICH ATTACH THE BRACKET THAT HOLDS THE LOCKSET IN THE DOOR ON THE REFUEL FLOOR SIDE OF THE AIRLOCK. THESE WELDS WERE BROKEN LOOSE DUE TO HEAVY USE OF THE AIRLOCK DOORS DURING THIS OUTAGE. THE LOCKSET BRACKET WAS REATTACHED AND OPERATION OF THE DOOR VERIFIED. SIGNS HAVE BEEN PLACED AT AIRLOCKS ABOUT PROPER USAGE. FUTURE MAINTENANCE REQUESTS WHICH AFFECT OPERABILITY OF SECONDARY CONTAINMENT DOORS WILL BE SCHEDULED FOR IMMEDIATE WORK. THE MAINTENANCE REQUEST PROCEDURE WILL ALSO BE REVISED TO ALLOW APPROPRIATE PRIORITIES TO BE PLACED ON MAINTENANCE REQUESTS UNDER ALL PLANT CONDITIONS. PERFORMANCE OF PREVENTIVE MAINTENANCE FOR SECONDARY CONTAINMENT AIRLOCK DOORS HAS BEEN INCREASED IN FREQUENCY. THE FEASIBILITY OF A

CHANGE TO THE CURRENT INTERLOCK SYSTEM DESIGN WILL BE EVALUATED. DURING THIS EVENT, UNIT 2 WAS IN COLD SHUTDOWN WITH IRRADIATED FUEL IN THE VESSEL. THE VESSEL WAS IN THE FLOODED UP CONDITION WITH THE FUEL POOL GATES INSTALLED. UNITS 1 AND 3 WERE DEFUELED.

[14] BROWNS FERRY 1 DOCKET 50-259 LER 89-022
INADVERTENT INSERTION OF RADIATION DETECTOR CHECKSOURCE RESULTS IN UNPLANNED ACTUATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM.
EVENT DATE: 081089 REPORT DATE: 090889 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 215235) ON 8/10/89 AT APPROXIMATELY 1000, TRAIN A OF THE CONTROL ROOM EMERGENCY VENTILATION (CREV) SYSTEM UNEXPECTEDLY ACTUATED DUE TO A SIGNAL FROM RADIATION MONITOR 3-RM-90-259B. TRAIN B OF THE CREV SYSTEM WAS OUT OF SERVICE AT THE TIME OF THIS EVENT. INVESTIGATION OF THE ACTUATION REVEALED THAT THE RADIATION DETECTOR INTERNAL CALIBRATION SOURCE (CHECKSOURCE) HAD BEEN INADVERTENTLY INSERTED DURING INSULATION WORK IN THE AREA. THE CAUSE OF THIS EVENT WAS THE INADVERTENT INSERTION OF THE RADIATION DETECTOR CHECKSOURCE DUE TO BUMPING OF THE CHECKSOURCE ASSEMBLY. THE ROOT CAUSE WAS HUMAN FACTORS IN THAT THE DETECTOR WAS NOT LABELED AS SENSITIVE EQUIPMENT; THEREFORE, IT WAS NOT PROTECTED FROM ONGOING WORK. FOLLOWING THIS EVENT, RADIATION MONITOR 3-RM-90-259B WAS FUNCTIONALLY TESTED TO ENSURE CORRECT OPERATION OF THE MONITOR. A TEMPORARY WOODEN STRUCTURE WAS PLACED AROUND THE DETECTOR ASSEMBLY TO PROTECT THE ASSEMBLY. THE DETECTOR ASSEMBLY WILL BE LABELED AS A SENSITIVE DEVICE. DURING THIS EVENT, UNIT 2 WAS IN THE COLD SHUTDOWN CONDITION WITH IRRADIATED FUEL IN THE REACTOR VESSEL. UNITS 1 AND 3 WERE DEFUELED.

[15] BROWNS FERRY 1 DOCKET 50-259 LER 89-023
PERFORMANCE OF SURVEILLANCE TESTING RESULTS IN INOPERABLE EMERGENCY EQUIPMENT COOLING WATER SYSTEM AND DIESEL GENERATORS.
EVENT DATE: 082489 REPORT DATE: 092289 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 215236) ON 8/15/89, TEST OF THE RESIDUAL HEAT REMOVAL (RHR) SERVICE WATER INITIATION LOGIC WAS PERFORMED TO SATISFY A TECH SPEC SURVEILLANCE REQUIREMENT. A REVIEW OF THE TEST PROCEDURE DETERMINED THAT IT REQUIRED THE DISABLING OF THE AUTOMATIC START LOGIC OF THE EMERGENCY EQUIPMENT COOLING WATER SYSTEM (EECW). EECW PROVIDES AN AUTOMATIC SUPPLY OF COOLING WATER TO ESSENTIAL EQUIPMENT, INCLUDING DIESEL GENERATORS. WITH EECW INOPERABLE, EQUIPMENT SERVED BY THE EECW IS ALSO INOPERABLE. AT THE TIME OF DISCOVERY OF THIS EVENT, UNIT 2 WAS IN COLD SHUTDOWN WITH IRRADIATED FUEL IN THE REACTOR, AND UNITS 1 AND 3 WERE DEFUELED. THE CAUSE OF THE EVENT WAS A DEFICIENT TEST PROCEDURE THAT WAS REVISED IN 1976 TO REQUIRE THE DISABLING OF THE EECWS AUTOMATIC START LOGIC. THE TEST PROCEDURE HAS BEEN PUT ON ADMINISTRATIVE HOLD TO PREVENT ITS USE AND WILL BE REVISED OR DELETED BEFORE ITS NEXT SCHEDULED PERFORMANCE DATE. A PROCEDURE REVIEW GROUP HAS BEEN CREATED TO REVIEW A SAMPLE OF TEST PROCEDURES TO CHECK IF OTHER TEST PROCEDURES CAUSE A SYSTEM OR DEVICE FROM PERFORMING ITS INTENDED SAFETY FUNCTION IN VIOLATION OF THE TECH SPECS. THE PROCEDURE VERIFICATION REVIEW CHECKLIST WAS REVISED TO ADD QUESTIONS TO IDENTIFY WHETHER A TEST PROCEDURE WOULD CAUSE A SYSTEM OR DEVICE FROM PERFORMING ITS INTENDED SAFETY FUNCTION IN VIOLATION OF THE TECH SPECS.

[16] BROWNS FERRY 1 DOCKET 50-259 LER 89-024
DEENERGIZATIONS OF REACTOR PROTECTION SYSTEM BUS BY MOTOR GENERATOR CIRCUIT PROTECTOR OPERATIONS CAUSED BY INADEQUATE DESIGN OF PROTECTOR SETPOINTS.
EVENT DATE: 082689 REPORT DATE: 092589 NSSS: GE TYPE: BWR

OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)
VENDOR: OHMITE

(NSIC 215237) ON 8/26/89 AT 1215 AND 9/20/89 AT 0400, UNIT 1 REACTOR PROTECTION SYSTEM (RPS) BUS 1B WAS DEENERGIZED RESULTING IN THE ACTUATION OF PLANT ENGINEERED SAFETY FEATURES INCLUDING STANDBY GAS TREATMENT ACTUATION, CONTROL ROOM EMERGENCY VENTILATION ACTUATION, AND PRIMARY CONTAINMENT ISOLATIONS. INVESTIGATIONS REVEALED THAT THE RPS MOTOR GENERATOR (MG) SET CIRCUIT PROTECTORS HAD TRIPPED BECAUSE OF SHORT DURATION OUTPUT VOLTAGE FLUCTUATIONS. IN THE AUGUST 26 EVENT, IT WAS DETERMINED THAT A DIRTY VOLTAGE ADJUSTMENT POTENTIOMETER CAUSED THE VOLTAGE FLUCTUATIONS. THE ROOT CAUSE OF THESE EVENTS IS AN INADEQUATE DESIGN OF THE SETPOINTS FOR THE CIRCUIT PROTECTORS FOR THE RPS MG SETS. THIS RESULTS IN THE LACK OF AN ADEQUATE OPERATING MARGIN FOR THE RPS MG SET VOLTAGE REGULATION. THEREFORE, NORMAL EQUIPMENT DEGRADATION FROM MINOR WEAR OR AGING CAN CAUSE INADVERTENT DEENERGIZATION OF A RPS BUS. AS CORRECTIVE ACTION, THE DEGRADED VOLTAGE ADJUSTMENT POTENTIOMETER WAS REPLACED. PRIOR TO THIS EVENT, THE CURRENT DESIGN BASIS FOR THE RPS MG SET CIRCUIT PROTECTION SETPOINTS WAS BEING EVALUATED. THIS EFFORT IS ONGOING AND APPROPRIATE DESIGN CHANGES WILL BE INITIATED BASED ON THE RESULTS OF THIS EVALUATION. AN EVALUATION WILL BE PERFORMED TO DETERMINE IF THE VOLTAGE ADJUSTMENT POTENTIOMETERS SHOULD BE REPLACED WITH A DIFFERENT TYPE.

[17] BROWNS FERRY 2 DOCKET 50-260 LER 89-008 REV 02
UPDATE ON ELECTRICAL FAULT ON TRANSFORMER CAUSES ENGINEERED SAFETY FEATURE ACTUATIONS.
EVENT DATE: 030989 REPORT DATE: 083089 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 1 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 215159) AT 1032 HOURS ON 3/9/89, AN ESF ACTUATION OCCURRED DUE TO AN ELECTRICAL FAULT ON THE UNIT STATION SERVICE TRANSFORMER (USST) 2B. THE FAULT LED TO THE LOSS OF SHUTDOWN BUS 2 (THE ALTERNATE FEED WAS TAGGED OUT FOR MAINTENANCE). AS A RESULT, 4KV SHUTDOWN BOARDS C AND D SENSED THE UNDERVOLTAGE CONDITION AND AUTOMATICALLY STARTED DIESEL GENERATORS C AND D. EMERGENCY EQUIPMENT COOLING WATER PUMPS C1 AND D3 ALSO AUTO STARTED PER DESIGN DUE TO THE DIESEL GENERATOR STARTS. THE TRANSFORMER FAULT OCCURRED BECAUSE 1) INADEQUATE INSULATION ABOVE THE BUS JOINT 2) THE DESIGN OF THE BUS DUCT ALLOWED COLLECTION OF CONDENSATION, AND 3) VENDOR RECOMMENDED PREVENTIVE MAINTENANCE WAS NOT PERFORMED. DURING THE POST EVENT RESTORATION OF POWER, ADDITIONAL ESF ACTUATIONS OCCURRED WHEN REACTOR PROTECTION SYSTEMS CIRCUIT PROTECTORS TRIPPED ON UNITS 1 AND 2. AS IMMEDIATE CORRECTIVE ACTION, STABLE ELECTRICAL POWER WAS RESTORED AND SAFETY SYSTEMS WERE RETURNED TO NORMAL. LONG TERM CORRECTIVE ACTION INCLUDES INSPECTION AND TESTING OF OTHER TRANSFORMERS OF THIS TYPE AND PREVENTIVE MAINTENANCE PRACTICES WERE REVIEWED. DESIGN CHANGES TO THE BUS CONNECTION AND BUS DUCT WERE INITIATED ON THE 2B USST. ADDITIONALLY, COMPREHENSIVE CORRECTIVE ACTION PLANS WERE DEVELOPED TO INCORPORATE LESSONS LEARNED FROM THIS EVENT. UNITS 1 AND 3 WERE DEFUELED AND UNIT 2 WAS IN COLD SHUTDOWN DURING THIS EVENT.

[18] BROWNS FERRY 2 DOCKET 50-260 LER 89-021
FOUR HOUR LCO EXCEEDED FOR SURVEILLANCE TESTING OF AVERAGE POWER RANGE MONITOR DUE TO PERSONNEL ERROR.
EVENT DATE: 071089 REPORT DATE: 080989 NSSS: GE TYPE: BWR

(NSIC 215397) ON 7/10/89, AT 1837 HOURS, IT WAS DETERMINED THAT A BROWNS FERRY UNIT 2 TECH SPEC (TS) FOUR-HOUR LIMITING CONDITION FOR OPERATION (LCO) HAD BEEN EXCEEDED BY 2 MINUTES. SURVEILLANCE TESTING OF AVERAGE POWER RANGE MONITOR (APRM) 2F REQUIRED INSTALLATION OF A SHORTING PLUG (JUMPER) TO RELAY 5A-K12F CONTACTS. THIS REMOVED APRM 2F AND INTERMEDIATE RANGE MONITOR (IRM) 2F INPUTS TO THE REACTOR PROTECTION SYSTEM AND MADE THESE INSTRUMENTS INOPERABLE. TECH SPEC

ALLOWED THESE INSTRUMENTS TO BE INOPERABLE FOR UP TO 4 HOURS FOR REQUIRED SURVEILLANCE UNDER THE EXISTING PLANT CONDITIONS. THIS VIOLATION OF TECH SPEC WAS THE RESULT OF PERSONNEL ERROR CAUSED BY INADEQUATE DIRECTIONS IN THE PROCEDURE AND INADEQUATE TRAINING. THE SURVEILLANCE PROCEDURE DID NOT PROVIDE ADEQUATE DIRECTION TO THE CREW PERFORMING THE TEST ON MONITORING THE LCO TIME FRAME OR WHEN TO NOTIFY THE SHIFT OPERATIONS SUPERVISOR (SOS) OF APPROACH TO THE TIME LIMIT. ALSO THE LEAD PERFORMER WAS UNAWARE THAT HIS RESPONSIBILITY FOR DIRECTING THE TEST INCLUDED MONITORING THE LCO TIME LIMIT. WRITERS GUIDES WILL BE REVIEWED AND REVISED AS NECESSARY TO PROVIDE ADEQUATE DIRECTION IN THE TEST PROCEDURES. AFFECTED TEST PROCEDURES WILL BE REVIEWED AND REVISED AS NECESSARY. RESPONSIBILITY FOR MONITORING THE LCO TIME LIMITS WILL BE SPECIFIED IN THE "CONDUCT OF TESTING" PROCEDURE.

[19]   BROWNS FERRY 2   DOCKET 50-260   LER 89-026
PERSONNEL ERROR DURING DIESEL GENERATOR AIR COMPRESSOR MAINTENANCE RESULTS IN FAILURE TO MEET TECH SPEC REQUIREMENTS.
EVENT DATE: 081089   REPORT DATE: 090889   NSSS: GE   TYPE: BWR
VENDOR: INGERSOLL-RAND CO.

(NSIC 215238) ON 8/10/89 AT 1750, THE "B" DIESEL GENERATOR (DG) WAS DECLARED INOPERABLE ALONG WITH ITS ASSOCIATED UNIT 2 RESIDUAL HEAT REMOVAL SYSTEM PUMP AND CORE SPRAY SYSTEM PUMP WHICH RESULTED IN NOT MEETING THE MINIMUM CORE COOLING SYSTEM REQUIREMENTS OF TECH SPECS 3.5. THE "B" DG WAS DECLARED INOPERABLE WHEN MINIMUM AIR START SYSTEM PRESSURE REQUIREMENTS COULD NOT BE MET DUE TO AN AIR LEAK ON THE HIGH PRESSURE HEAD OF THE RIGHT BANK AIR COMPRESSOR FOR THE DG STARTING AIR SYSTEM. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR DURING THE REASSEMBLY OF THE HIGH PRESSURE HEAD OF THE AIR COMPRESSOR FOLLOWING MAINTENANCE EARLIER IN THE DAY. FAILURE TO REASSEMBLE THE HEAD CORRECTLY AND ALLOWING DEBRIS TO REMAIN IN THE HEAD BOLT HOLES PREVENTED PROPER TORQUING OF THE HEAD BOLTS. ALSO CONTRIBUTING TO THIS EVENT WAS A MALFUNCTIONING UNLOADER VALVE. THIS MALFUNCTION INCREASED THE STRESSES ON THE HEAD GASKET. THESE PROBLEMS RESULTED IN THE FAILURE OF HEAD GASKET AND SUBSEQUENT INOPERABILITY OF THE DG. AS A RESULT OF THIS EVENT, APPROPRIATE PERSONNEL CORRECTIVE ACTION HAS BEEN INITIATED FOR THE MAINTENANCE PERSONNEL INVOLVED IN THIS EVENT. OTHER APPROPRIATE MAINTENANCE PERSONNEL WILL ALSO BE MADE AWARE OF THIS EVENT. ADDITIONALLY, THE PROCEDURE USED TO REPAIR THE AIR COMPRESSOR IN THIS EVENT WILL BE ENHANCED TO PREVENT RECURRENCE OF THESE PROBLEMS DURING FUTURE REPAIRS.

[20]   BRUNSWICK 1   DOCKET 50-325   LER 89-018
EXCEEDED TECH SPEC 3.6.5.2 REQUIRED ACTION AS RESULT OF UNRECOGNIZED DESIGN LOGIC INTERFACE WITH THE STANDBY GAS TREATMENT SYSTEM.
EVENT DATE: 081789   REPORT DATE: 091589   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)

(NSIC 215260) UNITS 1 AND 2 WERE SYNCHRONIZED TO THE MAIN POWER GRID WITH UNIT 1 AT 100% AND UNIT 2 AT 86%, WHILE IN A FUEL DEPLETION COAST DOWN TO A SCHEDULED SEPTEMBER 1989 REFUEL/MAINTENANCE OUTAGE. ON 8/17/89, IT WAS DETERMINED THAT DEENERGIZING THE STARTER CIRCUITRY OF A STANDBY GAS TREATMENT (SBGT) TRAIN WILL CAUSE INOPERABILITY OF SELECTED ISOLATION LOGIC (DIVISION I OR II) TO THE SECONDARY CONTAINMENT ISOLATION DAMPERS. THESE DAMPERS ARE DESIGNED TO ISOLATE ON A DRYWELL HIGH PRESSURE/REACTOR LOW LEVEL NO. 2 SIGNAL. THIS CONDITION IS A PREVIOUSLY UNRECOGNIZED DESIGN CONCERN FROM ORIGINAL PLANT CONSTRUCTION. CAUTION TAGS WERE PLACED ON THE INVOLVED POWER SUPPLY BREAKERS. SUBSEQUENT REVIEW DETERMINED, ON 8/30/89, THAT THE ISOLATION LOGIC WILL ALSO BE LOST DUE TO POSITIONING THE SBGT TRAIN INLET/OUTLET VALVES IN OTHER THAN FULL OPEN, SELECTION OF A SBGT TRAIN CONTROL SWITCH TO OTHER THAN "PREFERRED" OR "ON", AND A HIGH TEMPERATURE ISOLATION OF THE TRAIN. ON 8/17/89 AND 8/30/89, MEMORANDUMS WERE DISTRIBUTED TO PLANT OPERATIONS ADVISING THEM OF THESE FINDINGS. PROJECT IDENTIFICATION NO. G0098A WAS INITIATED ON 8/24/89 TO RESOLVE THE IDENTIFIED

DESIGN CONCERN. THIS EVENT WILL BE REVIEWED BY APPROPRIATE PERSONNEL. PRIOR SIMILAR EVENTS HAVE BEEN REPORTED IN LERS 1-88-032 AND 034.

[21] BYRON 1 DOCKET 50-454 LER 89-008
AUXILIARY FEEDWATER SUCTION PRESSURE SWITCHES FOUND OUT OF CALIBRATION DUE TO FAILURE TO CONSIDER HEAD CORRECTION.
EVENT DATE: 083089 REPORT DATE: 092989 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 2 (PWR)

(NSIC 215388) THERE WERE TWO RELATED EVENTS INVOLVING THE FUNCTION AND DESIGN OF THE AUXILIARY FEEDWATER (AF) PUMP SUCTION PRESSURE SWITCHES (AFSPSS). DURING AN INSTITUTE OF NUCLEAR POWER OPERATIONS EVALUATION OF THE AUXILIARY FEEDWATER SYSTEM IT WAS NOTED THAT THE FOUR AF SUCTION PRESSURE TRANSMITTERS (AFSPTS) (TWO TRANSMITTERS PER UNIT) WERE LOCATED AT DIFFERENT ELEVATIONS. HOWEVER, THEIR CALIBRATIONS WERE NOT HEAD CORRECTED. A NUCLEAR WORK REQUEST WAS WRITTEN 2/9/89 TO MAKE THE HEAD CORRECTIONS. THE MORNING OF 8/30/89, THE AFSPTS WERE HEAD CORRECTED TO THE MOST LIMITING CASE AS A CONSERVATIVE MEASURE. PER A LETTER FROM SARGENT AND LUNDY, RECEIVED ON 8/30 AT APPROXIMATELY 1500, IT WAS DETERMINED THAT THE SWITCH SETPOINTS, CORRESPONDING TO THREE OF THE FOUR AFSPTS, HAD NOT BEEN WITHIN THE TECH SPEC ALLOWABLE VALUES. WHILE INVESTIGATING THE IMPACT OF THE FIRST EVENT ON SAFETY, THE BASES OF THE ORIGINAL SETPOINTS WERE QUESTIONED. THIS INVESTIGATION IS STILL IN PROGRESS. THE RESULTS OF THIS INVESTIGATION WILL BE SUBMITTED IN A SUPPLEMENTAL REPORT. THE ROOT CAUSE OF THE FIRST EVENT WAS THAT DESIGN DOCUMENTS AND PROCEDURES ADDRESSING INSTRUMENT INSTALLATION AND CALIBRATION DID NOT CLEARLY INDICATE THAT THE AFSPTS REQUIRED HEAD CORRECTION. THE ROOT CAUSE OF THE SECOND EVENT IS STILL UNDER EVALUATION.

[22] CALLAWAY 1 DOCKET 50-483 LER 89-010
ENGINEERED SAFETY FEATURES ACTUATIONS DUE TO A FAILED POWER SUPPLY AND AN AUXILIARY FEEDWATER ACTUATION WITH SWAPOVER TO ESSENTIAL SERVICE WATER DUE TO IMPROPER OPERATOR ACTION.
EVENT DATE: 090589 REPORT DATE: 092789 NSSS: WE TYPE: PWR
VENDOR: CONSOLIDATED CONTROLS CORP.

(NSIC 215391) ON 9/5/89 AT 1531 CDT, ENGINEERED SAFETY FEATURES (ESF) CONTAINMENT PURGE ISOLATION, CONTROL ROOM VENTILATION ISOLATION AND FUEL BUILDING VENTILATION ISOLATION ACTUATIONS WERE RECEIVED WHEN A 15VDC POWER SUPPLY FAILED IN THE ESF CABINET FOR 'B' TRAIN. AT 1939, WHILE ATTEMPTING TO RESTORE THE ESF CABINET TO SERVICE FOLLOWING THE POWER SUPPLY REPLACEMENT, A 'B' TRAIN AUXILIARY FEEDWATER ACTUATION (AFAS) WITH SWAPOVER TO ESSENTIAL SERVICE WATER (ESW) WAS RECEIVED. THE PLANT WAS IN MODE 1 - POWER OPERATION AT 100% POWER. THE SECOND ESF ACTUATIONS WERE CAUSED BY IMPROPER OPERATOR ACTION. THE RESTORATION OF ESF PROCEDURE REQUIRES TRIPPED LOGIC TO BE RESET PRIOR TO PLACING RELAY POWER SUPPLY TO ON. WHEN THE LICENSED REACTOR OPERATOR WAS UNABLE TO RESET THE TRIPPED LOGIC HE MISINTERPRETED THE PROCEDURE AND TURNED ON THE POWER SUPPLY THUS ACTUATING THE AFAS AND VALVE SWAPOVER TO ESW. PLANT PROCEDURES REQUIRED A PLANT SHUTDOWN TO MODE 2 - START UP TO RESTORE STEAM GENERATOR CHEMISTRY. MODE 1 WAS RESUMED AT 2304 ON 9/7/89. THE FAILED POWER SUPPLY WAS REPLACED. THE IMPORTANCE OF CREW COMMUNICATION AND CONSERVATIVE JUDGEMENT WAS DISCUSSED WITH THOSE INVOLVED AND WILL BE COVERED IN TRAINING. THE FAILURE OF THE TRIPPED LOGIC TO RESET WAS EVALUATED. A REVIEW FOR SIMILAR INDUSTRY EXPERIENCE WILL BE PERFORMED.

[23] CALLAWAY 1 DOCKET 50-483 LER 89-011
STEAM GENERATOR LO-LO TRIP DELTA T SIGNALS WERE PUT INTO SERVICE WITHOUT A CURRENT TECH SPEC SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 090689 REPORT DATE: 092789 NSSS: WE TYPE: PWR

(NSIC 215392) ON 9/6/89, THE PLANT WAS IN MODE 1 AT APPROXIMATELY 30% POWER AND

DECREASING. AT 0040 CDT, THE STEAM GENERATOR LOW-LOW LEVEL TRIP TIME DELAYS (TTD'S) WERE REINSTATED WITHOUT BEING VERIFIED OPERABLE BY PERFORMANCE OF TECH SPEC SURVEILLANCE REQUIREMENT 4.3.2.1(6)(D)(3). THE NON-LICENSED SHIFT TECHNICAL ADVISOR (STA) REVIEWED THE SURVEILLANCE REQUIREMENT TRACKING SCHEDULE AND INCORRECTLY CONCLUDED THAT THE SURVEILLANCES HAD BEEN SATISFACTORILY PERFORMED ON 8/18/89. HE FAILED TO NOTE THE CODE 'N' IN THE ADJACENT COLUMN, DENOTING THE SURVEILLANCES HAD NOT BEEN COMPLETED SATISFACTORILY. HE LATER DISCOVERED AN APPARENT INCONSISTENCY WHEN HE REVIEWED THE EQUIPMENT OUT-OF-SERVICE LOG (EOSL) AND FOUND TWO EOSL SHEETS FOR THE TTD'S. AT 0600, HE DISCUSSED THIS WITH A SURVEILLANCE SCHEDULING ENGINEER WHO VERIFIED THAT THE SURVEILLANCES HAD IN FACT LAPSED. THEREFORE, THE TTD'S WERE TRIPPED UNTIL THE REQUIRED SURVEILLANCES WERE SATISFACTORILY COMPLETED AT 1248. THE ROOT CAUSE OF THIS EVENT WAS THE FAILURE OF THE STA TO ADEQUATELY REVIEW THE SURVEILLANCE REQUIREMENT TRACKING SCHEDULE. PROCEDURE OTG-ZZ-00004, POWER OPERATIONS, HAS BEEN REVISED TO REMIND OPERATORS THAT THE TTD SURVEILLANCES ARE NOT KEPT CURRENT DURING NORMAL POWER OPERATIONS ABOVE 20% POWER. THIS EVENT WILL BE COVERED IN LICENSED OPERATOR AND STA RETRAINING.

[24]   CALVERT CLIFFS 1   DOCKET 50-317   LER 88-014 REV 01
UPDATE ON LEAKING STEAM ISOLATION VALVE CAUSES EXCESSIVE CHECK VALVE CYCLING RESULTING IN CHECK VALVE MISALIGNMENT AND LEAKAGE.
EVENT DATE: 102988   REPORT DATE: 092089   NSSS: CE   TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 215294) ON 10/29/88, UNIT 1 WAS IN MODE 5, COLD SHUTDOWN, FOR A MAINTENANCE OUTAGE. WHILE PERFORMING A SURVEILLANCE TEST PROCEDURE TO REVERSE-FLOW TEST CHECK VALVES, MS-103 LEAKED BY. MS-103 IS A SIX INCH CHECK VALVE WHICH PROVIDES ISOLATION OF NO. 12 STEAM GENERATOR FROM NO. 11 STEAM GENERATOR IN THE EVENT OF A MAIN STEAM LINE BREAK UPSTREAM OF THE MAIN STEAM ISOLATION VALVES. MS-103 WAS DISASSEMBLED AND THE DISK WAS FOUND TO BE MISALIGNED APPROXIMATELY ONE-QUARTER INCH DUE TO EXCESSIVE WEAR OF THE HINGE PINS AND BUSHING AREA. THE VALVE HAD BEEN INSTALLED IN APRIL 1988. IT WAS AN ANCHOR/DARLING 6"-900# TILTING DISK CHECK VALVE. THE CAUSE OF THE VALVE FAILURE WAS EXCESSIVE CYCLING, INITIALLY THOUGHT TO BE DUE TO LEAK-BY OF AN UPSTREAM CONTROL VALVE, CV-4070, BUT LATER DETERMINED TO BE DUE TO LEAK-BY OF AN UPSTREAM ISOLATION VALVE, MS-102, TWO FEET AWAY. CHECK VALVE MS-103 WAS REPLACED WITH AN IDENTICAL VALVE. THE WORN PINS AND BUSHINGS WERE SENT FOR LABORATORY ANALYSIS AND FOUND TO SHOW WEAR PATTERNS REPRESENTATIVE OF MECHANICAL WEAR. A CHANGE TO THE ORIENTATION AND LOCATION OF MS-103 IS PLANNED. CV-4070 WAS OVERHAULED. MS-102 WAS FOUND TO HAVE STEAM CUTS ON THE DISK AND SEAT AND WAS REPAIRED.

[25]   CALVERT CLIFFS 1   DOCKET 50-317   LER 89-013
MISSING STEPS IN SURVEILLANCE TEST PROCEDURE.
EVENT DATE: 073189   REPORT DATE: 082889   NSSS: CE   TYPE: PWR

(NSIC 215130) ON JULY 31, 1989, THE CALVERT CLIFFS SURVEILLANCE TEST PROCEDURE (STP) PROGRAM MANAGER DISCOVERED THE FAILURE TO FULLY COMPLY WITH TECHNICAL SPECIFICATION SURVEILLANCE 4.9.12.9.A. THE SURVEILLANCE REQUIRES BOTH SPENT FUEL EXHAUST FANS TO BE RUN EVERY 31 DAYS FOR AT LEAST 15 MINUTES THE SURVEILLANCE TEST PROCEDURE USED TO SATISFY 4.9.12.A. STP.0.7.1, DID NOT CONTAIN STEPS TO TEST THE FANS. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR IN FAILING THE FAILURE TO INCORPORATE STEPS IN STP.0.7.1 OR ANY OTHER SURVEILLANCE PROCEDURE TO TEST THE FANS. A SECONDARY CAUSE WAS THE FAILURE OF THE PROCEDURE REVIEW PROCESS TO DETECT THE ERROR. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDE: 1. IMMEDIATELY SWAPPING IN-SERVICE FANS AND RUNNING THE NEW IN-SERVICE FAN FOR 15 MINUTES. 2. INCORPORATING STEPS FOR TESTING THE SPENT FUEL EXHAUST CHARCOAL FILTERS AND FANS IN A NEW TEST PROCEDURE. 3. PERFORMING A DETAILED REVIEW OF TEST PROCEDURES USED TO SATISFY TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS. 4. IMPLEMENTATION

OF A UPGRADED STP PROGRAM. 5. IMPROVEMENT OF THE QUALITY ASSURANCE TECHNICAL SPECIFICATION AUDIT PROCESS.

[26] CALVERT CLIFFS 1 DOCKET 50-317 LER 89-015
IODINE FILTER DOUSING SYSTEM NOT ENVIRONMENTALLY QUALIFIED.
EVENT DATE: 082289 REPORT DATE: 092089 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 2 (PWR)

(NSIC 215308) ON 8/22/89, WHILE UNIT 1 WAS IN COLD SHUTDOWN AND UNIT 2 WAS DEFUELED, IT WAS ESTABLISHED THAT CALCULATIONS USED TO DETERMINE THAT THE DOUSING SYSTEM FOR THE CONTAINMENT IODINE FILTERS IS NOT REQUIRED TO BE ENVIRONMENTALLY QUALIFIED WERE IN ERROR. THE COMPONENTS WHICH ARE NOT QUALIFIED INCLUDE SOLENOID VALVES, POSITION SWITCHES, THERMISTORS, AND INTERNAL WIRING. THE MIS-OPERATION OF THESE COMPONENTS COULD RESULT IN PREVENTING THE IODINE FILTERS FROM PERFORMING THEIR SAFETY FUNCTION. A SUPPLEMENTAL REPORT WILL BE PREPARED WHEN THE RESULTS OF THE OFF-SITE DOSE CALCULATIONS BECOME AVAILABLE. THE SUPPLEMENTAL REPORT WILL DISCUSS THE SAFETY SIGNIFICANCE OF THE EVENT. PRIOR TO RESTARTING EITHER UNIT, WE WILL ESTABLISH A JUSTIFICATION FOR CONTINUED OPERATION (JCO) WHICH ADDRESSES THE INOPERABILITY OF THE DOUSING SYSTEM. THE JCO IS NEEDED TO ALLOW OPERATION UNTIL AN EVALUATION CAN BE MADE TO DETERMINE THE APPROPRIATE PERMANET SOLUTION.

[27] CATAWBA 1 DOCKET 50-413 LER 89-021
TECH SPEC REQUIRED POWER REDUCTION DUE TO FAILURE OF T2 REFUELING WATER STORAGE TANK LEVEL CHANNELS CAUSED BY LIGHTNING STRIKE.
EVENT DATE: 072789 REPORT DATE: 082289 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215146) ON JULY 27, 1989, UNIT 1 WAS IN MODE 1 POWER OOPERATION AT 100% POWER. THUNDERSTORMS WERE OCCURRING IN THE VICINITY OF CATAWBA NUCLEAR STATION. AT 1948 HOURS, UNIT 1 REFUELING WATER STORAGE TANK (FWST) LOW LEVEL COMPUTER ALARMS WERE RECEIVED ON CHANNELS 1 AND 4. CHANNEL 1 WAS INDICATING OFF SCALE HIGH AND CHANNEL 4 WAS INDICATING OFF SCALE LOW. AT 2006 HOURS, CHANNELS 1 AND 4 OF THE FWST LEVEL INSTRUMENTATION WERE DECLARED INOPERABLE PER TECH SPEC 3.3.2 AND THE UNIT ENTERED TECH SPEC 3.0.3 DUE TO 2 OF 4 CHANNELS BEING INOPERABLE. A PRIORITY 1 WORK REQUEST WAS ISSUED FOR INSTRUMENTATION AND ELECTRICAL (IAE) TO INVESTIGATE AND REPAIR THE INOPERABLE CHANNELS. AT 2103 HOURS, THE CONTROL ROOM OPERATORS (CROS) BEGAN POWER REDUCTION TO COMPLY WITH TECH SPEC ACTION ITEM 3.0.3. AT 2150 HOURS, CHANNEL 1 WAS RECALIBRATED AND RETURNED TO SERVICE AND UNIT 1 EXITED TECHNICAL SPECIFICATION 3.0.3 AFTER REDUCING POWER TO 89%. UNIT 1 WAS SUBSEQUENTLY RETURNED TO 100% POWER, AND CHANNEL 4 WAS DECLARED OPERABLE ON JULY 28, 1989 AT 1717 HOURS, AFTER THE LEVEL TRANSMITTER WAS REPLACED. THE SIMULTANEOUS FAILURE OF THE TWO FWST LEVEL CHANNELS IS BELIEVED TO HAVE BEEN CAUSED BY A LIGHTNING STRIKE ON OR NEAR THE FWST.

[28] CATAWBA 1 DOCKET 50-413 LER 89-022
MANUAL REACTOR TRIP DUE TO FAILURE OF A GASKET ON THE MAIN FEEDWATER VALVE POSITIONER CONTROL AIR MANIFOLD.
EVENT DATE: 082389 REPORT DATE: 092189 NSSS: WE TYPE: PWR
VENDOR: MOORE PRODUCTS COMPANY

(NSIC 215384) ON 8/24/89, UNIT 1 WAS IN MODE 1, POWER OPERATION, AT 100% POWER, AT 1930 HOURS, 1CF28, STEAM GENERATOR (S/G) 1A MAIN FEEDWATER (CF) CONTROL VALVE, BEGAN CLOSING, CAUSING A STEAM GENERATOR 1A LEVEL DEVIATION ALARM. THE CONTROL ROOM OPERATORS (CROS) RESPONDED BY OPENING 1CF30, S/G 1A CF CONTROL BYPASS VALVE, AND DISPATCHING OPERATIONS PERSONNEL AND INSTRUMENTATION AND ELECTRICAL (IAE) PERSONNEL TO INVESTIGATE THE PROBLEM. THE CRO BEGAN POWER REDUCTION TO COMPENSATE FOR THE REDUCED CF FLOW TO S/G 1A. AT 1944 HOURS, S/G 1A LEVEL BEGAN DROPPING RAPIDLY. THE CRO THEN INITIATED A MANUAL REACTOR TRIP AND ENTERED

EMERGENCY PROCEDURE EP/1/A/5000/01, REACTOR TRIP OR SAFETY INJECTION. BY 2110 HOURS, S/G 1A WAS RETURNED TO THE NORMAL OPERATING LEVEL. 1CF28 MALFUNCTIONED DUE TO A FAILED GASKET IN THE POSITIONER CONTROL AIR MANIFOLD. THE GASKET APPEARED TO HAVE FAILED DUE TO IMPROPER DESIGN AND/OR INSTALLATION DEFICIENCY. THE POSITIONER WAS REPLACED ON 1CF28 AND ON THE REMAINING THREE UNIT 1 CONTROL VALVES BEFORE UNIT 1 RETURNED TO MODE 1, POWER OPERATION, AT 0753 HOURS ON 8/25/89. THE CF CONTROL VALVE POSITIONERS ARE BEING INSPECTED WEEKLY FOR AIR LEAKS, AND AN IMPROVED GASKET DESIGN HAS BEEN IMPLEMENTED ON UNIT 1.

[29] CATAWBA 2 DOCKET 50-414 LER 88-033 REV 02
UPDATE ON INOPERABLE CONTAINMENT AIR RETURN TRAIN A BECAUSE OF SWAPPED CONTROL WIRING DUE TO AN INAPPROPRIATE ACTION.
EVENT DATE: 040188 REPORT DATE: 052689 NSSS: WE TYPE: PWR

(NSIC 214081) ON 3/31/88, AT 0400 HRS, WHILE PERFORMING AUX. SAFEGUARDS TEST CABINET PERIODIC TEST, 2ARF-D-2, UNIT 2 CONTAINMENT AIR RETURN FAN DAMPER 2A, FAILED TO OPEN. A WORK REQUEST WAS WRITTEN TO INVESTIGATE AND REPAIR 2ARF-D-2, AND THE DAMPER WAS DECLARED INOPERABLE. WHILE PERFORMING THE WORK REQUEST, IT WAS DISCOVERED THAT 2 WIRES IN THE DAMPER WAS DECLARED INOPERABLE. WHILE PERFORMING THE WORK REQUEST, IT WAS DISCOVERED THAT 2 WIRES IN THE DAMPER CONTROL CIRCUIT WERE SWAPPED. THIS WIRING ERROR SWAPPED THE POSITIVE LEG OF 2 SEPARATE 120 VAC POWER SOURCES IN THE DAMPER CONTROL CIRCUIT AND EFFECTIVELY REMOVED THE INTERLOCKS TO OPEN THE DAMPER ON A SAFETY SYSTEM ACTUATION. AFTER THE WIRING WAS CORRECTED, A FUNCTIONAL TEST WAS PERFORMED ON THE DAMPER AND IT WAS DECLARED OPERABLE. THIS EVENT WAS DISCOVERED ON 4/1/88 AND DETERMINED REPORTABLE ON 12/14/88 BUT DELAYED UNTIL 1/27/89 PER 1/13/89 LETTER TO NRC. UNIT 2 HAD BEEN IN ALL MODES OF OPERATION WHILE THE DAMPER OPERABILITY WAS IN QUESTION. UNIT 2 WAS AT 100% POWER AT THE TIME OF THIS EVENT. THIS INCIDENT HAS BEEN ATTRIBUTED TO AN INAPPROPRIATE ACTION. THE WIRES WERE INCORRECTLY INSTALLED DURING A NUCLEAR STATION MODIFICATION. POST MODIFICATION TESTING VERIFIED DAMPER OPERABILITY. TESTING REVERIFIED THAT THE DAMPER WOULD OPEN WITH THE INCORRECT WIRING IN PLACE. A WORK REQUEST WAS COMPLETED TO FURTHER INVESTIGATE THE CIRCUITRY.

[30] CATAWBA 2 DOCKET 50-414 LER 89-017
POTENTIAL TECH SPEC VIOLATION DUE TO INOPERABILITY OF THE TURBINE DRIVEN AUX. FEEDWATER PUMP DUE TO CONTROL VALVE STEM CORROSION.
EVENT DATE: 073189 REPORT DATE: 083089 NSSS: WE TYPE: PWR
VENDOR: BINGHAM-WILLAMETTE CO.
KEROTEST MANUFACTURING CORP.
TERRY STEAM TURBINE COMPANY
WOODWARD GOVERNOR COMPANY

(NSIC 215281) ON 7/31/89, AT 1310 HOURS, WITH UNIT 2 IN MODE 1, POWER OPERATION, DURING MONTHLY SURVEILLANCE TESTING ON UNIT 2 TURBINE DRIVEN AUXILIARY FEEDWATER (CA) PUMP (CAPT), THE TURBINE REPETITVELY TRIPPED ON MECHANICAL OVERSPEED. THE TURBINE CONTROL VALVE LINKAGE WAS MANUALLY EXERCISED, AND SUBSEQUENT PUMP START WAS SUCCESSFUL. ON 8/7/89, AT 0950 HOURS, WITH UNIT 2 IN MODE 1, THE UNIT 2 CAPT TRIPPED ON ELECTRICAL OVERSPEED DURING A PRE-MAINTENANCE TEST START AT MAXIMUM SPEED. DISASSEMBLY AND INSPECTION OF THE CONTROL VALVE AND LINKAGE REVEALED THAT PITTING AND CORROSION ON THE CONTROL VALVE STEM AND PACKING WASHERS WERE CAUSING EXCESSIVE FRICTION DURING MOVEMENT. THE VALVE STEM, PACKING, AND WASHERS WERE REPLACED AND CONTROL LINKAGE WAS CLEANED AND RELUBRICATED. POST-MAINTENANCE TESTING OF THE UNIT 2 CAPT AT MAXIMUM SPEED WAS SUCCESSFUL. BETWEEN JULY 31, WHEN THE TURBINE WAS RETURNED TO SERVICE AFTER INITIALLY TRIPPING ON MECHANICAL OVERSPEED, AND AUGUST 7, WHEN THE ROOT CAUSE OF THE TRIPS WAS IDENTIFIED, THE ABILITY OF THE UNIT 2 CAPT TO RESPOND TO AN AUTOMATIC START SIGNAL WITHOUT OPERATOR ACTION IS QUESTIONABLE. SINCE THE PUMP WAS CAPABLE OF PERFORMING ITS INTENDED SAFETY FUNCTION ON 7/31/89, (AS EVIDENCED BY SUCCESSFULLY COMPLETING THE

PERFORMANCE TEST) AND TRIPPED A WEEK LATER IN A PRE-MAINTENANCE TEST, THE ACTUAL DATE OF INOPERABILITY CANNOT BE DETERMINED.

[31] CATAWBA 2 DOCKET 50-414 LER 89-018
COLD LEG ACCUMULATOR BORON ANALYSIS SURVEILLANCE MISSED DUE TO INAPPROPRIATE ACTION RESULTING IN A TECH SPEC VIOLATION.
EVENT DATE: 080789 REPORT DATE: 090789 NSSS: WE TYPE: PWR
VENDOR: KEROTEST MANUFACTURING CORP.

(NSIC 215282) ON 8/7/89, AT 1821 HOURS, WITH UNIT 2 IN MODE 1, POWER OPERATION, OPERATIONS BEGAN MAKEUP TO COLD LEG ACCUMULATOR (CLA) 2C. AT 1825 HOURS, A CONTROL ROOM OPERATOR (CRO) NOTIFIED CHEMISTRY SPECIALIST A THAT CLA 2C REQUIRED A SAMPLE ANALYSIS FOR BORON CONCENTRATION TO SATISFY TECH SPECS. SPECIALIST A RECORDED THIS REQUIREMENT IN THE UNIT 2 PRIMARY CHEMISTRY LOGBOOK AS A TURNOVER ITEM. CHEMISTRY SPECIALIST B, UPON ARRIVAL ON SHIFT, DISCUSSED THE TURNOVER ITEMS WITH SPECIALIST A. SPECIALIST B BELIEVED THAT SPECIALIST A HAD ALREADY TAKEN AND ANALYZED THE SAMPLE AND DID NOT NOTE THE LOGBOOK ITEM. THEREFORE, NO FURTHER ACTIONS WERE TAKEN BY SPECIALIST B. AT 0502 HOURS ON 8/8/89, OPERATIONS MADE UP TO CLA 2C AGAIN. SPECIALIST B COMPLETED SAMPLE COLLECTION AND ANALYSIS AND CALLED OPERATIONS WITH THE RESULTS AT 0544 HOURS. AT 0830 HOURS, CHEMISTRY SPECIALIST C REVIEWED DATA FROM THE PRIMARY CHEMISTRY LOGBOOK AND NOTED THAT THE BORON ANALYSIS FOR THE 1821 HOUR MAKEUP HAD NOT BEEN COMPLETED. AT 0925 HOURS, THE OPERATIONS SHIFT SUPERVISOR WAS NOTIFIED THAT THE CLA 2C SAMPLE HAD NOT BEEN COLLECTED WITHIN THE SPECIFIED SIX HOUR TIME PERIOD. THE SAMPLE TAKEN AT 0528 HOURS VERIFIED THE OPERABILITY OF CLA 2C. CREW MEETINGS WERE HELD BY EACH CHEMISTRY SHIFT TO EMPHASIZE THE IMPORTANCE OF CLEAR AND CONCISE TURNOVER INFORMATION. THIS INCIDENT IS A RESULT OF INAPPROPRIATE ACTIONS.

[32] CLINTON 1 DOCKET 50-461 LER 89-031
LACK OF UNDERSTANDING OF TECH SPECS AND FAILURE TO ENSURE EQUIPMENT OPERABILITY RESULTS IN ENTRY INTO STARTUP MODE WITHOUT MEETING LIMITING CONDITIONS FOR OPERATION.
EVENT DATE: 072489 REPORT DATE: 082289 NSSS: GE TYPE: BWR

(NSIC 215151) ON 7/24/89, AT 1147 HOURS, THE PLANT ENTERED MODE 2 (STARTUP) WITHOUT MEETING THE PROVISIONS OF TECHNICAL SPECIFICATION (TS) 3.0.4. TS 3.0.4 REQUIRES THAT THE CONDITIONS OF THE LIMITING CONDITION FOR OPERATION (LCO) OF A TS BE MET PRIOR TO ENTRY INTO AN OPERATIONAL CONDITION. THE LCOS FOR TSS 3.5.1 AND 3.3.7.5 WERE NOT MET PRIOR TO ENTERING MODE 2. LOOP B OF THE RESIDUAL HEAT REMOVAL SYSTEM WAS OPERATING IN THE SHUTDOWN COOLING (SDC) MODE WHEN THE PLANT ENTERED MODE 2. THE LCO OF TS 3.5.1 REQUIRES THAT LOW PRESSURE COOLANT INJECTION (LPCI) BE OPERABLE IN MODE 2. WHEN OPERATING IN THE SDC MODEL THE LPCI MODE IS INOPERABLE. DUE TO A LACK OF UNDERSTANDING OF TS 3.5.1 THE LCO WAS NOT MET. THE LCO OF TS 3.3.7.5 REQUIRES THAT THE ACOUSTIC MONITOR FOR EACH SAFETY RELIEF VALVE (SRV) BE OPERABLE IN MODE 2. IF THE ACOUSTIC MONITOR IS NOT AVAILABLE, THE TAIL PIPE TEMPERATURE MAY BE MONITORED TO MEET THE TS REQUIREMENT. WHEN THE PLANT ENTERED MODE 2 THE ACOUSTIC MONITOR AND THE TAIL PIPE TEMPERATURE RECORDER FOR ONE SRV WERE INOPERABLE THEREFORE, THE LCO OF TS 3.3.7.5 WAS NOT MET. PERSONNEL FAILED TO RECOGNIZE THAT THE TEMPERATURE RECORDER WAS INOPERABLE. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDE INCREASED TRAINING ON TSS AND REMINDING PERSONNEL OF THE NEED TO TRACK THE STATUS OF ALL EQUIPMENT WHICH COULD AFFECT OPERABILITY.

[33] CLINTON 1 DOCKET 50-461 LER 89-032
FAILURE TO MATCH MANUAL CONTROL TO AUTOMATIC CONTROL PRIOR TO TRANSFERRING FEEDWATER PUMP TO MANUAL RESULTS IN INCREASE IN REACTOR WATER LEVEL AND MANUAL SCRAM.
EVENT DATE: 073189 REPORT DATE: 083089 NSSS: GE TYPE: BWR
VENDOR: CROSBY VALVE & GAGE CO.

FISHER CONTROLS CO.

(NSIC 215152) ON JULY 31, 1989, WITH THE PLANT AT APPROXIMATELY 25 PERCENT POWER, OPERATORS INITIATED A MANUAL SCRAM OF THE REACTOR. PRIOR TO THE SCRAMS THE PLANT HAD BEEN OPERATING AT 100 PERCENT POWER WHEN DIFFICULTIES WERE EXPERIENCED WITH THE MOISTURE SEPARATOR REHEATER (MSR) AND THE HIGH PRESSURE FEEDWATER HEATER SYSTEMS AND THEIR VENT AND DRAIN SYSTEM. IN RESPONSE TO THESE DIFFICULTIES, OPERATORS BEGAN REDUCING POWER TO REMOVE THE MSRS FROM SERVICE. AT THIS POINT OPERATORS NOTED AN INCREASE IN WATER LEVEL IN THE REHEATER DRAIN TANK, AN INCREASE IN OFF-GAS SYSTEM FLOW, AND A DECREASE IN THE MAIN CONDENSER VACUUM. IN RESPONSE OPERATORS CONTINUED TO DECREASE REACTOR POWER. AT APPROXIMATELY TWENTY-FIVE PERCENT REACTOR POWER, WHILE REMOVING ONE OF THE TURBINE DRIVEN REACTOR FEEDWATER PUMPS FROM SERVICE, AN OPERATOR FAILED TO MATCH THE MANUAL FEEDWATER CONTROL TO THE AUTOMATIC FEEDWATER CONTROL PRIOR TO TRANSFERRING THE PUMP TO MANUAL. THIS CAUSED REACTOR WATER LEVEL TO INCREASE AND APPROACH THE HIGH WATER LEVEL SCRAM SETPOINT THEREFORE. OPERATORS INITIATED A MANUAL SCRAM. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO OPERATOR ERROR DURING THE TRANSFER OF FEEDWATER PUMP CONTROL. BECAUSE THE OPERATOR RECOGNIZED HIS ERROR AND IDENTIFIED IT TO HIS SUPERVISION, NO CORRECTIVE ACTION IS REQUIRED.

[34]   CONNECTICUT YANKEE   DOCKET 50-213   LER 89-013
DESIGN DEFICIENCY INDENTIFIED IN CHARGING PUMP LUBE OIL COOLER CIRCUIT.
EVENT DATE: 081189   REPORT DATE: 090889   NSSS: WE   TYPE: PWR

(NSIC 215203) ON 8/11/89, AT APPROXIMATELY 1730 HOURS, WITH THE PLANT IN MODE 1 AT 98% POWER, AN ENGINEERING ANALYSIS IDENTIFIED AN UNDERRATED CIRCUIT BREAKER IN CIRCUITRY PARALLEL WITH THE POWER SUPPLY TO THE 'B' CHARGING PUMP LUBE OIL COOLER FAN MOTOR. IT WAS DETERMINED THAT A FAIURE OF THIS CIRCUIT BREAKER COULD RENDER THE 'B' CHARGING PUMP INOPERATIVE DURING LOSS OF OFFSITE POWER SCENARIOS. THE ROOT CAUSE OF THIS EVENT WAS A DESIGN DEFICIENCY. CORRECTIVE ACTION INVOLVED SEPARATING THE UNDERRATED CIRCUIT BREAKER FROM THE POWER SUPPLY TO THE LUBE OIL COOLER FAN MOTOR. THE 'A' CHARGING PUMP WAS NOT AFFECTED BY THE ABOVE CONDITION. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(V)(D) SINCE A CONDITION EXISTED WHICH ALONE COULD HAVE PREVENTED THE FULFILLMENT OF A SAFETY FUNCTION OF A SYSTEM NEEDED TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT.

[35]   COOK 1   DOCKET 50-315   LER 89-004 REV 01
UPDATE ON CONTAINMENT TYPE B AND C LEAKAGE EXCEEDS L.C.O. VALUE DUE TO DEGRADATION OF ISOLATION VALVE SEATING SURFACES.
EVENT DATE: 032489   REPORT DATE: 083189   NSSS: WE   TYPE: PWR
VENDOR: ITT GRINNELL
ITT HAMMEL DAHL CONOFLOW

(NSIC 215161) THIS IS A SUPPLEMENTAL REPORT SUBMITTED TO PROVIDE ADDITIONAL INFORMATION REGARDING THE TYPE B AND C LEAK RATE TESTING AS REPORTED ON 4/21/89. ON 3/24/89, WITH THE REACTOR COOLANT SYSTEM 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 TECH SPEC 3.6.1.2.B. THE B AND C LEAK RATE TEST WAS APPROXIMATELY 50% COMPLETE AT THAT TIME. THE TOTAL AS-FOUND B AND C LEAK RATE WAS 3.08 LA. TEST DATA REVEALED THAT 2.7 LA WAS ATTRIBUTABLE TO THE LEAK RATES OF THE THREE VALVES. IT WAS ALSO DETERMINED THAT TWO VALVES HAD THE VALVE STEM PACKING REPLACED PRIOR TO OBTAINING AS-FOUND LEAK RATE DATA. THIS ACTION IS CONTRARY TO 10 CFR 50 APPENDIX J REQUIREMENTS FOR THIS VALVE DESIGN. THOSE CONTAINMENT ISOLATION VALVES THAT EXHIBITED LEAK RATES IN EXCESS OF ACCEPTANCE CRITERIA HAVE BEEN REPAIRED AND RETESTED TO ENSURE THE LEAK RATES ARE WITHIN ALLOWABLE LIMITS. THE AS-LEFT LEAK RATE FOR TYPE B AND C TESTING WAS 0.142 LA, WHICH IS WELL BELOW THE TECH SPEC LIMIT OF 0.60 LA.

[36] COOK 1 DOCKET 50-315 LER 89-009 REV 01
UPDATE ON REQUIRED POST MAINTENANCE TESTING NOT PERFORMED DUE TO PERSONNEL ERROR PRIOR TO ENTRY INTO A MODE FOR WHICH THE EQUIPMENT WAS REQUIRED TO BE OPERABLE.
EVENT DATE: 062089 REPORT DATE: 091489 NSSS: WE TYPE: PWR

(NSIC 215256) ON 5/2/89, WITH REACTOR CORE UNLOADED, MAINTENANCE WAS COMPLETED ON 1-CS-442-2 (THE REACTOR COOLANT PUMP SEAL WATER INJECTION CONTAINMENT ISOLATION CHECK VALVE). ASSOCIATED JOB ORDER WAS SUBMITTED FOR CLOSEOUT WITH THE UNDERSTANDING THAT POST MAINTENANCE TESTING, (APPENDIX J, TYPE C), REQUIRED FOLLOWING REPAIRS FOR VALVE OPERABILITY IN MODES 4 THROUGH 1, WOULD BE COMPLETED UNDER A SEPARATE TESTING JOB ORDER. THE UNIT ENTERED POWER OPERATION ON 6/20/89. ON 7/5/89 WITH THE UNIT OPERATING AT 32% POWER A JOB ORDER REVIEW REVEALED THAT NO DOCUMENTATION FOR TYPE C LEAKAGE RATE TEST EXISTED. UNIT WAS SHUT DOWN; TESTS WERE PERFORMED WITH SATISFACTORY RESULTS AND UNIT WAS RETURNED TO SERVICE ON 7/8/89. A DOCUMENT SEARCH OF APPROX. 4000 JOB ORDERS WAS CONDUCTED DURING UNIT SHUTDOWN. IT REVEALED ONE OTHER UNTESTED VALVE ON WHICH LEAKAGE RATE TESTING (NON-APPENDIX J) WAS REQUIRED AS RESULT OF REPAIR WORK (1-CTS-131W, WEST CONTAINMENT SPRAY TO UPPER COMPARTMENT RING HEADER CONTAINMENT ISOLATION CHECK VALVE). THIS VALVE ALSO TESTED SATISFACTORILY. THIS EVENT WAS CAUSED BY PERSONNEL ERROR IN THAT TESTING WAS NOT VERIFIED COMPLETE PRIOR TO DECLARING THE VALVES OPERABLE. A SIGNIFICANT CONTRIBUTING FACTOR WAS THAT THE TESTING JOB ORDERS WERE NOT KEYED TO MODE CHANGES; THUS, NO INDEPENDENT MEANS OF DETECTING OVERSIGHTS.

[37] COOK 2 DOCKET 50-316 LER 88-003 REV 04
UPDATE ON REPETITIVE VIOLATION OF ESF INSTRUMENTATION LIMITING CONDITIONS FOR OPERATION TOLERANCES DUE TO HIGHLY RESTRICTIVE ALLOWABLE VALUES.
EVENT DATE: 031188 REPORT DATE: 091189 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 1 (PWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215225) THIS REVISION IS BEING SUBMITTED TO REFLECT AN UPDATE ON THE RESULTS OF THE INCREASED FREQUENCY (MONTHLY) CALIBRATION CHECKS PERFORMED TO DATE. ON 3/11/88 AN EQUIPMENT TREND INVESTIGATION WAS BEING PERFORMED ON 4KV BUS LOSS OF VOLTAGE RELAYS AND THE 4KV BUS DEGRADED VOLTAGE RELAYS (EIIS/EK-27). THE 'AS FOUND' CONDITION OF THESE RELAYS DURING PAST CALIBRATION CHECKS HAS GENERALLY BEEN FOUND TO BE BEYOND THE TECH SPEC (T.S.) ALLOWABLE VALUES. EACH RELAY WAS ADJUSTED TO WITHIN ALLOWABLE VALUES AT THE TIME IT WAS DISCOVERED OUT OF SPECIFICATION. ALL RELAYS WERE FUNCTIONAL AND WOULD HAVE PERFORMED THE ESF FUNCTION, ALTHOUGH AT A SLIGHTLY DIFFERENT VOLTAGE THAN SPECIFIED IN TECH SPEC. AN ENGINEERING REVIEW HAS DETERMINED A PLUS OR MINUS 3% TOLERANCE (AS OPPOSED TO THE CURRENT 0.5%) TO BE ACCEPTABLE FOR THE LOSS OF VOLTAGE APPLICATION. THE DEGRADED VOLTAGE APPLICATION WILL ACCEPT A PLUS OR MINUS 1.5% TOLERANCE AND WILL REQUIRE INSTALLATION OF MORE ACCURATE UNDERVOLTAGE RELAYS (DESIGN CHANGE CURRENTLY UNDERWAY). A T.S. CHANGE REQUEST HAS BEEN SUBMITTED. AS STATED IN THE ORIGINAL LER, WE HAVE INCREASED THE CALIBRATION FREQUENCY FROM EVERY EIGHTEEN MONTHS TO MONTHLY.

[38] COOK 2 DOCKET 50-316 LER 89-014
REACTOR PROTECTION SYSTEM ACTUATION DUE TO MALFUNCTION OF CONTROL ROOM INSTRUMENTATION DISTRIBUTION INVERTER.
EVENT DATE: 081489 REPORT DATE: 091389 NSSS: WE TYPE: PWR
VENDOR: SOLID STATE CONTROLS, INC.

(NSIC 215257) ON 8/14/89 AT 1601 HOURS, A REACTOR PROTECTION SYSTEM (RPS) ACTUATION (REACTOR TRIP) OCCURRED WHEN OPERATORS TRANSFERRED THE CONTROL ROOM INSTRUMENTATION DISTRIBUTION (CRID) IV (VITAL BUS) INVERTER TO ITS NORMAL CLASS 1E POWER SUPPLY AND THE INVERTER FAILED. WHEN THE CRID IV INVERTER FAILED, A REACTOR TRIP SIGNAL WAS INITIATED DUE TO THE REACTOR COOLANT PUMP (RCP) CIRCUIT

BREAKER POSITION INDICATION OPEN (FED FROM CRID IV). PRIOR TO THE TRIP (AT APPROXIMATELY 1540 HOURS), THE CRID INVERTER HAD TRANSFERRED TO ITS ALTERNATE NON-CLASS 1E POWER SUPPLY AT THE SAME TIME THAT A CONTROL POWER FUSE HAD BLOWN ON POWER RANGE NUCLEAR INSTRUMENTATION SYSTEM (NIS) CHANNEL IV (N-44). SUBSEQUENT INVESTIGATION DETERMINED THAT THE CRID INVERTER FAILURE WAS DUE TO A FAILED SILICON CONTROLLED RECTIFIER (SCR) IN THE STATIC TRANSFER SWITCH. THIS ALSO RESULTED IN THE FAILURE OF FUSES AND POWER SUPPLIES IN VARIOUS COMPONENTS FED FROM THE CRID. THE FAULTED SCR'S WERE REPLACED AND THE CRID INVERTER DECLARED OPERABLE. ALL COMPONENTS FED FROM THE CRID WERE INSPECTED AND, WHERE NECESSARY, FUSES AND/OR POWER SUPPLIES WERE REPLACED.

[39] COOK 2 DOCKET 50-316 LER 89-015
FIRE DAMPERS INOPERABLE WITHOUT REQUIRED COMPENSATORY ACTION AS A RESULT OF AN ERRONEOUS INSPECTION PROCEDURE.
EVENT DATE: 082189 REPORT DATE: 091889 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 1 (PWR)

(NSIC 215307) ON 8/21/89 AN INOPERABLE OPEN FIRE DAMPER SUBJECT TO THE CONSTRAINTS OF TECH SPEC 3.7.10 WAS FOUND FOR WHICH THE ACTION STATEMENT REQUIREMENTS HAD NOT BEEN IMPLEMENTED. IT HAD BEEN INSPECTED JULY 18, 1989, AND WAS IDENTIFIED ON A TASK SHEET DEVELOPED FOR PREVENTIVE MAINTENANCE ON DAMPERS WHICH WERE NOT SUBJECT TO THE TECH SPEC. HOWEVER, AN ERROR WAS MADE IN IDENTIFYING NON-TECH SPEC DAMPERS DURING DEVELOPMENT OF THE TASK SHEET AND SOME TECH SPEC DAMPERS WERE INCORRECTLY INCLUDED. TECH SPEC DAMPERS WERE ALREADY INCLUDED IN A QUALITY CONTROL PROCEDURE WHICH CONTAINED NECESSARY INSTRUCTIONS. THE MAINTENANCE INSPECTION OVERALPPED THE QUALITY CONTROL INSPECTION WITHOUT CONTAINING PERTINENT INFORMATION NEEDED FOR TECH SPEC DAMPERS. ALL TASK SHEET JOB ORDERS WERE REVIEWED. TWO UNIT 1 DAMPERS WERE FOUND WHICH HAD ALREADY BEEN REPAIRED WITHOUT ACTION STATEMENT IMPLEMENTATION. FOUR MORE UNIT 1 DAMPERS WERE FOUND WITH OPEN JOB ORDERS. AFTER IDENTIFICATION OF THE TECH SPEC DAMPERS, THE ACTION STATEMENT WAS ENTERED AND REPAIRS MADE. THE TASK SHEET WAS REVISED TO DELETE TECH SPEC DAMPERS AS ORIGINALLY INTENDED.

[40] COOPER DOCKET 50-298 LER 89-020
UNPLANNED GROUP ISOLATIONS DURING DESIGN CHANGE ACCEPTANCE TESTING DUE TO WIRING ERRORS.
EVENT DATE: 052989 REPORT DATE: 062889 NSSS: GE TYPE: BWR

(NSIC 215398) ON 5/29/89, AT 9:11 A.M. DURING ACCEPTANCE TESTING OF A DESIGN CHANGE ASSOCIATED WITH THE DIESEL GENERATOR AUTOMATIC STARTING CIRCUIT, POWER TO THE 4160V 1F CRITICAL BUS WAS LOST. THIS RESULTED IN UNPLANNED ACTUATIONS OF GROUP 2, 3, 6, AND 7 ISOLATIONS (PRIMARY CONTAINMENT, REACTOR WATER CLEANUP (RWCU), SECONDARY CONTAINMENT, INCLUDING STANDBY GAS TREATMENT SYSTEM (SGTS) INITIATION, AND REACTOR WATER SAMPLE, RESPECTIVELY). AT THE TIME OF THIS EVENT, THE PLANT WAS SHUT DOWN FOR THE 1989 REFUELING/MAINTENANCE OUTAGE. THE ROOT CAUSE OF THIS EVENT IS A RESULT OF BOTH DESIGN AND PROGRAMMATIC INADEQUACIES. WITH REGARD TO DESIGN, WIRING CHANGES MADE DURING IMPLEMENTATION OF THE DESIGN CHANGE (DC) INADVERTENTLY DEFEATED A TEST SWITCH THAT IS USED TO PREVENT 4160V BREAKER 1FA ACTUATION DURING RELAY TESTING. THE PROGRAMMATIC DEFICIENCY IS ASSOCIATED WITH THE CORRECTNESS AND LEVEL OF DETAIL PROVIDED IN EXISTING DESIGN DRAWINGS. IN THIS CASE, DESIGN DRAWINGS USED FOR DC DEVELOPMENT HAD NOT YET BEEN VERIFIED AS PART OF THE DESIGN BASIS RECONSTITUTION EFFORT AND WERE NOT FIELD VERIFIED PRIOR TO DC IMPLEMENTATION. CORRECTIVE ACTION WAS IMMEDIATELY TAKEN TO RESTORE POWER TO THE 4160V BUS. SUBSEQUENTLY, THE GROUP ISOLATIONS WERE RESET AND THE NECESSARY SYSTEMS RESTORED TO OPERATION. THE WIRING PROBLEM WAS CORRECTED AND ACCEPTANCE TESTING WAS SATISFACTORILY COMPLETED.

[41] COOPER DOCKET 50-298 LER 89-024
DIESEL GENERATOR INJECTION PUMP PEDESTAL CRACK WHICH WAS CAUSED BY UNUSUAL HYDRAULIC FORCES FOUND DURING TROUBLESHOOTING SUBSEQUENT TO SURVEILLANCE TESTING.
EVENT DATE: 073189 REPORT DATE: 083189 NSSS: GE TYPE: BWR
VENDOR: COOPER-BESSEMER CO.

(NSIC 215251) ON 7/31/89, DURING PERFORMANCE OF MONTHLY DIESEL GENERATOR (DG) SURVEILLANCE TESTING, A SMALL AMOUNT OF OIL WAS NOTED TO BE EMANATING FROM A CRACK IN THE SIDE OF THE NUMBER 3 RIGHT CYLINDER INJECTION PUMP PEDESTAL CASTING ON DIESEL GENERATOR (DG) NO. 1. OPERABILITY OF THE DG WAS NOT AFFECTED; THEREFORE, THE SURVEILLANCE TEST WAS COMPLETED SATISFACTORILY. UPON INSPECTION OF THE PEDESTAL INTERNALS, THE CYLINDER SLEEVE WAS FOUND MISPOSITIONED AND ITS LOCKING DEVICE WAS FOUND SHEARED. THE PLANT WAS IN OPERATION AT 100% POWER. THE DIESEL ENGINE IS A MODEL KSV-16-T ENGINE, MANUFACTURED BY COOPER-BESSEMER. TROUBLESHOOTING LED TO THE CONCLUSION THAT THE PEDESTAL CRACK WAS CAUSED BY UNUSUAL HYDRAULIC FORCES CREATED BY LUBE OIL "TRAPPED" IN THE PEDESTAL CASTING. THE CAM FOLLOWER AND PUSHROD ASSEMBLY IN THE PEDESTAL CYLINDER WAS FUNCTIONING AS A POSITIVE DISPLACEMENT PUMP, ATTEMPTING TO PUMP THIS OIL. INTERNAL DRAIN HOLES IN THE PEDESTAL, WHICH ALLOW THIS OIL TO DRAIN TO THE CAMSHAFT TROUGH, WERE EFFECTIVELY "BLOCKED" DUE TO THE TROUGH BEING "FLOODED". THIS CONDITION WAS CAUSED BY A STREAM OF OIL FROM A THROUGH-WALL DEFECT IN A POROUS AREA ADJACENT TO AN OIL SUPPLY PORT IN THE ENGINE BLOCK CASTING. A SECTION OF 316 STAINLESS STEEL TUBING WAS INSERTED INTO THE 3/8 INCH DIAMETER BEARING OIL SUPPLY PORT, RESTORING ITS INTEGRITY.

[42] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-017 REV 01
UPDATE ON ENTRY INTO HOT SHUTDOWN FORCED BY TECHNICAL SPECIFICATIONS.
EVENT DATE: 080887 REPORT DATE: 090589 NSSS: BW TYPE: PWR

(NSIC 215200) ON 8/8/87, CRYSTAL RIVER UNIT 3 WAS SHUT DOWN TO HOT STANDBY CONDITIONS TO ALLOW MAINTENANCE ON A FAILED CONTROL ROD DRIVE MECHANISM (CRDM). OPERATORS HAD SECURED THE STEAM SUPPLY TO THE TURBINE DRIVEN EMERGENCY FEEDWATER PUMP (EFP) DURING COOLDOWN IN ACCORDANCE WITH PLANT PROCEDURES. PLANT TECH SPECS REQUIRE TWO OPERABLE EFPS DURING POWER OPERATION, STARTUP, AND HOT STANDBY OPERATING MODES. PROBLEMS ENCOUNTERED DURING CRDM REPAIRS DELAYED PLANT HEATUP. THIS PREVENTED RESTORATION OF THE PUMP TO OPERABLE STATUS WITHIN THE TIME ALLOWED BY TECH SPECS. THEREFORE, OPERATORS COOLED THE PLANT TO HOT SHUTDOWN CONDITIONS IN ORDER TO COMPLY WITH APPLICABLE TECH SPEC ACTION STATEMENTS. THE UTILITY HAS OBTAINED DETAILED ANALYSIS OF EMERGENCY FEEDWATER FLOW REQUIREMENTS DURING PLANT COOLDOWN. BASED ON THE RESULTS OF THIS ANALYSIS, UTILITY PERSONNEL HAVE SUBMITTED A TECH SPEC REVISION (TECH SPEC IMPROVEMENT PROGRAM SUBMITTAL) WHICH WILL ALLOW THE TURBINE DRIVEN EFP TO REMAIN OPERABLE UNTIL MODE 4 WHERE THE PUMP IS NOT REQUIRED.

[43] CRYSTAL RIVER 3 DOCKET 50-302 LER 89-029
PERSONNEL ERROR RESULTS IN OPERATION WITH TWO REACTOR COOLANT SYSTEM LEAKAGE DETECTION SYSTEMS INOPERABLE IN VIOLATION OF TECH SPECS.
EVENT DATE: 081289 REPORT DATE: 091389 NSSS: BW TYPE: PWR

(NSIC 215252) ON 8/12/89, WHILE OPERATING IN MODE 1 (POWER OPERATION) AT 97% RATED THERMAL POWER, A NON-LICENSED ASSISTANT NUCLEAR OPERATOR (ANO) SHUT DOWN THE NORMAL DUTY VACUUM PUMP FOR THE CONTAINMENT ATMOSPHERE RADIOACTIVITY MONITOR (RM-A6) AND DID NOT START THE BACKUP VACUUM PUMP. THIS RENDERED TWO OF THE THREE REACTOR COOLANT SYSTEM LEAKAGE DETECTION SYSTEMS REQUIRED BY TECH SPEC 3.4.6.1 INOPERABLE, A CONDITION PROHIBITED BY PLANT TECH SPECS. THE NORMAL DUTY VACUUM PUMP WAS RETURNED TO SERVICE WITHIN APPROXIMATELY 10 HOURS. COMPLIANCE WITH TECH SPEC 3.4.6.1 WAS THEN ESTABLISHED. THE ERROR WAS DETECTED AND THE EVENT DETERMINED TO BE REPORTABLE AS A LICENSEE EVENT REPORT ON 8/14/89. THE CAUSE OF THIS EVENT WAS THE ASSIGNMENT OF AN UNTRAINED AND UNQUALIFIED PERSON TO PERFORM

THE TASK. A MALFUNCTION OF THE LOW FLOW ALARM ON THE RADIATION MONITORING PANEL CONTRIBUTED TO THE EVENT. TO PRECLUDE RECURRENCE OF THIS EVENT, PLANT OPERATING AND TRAINING PROCEDURES WILL BE REVISED. THE MALFUNCTIONING LOW FLOW ALARM WAS REPAIRED.

[44] CRYSTAL RIVER 3 DOCKET 50-302 LER 89-030
DELIVERY, INSTALLATION, AND ACCEPTANCE OF INCORRECT IMPELLER LEADS TO ENGINEERED SAFEGUARDS PUMP INOPERABILITY, OPERATION OUTSIDE PLANT DESIGN BASIS, AND PLANT SHUTDOWN.
EVENT DATE: 082689 REPORT DATE: 092689 NSSS: BW TYPE: PWR

(NSIC 215356) ON 8/24/89, CRYSTAL RIVER UNIT 3 WAS OPERATING AT 95% RATED THERMAL POWER, GENERATING 823 MEGAWATTS ELECTRIC. TESTING OF THE "B" EMERGENCY NUCLEAR SERVICES SEAWATER PUMP (RWP-2B) WAS IN PROGRESS. IT WAS DETERMINED THAT THE PUMP DISCHARGE PRESSURE AND FLOW WERE LESS THAN REQUIRED AND THE PUMP WAS DECLARED INOPERABLE. RWP-2B IS AN ENGINEERED SAFEGUARDS PUMP, REQUIRED TO BE OPERABLE BY CRYSTAL RIVER UNIT 3 TECH SPECS. A PLANT SHUTDOWN WAS BEGUN. AT 1115 ON 8/26 THE PLANT ENTERED HOT STANDBY, COMPLETING A SHUTDOWN REQUIRED BY THE TECH SPECS. THE ROTATING ASSEMBLY OF RWP-2B WAS EXAMINED ON 9/4/89, AND FOUND TO BE EQUIPPED WITH AN INCORRECT IMPELLER. THE IMPELLER WAS INSTALLED IN 4/89. THIS WAS A CONDITION OUTSIDE THE PLANT'S DESIGN BASIS. THE CONDITION WAS THE RESULT OF TWO EVENTS, INSTALLATION OF, AND FAILURE TO DETECT THE INCORRECT IMPELLER. THE INCORRECT IMPELLER HAS BEEN REMOVED FROM THE PUMP AND THE ORIGINAL IMPELLER WAS SENT TO THE VENDOR FOR REPAIR. THE PLANT WILL REMAIN SHUT DOWN UNTIL RWP-2B IS RETURNED TO SERVICE AND SUCCESSFULLY TESTED. IMPROVED METHODS FOR DETERMINING PUMP PERFORMANCE IN THE RW SYSTEM ARE BEING INVESTIGATED. THE PROCUREMENT PROCESS WHICH LED TO OBTAINING THE INCORRECT IMPELLER FROM THE PUMP VENDOR WAS INVESTIGATED.

[45] CRYSTAL RIVER 3 DOCKET 50-302 LER 89-031
FAILURE OF A 480 VOLT ENGINEERED SAFEGUARDS TRANSFORMER CAUSES TEMPORARY INTERRUPTION OF DECAY HEAT COOLING AND A PLANT OPERATIONAL MODE CHANGE.
EVENT DATE: 082889 REPORT DATE: 092789 NSSS: BW TYPE: PWR
VENDOR: I-T-E CIRCUIT BREAKER
ROTO HAMMER COMPANY, INC.

(NSIC 215357) CRYSTAL RIVER UNIT 3 WAS IN MODE 5 (COLD SHUTDOWN) TO PERFORM REPAIRS TO A NUCLEAR SERVICES RAW WATER PUMP WITH THE B DECAY HEAT TRAIN (DH) REMOVED FROM SERVICE. THE A DH TRAIN WAS OPERATING AND THE REACTOR COOLANT SYSTEM WAS FILLED AND INTACT WITH AN OPERABLE STEAM GENERATOR. AT 0237 ON 8/28/89, THE A 480 VOLT ENGINEERED SAFEGUARDS (ES) STEPDOWN TRANSFORMER FAULTED CAUSING DH CLOSED CYCLE COOLING PUMP A (DCP-1A) TO DE-ENERGIZE. LOSS OF DCP-1A REMOVED THE COOLING FOR DH TRAIN A, RENDERING THIS TRAIN INOPERABLE. AT 0256, DH TRAIN B AND ITS SUPPORT SYSTEMS WERE STARTED. THE A AND B 480 VOLT ES BUSES WERE THEN CROSS-TIED TO PROVIDE AC POWER TO THE A BUS. THE LOSS OF DH COOLING CAUSED AN INCREASE IN RCS TEMPERATURE AND AN INADVERTENT ENTRY INTO MODE 4 (HOT SHUTDOWN). THE TRANSFORMER HAS BEEN REPLACED. A FAILURE ANALYSIS WILL BE PERFORMED TO DETERMINE THE CAUSE OF THE TRANSFORMER FAILURE. DURING THE EVENT, OPERATORS ATTEMPTED TO OPERATE THE B TRAIN DH SUCTION ISOLATION VALVE (DHV-40) FROM THE CONTROL ROOM. THIS VALVE FAILED BECAUSE THE NUT SECURING THE LINKAGE BETWEEN THE MOTOR OPERATOR AND THE VALVE STEM HAD BACKED OFF FROM NORMAL OPERATION. DHV-40 HAS BEEN REPAIRED AND THE NUT ON THE LINKAGE HAS BEEN STAKED TO PREVENT IT FROM BACKING OFF. THE REDUNDANT VALVE, DHV-39, WILL BE STAKED.

[46] CRYSTAL RIVER 3 DOCKET 50-302 LER 89-032
DESIGN ENGINEERING DEFICIENCY RESULTS IN STATION BATTERY INOPERABILITY AND OPERATION PROHIBITED BY PLANT TECH SPECS.
EVENT DATE: 083189 REPORT DATE: 092989 NSSS: BW TYPE: PWR

(NSIC 215358) AT 1300 ON 8/31/89 WHILE CRYSTAL RIVER UNIT 3 WAS IN MODE 5 (COLD SHUTDOWN), IT WAS DETERMINED THAT, DUE TO A DESIGN ENGINEERING DEFICIENCY, THE 'B' STATION BATTERY WAS INOPERABLE DUE TO INADEQUATE TESTING LOAD PROFILE. THIS CAUSED THE PLANT TO BE OPERATED IN A CONDITION PROHIBITED BY PLANT TECH SPECS. THE LOAD PROFILE USED TO DEVELOP SURVEILLANCE TEST ON THE BATTERY HAD ASSUMED GENERATOR AIR SIDE SEAL OIL BACKUP PUMP (TBP-10) WOULD NOT START UNTIL SIXTY MINUTES FOLLOWING A LOSS OF OFF-SITE POWER. VENDOR INFORMATION INDICATES IT COULD START IN APPROXIMATELY ONE TO THREE MINUTES FOLLOWING A LOSS OF OFF-SITE POWER AND LOSS OF VACUUM. DURING SUBSEQUENT REVIEW, IT WAS ALSO DISCOVERED THAT ONE BANK (3B2) OF THE 'B' BATTERY WAS OVERLOADED DUE TO NOT BALANCING THE 125 VOLT LOADS BETWEEN THE TWO BANKS. ENGINEERING CALCULATION REVIEWS WILL NOW EMPHASIZE ACCURACY OF INPUTS TO THE CALCULATION. THE TBP-10 LOAD HAS BEEN REMOVED FROM THE STATION BATTERY AND WILL BE POWERED FROM A TEMPORARY BATTERY. THE 125V LOADS ON THE B BATTERY WILL BE REASSIGNED TO ACHIEVE A BETTER BALANCE. THESE ACTIONS WILL BE PERFORMED TO INSURE OPERATIONS WITHIN THE DESIGN BASIS AND WILL BE COMPLETED PRIOR TO START-UP.

[47] DAVIS-BESSE 1 DOCKET 50-346 LER 89-012
UNPLANNED RELEASE OF 1700 GALLONS FROM THE CLEAN WASTE RECEIVER TANK 1-1 DUE TO PROCEDURAL DEFICIENCIES.
EVENT DATE: 071489 REPORT DATE: 090189 NSSS: BW TYPE: PWR

(NSIC 215133) ON JULY 14, 1989, AT 1735 HOURS, AN UNPLANNED RELEASE OCCURRED FROM CLEAN WASTE MONITOR TANK (CWMT) 1.1 WHILE INTENDING TO RELEASE FROM CWMT 1.2. THIS RELEASE TO THE COLLECTION BOX LASTED FOR ABOUT TEN MINUTES THERE WERE 1700 GALLONS AND 2.14E.3 CI OF ENTRAINED NOBLE GASES RELEASED THE CAUSE WAS AN ERROR IN PROCEDURE DB.OP.03011, RADIOACTIVE LIQUID BATCH RELEASE, WHICH DIRECTED THE OPERATOR TO THE WRONG COLUMN IN PROCEDURE SP 1104.83, CLEAN WASTE MONITOR TANK OPERATIONS. THIS CAUSED FOUR VALVES TO BE OPENED THAT OTHERWISE WOULD BE CLOSED. WHEN THE #2 TRANSFER PUMP WAS STARTED, CWMT 1.1 WAS BEING RELEASED. THE APPROPRIATE SAMPLING HAD BEEN CONDUCTED IN ANTICIPATION OF RELEASING FROM CWMT 1.2. HOWEVER, BECAUSE THE RELEASE FROM CWMT 1.1 WAS UNPLANNED, THE SURVEILLANCE REQUIREMENT 4.11.1.1 TO SAMPLE THE TANK PRIOR TO THE RELEASE WAS NOT SATISFIED. THIS IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B) AS A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS. THE RELEASE WAS BELOW THE LIMITS OF 10CFR50 73(A)(2)(VIII)(B). TEMPORARY CHANGE TA89.4775 TO DB.OP.03011 WAS ISSUED JULY 15, 1989. A RE-VALIDATION OF THE ENTIRE PROCEDURE WAS COMPLETED ON AUGUST 12, 1989. CHANGES FROM THIS VALIDATION WERE IMPLEMENTED ON AUGUST 21, 1989, AS TA89.9513. THE UNPLANNED RELEASE WILL ALSO BE INCLUDED IN THE NEXT SEMI-ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT.

[48] DIABLO CANYON 1 DOCKET 50-275 LER 89-007
MISSED SURVEILLANCE FOR CONTAINMENT ISOLATION VALVE DUE TO AN INADEQUATE PROCEDURE.
EVENT DATE: 070289 REPORT DATE: 091389 NSSS: WE TYPE: PWR

(NSIC 215244) ON JULY 2, 1989, THE SURVEILLANCE REQUIREMENT OF TECH SPEC 4.6.1.1.A INCLUDING THE ALLOWED 25% EXTENSION OF TECH SPEC 4.0.2 WAS EXCEEDED FOR CONTAINMENT ISOLATION VALVE NSS-1-9356B. NSS-1-9356A, A REACTOR COOLANT HOT LEG SAMPLE ISOLATION AND CONTAINMENT ISOLATION VALVE INSIDE CONTAINMENT, WAS DECLARED INOPERABLE ON 5/24/89 DUE TO EXCESSIVE PACKING LEAKAGE. VALVE NSS-1-9356B, THE OUTSIDE CONTAINMENT ISOLATION VALVE, WAS DEACTIVATED TO ISOLATE THE PENETRATION IN ACCORDANCE WITH TECH SPEC 3.6.3.B. REPAIR OF VALVE NSS-1-9356A IS PLANNED FOR THE NEXT UNIT 1 MODE 5 (COLD SHUTDOWN) OUTAGE. ON 8/16/89, OPERATIONS PERSONNEL DETERMINED THAT TECH SPEC 4.6.1.1.A REQUIRES A 31 DAY VERIFICATION OF THE CLOSED POSITION FOR AUTOMATIC CONTAINMENT ISOLATION VALVES DEACTIVATED IN ACCORDANCE WITH TECH SPEC 3.6.3.B. THE VALVE WAS VERIFIED CLOSED AND WAS ADDED TO THE CONTROL ROOM CONDITIONAL SURVEILLANCE PROGRAM. SURVEILLANCE TEST PROCEDURE I-1D, "ROUTINE MONTHLY CHECKS REQUIRED BY LICENSES", WILL BE REVISED TO REQUIRE THAT

ANY VALVE CLOSED TO SATISFY AN ACTION ASSOCIATED WITH TECH SPEC ACTION STATEMENT 3.6.3 TO BE VERIFIED CLOSED EVERY 31 DAYS.

[49] DIABLO CANYON 2 DOCKET 50-323 LER 88-016 REV 01
UPDATE ON SAFETY INJECTION PUMP 2-2 FAILED AFTER THE SUCTION VALVE WAS CLOSED DUE TO THE EFFECTS FROM AN INADEQUATE ASSESSMENT OF CLEARANCE REMOVAL.
EVENT DATE: 111188 REPORT DATE: 091389 NSSS: WE TYPE: PWR
VENDOR: PACIFIC PUMPS

(NSIC 215226) ON 11/11/88, AT APPROX. 0040 PST, WHILE FILLING THE ACCUMULATORS, THE ASSISTANT CONTROL ROOM OPERATOR (ACO) NOTICED THAT THE FILL WAS NOT PROGRESSING AS FAST AS EXPECTED. AT 0050 PST, THE DISCHARGE PRESSURE FROM SAFETY INJECTION PUMP 2-2 AND MOTOR CURRENT WERE NOTED AS LOW. THE PUMP WAS IMMEDIATELY SECURED. THE SENIOR CONTROL OPERATOR (SCO) WAS DISPATCHED FOR A LOCAL CHECK OF THE PUMP. AT APPROXIMATELY 0055 PST, THE SCO REPORTED THAT THE SUCTION VALVE WAS CLOSED AND THAT SOME OF THE PAINT ON THE PUMP WAS BURNT. THE PUMP WAS REPLACED, TESTED SATISFACTORILY, AND RETURNED TO SERVICE BY 11/23/88. THIS EVENT WAS CAUSED BY PERSONNEL ERROR, COGNITIVE, IN THAT THE SUCTION VALVE BREAKER WAS RACKED IN WITH THE CONTROL ROOM VALVE CONTROL SWITCH IN THE CLOSED POSITION DUE TO AN INADEQUATE REVIEW OF A CLEARANCE. THE VALVE CLOSED CUTTING OFF SUCTION FLOW TO SI PUMP 2-2. APPLICABLE CLEARANCE AND TAGGING PROCEDURES WERE REVISED AND A NEW PROCEDURE WRITTEN TO PROVIDE GUIDANCE DURING FILLING OPERATIONS. OPERATORS WERE COUNSELED WITH REGARD TO PROPER VALVE AND SWITCH ALIGNMENT WITH RETURNING EQUIPMENT TO SERVICE.

[50] DIABLO CANYON 2 DOCKET 50-323 LER 89-006
LEAKAGE PAST PRESSURIZER SAFETY RELIEF VALVE CAUSES TEMPERATURE RISE.
EVENT DATE: 032089 REPORT DATE: 092689 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE

(NSIC 215372) THIS VOLUNTARY LER IS BEING SUBMITTED FOR INFORMATION PURPOSES ONLY AS DESCRIBED IN ITEM 19 OF SUPPLEMENT NUMBER 1 TO NUREG-1022. ON 3/20/89, A PRESSURIZER SAFETY RELIEF VALVE TAILPIPE TEMPERATURE MONITOR ALARMED ON HIGH TEMPERATURE. INVESTIGATION DETERMINED THAT THE TEMPERATURE RISE WAS DUE TO LEAKAGE ("WEEPING" AND BURPING") PAST PRESSURIZER SAFETY RELIEF VALVE 8010A, WHICH ALLOWED REACTOR COOLANT LEAKAGE TO THE PRESSURIZER RELIEF TANK. THE LEAKAGE WAS WITHIN THE ALLOWABLE 10 GPM TECH SPEC 3.4.6.2 LIMIT. ON 4/8/89, UNIT 2 WAS SHUTDOWN TO REPLACE VALVE 8010A. FOLLOWING VALVE REPLACEMENT, OFFSITE TESTING OF VALVE 8010A IDENTIFIED SEVERAL POSSIBLE SIGNIFICANT CONTRIBUTORS TO VALVE LEAKAGE. PG&E WILL PERFORM SEVERAL ACTIONS AND INVESTIGATIONS TO FURTHER STUDY POSSIBLE CONTRIBUTING EFFECTS TO SAFETY VALVE LEAKAGE. IF ADDITIONAL INFORMATION IS OBTAINED FROM THE INVESTIGATION REGARDING THE CAUSE OF VALVE LEAKAGE, IT WILL BE REPORTED IN A SUPPLEMENTAL REPORT.

[51] DIABLO CANYON 2 DOCKET 50-323 LER 89-008
MANUAL REACTOR TRIP DUE TO REACTOR COOLANT PUMP ELECTRICAL FAULT DUE TO A FAILURE OF A LOAD SIDE BOLTED TERMINATION.
EVENT DATE: 082889 REPORT DATE: 092789 NSSS: WE TYPE: PWR
VENDOR: AMERACE CORP.

(NSIC 215359) ON 8/28/89, AT 2057 PDT, WITH UNIT 2 IN MODE 1 AT 100% POWER, OPERATORS INITIATED A MANUAL REACTOR TRIP AFTER OBSERVING ELECTRICAL GROUND ALARMS FOR REACTOR COOLANT PUMP (RCP) 2-1, CIRCULATING WATER PUMPS (CWPS) 2-1 AND 2-2, AND THE AUXILIARY TRANSFORMER, AND ELEVATED, FLUCTUATING MOTOR CURRENT FOR RCP 2-1. OPERATORS TRIPPED THE REACTOR AND THEN TRIPPED RCP 2-1. FEEDER GROUND ALARMS FOR RCP 2-1 AND BOTH CWPS CLEARED. AT 2115 PDT, THE UNIT WAS STABILIZED IN MODE 3 WITH AN RCS TEMPERATURE OF APPROXIMATELY 525F. IN ACCORDANCE WITH 10 CFR 50.72(B)(2)(II) A 4-HOUR NON-EMERGENCY REPORT WAS COMPLETED AT 2205 PDT ON

8/??/8?. THE EVENT WAS CAUSED BY AN INADEQUATE ELECTRICAL CONNECTION ON RCP 2-1. ALL 12KV ELECTRICAL CONNECTORS, INCLUDING THE FAILED CONNECTOR, WERE REPLACED ON RCP 2-1. ALL OTHER 12KV CONNECTORS FOR UNIT 2 RCPS WERE EXAMINED TO THE EXTENT PRACTICABLE WITH NO ABNORMALITIES FOUND.

[52] DRESDEN 2 DOCKET 50-237 LER 89-021
INADVERTENT GROUP V PRIMARY CONTAINMENT ISOLATION DUE TO WIRE LUG FAILURE.
EVENT DATE: 080989 REPORT DATE: 090689 NSSS: GE TYPE: BWR
VENDOR: AMP SPECIAL INDUSTRIES

(NSIC 215205) AT 0500 HOURS ON 8/9/89 WITH UNIT 2 AT 1% RATED CORE THERMAL POWER, THE UNIT 2 NUCLEAR STATION OPERATOR (NSO) WAS PREPARING AN EQUIPMENT OUT OF SERVICE (OOS) FOR THE MAIN STEAM ISOLATION VALVES (MSIVS). AS THE NSO WAS MOVING LEAD WIRES BEHIND CONTROL ROOM PANEL 902-3 IN ORDER TO READ THE LABELS FOR THE MSIV CONTROL FUSES BEING TAKEN OUT OF SERVICE, A WIRE LUG CONNECTOR INADVERTENTLY FAILED RESULTING IN AN OPEN CONTROL POWER CIRCUIT FOR THE PRIMARY CONTAINMENT GROUP I AND ISOLATION CONDENSER ISOLATION LOGIC. THE INBOARD MSIVS HAD BEEN CLOSED PRIOR TO THIS EVENT; HOWEVER, A HALF PRIMARY CONTAINMENT GROUP I ISOLATION AND A COMPLETE PRIMARY CONTAINMENT GROUP V ISOLATION RESULTED FROM THE OPEN CIRCUIT. BECAUSE THE GROUP V ISOLATION RESULTED IN AUTOMATIC CLOSURE OF THE ISOLATION CONDENSER ISOLATION VALVES, THE ISOLATION CONDENSER WAS DECLARED INOPERABLE. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE COMPONENT FAILURE OF THE WIRE LUG. THE IMMEDIATE CORRECTIVE ACTIONS WERE TO VERIFY THAT ACTUAL GROUP V INITIATING CONDITIONS (ISOLATION CONDENSER LINE BREAK) HAD NOT OCCURRED AND TO INITIATE REPAIR OF THE WIRE LUG. REPAIRS WERE COMPLETED AT 0630 HOURS ON 8/9/89 AND THE ISOLATION CONDENSER WAS THEN DECLARED OPERABLE. SAFETY SIGNIFICANCE WAS MINIMUM BECAUSE THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM AND THE RELIEF VALVES WERE OPERABLE.

[53] DRESDEN 2 DOCKET 50-237 LER 89-023
POSSIBLE SINGLE FAILURE LOSS OF UNIT 2 ACAD/CAM OR UNIT 3 CAM DUE TO A DESIGN DEFICIENCY.
EVENT DATE: 081189 REPORT DATE: 091189 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 215231) ON 8/11/89, DURING NORMAL UNIT 2 AND UNIT 3 OPERATION AT 68% AND 99% RATED CORE THERMAL POWER RESPECTIVELY, IT WAS DETERMINED THAT A SINGLE FAILURE COULD RENDER THE UNIT 2 ATMOSPHERIC CONTAINMENT ATMOSPHERE DILUTION (ACAD) SYSTEM OR CONTAINMENT ATMOSPHERE MONITORING (CAM) SYSTEM OR THE UNIT 3 CAM SYSTEM INOPERABLE. THE ROOT CAUSE HAS BEEN ATTRIBUTED TO A DESIGN DEFICIENCY IN THE AC POWER SUPPLY TO THESE SYSTEMS. ACAD AND CAM OPERATING PROCEDURES HAVE HAD TEMPORARY PROCEDURE CHANGES IMPLEMENTED TO INFORM OPERATORS OF THE LOSS OF THESE SYSTEMS DURING THE POSTULATED EVENTS DESCRIBED. ENGINEERING IS EVALUATING POTENTIAL FIXES TO THE DESIGN DEFICIENCY. A SIMILAR PREVIOUS OCCURRENCE WAS REPORTED IN LER 89-013, POSSIBLE SINGLE FAILURE LOSS OF BOTH STANDBY GAS TREATMENT SYSTEMS DUE TO A DESIGN DEFICIENCY. THIS REPORT IS BEING SUBMITTED TO PROVIDE ADDITIONAL INFORMATION REGARDING A "COURTESY" ENS PHONE CALL MADE ON 8/11/89, AT 1420 HOURS.

[54] DRESDEN 2 DOCKET 50-237 LER 89-022
HIGH PRESSURE COOLANT INJECTION SYSTEM INOPERABLE DUE TO ROOM COOLER BROKEN DRIVE BELTS.
EVENT DATE: 082789 REPORT DATE: 092289 NSSS: GE TYPE: BWR

(NSIC 215299) AT 0015 HOURS ON 8/27/89 DURING NORMAL UNIT 2 OPERATION AT 93% RATED CORE THERMAL POWER, INCREASING HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM ROOM AMBIENT TEMPERATURES WERE OBSERVED. UPON VISUAL INSPECTION, IT WAS DISCOVERED THAT THE HPCI ROOM COOLER DRIVE BELTS HAD BROKEN. THE HPCI SYSTEM WAS

THEN DECLARED INOPERABLE. THE IMMEDIATE CORRECTIVE ACTIONS WERE TO REPLACE THE DRIVE BELTS. IN ADDITION, LONG TERM CORRECTIVE ACTIONS HAVE BEEN INITIATED TO PREVENT RECURRENCE. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS DETERMINED TO BE MINIMAL SINCE THE HPCI SYSTEM REMAINED CAPABLE OF AUTOMATICALLY INITIATING AND FULFILLING ITS DESIGN BASIS OPERATIONAL REQUIREMENTS BEFORE A POSTULATED AUTOMATIC HPCI SYSTEM ISOLATION ON HIGH AREA TEMPERATURE. A SIMILAR PREVIOUS OCCURRENCE WAS REPORTED BY LER 87-18 ON DOCKET 50-237.

[55] DRESDEN 2 DOCKET 50-237 LER 89-024
DOWNSCALE TRIP NOT INSERTED DURING EMERGENCY CORE COOLING SYSTEM INITIATING INSTRUMENT REPAIRS DUE TO MANAGEMENT DEFICIENCY.
EVENT DATE: 083089 REPORT DATE: 092789 NSSS: GE TYPE: BWR

(NSIC 215339) AT 1925 HOURS ON 8/30/89, WITH UNIT 2 AT 100% RATED CORE THERMAL POWER, LEVEL INDICATING SWITCH (LIS) 2-263-72C WAS TAKEN OUT-OF-SERVICE (OOS) FOR THE INSTRUMENT MAINTENANCE DEPARTMENT (IMD) TO REPAIR A SENSING LINE. THIS SWITCH IS PART OF THE LOW LOW REACTOR WATER LEVEL EMERGENCY CORE COOLING SYSTEM (ECCS) INITIATION LOGIC AND DIESEL GENERATOR (DG) AUTO START LOGIC. AT APPROXIMATELY 2050 HOURS, IT WAS DISCOVERED THAT THE INSTRUMENT WAS IN THE UPSCALE POSITION, CONTRARY TO TECH SPEC TABLE 3.2.2, WHICH REQUIRED INSERTION OF A DOWNSCALE TRIP. THE CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO A MANAGEMENT DEFICIENCY BECAUSE THE NEED FOR INSERTING THE DOWNSCALE TRIP DURING THE REPAIRS WAS NOT ADEQUATELY COMMUNICATED. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL BECAUSE THE REMAINING THREE LOW LOW REACTOR WATER LEVEL SWITCHES WERE AVAILABLE TO PERFORM THE ECCS AND DG START LOGIC FUNCTIONS. THE IMMEDIATE CORRECTIVE ACTION WAS TO INITIATE THE DOWNSCALE TRIP. THIS EVENT HAS BEEN REVIEWED WITH THE INDIVIDUALS INVOLVED AND ADDITIONAL CORRECTIVE ACTIONS WERE INITIATED TO PREVENT RECURRENCE. A PREVIOUS EVENT INVOLVING FAILURE TO PROPERLY TRIP A TECH SPEC INSTRUMENT DURING REPAIRS IS DOCUMENTED BY LER 87-21/050237.

[56] FERMI 2 DOCKET 50-341 LER 89-015 REV 01
UPDATE ON LOSS OF POWER TO DIVISION I REACTOR PROTECTION SYSTEM DUE TO OVERVOLTAGE ON THE MOTOR GENERATOR.
EVENT DATE: 070789 REPORT DATE: 091589 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215264) ON 7/7/89, POWER WAS LOST TO THE DIVISION I REACTOR PROTECTION SYSTEM BUS A WHEN ITS MOTOR GENERATOR (MG) SET EXPERIENCED AN OVERVOLTAGE CONDITION. INVESTIGATION REVEALED THAT VOLTAGE METER READINGS HAD DRIFTED UP FROM 120 VOLTS TO 128 VOLTS FOLLOWING ADJUSTMENT OF THE VOLTAGE ADJUSTMENT POTENTIOMETER. A PROBLEM WITH THE STABILITY OF THE INSTALLED VOLTAGE METER INDICATION WAS IDENTIFIED. BOTH THE POTENTIOMETER AND THE VOLTAGE METER WERE REPLACED PRIOR TO PLACING THE MG SET BACK IN SERVICE. ON 7/18/89, A SIMILAR EVENT OCCURRED AND THE VOLTAGE REGULATOR WAS REPLACED. WHILE THE MG SET WAS RUNNING, A VOLTAGE FLUCTUATION OCCURRED ON 7/1?/89. SUBSEQUENT TESTING INDICATES THAT THE MOST PROBABLE CAUSE WAS THE SINGLE PHASE SENSING POTENTIOMETER. THIS COMPONENT WAS REPLACED WITH A FIXED RESISTOR. FOLLOWING FURTHER TESTING AND MONITORING WITHOUT DISCOVERING ANY OTHER PROBLEMS, THE MOTOR GENERATOR SET WAS RETURNED TO SERVICE ON 8/16/89.

[57] FERMI 2 DOCKET 50-341 LER 89-017
ANALYSIS OF SINGLE FAILURE OF FEEDWATER PUMP DISCHARGE VALVES NON-CONSERVATIVE.
EVENT DATE: 081089 REPORT DATE: 091189 NSSS: GE TYPE: BWR

(NSIC 215265) THE ANALYSIS OF THE POSTULATED FEEDWATER LINE BREAK IN THE STEAM TUNNEL IS DESCRIBED IN UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) SECTION 3.6.2.2.2. THE UFSAR EVALUATION WAS COMPLETED IN THE EARLY 1970'S. SINCE DOCUMENTATION FOR THE UFSAR CONCLUSIONS CANNOT BE LOCATED, A NEW ANALYSIS WAS

CONDUCTED. THE DIFFERENCE BETWEEN THE NEW ANALYSIS AND THE UFSAR IS THAT THE SINGLE FAILURE OF THE FAST CLOSING FEEDWATER PUMP DISCHARGE VALVES AS DESCRIBED IN THE UFSAR IS NON-CONSERVATIVE. THE SINGLE FAILURE IN THE NEW FEEDWATER LINE BREAK SCENARIO IS A FAILURE IN THE OPEN POSITION OF THE FEEDWATER START-UP CONTROL VALVE. WITH THIS FAILURE OF THE FEEDWATER START-UP CONTROL VALVE, THE BREAK WILL NOT BE ISOLATED UNTIL THE TRIPPING OF THE CONDENSATE AND HEATER FEED PUMPS ON LOW HOTWELL LEVEL. THEREFORE THE CONSEQUENCES OF THE REVISED ANALYSIS ARE ADDITIONAL FLOODING BEYOND THAT PREVIOUSLY ANALYZED. THE FEEDWATER AND MAIN STEAM LINE BREAK ANALYSES HAVE BEEN REVIEWED AND ARE IN THE COMMENT INCORPORATION STAGE. THE FINAL REPORT WILL BE ISSUED BY 11/1/89. AFTER APPROVAL OF THE FINAL REPORT, A UFSAR CHANGE WILL BE SUBMITTED WITH THE NEXT ANNUAL UFSAR UPDATE IN MARCH 1990.

[58] FERMI 2 DOCKET 50-341 LER 89-018
CONTROL CENTER HEATING VENTILATION AND AIR CONDITIONING SYSTEM SHIFT TO RECIRCULATION BECAUSE OF BLOWN FUSE.
EVENT DATE: 081889 REPORT DATE: 091889 NSSS: GE TYPE: BWR

(NSIC 215266) ON 8/18/89, WHILE PERFORMING A POST MAINTENANCE TEST ON THE CONTROL CENTER HEATING VENTILATION AND AIR CONDITIONING SYSTEM (CCHVAC)(VI) AN INSTRUMENTATION AND CONTROL (I&C) TECHNICIAN INADVERTENTLY STEPPED ON A TEST LEAD WHICH WAS CONNECTED TO A TEST RECORDER. THE LEAD DISCONNECTED FROM THE RECORDER SHORTING OUT THE CIRCUIT UNDER TEST AND BLOWING THE ASSOCIATED CIRCUIT'S FUSE. THE BLOWN FUSE REMOVED POWER FROM THE LOGIC STRING FOR THE CCHVAC RADIATION MONITORING SUBSYSTEM. AS DESIGNED, BOTH DIVISIONS OF CCHVAC SHIFTED FROM THEIR NORMAL MODE OF OPERATION TO THEIR RECIRCULATION MODE. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR. THE TECHNICIAN INADVERTENTLY STEPPED ON A TEST LEAD WHICH RESULTED IN THE SHORT CIRCUIT AND SUBSEQUENT INITIATION OF THE RECIRCULATION MODE OF CCHVAC. AN ACCOUNTABILITY MEETING WAS HELD ON 9/5/89, WITH THE I&C TECHNICIAN INVOLVED, I&C SUPERIVISION AND PLANT MANAGEMENT TO DISCUSS THE EVENT TO PREVENT ITS REOCCURRENCE. ASSIGNED READING OF DOCUMENTS RELATED TO THIS EVENT WILL BE COMPLETED BY I&C SHOP PERSONNEL AND LONGER TEST LEADS WILL BE FABRICATED SUCH THAT TEST EQUIPMENT CAN BE PLACED IN LESS CUMBERSOME LOCATIONS.

[59] FERMI 2 DOCKET 50-341 LER 89-019
FAILURE OF DIVISION I CONTROL CENTER HEATING, VENTILATING AND AIR CONDITIONING RECIRCULATION FAN.
EVENT DATE: 081989 REPORT DATE: 091889 NSSS: GE TYPE: BWR
VENDOR: BUFFALO FORGE COMPANY

(NSIC 215315) ON 8/19/89, DIVISION I OF THE CONTROL CENTER HEATING, VENTILATING AND AIR CONDITIONING SYSTEM (CCHVAC) WAS OPERATING IN RECIRCULATION MODE. AT 1010 HOURS, PRESSURE CONTROL WAS LOST WHICH RESULTED IN A DECREASE IN CONTROL ROOM AIR PRESSURE. IT WAS DISCOVERED THAT THE DIVISION I RECIRCULATION FAN WAS NOT ROTATING AND DIVISION I OF CCHVAC WAS PLACED IN ITS NORMAL MODE OF OPERATION IN ORDER TO RESTORE SYSTEM PRESSURE. IN ORDER TO REPAIR THE FAN, IT WAS NECESSARY TO BREACH THE EMERGENCY FILTER HOUSING WHICH IS COMMON TO BOTH DIVISIONS OF CCHVAC. DIVISION II OF CCHVAC WAS PLACED IN NORMAL MODE OF OPERATION. DIVISION I OF CCHVAC WAS SHUTDOWN AND TECH SPEC 3.0.3 WAS ENTERED. THE FILTER HOUSING WAS RESTORED WITHIN AN HOUR. ON 8/20/89, THE FILTER WAS AGAIN BREACHED TO COMPLETE REPAIRS ON THE FAN. THE CAUSE OF THE FAILURE WAS THAT ONE OF THE BEARINGS SEIZED. THE DAMAGED COMPONENTS HAVE BEEN SENT OFF SITE FOR ANALYSIS. THIS REPORT WILL BE REVISED WITHIN 30 DAYS AFTER THE REVIEW OF THE ANALYSIS RESULTS.

[60] FERMI 2 DOCKET 50-341 LER 89-020
REMOVAL OF AN INCORRECT FUSE CAUSED THE FUEL POOL EXHAUST RADIATION MONITOR TO BECOME DE-ENERGIZED.
EVENT DATE: 083089 REPORT DATE: 092989 NSSS: GE TYPE: BWR

(NSIC 215367) AT 1353 HOURS, ON 8/30/89, REMOVAL OF AN INCORRECT FUSE CAUSED THE ACTUATION OF STANDBY GAS TREATMENT SYSTEM DIVISION I, THE ISOLATION OF REACTOR BUILDING HEATING VENTILATION AND AIR CONDITIONING, THE SHIFTING OF CONTROL CENTER HEATING VENTILATION AND AIR CONDITIONING TO THE RECIRCULATION MODE, AND AN ISOLATION SIGNAL TO THE CONTAINMENT ISOLATION VALVES GROUPS 14 AND 16. THE GROUP 14 ISOLATION SIGNAL WAS FOR THE DRYWELL AND SUPPRESSION POOL VENTILATION SYSTEM, AND THE GROUP 16 ISOLATION SIGNAL WAS FOR THE NITROGEN INERTING SYSTEM. AT 1440 HOURS ALL THE AFFECTED SYSTEMS WERE RETURNED TO NORMAL. THIS EVENT WAS CAUSED BY PERSONNEL ERROR. A LICENSED OPERATOR PULLED THE WRONG FUSE WHEN ASSISTING INSTRUMENT & CONTROL (I&C) REPAIRMEN DURING INVESTIGATIVE TROUBLESHOOTING. A CRITIQUE OF THIS EVENT WILL BE INCLUDED IN THE OPERATIONS AND I&C REQUIRED READING PROGRAMS. THIS REQUIRED READING IS A SUMMARY OF THE CRITIQUE AND IS SCHEDULED TO BE ISSUED BY 10/16/89.

[61] FITZPATRICK DOCKET 50-333 LER 89-013
PRIMARY CONTAINMENT PURGE VALVES ISOLATED BY ACCIDENTAL GROUND OF RELAY JUMPER WIRE CONTRIBUTED TO BY INADEQUATE WORK SPACE DUE TO BAD CABLE ROUTING.
EVENT DATE: 080389 REPORT DATE: 090189 NSSS: GE TYPE: BWR

(NSIC 215214) AT 1:30 P.M. ON 8/3/89 A TECHNICIAN WAS REMOVING A TEMPORARY BYPASS TEST JUMPER FROM A RELAY IN ACCORDANCE WITH AN APPROVED PROCEDURE. THE RELAY WAS LOCATED IN A CONFINED SPACE (JL). A SAFETY SYSTEM DIVISIONAL SEPARATION CONDUIT (JL) BLOCKED DIRECT ACCESS TO THE RELAY. THE CONDUIT CONNECTOR IS JOINED TO AN ADJACENT DIVISIONAL SEPARATION RELAY ENCLOSURE BOX. THE CONNECTOR AND ENCLOSURE BOX ARE NOT INSULATED AND ARE AT ELECTRICAL GROUND. ONE END OF THE JUMPER WAS CONNECTED TO A CONTACT ON THE RELAY. THE OTHER END WAS ACCIDENTALLY BRUSHED AGAINST A CONDUCTIVE GROUNDED SURFACE. THIS CAUSED THE PROTECTIVE FUSE TO CLEAR AND DEENERGIZE THE ISOLATION CIRCUIT. DEENERGIZING THE CIRCUIT CAUSED 2 SOLENOID OPERATED GROUP II PRIMARY CONTAINMENT PURGE SYSTEM ISOLATION VALVES (JM) TO CLOSE. THE FUSE WAS REPLACED AND THE 2 VALVES REOPENED WITHIN 15 MINS. THE CLOSING OF 2 VALVES, INSTEAD OF ONLY 1 VALVE, FOR A SINGLE FUSE REVEALED A WIRING ERROR. SHORT-TERM CORRECTIVE ACTION ADDED A CAUTION STATEMENT TO THE PROCEDURE AND APPLIED ELECTRICAL INSULATING TAPE TO THE CONDUIT CONNECTORS. LONG-TERM CORRECTIVE ACTION WILL INSTALL CONVENIENTLY LOCATED REMOTE BYPASS TERMINALS FOR THE RELAY. THE WIRING ERROR WILL BE CORRECTED TO PROVIDE A SINGLE FUSE FOR EACH ISOLATION VALVE. LER-86-014 AND 86-019 SHARE COMMON ELEMENTS WITH THIS EVENT.

[62] FITZPATRICK DOCKET 50-333 LER 89-014
HIGH PRESSURE COOLANT INJECTION PUMP TURBINE DECLARED INOPERABLE DUE TO WATER IN LUBRICATING OIL DUE TO LEAKING STEAM SUPPLY ISOLATION VALVE.
EVENT DATE: 081789 REPORT DATE: 091589 NSSS: GE TYPE: BWR

(NSIC 215313) THE STEAM SUPPLY ISOLATION VALVE FOR THE HIGH PRESSURE COOLANT INJECTION (HPCI) (BJ) SYSTEM TURBINE HAS BEEN LEAKING A SMALL QUANTITY OF STEAM TO THE TURBINE, WHILE THE VALVE IS IN THE CLOSED POSITION, SINCE DECEMBER 1988. BECAUSE THE TURBINE IS IDLE, THE STEAM CONDENSES IN THE TURBINE CASING AND SOME OF THE ACCUMULATED WATER IS PUSHED THROUGH THE SHAFT SEALS AND ENTERS THE BEARING LUBRICATING OIL SYSTEM WHEN HPCI IS STARTED. DURING REPEATED IDLE AND STARTUP CYCLES THE WATER ACCUMULATES IN THE TURBINE OIL SUMP. BECAUSE OF THE CONCERN WITH THE LEAKAGE, SAMPLES WERE DRAWN FROM THE BOTTOM OF THE SUMP ON 8/15/89 AND AGAIN ON 8/17/89. BOTH SAMPLES CONTAINED SLIGHTLY LESS THAN 50% WATER. ALTHOUGH, A BOTTOM SAMPLE FROM THE 155-GALLON SUMP IS EXPECTED TO CONTAIN WATER AND IS NOT REPRESENTATIVE OF THE ENTIRE SUMP. HPCI WAS MADE INOPERABLE ON 8/17/89 TO REPLACE THE LUBRICATING OIL AND THE SYSTEM WAS RETURNED TO SERVICE APPROXIMATELY 24 HOURS LATER. CORRECTIVE ACTION DURING THE SEPTEMBER 1989 MAINTENANCE OUTAGE WILL INCLUDE DISASSEMBLY, INSPECTION, AND REPAIR OF THE LEAKING STEAM VALVE AND REPLACEMENT OF THE TURBINE SHAFT SEALS. LONG-TERM CORRECTIVE ACTION WILL EXPAND THE LUBRICATION OIL PROGRAM TO INCLUDE SAMPLING THE HPCI LUBRICATING OIL ON A QUARTERLY BASIS TO PROVIDE EARLY DETECTION OF ACCUMULATION OF WATER IN THE SUMP.

[63] FT. CALHOUN 1 DOCKET 50-285 LER 89-014 REV 01
UPDATE ON AUXILIARY FEEDWATER PANEL INSTRUMENTATION OUTSIDE DESIGN BASIS.
EVENT DATE: 051989 REPORT DATE: 092989 NSSS: CE TYPE: PWR

(NSIC 215353) DESIGN BASIS RECONSTITUTION FOR THE AUXILIARY FEEDWATER PANEL REVEALED DEFICIENCIES IN THE INSTRUMENTATION ON 5/19/89, AT APPROXIMATELY 1750 HOURS WHILE THE PLANT WAS IN MODE 1 AT 100% POWER. THE WIRING FOR THE WIDE RANGE STEAM GENERATOR PRESSURE INDICATION WAS FOUND TO BE IN NON-COMPLIANCE WITH 10 CFR 50 APPENDIX R. THIS WIRING WAS ROUTED THROUGH THE CONTROL ROOM ENVELOPE WHICH COULD AFFECT THE OPERABILITY OF THE INSTRUMENTATION IN THE EVENT OF A FIRE IN THE CONTROL ROOM. THE DESIGN BASIS RECONSTITUTION ALSO REVEALED THAT PRESSURIZER PRESSURE AND STEAM GENERATOR NARROW RANGE LEVEL INDICATIONS ON THE AUXILIARY FEEDWATER PANEL RECEIVE POWER FROM INVERTER "C" WHICH IS POWERED BY BATTERY #1. USE OF BATTERY #1 WAS NOT ANALYZED IN THE SAFE SHUTDOWN ANALYSIS WITH RESPECT TO THE WORST CASE CONTROL ROOM FIRE. THESE DEFICIENCIES WERE REPORTED ON 5/19/89, AT 1820 HOURS TO THE NRC PURSUANT TO 10 CFR 50.72(B)(1)(II)(B). IT HAS BEEN DETERMINED THAT THE DEFICIENCIES WERE THE RESULT OF INADEQUACIES IN THE FORMER DESIGN MODIFICATION PROCESS AND IN THE ORIGNAL ANALYSIS ASSOCIATED WITH APPENDIX R COMPLIANCE. PRESENT DAY DESIGN PROCEDURES REQUIRE THAT POWER SUPPLY AND CABLE ROUTING BE CONSIDERED AS PART OF THE DESIGN TO ENSURE THAT APPENDIX R REQUIREMENTS ARE MET. A DESIGN MODIFICATION WILL, WHEN COMPLETED, PROVIDE APPROPRIATE POWER SUPPLIES FOR THE AUX. FEEDWATER PANEL INDICATION.

[64] FT. CALHOUN 1 DOCKET 50-285 LER 89-018
FAILURE TO CONDUCT HOURLY FIREWATCH PATROL DUE TO PROCEDURAL INADEQUACIES.
EVENT DATE: 080389 REPORT DATE: 090589 NSSS: CE TYPE: PWR

(NSIC 215249) ON 8/3/89 AT 0920 HOURS, WITH FORT CALHOUN STATION UNIT 1 OPERATING AT 100% POWER, THE FIRE PROTECTION SYSTEM ENGINEER DETERMINED THAT THE REQUIRED HOURLY FIREWATCH PATROL AT DOOR 989-14 WAS NOT BEING PERFORMED. THIS DOOR HAD BEEN ERRONEOUSLY REMOVED FROM THE HOURLY FIREWATCH LOG USED BY SECURITY ON 7/29/89 AT APPROXIMATELY 0400 HOURS. UPON DISCOVERY OF THE PROBLEM, THE SYSTEM ENGINEER CONTACTED THE SHIFT SECURITY SUPERVISOR, WHO REINSTATED THE HOURLY PATROL FOR DOOR 989-14 AND RESTORED THE DOOR TO THE HOURLY FIREWATCH LOG. THIS EVENT RESULTED PRIMARILY FROM INADEQUATE PROCEDURAL CONTROLS; INEFFECTIVE COMMUNICATIONS AND PROCEDURAL NON-COMPLIANCE WERE CONTRIBUTING CAUSES. REVISIONS HAVE BEEN MADE TO STATION PROCEDURES TO BETTER CONTROL THE FIREWATCH PATROL PROCESS. FAILURE TO PERFORM THE FIREWATCH PATROL WAS A VIOLATION OF TECH SPEC 2.19(7). THIS REPORT IS MADE PURSUANT TO 10 CFR 50.73(A)(2)(I)(B).

[65] GRAND GULF 1 DOCKET 50-416 LER 89-002 REV 01
UPDATE ON CONTAINMENT ISOLATION VALVES EXCEED TECH SPEC LEAKAGE LIMITS.
EVENT DATE: 033189 REPORT DATE: 083089 NSSS: GE TYPE: BWR
VENDOR: ATWOOD & MORRILL CO., INC.
POWELL, WILLIAM COMPANY, THE

(NSIC 215174) DURING ROUTINE LOCAL LEAK RATE TESTING, THE LEAKAGE RATES OF FIVE CONTAINMENT ISOLATION VALVES COULD NOT BE QUANTIFIED. THIS WAS DUE TO EXCESSIVE LEAKAGE BEYOND THE CAPABILITY OF THE TEST EQUIPMENT TO PRESSURIZE THE TEST VOLUME TO THE REQUIRED TEST PRESSURE. THUS, THE LEAKAGE IS CONSIDERED TO EXCEED THE TECH SPEC ALLOWABLE LEAKAGE LIMIT. THE SUBJECT VALVES ARE: MAIN STEAM LINE "C" INBOARD ISOLATION VALVE, B21F022C; FEEDWATER LINE "A" INBOARD AND OUTBOARD ISOLATION CHECK VALVES, B21F010A AND B21F032A; FEEDWATER LINE "B" INBOARD AND OUTBOARD ISOLATION CHECK VALVES, B21F010B AND B21F032B. ALL CONDITIONS WHICH ATTRIBUTED TO THE EXCESSIVE LEAKAGE HAVE BEEN CORRECTED AND THE VALVES HAVE BEEN RETESTED SATISFACTORILY. AN EVALUATION OF THE SEAT MATERIAL FAILURE OF THE FEEDWATER CHECK VALVES WILL BE PERFORMED PRIOR TO THE NEXT REFUELING OUTAGE. IN THE INTERIM, SEATS OF A MATERIAL TYPE THAT HAVE DEMONSTRATED ACCEPTABILITY FOR AN OPERATING CYCLE WILL BE USED. THE OVERALL LEAKAGE FOR EACH OF THE SUBJECT

PENETRATIONS WAS WITHIN ALLOWABLE LEAKAGE LIMITS CONSIDERING MINIMUM PATHWAY LEAKAGE METHODOLOGY.

[66] GRAND GULF 1 DOCKET 50-416 LER 89-012
REACTOR SCRAM DUE TO CONDENSER EXPANSION JOINT FAILURE.
EVENT DATE: 081489 REPORT DATE: 091389 NSSS: GE TYPE: BWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)
PROCESS ENGINEERING, INC.
REACTOR CONTROLS INC.

(NSIC 215283) ON 8/14/89, A MAIN CONDENSER EXPANSION JOINT FAILED CAUSING CONDENSER VACUUM TO DECREASE. THE MAIN TURBINE TRIPPED ON CONDENSER LOW VACUUM RESULTING IN A REACTOR SCRAM DUE TO THE FAST CLOSURE OF THE MAIN TURBINE STOP AND CONTROL VALVES. DURING THE COURSE OF THE EVENT TWO SUBSEQUENT RPS ACTUATIONS OCCURRED ON LOW REACTOR WATER LEVEL SIGNALS. A MAIN STEAM ISOLATION VALVE (MSIV) ISOLATION SIGNAL WAS ACTUATED ON LOW CONDENSER VACUUM. THE REACTOR CORE ISOLATION COOLING SYSTEM WAS MANUALLY INITIATED TO CONTROL REACTOR WATER LEVEL. NO ECCS WERE INITIATED MANUALLY OR AUTOMATICALLY. ALL CONTROL RODS FULLY INSERTED WITH THE EXCEPTION OF CONTROL ROD 32-45 WHICH STOPPED AT POSITION 08. ALL MSIVS PROPERLY CLOSED WITH THE EXCEPTION OF B21-F022B WHICH CLOSED APPROXIMATELY 35 MINUTES FOLLOWING A MANUAL CLOSURE SIGNAL. COMPONENTS IN THE CONTROL ROD HYDRAULIC CONTROL UNIT WERE REPLACED TO CORRECT THE CONTROL ROD MALFUNCTION. THE FAILURE MECHANISM OF THE MSIV WAS DETERMINED TO BE GENERIC TO OTHER INSTALLED MSIVS AND IS REPORTED SEPARATELY IN LER 89-013. A 3 FOOT SECTION OF THE CONDENSER EXPANSION BELT HAD TORN HORIZONTALLY NEAR THE LOWER RETAINING CLAMPS. THE HEMP REINFORCEMENT FIBERS HAD APPARENTLY DEGRADED DUE TO MOISTURE INTRUSION THROUGH THE RUBBER. THE BELTS ON ALL 3 CONDENSER SECTIONS WERE REPLACED WITH BELTS USING A POLYESTER REINFORCEMENT FIBER.

[67] GRAND GULF 1 DOCKET 50-416 LER 89-013
MAIN STEAM ISOLATION VALVE FAILED TO CLOSE DUE TO EXTRUDED ELASTOMER SEAT MATERIAL.
EVENT DATE: 081489 REPORT DATE: 091389 NSSS: GE TYPE: BWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)

(NSIC 215284) ON 8/14/89, FOLLOWING A REACTOR SCRAM, MAIN STEAM ISOLATION VALVE (MSIV) B21F022B FAILED TO CLOSE UPON DEMAND FROM AUTOMATIC AND REMOTE MANUAL ACTUATION SIGNALS. THE REDUNDAT MSIV FUNCTIONED AS REQUIRED. APPROXIMATELY THIRTY MINUTES FOLLOWING THE INITIAL CLOSURE ATTEMPT, VALVE B21F022B CLOSED WITH NO ADDITIONAL OPERATOR ACTIONS. AN EXAMINATION OF SOLENOID VALVE B21SVF501B REVEALED THAT A PIECE OF ELASTOMER SEAT MATERIAL HAD EXTRUDED INTO THE EXHAUST PORT VENT HOLE, PREVENTING THE SOLENOID VALVE FROM EXHAUSTING AIR. THE MATERIAL WAS LATER DISLODGED BY AIR PRESSURE WHICH ALLOWED VALVE B21F022B TO VENT AND CLOSE. THE ASCO NP8323A20E DUAL SOLENOID VALVES ON ALL EIGHT MSIVS WERE REPLACED ORE REFURBISHED PRIOR TO STARTUP FOLLOWING THE REACTOR SCRAM. SYSTEM ENERGY RESOURCES, INC. (SERI) WILL CONTINUE AN ACCELERATED REPLACEMENT OR REFURBISHMENT SCHEDULE IN PARALLEL WITH PURSUING DESIGN CHANGE OPTIONS.

[68] HATCH 1 DOCKET 50-321 LER 89-009
PLANT PROCEDURES INADEQUATELY IMPLEMENT TECH SPECS SURVEILLANCE REQUIREMENTS.
EVENT DATE: 082389 REPORT DATE: 092189 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 2 (BWR)

(NSIC 215309) ON 08/23/89, AT APPROXIMATELY 1335 CDT, UNIT 1 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2436 CMWT (APPROXIMATELY 100% OF RATED THERMAL POWER). AT THAT TIME, IT WAS DETERMINED THAT THE SURVEILLANCE REQUIREMENTS OF UNIT 1 TECH SPECS TABLE 4.2.9, ITEM 3(C) FOR END OF CYCLE-RECIRCULATION PUMP TRIP (EOC-RPT, EIIS CODE JC) RESPONSE TIME TESTING HAD NOT BEEN ADEQUATELY IMPLEMENTED

IN PLANT PROCEDURES. ON 8/30/89, AT APPROXIMATELY 1115 CDT, UNIT 2 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 1725 CMWT (APPROXIMATELY 71% OF RATED THERMAL POWER). AT THAT TIME, IT WAS DETERMINED THAT THE SURVEILLANCE REQUIREMENTS OF UNIT 2 TECH SPECS SECTION 4.3.9.2.3 FOR EOC-RPT RESPONSE TIME TESTING HAD NOT BEEN ADEQUATELY IMPLEMENTED IN PLANT PROCEDURES. THE ROOT CAUSE OF THE EVENTS WAS COGNITIVE PERSONNEL ERROR IN THAT PERSONNEL FAILED TO COMPLETELY INCORPORATE TECH SPECS SURVEILLANCE REQUIREMENTS INTO PLANT PROCEDURES UPON ISSUANCE OF THE ASSOCIATED TECH SPECS AMENDMENTS. CORRECTIVE ACTIONS INCLUDE PERFORMANCE OF REQUIRED TESTING, COUNSELING OF PERSONNEL, ISSUING A DEPARTMENTAL DIRECTIVE, IMPLEMENTING PERMANENT PLANT PROCEDURES, PERFORMING A REVIEW OF A SAMPLE OF TECH SPECS AMENDMENTS TO ENSURE COMPLIANCE WITH THEIR SURVEILLANCE REQUIREMENTS, AND CLARIFYING (BY REVISION) TECH SPECS.

[69] HATCH 1 DOCKET 50-321 LER 89-010
TWO REACTOR WATER CLEANUP ISOLATIONS IN 24 HOUR PERIOD.
EVENT DATE: 082389 REPORT DATE: 092189 NSSS: GE TYPE: BWR

(NSIC 215310) ON 08/23/89, AT 1326 CDT, UNIT 1 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2431 MWT (APPROXIMATELY 100% OF RATED THERMAL POWER). AT THAT TIME, AN 18-MONTH SURVEILLANCE WAS BEING PERFORMED ON THE REACTOR WATER CLEANUP (RWCU) SYSTEM DIFFERENTIAL FLOW INSTRUMENTATION. FOLLOWING COMPLETION OF A MODULE OF THE SURVEILLANCE, THE RWCU INBOARD ISOLATION VALVE ISOLATED, TRIPPING THE RWCU PUMP. NO VALID PLANT CONDITION NECESSITATING THE ISOLATION COULD BE FOUND AND THE RWCU SYSTEM WAS RETURNED TO SERVICE AT 2020 CDT. AT 0331 CDT ON 08/24/89, AS THE DIFFERENTIAL FLOW SURVEILLANCE WAS BEING COMPLETED, ANOTHER ISOLATION OF RWCU OCCURRED. AGAIN, WHEN NO VALID PLANT CONDITION WAS FOUND WHICH WOULD HAVE NECESSITATED THE ISOLATION, RWCU WAS RETURNED TO SERVICE. THE ROOT CAUSE OF THIS EVENT COULD NOT BE CONCLUSIVELY DETERMINED. POSSIBLE ROOT CAUSES NOT ALREADY RULED OUT WILL BE INVESTIGATED FURTHER. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDE: 1) INSTALLING, TEMPORARILY, A DATA ACQUISITON AND ANALYSIS SYSTEM (DAAS) TO IDENTIFY ANY CONTROL ROOM PANEL CIRCUIT OR RELAY CONTACT PROBLEM, 2) PERFORMING A FUNCTIONAL TEST AND CALIBRATION ON THE REMOTELY LOCATED TEMPERATURE AND DIFFERENTIAL FLOW SENSING EQUIPMENT, AND 3) CLEANING CERTAIN RELAY CONTACTS IN THE RWCU TRIP CIRCUITRY.

[70] HATCH 1 DOCKET 50-321 LER 89-011
DEFICIENT PROCEDURES RESULT IN INADEQUATE TECH SPECS SURVEILLANCE.
EVENT DATE: 090189 REPORT DATE: 092689 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 2 (BWR)

(NSIC 215363) ON 9/1/89, AT APPROXIMATELY 2246 CDT, UNIT 1 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2431 CMWT (APPROXIMATELY 100% OF RATED THERMAL POWER) AND UNIT 2 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 1740 CMWT (APPROXIMATELY 69.9% OF RATED THERMAL POWER). AT THAT TIME ON-SHIFT LICENSED PERSONNEL WERE MADE AWARE THAT PROCEDURE 34GO-OPS-013-1S AND 2S, "NORMAL PLANT SHUTDOWN," DID NOT FULLY VERIFY THE OPERABILITY OF THE ROD WORTH MINIMIZER (RWM, EIIS CODE ID) AS REQUIRED BY UNIT 1 TECH SPEC SECTION 4.3.G.1.D AND BY UNIT 2 TECH SPECS SECTION 4.1.4.1.A.2. ADEQUATE PROCEDURES TO PROPERLY TEST THE RWM DURING PLANT STARTUPS WERE IN PLACE. THUS, THE RWM WAS TECHNICALLY OPERABLE AT THE TIME OF THE EVENT (THE SOFTWARE MAKES NO DISTINCTION BETWEEN POWER ASCENSION AND POWER DESCENT). HOWEVER, THE RWM WAS DECLARED INOPERABLE ON EACH UNIT AND LIMITING CONDITIONS OF OPERATION WERE INITIATED TO ENSURE COMPLIANCE WITH TECH SPECS REQUIREMENTS DURING POWER DESCENT. THE CAUSE OF THIS EVENT IS A PROCEDURAL DEFICIENCY. THE PROCEDURES WERE WRITTEN SUCH THAT THE OPERABILITY OF THE RWM WAS NOT COMPLETELY TESTED. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDE REVISING PROCEDURES 34GO-OPS-013-1S AND 2S, COUNSELING INVOLVED PERSONNEL, AND ISSUING A DEPARTMENTAL DIRECTIVE.

[71] HATCH 2 DOCKET 50-366 LER 89-004
PERSONNEL ERROR RESULTS IN AN INADEQUATE PROCEDURE AND AN ACTUATION OF AN ESF.
EVENT DATE: 090289 REPORT DATE: 092789 NSSS: GE TYPE: BWR

(NSIC 215371) ON 9/2/89, AT APPROXIMATELY 1220 CDT, UNIT 2 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 1704 CMWT (APPROXIMATELY 70% OF RATED THERMAL POWER). AT THAT TIME, AUTOMATIC CLOSURE OF PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS, EIIS CODE JM) VALVE 2E41-F002 OCCURRED DUE TO AN ISOLATION SIGNAL. THIS VALVE FUNCTIONS AS THE INBOARD STEAM SUPPLY ISOLATION VALVE FOR THE HIGH PRESSURE COOLANT INJECTION (HPCI, EIIS CODE BJ) SYSTEM. AT THE TIME OF THE EVENT, PLANT PERSONNEL WERE IN THE PROCESS OF PERFORMING PROCEDURE 57SV-SUV-004-2S, "EXCESS FLOW CHECK VALVE OPERABILITY." UPON VALVING IN DIFFERENTIAL PRESSURE TRANSMITTER 2E41-N057A, A PCIS SIGNAL WAS GENERATED RESULTING IN CLOSURE OF VALVE 2E41-F002. THE ROOT CAUSE OF THIS EVENT IS COGNITIVE PERSONNEL ERROR. THE PROCEDURE WRITER INCORRECTLY UNDERSTOOD THE INSTRUMENT LOGIC WIRING CONFIGURATION ASSOCIATED WITH INSTRUMENT 2E41-N057A. THIS RESULTED IN A DEFICIENT PROCEDURE IN THAT ADEQUATE PROVISIONS FOR ELECTRICALLY ISOLATING THE INSTRUMENT WERE NOT SPECIFIED. THE CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED SUSPENDING THE PERFORMANCE OF THE PROCEDURE WHILE INVESTIGATING THE HPCI ISOLATION VALVE CLOSURE, TEMPORARILY REVISING THE PROCEDURE, AND INITIATING A PERMANENT REVISION TO THE PROCEDURE.

[72] HATCH 2 DOCKET 50-366 LER 89-005
FEEDWATER CONTROLLER FAILURE CAUSES REACTOR SCRAM ON LOW WATER LEVEL.
EVENT DATE: 090389 REPORT DATE: 092789 NSSS: GE TYPE: BWR
VENDOR: GEN ELEC CO (STEAM TURB/ENGRD PROD)

(NSIC 215373) ON 9/3/89 AT APPROXIMATELY 2239 CDT, UNIT 2 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 1695 CMWT (APPROXIMATELY 70% OF RATED THERMAL POWER). AT THAT TIME, LICENSED OPERATIONS PERSONNEL WERE CHANGING REACTOR VESSEL WATER LEVEL CONTROL FROM SINGLE ELEMENT TO THREE ELEMENT CONTROL FOLLOWING COMPLETION OF PROCEDURE 57SV-SUV-004-2S, "EXCESS FLOW CHECK VALVE OPERABILITY," FOR THE MAIN STEAM LINE FLOW INSTRUMENTS' EXCESS FLOW CHECK VALVES. WHEN THE MASTER CONTROLLER WAS PLACED IN AUTOMATIC FOLLOWING THE CHANGE FROM SINGLE ELEMENT TO THREE ELEMENT CONTROL, THE CONTROLLER'S OUTPUT SIGNAL SUDDENLY WENT TO ZERO. BOTH REACTOR FEED PUMPS DECREASED FEEDWATER FLOW TO THE REACTOR VESSEL IN RESPONSE TO THE CONTROLLER'S ZERO OUTPUT SIGNAL. REACTOR VESSEL WATER LEVEL DECREASED AND THE REACTOR SCRAMMED ON LOW WATER LEVEL. THE ROOT CAUSE OF THIS EVENT IS COMPONENT FAILURE. THE SELF SYNCHRONIZED CONTROL UNIT, THE MAIN OPERATING UNIT OF THE MASTER CONTROLLER, FAILED WHEN THE MASTER CONTROLLER WAS PLACED IN AUTOMATIC. THE FAILURE OF THE SELF SYNCHRONIZED CONTROL UNIT CAUSED THE MASTER CONTROLLER'S OUTPUT SIGNAL TO GO TO ZERO. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED REPLACING THE SELF SYNCHRONIZED CONTROL UNIT.

[73] HOPE CREEK 1 DOCKET 50-354 LER 89-016
NON-ENVIRONMENTALLY QUALIFIED (EQ) LIMIT SWITCH ASSEMBLY UTILIZED IN AN EQ APPLICATION DUE TO INSTALLATION DEFICIENCIES DURING PRE-COMMERCIAL CONSTRUCTION ACTIVITIES.
EVENT DATE: 080289 REPORT DATE: 090289 NSSS: GE TYPE: BWR
VENDOR: CONAX CORP.
HILLER, RALPH A., CO.
NAMCO CONTROLS

(NSIC 215170) ON 8/2/89 DURING PRE-REFUEL OUTAGE PLANNING, IT WAS DISCOVERED THAT THE LIMIT SWITCH ASSEMBLY ON PRIMARY CONTAINMENT INSTRUMENT GAS (PCIG) SYSTEM VALVE HV-5154 DID NOT MEET EQ REQUIREMENTS. THIS VALVE IS THE OUTBOARD TORUS ISOLATION ON THE PCIG SUPPLY TO THE TORUS VACUUM RELIEF VALVES. INVESTIGATION SUBSEQUENT TO DISCOVERY DETERMINED THAT THE ENTIRE VALVE ASSEMBLY, INCLUDING ACTUATOR AND LIMIT SWITCHES, HAD BEEN REPLACED DURING PRE-COMMERCICAL CONSTRUCTION ACTIVITIES IN 1985. THE CONSTRUCTION ORGANIZATION RESPONSIBLE FOR

REPLACING THE VALVE DID NOT PROPERLY REPLACE THE VALVE IN ACCORDANCE WITH FIELD INSTRUCTIONS ISSUED IN DECEMBER OF 1985. CORRECTIVE ACTIONS CONSISTED OF REPLACING THE NON-EQ LIMIT SWITCH WITH AN ENVIRONMENTALLY QUALIFIED SWITCH, AND REVIEWING ALL SIMILAR VALVES TO ENSURE EQ COMPLIANCE.

[74] HOPE CREEK 1 DOCKET 50-354 LER 89-017
REACTOR SCRAM CAUSED BY FAILURE OF SOLDERED SCRAM VALVE PILOT AIR LINE DUE TO INSTALLATION DEFICIENCY.
EVENT DATE: 083089 REPORT DATE: 093089 NSSS: GE TYPE: BWR
VENDOR: BECHTEL CORP.

(NSIC 215369) ON 8/30/89 AT 0112, A REACTOR SCRAM OCCURRED AS A RESULT OF A LOW REACTOR PRESSURE VESSEL (RPV) WATER LEVEL (+12.5") SIGNAL. THE TRANSIENT WAS INITIATED BY THE FAILURE OF A 1/2" SOLDERED CONNECTION WHERE THE SCRAM PILOT AIR LINE FROM CONTROL ROD DRIVE (CRD) HYDRAULIC CONTROL UNIT (HCU) 34-59 JOINED A 1-1/2" AIR HEADER INTERCONNECTING A BANK OF HCU'S ON THE SOUTH SIDE OF THE REACTOR BUILDING. THIS FAILURE RESULTED IN THE RAPID DEPRESSURIZATION OF THE SCRAM AIR HEADER, AND IN RESPONSE, CONTROL RODS BEGAN TO INSERT. THE ENSUING VOID COLLAPSE RESULTED IN AN UNRECOVERABLE LOW LEVEL TRANSIENT AND THE SUBSEQUENT SCRAM. FAILURE ANALYSIS FOLLOWING THE SCRAM DETERMINED THE ROOT CAUSE OF THE INITIAL AIR LINE FAILURE TO BE DEFICIENCIES IN INITIAL INSTALLATION OF THE AIR LINE DURING PLANT CONSTRUCTION. IMMEDIATE CORRECTIVE ACTIONS INCLUDED RADIOGRAPHY OF SIMILAR JOINTS PRIOR TO RESTARTING THE PLANT, LEAK CHECKING ALL SOLDERED CONNECTIONS AT ALL 185 SCRAM PILOT AIR HEADER RISERS, AND PULL TESTING ALL 185 TEE JOINTS TO DEMONSTRATE THE ABILITY OF THE JOINTS TO WITHSTAND NORMAL OPERATING STRESSES. LONG TERM CORRECTIVE ACTIONS INCLUDE COMPLETION OF AN ONGOING TESTING PROGRAM FOR INSTRUMENT AIR PIPING PRIOR TO THE END OF THE CURRENT REFUELING OUTAGE.

[75] INDIAN POINT 2 DOCKET 50-247 LER 89-008
TECH SPEC LIMIT EXCEEDED ISOLATION VALVE SEAL WATER SYSTEM.
EVENT DATE: 071989 REPORT DATE: 081989 NSSS: WE TYPE: PWR
VENDOR: ANCHOR VALVE CO.

(NSIC 215110) ON JULY 19, 1989, DURING A REVIEW OF COMPLETED TESTS RESULTS, IT WAS DETERMINED THAT A TECHNICAL SPECIFICATION LIMIT HAD BEEN EXCEEDED. DURING THE 1989 REFUELING OUTAGE, SURVEILLANCE TESTS OF TYPE "B" AND "C" (APPENDIX J) VALVES WERE CONDUCTED FROM MAY THROUGH JUNE OF 1989. WHEN TEST RESULTS WERE TOTALLED, IT WAS DISCOVERED THAT THE AS FOUND TOTAL OF INDIVIDUAL VALVE LEAKAGE EXCEEDED THE 14,700 CUBIC CENTIMETERS PER HOUR PERMITTED BY THE TECHNICAL SPECIFICATION FOR THE ISOLATION VALVE SEAL WATER SYSTEM. EXCESSIVE LEAKAGE OCCURRED ACROSS TEN VALVES. THE VALVES WERE PROMPTLY REPAIRED AND LEAKAGE BROUGHT TO WELL WITHIN SPECIFICATION PRIOR TO HEATING ABOVE COLD SHUTDOWN.

[76] INDIAN POINT 2 DOCKET 50-247 LER 89-011
DIFFERENTIAL PRESSURE FOR 2 NON-ESSENTIAL SERVICE WATER PUMP STRAINERS ABOVE LIMITS.
EVENT DATE: 080189 REPORT DATE: 083189 NSSS: WE TYPE: PWR
VENDOR: ZURN INDUSTRIES, INC.

(NSIC 215232) ON TUESDAY, 8/1/89, IP-2 WAS OPERATING WITH EMERGENCY DIESEL GENERATOR #23 AND SERVICE WATER PUMP #23 REMOVED FROM SERVICE. AT APPROXIMATELY 0815, THE DIFFERENTIAL PRESSURE FOR THE REMAINING TWO NON-ESSENTIAL SERVICE WATER PUMP STRAINERS INCREASED ABOVE THE ALLOWABLE VALUE. AN EVALUATION BY THE OPERATORS BEGAN, AND AT 0840 THE ENTIRE NON-ESSENTIAL SERVICE WATER SYSTEM WAS DECLARED INOPERABLE PLACING THE PLANT INTO THE TECH SPEC (TS) 3.0.1, AND REQUIRING A 1 HOUR NOTIFICATION AND SUBSEQUENT PLANT SHUTDOWN TO HOT SHUTDOWN WITHIN 7 HOURS. AT 0940 NON-ESSENTIAL SERVICE WATER PUMP #23 WAS MADE OPERABLE

BY STRAINER CLEANING, AND AT 1330 A SECOND NON-ESSENTIAL SERVICE WATER PUMP (#22) BECAME OPERABLE. TS 3.0.1 WAS INCORRECTLY INTERPRETED TO BE SATISFIED AT 0940 ALTHOUGH IT WAS ACTUALLY SATISFIED AT 1330, 4 HOURS AND 50 MINUTES AFTER TECH SPEC 3.0.1 WAS ENTERED.

[77] KEWAUNEE DOCKET 50-305 LER 89-014
THE AGE AND DESIGN OF THE RADIATION MONITORING SYSTEM RESULT IN ACTUATION OF THE AUX. BLDG SPECIAL VENTILATION SYSTEM.
EVENT DATE: 072489 REPORT DATE: 082389 NSSS: WE TYPE: PWR

(NSIC 215128) THIS REPORT DESCRIBES 2 SEPARATE BUT RELATED UNPLANNED ACTUATIONS OF THE AUX. BLDG. SPECIAL VENTILATION (ASV) SYSTEM, AN ENGINEERED SAFETY FEATURE. ON 7/24/89, AND ON 8/3/89, WHILE THE PLANT WAS AT 100% POWER, THE ASV SYSTEM STARTED WHEN AN I&C TECHNICIAN INADVERTENTLY GROUNDED ONE OF THE POWER LEADS TO R-24 (THE CIRCULATING WATER DISCHARGE RADIATION MONITOR). WHEN THE LEAD BECAME GROUNDED ON JULY 24, A PROTECTIVE FUSE BLEW, CAUSING DOWN SCALE FAILURES OF THE 5 OTHER RADIATION MONITORS ON THE SAME CIRCUIT AS R-24. THESE DOWN SCALE FAILURES RESULTED IN THE ACTUATION OF THE ASV SYSTEM. WHEN THE POWER SUPPLY BECAME GROUNDED ON AUGUST 3, THE VOLTAGE ON THE CIRCUIT MOMENTARILY DIPPED CAUSING THE ASV SYSTEM TO ACTUATE AND THE INSTRUMENT BUS FOR R-24 TO SWITCH FROM ITS PREFERRED TO ALTERNATE POWER SOURCE. THE ROOT CAUSES OF THESE EVENTS ARE: 1) THE AGE OF THE RADIATION MONITORING SYSTEM. AS A RESULT OF THE SYSTEM'S AGE, THE SYSTEM REQUIRES FREQUENT MAINTENANCE AND IDENTICAL REPLACEMENT PARTS ARE SCARCE, 2) DESIGN OF THE SYSTEM DOES NOT FACILITATE DE-ENERGIZING INDIVIDUAL DRAWERS OR REMOVING THE DRAWERS FOR MAINTENANCE. THUS, WORK IS PERFORMED ON ENERGIZED COMPONENTS IN A CRAMPED ENVIRONMENT. MODIFICATIONS WILL BE MADE TO THE EXISTING RADIATION MONITORS TO ALLOW THEM TO BE INDIVIDUALLY DE-ENERGIZED PRIOR TO STARTING MAINTENANCE.

[78] KEWAUNEE DOCKET 50-305 LER 89-015
PIPE RUPTURE RESTRAINT NOT INSTALLED AS DESIGNED RESULTS IN A CONDITION OUTSIDE THE PLANT'S DESIGN BASIS.
EVENT DATE: 081689 REPORT DATE: 091589 NSSS: WE TYPE: PWR

(NSIC 215253) AT 1120, ON 8/16/89, WITH THE PLANT AT 100% POWER, WISCONSIN PUBLIC SERVICE CORPORATION WAS NOTIFIED THAT RUPTURE RESTRAINT 2152, AS INSTALLED, WOULD NOT HAVE MET ITS DESIGN BASIS. THE RESTRAINT IS LOCATED INSIDE CONTAINMENT ON THE RESIDUAL HEAT REMOVAL (RHR) RETURN LINE TO THE REACTOR COOLANT SYSTEM (RCS). TWO BOLTS AND TWO 1/2 INCH THICK STEEL PLATES WERE DISCOVERED MISSING FROM THE RESTRAINT ON 3/20/89, WHILE THE PLANT WAS IN REFUELING SHUTDOWN. UPON DISCOVERING THAT THE RESTRAINT WAS NOT BUILT AS DESIGNED, THE PERSONNEL INVOLVED INITIATED A WORK REQUEST TO INSTALL THE MISSING PARTS. AS PART OF THE INVESTIGATION INTO THE INCIDENT, KEWAUNEE'S ARCHITECT ENGINEER (FLUOR DANIEL, FORMERLY PIONEER SERVICES AND ENGINEERING) WAS CONTRACTED TO ANALYZE THE RESTRAINT'S AS BUILT CONDITION. IT WAS FLUOR DANIEL WHO CONTACTED WPSC ON AUGUST 16. THE RUPTURE RESTRAINT WAS INSTALLED DURING ORIGINAL PLANT CONSTRUCTION. IT APPEARS THAT THE MISSING PARTS WERE NOT INSTALLED DURING ORIGINAL CONSTRUCTION. DUE TO THE LENGTH OF TIME BETWEEN PLANT CONSTRUCTION AND THE DISCOVERY THAT THE PARTS WERE MISSING (APPROXIMATELY 15 YEARS), THE CAUSE OF THIS EVENT CAN NOT BE DETERMINED. IN ORDER TO CORRECT THIS NONCONFORMANCE WITH THE PLANTS DESIGN BASIS, THE MISSING BOLTS AND PLATES WERE INSTALLED PRIOR TO THE END OF THE 1989 REFUELING OUTAGE.

[79] LA SALLE 1 DOCKET 50-373 LER 89-009 REV 01
UPDATE ON REACTOR SCRAM DUE TO LOSS OF MAIN GENERATOR RESULTING IN LOSS OF UNIT 2 SYSTEM AUXILIARY TRANSFORMER CAUSED BY INADVERTENT PHASE TO GROUND FAULT DURING HIGH WIND CONDITIONS.
EVENT DATE: 030289 REPORT DATE: 062789 NSSS: GE TYPE: BWR

OTHER UNITS INVOLVED: LA SALLE 2 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 214561) ON 3/3/89, AT 2302 HRS A PHASE TO GROUND FAULT OCCURRED AT "C" PHASE LIGHTNING ARRESTOR ON THE PRIMARY SIDE OF THE UNIT 2 SYSTEM AUX. TRANSFORMER. FAULT WAS AUTOMATICALLY ISOLATED BY THE TRIPPING OF SWITCHYARD OIL CIRCUIT BREAKERS (OCB) 4-6 AND 6-1 AND UNIT 2 FEEDER BREAKERS. ALL LOADS BEING FED FROM THE SAT TRANSFERRED TO UNIT 2 UNIT AUX. TRANSFORMER EXCEPT FOR BUS 243 WHICH WAS SUPPLIED BY 2B DIESEL GENERATOR WHICH SATISFACTORILY AUTO-STARTED ON UNDERVOLTAGE. UNIT 2 REMAINED ON-LINE AFTER THE INCIDENT. AS A RESULT, UNIT 1 GENERATOR PROTECTIVE RELAYING SENSED A HIGH GENERATOR DIFFERENTIAL CURRENT ON PHASE A AND ISOLATED UNIT 1 GENERATOR. UNIT 1 TURBINE TRIPPED ON LOAD REJECTION RESULTING IN A REACTOR SCRAM FROM TURBINE CONTROL VALVE FAST CLOSURE. UNIT 1 PROCEEDED INTO NORMAL POST-SCRAM CONDITIONS WITH THE EXCEPTION OF TEMPORARY LOSS OF THE SERVICE AIR COMPRESSOR AND PLANT PROCESS COMPUTER. PROBLEMS WERE ALSO ENCOUNTERED WITH THE RESETTING OF THE SCRAM LOGIC. CAUSE OF THIS EVENT WAS THE PHASE TO GROUND FAULT THAT OCCURRED FROM THE LIGHTNING ARRESTOR TOP CAP TO A SPARGER HEAD ON THE TRANSFORMER DELUGE SYSTEM. THIS WAS EVIDENT FROM ARC BURNING IDENTIFIED AT TOP OF LIGHTNING ARRESTOR AND AT SPARGER HEAD. FAULT WAS CAUSED BY DEBRIS THAT HAD BLOWN ONTO THE LIGHTNING ARRESTOR LEAD.

[80] LA SALLE 1 DOCKET 50-373 LER 89-022
REACTOR CORE ISOLATION COOLING ISOLATION DURING WARMUP DUE TO A SPURIOUS HIGH STEAM FLOW SIGNAL.
EVENT DATE: 081089 REPORT DATE: 090889 NSSS: GE TYPE: BWR

(NSIC 215271) ON 8/10/89, AT 1755 HOURS THE REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) RECEIVED A DIVISION I AND DIVISION II (OUTBOARD AND INBOARD) ISOLATION ON RCIC HIGH STEAM LINE FLOW. UNIT 1 WAS IN OPERATIONAL CONDITION 1 AT 98% POWER. UPON COMPLETION OF LASALLE INSTRUMENT SURVEILLANCE LIS-RI-301, UNIT 1 STEAM LINE HIGH FLOW RCIC ISOLATION FUNCTIONAL TEST, RCIC WAS BEING RETURNED TO STANDBY PER LASALLE OPERATING PROCEDURE, LOP-RI-05, PREPARATION FOR STANDBY OF THE RCIC SYSTEM, WHEN THE OUTBOARD ISOLATION VALVE (1E51-F008), AND INBOARD ISOLATION VALVES (1E51-F076 AND 1E51-F063) ISOLATED DUE TO A SPURIOUS HIGH STEAM FLOW SIGNAL. THE RCIC STEAM LINE ISOLATION WAS NOT A VALID HIGH STEAM FLOW SIGNAL. A STEAM/WATER MIXTURE TRAPPED BETWEEN THE INBOARD VALVES AND THE OUTBOARD VALVE WAS RELEASED INTO THE PIPING DOWNSTREAM OF THE OUTBOARD VALVE WHEN IT WAS OPENED. THE SURGE FROM THIS RELEASE WAS SUFFICIENT TO EXCEED THE TRIP SETPOINT OF THE HIGH FLOW ISOLATION SWITCHES. WHEN THE ISOLATION SIGNAL WAS RECEIVED, THE OUTBOARD ISOLATION VALVE HAD BEEN ALREADY CLOSED, RCIC SYSTEM PIPING WAS VERIFIED AND THE ISOLATION LOGIC WAS RESET. WARMING UP OF THE RCIC SYSTEM PROCEEDED WITH NO FURTHER INCIDENTS. THIS EVENT IS REPORTABLE TO THE NRC AS A LER IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV) DUE TO AN ENGINEERED SAFETY FEATURE ACTUATION.

[81] LA SALLE 2 DOCKET 50-374 LER 88-014 REV 01
UPDATE ON TYPE B AND TYPE C TOTAL LEAKAGE EXCEEDED 0.6 LA DURING LEAK RATE TESTING DURING REFUEL OUTAGE TWO.
EVENT DATE: 102188 REPORT DATE: 092289 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 215296) ON 10/17/88, WITH UNIT 2 IN COLD SHUTDOWN DURING ITS REFUEL OUTAGE, IT WAS DETERMINED THAT THE TOTAL AS FOUND CONTAINMENT LEAKAGE RATE DUE TO TYPE B AND TYPE C TESTING EXCEEDED THE TECH SPEC LIMIT OF 0.6 LA (231.4 SCFH). THE SPECIFIC TESTS THAT IDENTIFIED LEAKAGES OVER THE 0.6 LA LIMIT INVOLVED THE HYDROGEN RECOMBINER COMBUSTIBLE GAS CONTROL "A" RETURN VALVES 2HG005A AND 2HG006A. THE LEAKAGE WAS DETERMINED TO BE IN THE 2HG006A VALVE. THE VALVE WAS FOUND TO HAVE DIRTY SEATS. THE SEATS WERE CLEANED AND LAPPED. THIS REPORT IS BEING SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(II) AND INCLUDES

OTHER CONTAINMENT LEAK RATE TEST FAILURES FROM UNIT 2 SECOND REFUEL OUTAGE, WHICH CONTRIBUTED TO EXCEEDING THE TECH SPEC LIMIT.

[82] LA SALLE 2 DOCKET 50-374 LER 89-009
REACTOR CORE ISOLATION COOLING HI STEAM FLOW ISOLATION SWITCH DIAPHRAGM LEAK.
EVENT DATE: 061989 REPORT DATE: 071989 NSSS: GE TYPE: BWR
VENDOR: STATIC-O-RING

(NSIC 214929) ON 6/19/89, AT 0905 HOURS, WHILE UNIT 2 WAS IN OPERATIONAL CONDITION 1 (RUN), REACTOR CORE ISOLATION COOLING (RCIC) STEAM LINE HIGH FLOW ISOLATION SWITCH PDS-2E31-N013AA WAS FOUND TO HAVE A DIAPHRAGM LEAK. THE SETPOINT FOR THIS SWITCH WAS FOUND WITHIN THE ACTION LIMIT AND THE LIMITING CONDITION FOR OPERATION. THE RCIC SYSTEM WAS MAINTAINED OPERABLE DURING THE EVENT. THE HIGH PRESSURE CORE SPRAY SYSTEM WAS INOPERABLE DUE TO THE REPLACEMENT OF THE 2B DIESEL GENERATOR AT THE TIME OF THE EVENT. THE PRESSURE DIFFERENTIAL SWITCH PDS-2E31-N013AA WAS REMOVED AND REPLACED, TESTED AND DECLARED OPERABLE ON 6/19/89 AT 1415 HOURS. THE FAILED PRESSURE DIFFERENTIAL SWITCH PDS-2E31-N013AA WILL BE DISASSEMBED AND INSPECTED WITH THE FINDINGS OF THE INSPECTION TO BE REPORTED IN A SUPPLEMENT TO THIS REPORT. THIS EVENT IS REPORTED TO THE NUCLEAR REGULATORY COMMISSION AS A VOLUNTARY LICENSEE EVENT REPORT IN ACCORDANCE WITH THE REQUIREMENTS OF I.E. BULLETIN 86-02, "STATIC-O-RING (SOR) DIFFERENTIAL PRESSURE SWITCHES."

[83] LA SALLE 2 DOCKET 50-374 LER 89-011
SPURIOUS REACTOR PROTECTION SYSTEM ACTUATION DUE TO UNKNOWN CAUSE.
EVENT DATE: 082689 REPORT DATE: 092589 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 215377) ON 8/26/89, A CONTROLLED SHUTDOWN WAS IN PROGRESS ON UNIT 2. WHILE CONDUCTING LOS-TG-SA2, "TURBINE VALVE LEAK TIGHTNESS SURVEILLANCE," A REACTOR PROTECTION SYSTEM (RPS) ACTUATION OCCURRED AT 0414 HOURS. WHEN THE ACTUATION SIGNAL WAS RECEIVED, TWO OF THE FOUR SCRAM GROUP LIGHTS OF THE RPS BUS A REMAINED ENERGIZED. THIS PREVENTED SOME OF THE RODS FROM RECEIVING THE NORMAL AUTOMATIC SCRAM ACTUATION. ROD MOTION APPEARS TO HAVE INITIATED FOR THESE RODS DUE TO THE CHANNEL A BACKUP SCRAM ACTUATION WHICH OCCURRED AT THE SAME TIME AS THE INITIAL EVENT. THE CONTROL ROOM OPERATOR MANUALLY INITIATED A NORMAL SCRAM SIGNAL A FEW SECONDS LATER USING THE A2 AND B2 SCRAM PUSHBUTTONS. AT THIS TIME ALL THE REMAINING SCRAM VALVES DEENERGIZED, INDICATING THAT THE SCRAM HAD OCCURRED. BEFORE THE TURBINE VALVE TEST WAS STARTED, THE HATHAWAY SEQUENCE OF EVENTS ALARM TYPER WAS TURNED OFF DUE TO ITS CONSTANT PRINTING CAUSED BY ALARM RELAY CHATTERING. AS A RESULT, THE MAIN SOURCE OF INFORMATION TO BE USED IN ANALYZING THE REACTOR TRIP WAS NOT AVAILABLE. AT THE TIME OF THE TRIP, NO PLANT PARAMETERS EXCEEDED THEIR TRIP SETPOINTS. AS A RESULT OF THE MISSING INFORMATION, SEVERAL SCENARIOS WERE DEVELOPED USING AVAILABLE INFORMATION IN AN ATTEMPT TO DETERMINE THE CAUSE OF THE REACTOR TRIP.

[84] LIMERICK 1 DOCKET 50-352 LER 89-050
MISSING RCIC PUMPS ALIGNMENT PINS CAUSING SYSTEM INOPERABILITY RESULTING IN A CONDITION PROHIBITED BY TECH SPEC.
EVENT DATE: 010385 REPORT DATE: 092189 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 215318) DURING THE PREPARATION FOR THE INSTALLATION OF THE UNIT 2 REACTOR CORE INJECTION COOLING (RCIC) SYSTEM TURBINE ALIGNMENT PINS, MAINTENANCE PERSONNEL NOTICED THAT NO ALIGNMENT PINS WERE TO BE INSTALLED IN THE RCIC PUMP MOUNTING BRACKETS. THE UNIT 1 RCIC PUMP WAS INSPECTED AND FOUND TO BE WITHOUT THESE ALIGNMENT PINS INSTALLED. IN RESPONSE TO A SEISMIC OR HYDRODYNAMIC EVENT, WITHOUT THE PINS IN PLACE, THE RCIC PUMPS COULD MOVE CAUSING RCIC SYSTEMS

INOPERABILITY. UPON DISCOVERY OF THIS CONDITION, ON 8/25/89, UNIT 1 AND UNIT 2 RCIC SYSTEMS WERE DECLARED INOPERABLE. THE APPROPRIATE ACTION STATEMENTS OF TECH SPECS WERE NOT TAKEN WITHIN THE SPECIFIED TIME PERIOD. THIS CONSTITUTES CONDITIONS PROHIBITED BY TECH SPEC AND IS REPORTABLE IN ACCORDANCE WITH 10CFR50.73(A)(2)(I)(B). THIS WAS A RESULT OF THE LACK OF CLARITY IN THE RCIC PUMP INSTALLATION INSTRUCTIONS. AS AN INTERIM MEASURE NEW TORQUE VALUES WERE PROVIDED BY THE NUCLEAR STEAM SUPPLY SYSTEM SUPPLIER AND USED FOR THE PUMP MOUNTING BRACKETS TO ENSURE SUFFICIENT BRACKET TO PEDESTAL SHEAR LOAD CAPABILITY. IMPLEMENTATION OF THE NEW TORQUE VALUES ALLOWED THE RETURN TO OPERABILITY OF UNIT 1 AND UNIT 2 RCIC SYSTEMS. WHEN THE ALIGNMENT PINS ARE OBTAINED, AND AFTER A HOT ALIGNMENT CHECK, THE RCIC PUMPS WILL BE PINNED. ALSO, THE RCIC PUMP MANUAL WILL BE MODIFIED TO INCLUDE ALIGNMENT PIN INSTALLATION DIRECTION.

[85] LIMERICK 1 DOCKET 50-352 LER 88-042 REV 05
UPDATE ON INOPERABILITY OF PLANT SYSTEMS DUE TO UNACCEPTABLE PHYSICAL SEPARATION BETWEEN SAFETY RELATED CABLING AS A RESULT OF PERSONNEL ERROR.
EVENT DATE: 122388 REPORT DATE: 092389 NSSS: GE TYPE: BWR

(NSIC 215228) BEGINNING ON DECEMBER 23, 1988, THE PLANT STAFF DETERMINED THAT CERTAIN PLANT SYSTEMS WERE INOPERABLE, DUE TO DEFICIENCIES IN THAT PHYSICAL SEPARATION BETWEEN CABLING IN SAFETY RELATED ELECTRICAL PANELS. THE APPROPRIATE CABLING WAS SLEEVED TO COMPLY WITH SEPARATION REQUIREMENTS AND THE SYSTEMS WERE DECLARED OPERABLE. WE HAVE CONCLUDED THAT THESE CONDITIONS, AFFECTING OPERABILITY, HAVE EXISTED SINCE 10/26/84, THE DATE OF ISSUANCE OF THE LOW POWER OPERATING LICENSE. THE CONSEQUENCES OF THIS CONDITION WERE MINIMAL IN THAT NO FAULT OR FIRE CONDITION OCCURRED CAUSING CABLE DEGRADATION. THERE WERE NO RADIOLOGICAL RELEASES TO THE ENVIRONMENT AS A RESULT OF THIS CONDITION. THIS CONDITION RESULTED FROM A PERSONNEL ERROR IN THAT THE CABLES WERE NOT PROPERLY INSTALLED. ADDITIONALLY, THE REQUIRED PHYSICAL SEPARATION INSPECTIONS OF COMMON PANELS WERE NOT ADEQUATELY DOCUMENTED AND TRACKED TO ENSURE THAT ALL INSPECTIONS WERE COMPLETED PRIOR TO UNIT 1 STARTUP. ALL FINAL PHYSICAL SEPARATION INSPECTIONS WERE SATISFACTORILY COMPLETED. TO FURTHER PREVENT THESE PHYSICAL SEPARATION DEFICIENCIES FROM OCCURRING THROUGH THE UNIT 2 CONSTRUCTION AND STARTUP PHASES, CONTROL OF THE CONSOLIDATED OPEN ITEMS LIST WILL BE INCORPORATED IN A PLANT PROCEDURE.

[86] LIMERICK 1 DOCKET 50-352 LER 89-034 REV 01
UPDATE ON PRESSURE LEVEL AND FLOW INDICATION WERE INOPERABLE DUE TO AN INADEQUATE MODIFICATION REVIEW.
EVENT DATE: 051289 REPORT DATE: 082589 NSSS: GE TYPE: BWR

(NSIC 215134) ON MAY 12, 1989, WITH UNIT 1 IN A REFUELING OUTAGE, STATION PERSONNEL WERE NOTIFIED THAT CERTAIN EXCESS FLOW CHECK VALVE TEST VALVES WOULD NOT REMAIN LEAK TIGHT DURING POST DESIGN BASIS ACCIDENT (DBA) CONDITIONS. NINETEEN (19) TEST VALVES IN VARIOUS APPLICATIONS WERE FOUND TO BE IN NONCONFORMANCE WITH REGULATORY GUIDE (RG) 1.97 REQUIREMENTS. THE TEST VALVE MATERIALS REVEALED TEFLON COATED VITON SEALS THAT FORMED THE PRESSURE BOUNDARY BETWEEN THE VALVE PLUG AND BODY WOULD BREAKDOWN DURING POST DBA CONDITIONS CAUSING A LOSS OF PRESSURE BOUNDARY AND INSTRUMENTATION. THE NINETEEN (19) VALVES WERE REMOVED BY MAY 13, 1989. ON MAY 22, 1989, SIX (6) OF THE NINETEEN TEST VALVES WERE IDENTIFIED BY PLANT STAFF AS REQUIRED TO BE OPERABLE BY R.G. 1.97 AND TECHNICAL SPECIFICATIONS (TS) SECTION 3.3.7.5. THE CONSEQUENCES OF THIS CONDITION WERE MINIMAL IN THAT A DBA CONDITION DID NOT OCCUR DURING THE TIME PERIOD THAT THE VALVES WERE INSTALLED AND, THEREFORE, NO DEGRADATION OF THE PRESSURE BOUNDARY OCCURRED. THE CAUSE OF THE EVENT WAS A RESULT OF INADEQUATE REVIEW OF SYSTEM DESIGN SPECIFICATIONS BY PLANT STAFF SUPERVISION PRIOR TO RECEIPT OF THE LOW POWER OPERATING LICENSE. THIS LER REPORTS A CONDITION PROHIBITED BY TS AND A CONDITION THAT COULD HAVE PREVENTED THE FULFILLMENT OF A SAFETY FUNCTION.

[87] LIMERICK 1 DOCKET 50-352 LER 89-038
REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION DUE TO TESTMETER NOT REMOVED FROM STEAM LEAK DETECTION SYSTEM DUE TO PERSONNEL ERROR.
EVENT DATE: 052889 REPORT DATE: 062789 NSSS: GE TYPE: BWR

(NSIC 215317) ON 5/28/89, AT 0615 HOURS, DURING PERFORMANCE OF A SURVEILLANCE TEST, THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM ISOLATED AND THE RCIC TURBINE RECEIVED AN TRIP SIGNAL ON RECEIPT OF A STEAM LEAK DETECTION SYSTEM HIGH DIFFERENTIAL TEMPERATURE ISOLATION SIGNAL. THE STEAM LEAK DETECTION SYSTEM IS AN ENGINEERED SAFETY FEATURE (ESF). THE ISOLATION WAS CAUSED BY PROCEDURAL NON-COMPLIANCE DUE TO PERSONNEL ERROR. A CONTRACT INSTRUMENTATION AND CONTROLS (I&C) TECHNICIAN WORKING UNDER LICENSEE CONTROL AND SUPERVISION DID NOT REMOVE A TEST METER AS REQUIRED BY THE PROCEDURE. THE CONSEQUENCES OF THE EVENT WERE MINIMAL. THE ISOLATION WAS RESET AT 0630 HOURS BY OPERATIONS PERSONNEL. THE TECHNICIANS INVOLVED HAVE BEEN DISCIPLINED AND THE NEED FOR PROCEDURAL COMPLIANCE RE-EMPHASIZED WITH ALL I&C TECHNICIANS. THE NRC WAS NOTIFIED IN ACCORDANCE WITH 10CFR 50.72(B)(2)(II) AT 0745 ON 5/28/89.

[88] LIMERICK 1 DOCKET 50-352 LER 89-041 REV 01
UPDATE ON ALL CHANNELS OF POST-LOCA RADIATION MONITORS INSTRUMENTATION INOPERABLE SINCE INITIAL LOW POWER OPERATION DUE TO DEFICIENT CALIBRATION PROCEDURE.
EVENT DATE: 061389 REPORT DATE: 082989 NSSS: GE TYPE: BWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 215135) ON JUNE 13, 1989, THE PLANT STAFF DETERMINED THAT THE FOUR CHANNELS OF THE UNIT 1 PRIMARY CONTAINMENT POST-LOCA (LOSS OF COOLANT ACCIDENT) RADIATION MONITOR (PLRM) INSTRUMENTATION WERE INOPERABLE FOR A TIME PERIOD IN EXCESS OF THE LIMERICK GENERATING STATION TECHNICAL SPECIFICATIONS (TS) SECTION 3.3.7.5 LIMIT WITHOUT THE ASSOCIATED REQUIRED ACTION BEING TAKEN. THIS RESULTED IN A CONDITION PROHIBITED BY TS. PLANT STAFF DISCOVERED THAT THE SURVEILLANCE REQUIREMENTS FOR CALIBRATION AS DEFINED BY TS SECTION 3.3.7.5 WERE NOT COMPLETELY SATISFIED BY THE ESTABLISHED 18-MONTH SURVEILLANCE TEST (ST). THE PRIMARY CAUSE OF THIS EVENT WAS A PROCEDURAL DEFICIENCY. THE SURVEILLANCE TESTS, INTENDED TO SATISFY THE CALIBRATION REQUIREMENTS, DID NOT ADDRESS THE ANALOG PORTION OF THE CHANNEL ELECTRONICS; ONLY THE DIGITAL PORTION WAS PROPERLY CALIBRATED. A CAUSAL FACTOR RESULTING IN THE PROCEDURAL DEFICIENCY WAS MIS-INTERPRETATION OF INFORMATION PROVIDED VERBALLY BY THE VENDOR/SUPPLIER OF THE PLRMS TO THE ST WRITER. A TEMPORARY PROCEDURE CHANGE (TPC) TO THE AFFECTED STS, WAS IMPLEMENTED INCORPORATING CALIBRATION OF THE ANALOG PORTION AND THE STS WERE THEN PERFORMED. THE TS REQUIRED MINIMUM NUMBER OF CHANNELS OF THE PLRMS WERE DECLARED OPERABLE ON JUNE 14, 1989. THE AFFECTED STS HAVE BEEN PERMANENTLY REVISED.

[89] LIMERICK 1 DOCKET 50-352 LER 89-045
PARTIAL ISOLATION OF NSSSS GROUP VI C CAUSED BY FAULTY HIGH RADIATION SIGNAL RESULTING FROM A FAILED GM TUBE.
EVENT DATE: 071689 REPORT DATE: 081689 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215026) ON 7/15/89, AT APPROX. 0900 HOURS, MAIN CONTROL ROOM PERSONNEL OBSERVED AN INCREASE IN THE UNIT 1 'C' REACTOR ENCLOSURE (RE) VENTILATION EXHAUST RADIATION MONITOR, RISH-026-1K609C, INDICATION TO APPROXIMATELY 0.6 MR/HR. THE INSTRUMENT AND CONTROLS (I&C) GROUP WAS NOTIFIED BY SHIFT SUPERVISION AND A MAINTENANCE REQUEST FORM (MRF) WAS INITIATED TO PERFORM AN INVESTIGATION. BEFORE THE MRF WAS PROCESSED, AT 0255 HOURS ON JULY 16, 1989, THE RADIATION MONITOR INDICATION DRIFTED ABOVE ITS ISOLATION SETPOINT OF 1.35 MR/HR. THIS CAUSED A PARTIAL ISOLATION OF THE FOLLOWING: NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) GROUP VI C, "PRIMARY CONTAINMENT SAMPLING AND RECOMBINER LINES", AND CONTAINMENT BYPASS LEAKAGE BARRIER BLOCKS AND VENTS, CONSTITUTING AN AUTOMATIC ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF). PROPER ISOLATION WAS VERIFIED, THE RADIATION

MONITOR WAS LEFT IN THE TRIPPED CONDITION, AND THE UNIT 1 'C' RADIATION SENSOR/CONVERTER ASSEMBLY WAS REPLACED FOLLOWING THE DISCOVERY OF A FAILED GEIGER/MUELLER (GM) TUBE. THE ISOLATION WAS RESET ON JULY 17, 1989, AT 0023 HOURS. THERE WERE 3 OTHER DETECTORS AVAILABLE WHICH WERE FUNCTIONING PROPERLY AND WERE CAPABLE OF PERFORMING THE DESIGN FUNCTION OF THE SYSTEM. THIS EVENT, CAUSED BY EQUIPMENT FAILURE, IS CONSIDERED TO BE AN ISOLATED OCCURRENCE AND, THEREFORE, NO ADDITIONAL CORRECTIVE ACTIONS ARE PLANNED.

[90]   LIME[illegible] 1   DOCKET 50-352   LER 89-046
NONCOMPLIANCE WITH TECH SPECS WHEN THE STANDBY GAS TREATMENT SYSTEM WAS INOPERABLE BECAUSE OF PERSONNEL ERROR.
EVENT DATE: 071989   REPORT DATE: 081889   NSSS: GE   TYPE: BWR

(NSIC 215136) ON 7/17/89, CHARCOAL FILTER SAMPLE CANISTERS WERE INCORRECTLY REMOVED FROM BOTH STANDBY GAS TREATMENT SYSTEM (SGTS) SUBSYSTEMS. ON 7/19/89, THE CANISTERS WERE INAPPROPRIATELY RE-INSTALLED. BECAUSE THE PHYSICAL CONFIGURATION OF CHARCOAL INSIDE THE CANISTER HAD BEEN ALTERED, STATION PERSONNEL DETERMINED THAT RE-INSTALLATION OF THE SAMPLE CANISTER PROVIDED THE POTENTIAL FOR SGTS FILTER BYPASS LEAKAGE TO BE GREATER THAN THE TECHNICAL SPECIFICATION (TS) ALLOWABLE LIMIT OF 0.05%. THIS ACTION CREATED A CONDITION THAT COULD HAVE PREVENTED THE SGTS FROM FULFILLING ITS SAFETY FUNCTION TO LIMIT THE RELEASE OF RADIOACTIVE MATERIAL. AT 1245 ON 7/19, THIS CONDITION WAS REALIZED AND BOTH SUBSYSTEMS OF THE SGTS WERE DECLARED INOPERABLE. THE SAMPLE CANISTERS WERE REMOVED AND SGTS TS OPERABILITY WAS RE-ESTABLISHED BY 1325. THE CONSEQUENCES OF THIS EVENT WERE MINIMAL IN THAT NO INCIDENT REQUIRING SGTS OCCURRED DURING THE SHORT EVENT TIME PERIOD OF LESS THAN FOUR (4) HOURS. ADDITIONALLY, THE ASSUMPTION THAT THE RE-INSTALLED CANISTERS COULD CAUSE THE SGTS FILTER BYPASS LEAKAGE TO BE GREATER THAN 0.05% WAS CONSERVATIVE. THIS EVENT WAS CAUSED BY PERSONNEL ERROR, INCORRECT REMOVAL AND INAPPROPRIATE RE-INSTALLATION OF SGTS CHARCOAL FILTER SAMPLE CANISTERS. DETAILS OF THIS EVENT AND THE JOB RESPONSIBILITIES OF THE CHEMISTRY ST COORDINATOR HAVE BEEN DISCUSSED WITH PERSONNEL INVOLVED.

[91]   LIMERICK 1   DOCKET 50-352   LER 89-047
SERVICE WATER SYSTEM WAS UNAVAILABLE TO PROVIDE COOLING WATER IF NEEDED TO EMERGENCY CORE COOLING SYSTEM COMPARTMENT COOLERS DUE TO AN INCORRECT VALVE POSITION.
EVENT DATE: 072089   REPORT DATE: 082989   NSSS: GE   TYPE: BWR

(NSIC 215137) ON JULY 20, 1989, PHILADELPHIA ELECTRIC IDENTIFIED THAT THE SERVICE WATER (SW) SYSTEM WAS UNAVAILABLE TO PROVIDE COOLING WATER IF NEEDED TO THE EMERGENCY CORE COOLING SYSTEM (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM COMPONENTS DUE TO AN INCORRECT VALVE POSITION. AN EVALUATION WAS INITIATED TO DETERMINE THE IMPACT OF SW BEING ISOLATED FROM THE EMERGENCY SERVICE WATER (ESW) SYSTEM. THE EVALUATION CONCLUDED THAT THE ESW SYSTEM MAY HAVE NOT BEEN PLACED IN SERVICE DURING SPECIFIC LOSS OF COOLANT ACCIDENTS (LOCA) WHICH INITIATE RCIC AND HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM WHICH DO NOT AUTOMATICALLY INITIATE THE ESW SYSTEM. THE HPCI AND RCIC SYSTEM RECEIVE A DIFFERENT AUTOMATIC START SIGNAL THAN THE ESW PUMP AUTOMATIC START SIGNAL. AS A RESULT, UNDER CERTAIN LOCA SCENARIOS, WHICH DO NOT RESULT IN AUTOMATIC START OF ESW, MANUAL INITIATION OF ESW IS REQUIRED. THE EMERGENCY OPERATING PROCEDURES DID NOT ADVISE THE OPERATORS TO CHECK FOR ESW OPERATION OR SUFFICIENT COOLING WATER FLOW. THE LACK OF COOLANT FLOW DURING ACCIDENT CONDITIONS COULD PREVENT THE HPCI AND RCIC SYSTEMS FROM PERFORMING THEIR INTENDED SAFETY FUNCTIONS SINCE HIGH TEMPERATURE IN THE AREAS COULD HAVE DAMAGED THE SAFETY RELATED EQUIPMENT. THE EMERGENCY OPERATING PROCEDURES HAVE BEEN REVISED TO CHECK FOR ESW OPERATION DURING LOCA CONDITIONS.

[92] LIMERICK 1 DOCKET 50-352 LER 89-048
TECHNICIAN TESTS THE INCORRECT INSTRUMENT RESULTING IN AN AUTOMATIC ACTUATION OF PORTIONS OF THE PRIMARY AND SECONDARY CONTAINMENT SYSTEM.
EVENT DATE: 080189 REPORT DATE: 083089 NSSS: GE TYPE: BWR

(NSIC 215167) ON 8/1/89, AT 0954 HOURS, VARIOUS UNIT 1 PRIMARY AND SECONDARY CONTAINMENT ISOLATION VALVES RECEIVED ISOLATION SIGNALS WHEN A UTILITY EMPLOYED INSTRUMENTATION AND CONTROLS (I&C) TECHNICIAN INCORRECTLY TESTED THE UNIT 1 'D' REFUEL AREA VENTILATION EXHAUST DUCT RADIATION MONITOR INSTEAD OF THE UNIT 1 'D' REACTOR ENCLOSURE VENTILATION EXHAUST DUCT RADIATION MONITOR, DURING THE PERFORMANCE OF A SURVEILLANCE TEST. TRIPPING OF THE INCORRECT MONITOR RESULTED IN AN AUTOMATIC ACTUATION OF CERTAIN CONTAINMENT ISOLATIONS, AN ENGINEERED SAFETY FEATURE (ESF), DUE TO THE UNANTICIPATED SIGNAL. THE CONSEQUENCES OF THIS EVENT WERE MINIMAL IN THAT NO RADIOACTIVE MATERIAL WAS RELEASED TO THE ENVIRONMENT AND NO ADVERSE CONSEQUENCES WOULD HAVE RESULTED HAD THERE BEEN AN ACTUAL REFUELING FLOOR HIGH RADIATION SITUATION, SINCE THE ISOLATIONS OCCURRED AS DESIGNED. THE 'D' REFUEL AREA RADIATION MONITOR CHANNEL TRIP SIGNAL AND THE ASSOCIATED ISOLATIONS WERE RESET AT 1038 HOURS. THE DURATION OF THE ISOLATION EVENT WAS 44 MINUTES. THE I&C TECHNICIANS PERFORMING THE TESTING WERE COUNSELED REGARDING PROPER WORK PRACTICES AND ATTENTION TO DETAIL.

[93] LIMERICK 1 DOCKET 50-352 LER 89-051
NORMAL MCR VENTILATION WAS IN OPERATION WITH THE CHLORINE DETECTION SYSTEM INOPERABLE RESULTING IN A CONDITION PROHIBITED BY TECH SPECS.
EVENT DATE: 083189 REPORT DATE: 100289 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 215368) ON 8/31/89, AT 0300 HOURS, WITH THE MAIN CONTROL ROOM (MCR) VENTILATION SYSTEM CHLORINE DETECTORS INOPERABLE, THE CONTROL ROOM EMERGENCY FRESH AIR SUPPLY SYSTEM WAS INADVERTENTLY SECURED FROM THE CHLORINE ISOLATION MODE OF OPERATION AND NORMAL MCR VENTILATION WAS RESTORED. THIS CONSTITUTES A CONDITION PROHIBITED BY TECH SPECS SINCE THE REQUIRED TECH SPEC ACTION WAS NOT SATISFIED WHEN OPERATING THE NORMAL MCR VENTILATION SYSTEM WITH THE CHLORINE DETECTION SYSTEM INOPERABLE. THE NORMAL MCR VENTILATION WAS RESTORED BY THE MCR CHIEF OPERATOR (CO). THE CO WAS UNAWARE THAT THE CHLORINE DETECTION SYSTEM HAD BEEN RESTORED TO A FULLY FUNCTIONAL STATUS DURING THE POST MODIFICATION TESTING OF THE ISOLATION LOGIC WHICH WAS PERFORMED BY THE FIELD ENGINEERS (FE). THE CAUSE OF THE EVENT WAS A PERSONNEL ERROR DUE TO INADEQUATE COMMUNICATION BETWEEN THE FE AND CO. THE FE FAILED TO INFORM THE CO OF THE CONFIGURATION OF THE MCR VENTILATION SYSTEM AFTER THE COMPLETION OF THE MODIFICATION TESTING. TO CORRECT THE DEFICIENCIES IN COMMUNICATION BETWEEN MCR OPERATORS AND FE, A MEMO WAS ISSUED TO ALL FES EMPHASIZING THE IMPORTANCE OF EFFECTIVELY COMMUNICATING THE STATUS AND CONFIGURATION OF SYSTEMS BEING TESTED TO THE MCR OPERATORS.

[94] LIMERICK 2 DOCKET 50-353 LER 89-003
INTERMEDIATE RANGE MONITOR UPSCALE TRIPS RESULTING IN RPS SCRAM SIGNALS/SINGLE CHANNEL SCRAM SIGNALS DUE TO A FAULTY PREAMPLIFIER/UNDERVESSEL WORK ACTIVITIES.
EVENT DATE: 071789 REPORT DATE: 081689 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215399) ON 7/17/89, AT 1306, WITH ALL CONTROL RODS FULLY INSERTED A FULL REACTOR PROTECTION SYSTEM (RPS) SCRAM SIGNAL WAS GENERATED. THE SCRAM SIGNAL WAS INITIATED BY AN UPSCALE TRIP OF THE "B" INTERMEDIATE RANGE MONITOR (IRM). ALSO ON 7/22/89, AT 1510 A SECOND SHORT DURATION "B" IRM UPSCALE TRIP OCCURRED WHICH INITIATED A FULL RPS SCRAM SIGNAL. THE RPS WAS IN A NON-COINCIDENT CONFIGURATION. NEITHER EVENT RESULTED IN ROD MOTION. OPERATIONS IDENTIFIED THE SCRAM SIGNALS AND PROMPTLY RESPONDED IN ACCORDANCE WITH PROCEDURES. THE SCRAM SIGNALS WERE RESET AND UNIT 2 WAS RETURNED TO THE PRE-EVENT CONDITIONS PER APPROPRIATE PROCEDURES. THE "B" IRM UPSCALE TRIPS WERE DUE TO A FAULTY

PREAMPLIFIER. THE "B" IRM PREAMPLIFIER WAS REPLACED AND CALIBRATED SATISFACTORILY ON 8/9/89. ADDITIONALLY, ON 7/21/89, AT 2202 WITH RPS IN A NON-COINCIDENT CONFIGURATION, A MOMENTARY UPSCALE TRIP OF THE "F" IRM RESULTED IN A SINGLE CHANNEL RPS TRIP AND DID NOT GENERATE A FULL SCRAM SIGNAL AS EXPECTED. THIS RESPONSE WAS INVESTIGATED AND FOUND TO BE IN ACCORDANCE WITH DESIGN. THE MOMENTARY UPSCALE TRIP ON THE "F" IRM WAS CAUSED BY INADVERTENT CONTACT TO THE "F" IRM ASSEMBLY BY MAINTENANCE PERSONNEL DURING WORK IN THE UNDERVESSEL AREA. THE MAINTENANCE PERSONNEL WERE EXERCISING CAUTION DURING UNDERVESSEL WORK ACTIVITIES DUE TO THE KNOWN SENSITIVITY OF EQUIPMENT IN THE AREA.

[95] LIMERICK 2 DOCKET 50-353 LER 89-005
SPECIAL REPORT FOR DIESEL GENERATOR SURVEILLANCE TEST FAILURE.
EVENT DATE: 080389 REPORT DATE: 090189 NSSS: GE TYPE: BWR

(NSIC 215168) ON 8/3/89, THE UNIT 2 D24 DIESEL GENERATOR WAS SHUT DOWN MANUALLY BEFORE THE TECH SPEC OPERABILITY SURVEILLANCE TEST WAS COMPLETED. APPROXIMATELY 15 MINS AFTER THE ENGINE START PER THE APPROVED PROCEDURE, A LOCAL OPERATOR DISCOVERED A LEAK (OIL SPRAY) FROM A CRACK IN A LUBE OIL PRESSURE GAGE SENSING LINE. THE TERMINATED TEST WAS CLASSIFIED AS A VALID TEST FAILURE PER REGULATORY GUIDE 1.08. THE D24 DIESEL GENERATOR WAS DECLARED INOPERABLE AND THE SENSING LINE WAS REPAIRED. ON 8/5/89, D24 DIESEL GENERATOR WAS RUN SUCCESSFULLY AND DECLARED OPERABLE. INSPECTIONS WERE PERFORMED AND NO SIGNS OF SIMILAR CRACKING WERE FOUND ON THE OTHER UNIT 2 AND UNIT 1 DIESEL GENERATORS. THE MOST LIKELY CAUSE OF THE CRACKED SENSING LINE IS THE RESULT OF INADVERTENT DAMAGE INITIATED DURING CONSTRUCTION OR PREOPERATIONAL TESTING ACTIVITIES AND SUBSEQUENT DIESEL GENERATOR OPERATION WHICH WEAKENED THE PIPING AT A THREADED CONNECTION, THUS CREATING A CRACK. THE CONSEQUENCES OF THIS EVENT WERE MINIMAL BECAUSE THE FAILURE WAS DISCOVERED DURING ROUTINE SURVEILLANCE AND THE REMAINING OPERABLE DIESEL GENERATORS COULD HAVE PROVIDED ADEQUATE POWER TO ENSURE SAFETY SHUTDOWN OF THE PLANT IN THE EVENT OF A LOSS OF OFFSITE POWER. THIS SPECIAL REPORT IS BEING SUBMITTED PURSUANT TO TECH SPEC 6.9.2 AND TECH SPEC 4.8.1.1.3 REPORTS - ALL DIESEL GENERATOR FAILURES.

[96] LIMERICK 2 DOCKET 50-353 LER 89-006
MISSED EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 080789 REPORT DATE: 090189 NSSS: GE TYPE: BWR

(NSIC 215169) ON 8/8/89, STATION PERSONNEL DETERMINED THAT SURVEILLANCE REQUIREMENTS FROM THE UNIT 2 TECH SPECS TABLE 4.3.3.1-1 (EMERGENCY CORE COOLING SYSTEM (ECCS) ACTUATION INSTRUMENTATION) HAD NOT BEEN MET BECAUSE THE SURVEILLANCE TEST (ST) HAD NOT BEEN PERFORMED RENDERING THE INSTRUMENTATION INOPERABLE. THE ASSOCIATED ACTIONS OF THE LIMITING CONDITION OF OPERATION HAD NOT BEEN TAKEN WITHIN THE SPECIFIED TIME FRAME RESULTING IN A CONDITION PROHIBITED BY TECH SPEC. THE CAUSE OF THE EVENT IS PERSONNEL ERROR IN THAT THE ST WAS PLACED ON A LIST SCHEDULED TO BE PERFORMED AFTER THE EXPIRATION OF THE SST'S SURVEILLANCE INTERVAL. REDUNDANT INSTRUMENTATION WAS IN SURVEILLANCE AND OPERABLE. THE ST WAS SATISFACTORILY COMPLETED AT 2113 ON 8/8/89. THE PERSONNEL INVOLVED WERE COUNSELED ON THE NEED FOR MAINTAINING ATTENTION TO DETAIL. ADDITIONAL CORRECTIVE ACTIONS IN THIS REGARD ARE; UNIT 2 INSTRUMENTATION AND CONTROLS (I&C) STS WILL BE PERFORMED FROM A SINGLE LIST AND THE ST COORDINATOR WILL NOTIFY THE I&C SUPERVISOR OF STS THAT ARE CLOSE TO EXCEEDING THEIR SURVEILLANCE INTERVAL.

[97] MCGUIRE 1 DOCKET 50-369 LER 89-021
BOTH TRAINS OF ANNULUS VENTILATION WERE FOUND TO BE INOPERABLE BECAUSE OF DESIGN DEFICIENCY IN SELECTION OF THE SYSTEM SETPOINTS.
EVENT DATE: 062389 REPORT DATE: 092589 NSSS: WE TYPE: PWR

OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 215375) IN 9/88, THE NRC ISSUED INFORMATION NOTICE 88-76 TO ALERT LICENSEES TO THE DISCOVERY OF THE POTENTIAL FOR INADEQUATE DESIGN OF THE SECONDARY CONTAINMENT PRESSURE CONTROL SYSTEM TO CAUSE THE SECONDARY CONTAINMENT PRESSURE TO RISE ABOVE ALLOWABLE VALUES. SUBSEQUENT EVALUATION BY DESIGN ENGINEERING PERSONNEL REVEALED THAT DESIGN OF THE SYSTEM DID NOT TAKE INTO ACCOUNT THE TEMPERATURE - INDUCED DIFFERENCE IN THE PRESSURE GRADIENTS INSIDE AND OUTSIDE THE ANNULUS. THEREFORE, CHANGES IN OUTSIDE AIR TEMPERATURE AND HUMIDITY COULD CAUSE THE DIFFERENTIAL PRESSURE AT HIGHER ELEVATIONS OF THE ANNULUS TO BE LESS NEGATIVE THAN THE TECH SPEC REQUIREMENT OF NEGATIVE 0.5 INCHES W.G. THIS EVENT IS ASSIGNED A CAUSE OF DESIGN DEFICIENCY BECAUSE OF UNANTICIPATED ENVIRONMENTAL INTERACTION. UNIT 1 WAS IN MODE 1, POWER OPERATION, AT THE TIME OF THE EVENT DISCOVERY. CONSEQUENTLY, OPERATIONS PERSONNEL LOGGED UNIT 1 INTO TECH SPEC 3.0.3; HOWEVER, AN EVALUATION OF THE CURRENT CONDITION OF THE ANNULUS VENTILATION SYSTEM AT THAT TIME CONCLUDED THAT CONTINUED OPERATION IN THE EXISTING CONFIGURATION WOULD BE AS SAFE OR SAFER THAN SHUTTING UNIT 1 DOWN. DISCRETIONARY ENFORCEMENT WAS REQUESTED AND SUBSEQUENTLY GRANTED BY NRC/REGION II. UNIT 2 WAS IN MODE 5 AND WAS NOT AFFECTED BY THE EVENT. SYSTEM SETPOINT MODIFICATIONS HAVE BEEN IMPLEMENTED TO PREVENT RECURRENCE OF THIS EVENT.

[98] MCGUIRE 1 DOCKET 50-369 LER 89-014 REV 01
UPDATE ON THE LAND USE CENSUS TECH SPEC REQUIREMENT WAS NOT COMPLETELY FULFILLED BECAUSE OF POOR INTERFACE BETWEEN GROUPS.
EVENT DATE: 071389 REPORT DATE: 081689 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 215171) DURING CLARIFICATION OF THE RESPONSIBILITIES FOR GENERAL OFFICE RADIATION PROTECTION (RP) PERSONNEL, STATION RP PERSONNEL, AND THE ENVIRONMENTAL SERVICES PERSONNEL ON THE LAND USE CENSUS (LUC), IT WAS DISCOVERED THAT PART OF TECH SPEC REQUIREMENT 3.12.2 HAD NEVER BEEN FULFILLED. TECH SPEC 3.12.2 STATES THAT THE LUC LOCATIONS ARE REQUIRED TO BE COMPARED WITH THE CURRENT RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (REMP) SAMPLING LOCATIONS OF THE SAME SAMPLE TYPE AND PATHWAY. IF THE LUC LOCATION CALCULATED DOSE OR DOSE COMMITMENT IS 20% GREATER THAN THE REMP SAMPLING LOCATIONS, THE NEW LUC LOCATION MUST BE IDENTIFIED GREATER THAN THE REMP SAMPLING LOCATIONS, THE NEW LUC LOCATION MUST BE IDENTIFIED IN THE NEXT SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (SERR)J. THE SERR SHALL CONTAIN THE FOLLOWING: (1) NEW LUC LOCATION, (2) CHANGES FOR THE OFFSITE DOSE CALCULATION MANUAL REFLECTING THE NEW LUC LOCATION, AND (3) IF SAMPLES CANNOT BE OBTAINED, AN EXPLANATION OF WHY SAMPLES ARE NOT OBTAINABLE. IF CERTAIN CONDITIONS ARE MET FOR SAMPLING (TECH SPEC 3.12.1) THEN THE NEW LUC LOCATION SHOULD BE ADDED TO THE REMP. UNIT 1 WAS IN MODE 1, POWER OPERATION, AT 100% POWER AND UNIT 2 WAS IN MODE 6, REFUELING, WHEN THIS EVENT WAS DISCOVERED. THIS EVENT IS ASSIGNED A CAUSE OF MANAGEMENT DEFICIENCY BECAUSE OF POOR INTERFACE BETWEEN GROUPS.

[99] MCGUIRE 1 DOCKET 50-369 LER 89-017
ANNULUS VENTILATION SYSTEM INOPERABLE BECAUSE WIRING DID NOT MEET ENVIRONMENTAL QUALIFICATION REQUIREMENTS AS A RESULT OF MANUFACTURING DEFICIENCY.
EVENT DATE: 071489 REPORT DATE: 082189 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)
VENDOR: ROCKBESTOS COMPANY

(NSIC 215141) DURING A MATERIALS EVALUATION, DESIGN ENGINEERING PERSONNEL DETERMINED ON JULY 14, 1989 THAT WIRING FOR THE ANNULUS VENTILATION FILTER UNIT PREHEATER CONTAINED A TEFLON JACKET. SINCE THE TEFLON JACKET WOULD BREAKDOWN WHEN SUBJECTED TO ACCIDENT RADIATION DOSE LEVELS, THE WIRE DID NOT MEET THE REQUIRED ENVIRONMENTAL QUALIFICATIONS. OPERATIONS PERSONNEL MADE APPROPRIATE PROCEDURE CHANGES TO ENSURE THAT THE ANNULUS VENTILATION SYSTEM WAS MAINTAINED

OPERABLE. A REPLACEMENT WIRE WAS IDENTIFIED AND REQUISITIONED. THIS EVENT IS ASSIGNED A CAUSE OF MANUFACTURING DEFICIENCY RESULTING FROM A MATERIAL DEFICIENCY BECAUSE OF IMPROPER SELECTION. UNIT 1 WAS IN MODE 1, POWER OPERATIONS, AT 100% POWER AND UNIT 2 WAS IN MODE 6, REFUELING, AT THE TIME THE WIRING PROBLEM WAS DISCOVERED. THIS EVENT IS PART 21 REPORTABLE.

[100] MCGUIRE 1 DOCKET 50-369 LER 89-015
THE CONTROL ROOM VENTILATION SYSTEM DID NOT MEET THE REQUIRED POSITIVE PRESSURE BECAUSE OF A DESIGN OVERSIGHT.
EVENT DATE: 072289 REPORT DATE: 091889 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 215319) ON 7/21/89, DURING POST MODIFICATION FLOW BALANCING OF OUTSIDE AIR PRESSURE FILTER TRAIN - 2, PERFORMANCE PERSONNEL MEASURED CONTROL ROOM PRESSURE RELATIVE TO OUTSIDE ATMOSPHERE AND DISCOVERED THAT NEUTRAL PRESSURE WAS THE BEST THAT COULD BE ACHIEVED IN SOME REQUIRED SYSTEM CONFIGURATIONS. THE APPLICABLE TECH SPEC SPECIFIES THAT THE CONTROL ROOM BE MAINTAINED AT A POSITIVE PRESSURE OF AT LEAST +0.125 INCHES WATER GAUGE RELATIVE TO OUTSIDE ATMOPSHERE. THE CONTROL ROOM PRESSURIZATION HAS BEEN TESTED RELATIVE TO THE PRESSURE IN THE CABLE SPREADING ROOM SINCE INITIAL TESTING AND STARTUP. THIS EVENT IS ASSIGNED A CAUSE OF DESIGN DEFICIENCY BECAUSE OF A DESIGN OVERSIGHT. ON 8/19/89, DESIGN ENGINEERING PERSONNEL ISSUED AN OPERABILITY EVALUATION FOR THE CONTROL AREA VENTILATION SYSTEM. THE OPERABILITY ISSUED AN OPERABILITY EVALUATION FOR THE CONTROL AREA VENTILATION SYSTEM. THE OPERABILITY EVALUATION STATED THAT THE CONTROL AREA VENTILATION SYSTEM IS CONDITIONALLY OPERABLE IF THE CONTROL ROOM DOORS ARE TAPED AND ALL FOUR OUTSIDE AIR INTAKES REMAIN OPEN AT ALL TIMES. A PERMANENT OUTSIDE AIR REFERENCE POINT WILL BE INSTALLED TO ENSURE THAT FUTURE TESTING IS ACCURATE. UNIT 1 WAS IN MODE 1, POWER OPERATION, AND UNIT 2 WAS IN MODE 6, REFUELING, AT THE TIME THIS EVENT WAS DISCOVERED.

[101] MCGUIRE 1 DOCKET 50-369 LER 89-020
ABNORMAL DEGRADATION OF STEEL CONTAINMENT VESSELS DUE TO CORROSION CAUSED BY STANDING WATER IN THE ANNULUS AREAS BECAUSE OF UNKNOWN REASONS.
EVENT DATE: 072789 REPORT DATE: 092589 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 215374) ON 7/27/89, DESIGN ENGINEERING PERSONNEL DISCOVERED ABNORMAL DEGRADATION OF THE UNIT 2 STEEL CONTAINMENT VESSEL (SCV) BECAUSE OF CORROSION. THIS DISCOVERY WAS MADE WHILE PERFORMING A PRELIMINARY INSPECTION OF THE SCV PRIOR TO INTEGRATED LEAK RATE TESTING. THE CORROSION WAS CAUSED BY STANDING WATER IN THE ANNULUS AREA. THE MOST SIGNIFICANT CORROSION OCCURRED IN AREAS WHERE BORIC ACID DEPOSITS WERE ALSO FOUND. THE BORIC ACID DEPOSITS RESULTED FROM LEAKING INSTRUMENTATION CONNECTIONS. THIS EVENT IS ASSIGNED A CAUSE OF TRULY UNKNOWN BECAUSE THE FAILURE OF THE DRAIN SYSTEM TO PREVENT STANDING WATER IN THE ANNULUS AREA COULD NOT BE DETERMINED. A CONTRIBUTING CAUSE OF IMPROPER INSTALLATION IS ASSIGNED BECAUSE THE JUNCTION BETWEEN THE SCV AND CONCRETE FLOOR WAS NOT SEALED. A SECOND CONTRIBUTING CAUSE OF UNANTICIPATED ENVIRONMENTAL INTERACTION IS ALSO ASSIGNED BECAUSE OF CHEMICAL REACTION BETWEEN THE BORIC ACID AND SCV. UNIT 1 WAS IN MODE 1, POWER OPERATION, AT 100% POWER AND UNIT 2 WAS IN NO MODE, NO FUEL IN THE CORE, WHEN THIS EVENT WAS DISCOVERED. UNIT 2 HAD PREVIOUSLY OPERATED IN MODE 1 AT 100% POWER WITH THESE CONDITIONS EXISTING. DESIGN ENGINEERING PERSONNEL PERFORMED A PRELIMINARY INSPECTION OF UNIT 1 AND DETECTED SIMILAR DEGRADATION. DESIGN ENGINEERING PERSONNEL EVALUATED THE EXTENT OF DEGRADATION AND PROVIDED AN OPERABILITY DETERMINATION.

[102] MCGUIRE 1 DOCKET 50-369 LER 89-019
AN INADVERTENT UNIT 2 ENGINEERED SAFETY FEATURES ACTUATION OCCURRED BECAUSE OF AN INAPPROPRIATE ACTION AND A DEFECTIVE PROCEDURE.
EVENT DATE: 082189 REPORT DATE: 092189 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 215269) ON 8/21/89, AT 0950, THERE WAS AN INADVERTENT ENGINEERED SAFETY FEATURES (ESF) ACTUATION FROM THE UNIT 2 TRAIN A DIESEL GENERATOR (D/G) LOAD SEQUENCER. INSTRUMENT AND ELECTRICAL (IAE) PERSONNEL WERE PERFORMING THE TRAIN A DIESEL SEQUENCER TIMER CALIBRATION PROCEDURE. THE APPLICABLE PROCEDURE SECTION DID NOT INCLUDE A STEP REQUIRING PLACING THE SEQUENCER INTO THE TEST POSITION. THE ESF SIGNAL CAUSED AN AUTOMATIC START OF VARIOUS EQUIPMENT. OPERATIONS PERSONNEL REALIGNED ALL COMPONENTS AND REMOVED POWER FROM THE NUCLEAR SERVICE WATER TRAIN B CROSSOVER VALVES ON UNIT 1. THIS ACTION RESULTED IN UNIT 1 ENTERING TECH SPEC 3.0.3. IAE PERSONNEL TERMINATED THE TEST AND RESTORED THE EQUIPMENT TO ITS ORIGINAL CONFIGURATION. OPERATIONS PERSONNEL RESTORED POWER TO THE TRAIN B CROSSOVER VALVES AND LOGGED UNIT 1 OUT OF TECH SPEC 3.0.3. UNIT 1 WAS IN MODE 1, POWER OPERATION, AT 100% POWER AND UNIT 2 WAS IN MODE 5, COLD SHUTDOWN, AT THE TIME OF THIS INCIDENT. THIS INCIDENT IS ASSIGNED A CAUSE OF INAPPROPRIATE ACTION BECAUSE THE ACTION TAKEN WAS NOT THE BEST ALTERNATIVE. THIS INCIDENT IS ALSO ASSIGNED A CONTRIBUTORY CAUSE OF DEFECTIVE PROCEDURE. PROCEDURE CHANGES WILL BE COMPLETED FOR THE TRAIN A AND TRAIN B DIESEL SEQUENCER TIMER CALIBRATION PROCEDURES.

[103] MCGUIRE 1 DOCKET 50-369 LER 89-022
REACTOR TRIP OCCURRED BECAUSE OF A FAILED UNIVERSAL BOARD IN THE SOLID STATE PROTECTION SYSTEM CABINET TRAIN "A".
EVENT DATE: 082689 REPORT DATE: 092589 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215376) ON 8/26/89 AT 0934, A UNIT 1 REACTOR TRIP OCCURRED BECAUSE OF A REACTOR COOLANT (NC) LOW FLOW SIGNAL THAT IS INTERLOCKED WITH A PERMISSIVE SIGNAL WHICH BLOCKS A REACTOR TRIP WHEN THE UNIT IS LESS THAN 48% POWER (P-8 PERMISSIVE). UNIT 1 WAS OPERATING IN MODE 1, POWER OPERATION, AT 100% POWER PRIOR TO THE TRIP. A LOW NC FLOW SIGNAL WITH P-8 PERMISSIVE CAUSED REACTOR TRIP BREAKER "A" TO TRIP THE UNIT. THE SIGNAL WAS CAUSED BY A FAILED UNIVERSAL BOARD IN THE SOLID STATE PROTECTION SYSTEM (SSPS) CABINET FOR TRAIN "A". THE TURBINE GENERATOR AUTOMATICALLY TRIPPED BECAUSE OF THE REACTOR TRIP. ALL SYSTEMS AND EQUIPMENT RESPONDED AS EXPECTED FOLLOWING THE TRIP WITH ONE EXCEPTION. OPERATIONS PERSONNEL IMPLEMENTED THE REACTOR TRIP RECOVERY PROCEDURE TO RECOVER FROM THE TRANSIENT. AT ABOUT 1000, OPERATIONS PERSONNEL MADE THE REQUIRED NOTIFICATION TO THE NRC. AT 2220, INSTRUMENTATION AND ELECTRICAL PERSONNEL DISCOVERED THE FAILED UNIVERSAL BOARD AND REPLACED IT. THE SSPS CABINET WAS TESTED AFTER THE BOARD WAS REPLACED TO ENSURE THE TRAIN WOULD OPERATE PROPERLY. UNIT 1 WAS RETURNED TO MODE 1, POWER OPERATION, ON 8/28/89 AT 1230. THIS EVENT IS ASSIGNED A CAUSE OF EQUIPMENT FAILURE/MALFUNCTION. THIS EVENT IS NUCLEAR PLANT RELIABILITY DATA SYSTEM REPORTABLE. THE BOARD WILL BE SENT TO WESTINGHOUSE FOR REPAIR AND FAILURE ANALYSIS.

[104] MCGUIRE 2 DOCKET 50-370 LER 89-007
UNIT 2 OPERATED IN VIOLATION OF TECH SPEC 3.1.2.1A AND 3.1.2.3 BECAUSE BOTH CENTRIFUGAL CHARGING PUMPS WERE INOPERABLE DURING REFUELING.
EVENT DATE: 081089 REPORT DATE: 091189 NSSS: WE TYPE: PWR

(NSIC 215270) ON 7/13/89, MECHANICAL MAINTENANCE PERSONNEL COMPLETED PREVENTATIVE MAINTENANCE ON CENTRIFUGAL CHARGING PUMP 2B. NO FUNCTIONAL VERIFICATION OR RETEST WAS PERFORMED AT THE TIME BECAUSE OF A LACK OF EQUIPMENT AVAILABILITY. THE WORK REQUEST WAS RETURNED TO THE PLANNING SECTION TO RESCHEDULE FOR PERFORMANCE OF THE FUNCTIONAL VERIFICATION. CONSEQUENTLY, PLANNER A CHANGED THE

STATUS OF THE WORK REQUEST TO "AWAITING FUNCTIONAL VERIFICATION" AND PLACED IT IN THE APPROPRIATE OUTAGE FILE. ON 8/1/89, OPERATIONS PERSONNEL DECLARED "B" TRAIN EQUIPMENT OPERABLE AND REMOVED "A" TRAIN EQUIPMENT FROM SERVICE. THEREFORE, UNIT 2 OPERATED IN MODE 6, REFUELING, WITH BOTH PUMPS INOPERABLE. ON 8/8/89, PLANNER B FOUND THAT THE WORK REQUEST HAD BEEN PLACED IN THE "AWAITING FUNCTIONAL VERIFICATION" FILE IN ERROR. THE WORK REQUEST WAS IMMEDIATELY FORWARDED TO THE PERFORMANCE GROUP FOR RETEST OF THE PUMP. THIS EVENT IS ASSIGNED A CAUSE OF MANAGEMENT/QUALITY ASSURANCE DEFICIENCY SINCE OPERATIONS PERSONNEL WERE UNAWARE THAT CENTRIFUGAL CHARGING PUMP 2B HAD NOT BEEN RETESTED BECAUSE OF THE LACK OF A TRACKING PROGRAM DURING OUTAGES. A CONTRIBUTORY CAUSE OF INAPPROPRIATE ACTION BECAUSE OF LACK OF ATTENTION TO DETAIL IS ASSIGNED SINCE PLANNER A FAILED TO SEND THE WORK REQUEST TO THE APPROPRIATE RETEST GROUP.

[105] MCGUIRE 2 DOCKET 50-370 LER 89-008
BORON SURVEILLANCES WERE MISSED BECAUSE OF A FAILURE TO FOLLOW PROCEDURE.
EVENT DATE: 082889 REPORT DATE: 092789 NSSS: WE TYPE: PWR

(NSIC 215360) ON 8/26-27/89, CHEMISTRY PERSONNEL OBTAINED CHEMICAL AND VOLUME CONTROL SYSTEM LETDOWN HEAT EXCHANGER OUTLET SAMPLES TO MEET THE TECH SPEC SURVEILLANCE FOR SHUTDOWN BORON MARGIN ON UNIT 2. THE SAMPLES THAT WERE OBTAINED WERE INVALID BECAUSE THERE WAS NO LETDOWN FLOW AT THE TIME THE SAMPLES WERE OBTAINED. THE SAMPLING ERROR WAS DISCOVERED AT APPROXIMATELY 0800 ON 8/28/89. CHEMISTRY PERSONNEL OBTAINED A RESIDUAL HEAT REMOVAL DISCHARGE SAMPLE AT 0830 ON 8/28/89 AND VERIFIED THAT THE SHUTDOWN BORON MARGIN WAS GREATER THAN OR EQUAL TO 2000 PARTS PER MILLION. THIS EVENT IS ASSIGNED A CAUSE OF INAPPROPRIATE ACTION BECAUSE CHEMISTRY TECHNICIAN A FAILED TO FOLLOW PROCEDURE. UNIT 2 WAS IN MODE 5, COLD SHUTDOWN, AT THE TIME OF THIS EVENT. THIS EVENT WILL BE REVIEWED WITH ALL APPROPRIATE CHEMISTRY PERSONNEL.

[106] MILLSTONE 1 DOCKET 50-245 LER 88-010 REV 01
UPDATE ON HYDRAULIC SNUBBER FAILURE DUE TO A SMALL CRACK IN THE PLASTIC RESERVOIR.
EVENT DATE: 111588 REPORT DATE: 092289 NSSS: GE TYPE: BWR
VENDOR: ITT GRINNELL

(NSIC 215292) ON 11/15/88, WHILE THE PLANT WAS IN COLD SHUTDOWN TO INVESTIGATE THE CAUSE FOR THE INCREASE IN UNIDENTIFIED LEAKAGE IN CONTAINMENT, IT WAS NOTICED THAT ONE OF THE HYDRAULIC SNUBBERS (HSS-73) HAD NO VISIBLE SIGNS OF FLUID IN ITS RESERVOIR. AS A RESULT OF THIS FINDING, A VISUAL INSPECTION OF ALL SAFETY RELATED HYDRAULIC SNUBBERS IN CONTAINMENT WAS PERFORMED TO ENSURE THAT THIS CONDITION DID NOT EXIST ON ANY OF THE REMAINING SUBBERS. THE REMAINING SAFETY RELATED SNUBBERS WERE FOUND TO BE SATISFACTORY. THE LOSS OF FLUID IN SUBBER HSS-73 WAS ATTRIBUTED TO A SMALL CRACK AT THE BASE OF THE PLASTIC RESERVOIR.

[107] MILLSTONE 1 DOCKET 50-245 LER 89-018
MISSING HYDRAULIC CONTROL UNIT RESTRAINING STRAPS.
EVENT DATE: 083189 REPORT DATE: 092889 NSSS: GE TYPE: BWR

(NSIC 215340) ON 8/22/89 AT APPROXIMATELY 1400, WHILE OPERATING AT 100% POWER, A SITE NRC INSPECTOR REPORT TO STATION MANAGEMENT INDICATED THAT TWO RESTRAINING METAL STRAPS ON CONTROL ROD DRIVE SYSTEM HYDRAULIC CONTROL UNITS (HCU) 30-31 AND 18-47 WERE FOUND MISSING. A COMPLETE INSPECTION OF ALL HCU'S WAS INITIATED BY PLANT PERSONNEL AND TWO ADDITIONAL STRAPS WERE FOUND MISSING. NO SITE SPECIFIC ANALYSIS COULD BE LOCATED FOR THE HCU'S AND THUS ON 8/31/89, AT 1130, THIS EVENT WAS CONSIDERED REPORTABLE. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(V) AND 10CFR50.73(A)(2)(II)(A) DUE TO THE FACT THAT SPECIFIC SITE ANALYSIS IS NOT AVAILABLE TO DETERMINE THE CAPABILITY OF THE HCU'S TO REMAIN FUNCTIONAL DURING A SEISMIC EVENT WITHOUT THE STRAPS AND THUS IT IS AN UNANALYZED CONDITION. THE STRAPS WERE REINSTALLED ON THE AFFECTED UNITS. EXISTING PROCEDURES PROVIDE

ADEQUATE DIRECTION FOR INSTALLATION OF THE STRAPS AND THUS THIS EVENT IS ATTRIBUTED TO PERSONNEL ERROR. ALL SYSTEMS WERE FUNCTIONAL AND NO SAFETY CONSEQUENCES RESULTED FROM THIS EVENT.

[108] MILLSTONE 2 DOCKET 50-336 LER 86-007 REV 01
UPDATE ON STEAM GENERATOR LEVEL TRANSMITTER FAILURE DUE TO AMPLIFIER ASSEMBLY FAILURE.
EVENT DATE: 090186 REPORT DATE: 092889 NSSS: CE TYPE: PWR
VENDOR: FOXBORO CO., THE

(NSIC 215332) ON 9/1/86 THE PLANT WAS OPERATING IN MODE 1 AT 99.8% POWER. AT 1820 DURING THE MONTHLY CONTROL ROOM SURVEILLANCE CHECK THE NUMBER ONE STEAM GENERATOR LEVEL INDICATION ON THE REMOTE SHUTDOWN PANEL INDICATED 0%. THE LEVEL TRANSMITTER LT-1113A HAD BEEN OUT OF SERVICE SINCE 0935 ON 8/18/86 AND THE PLANT WAS OPERATING IN ACCORDANCE WITH TECH SPEC 3.3.1.1. IT WAS NOT REALIZED THAT LT-1113A ALSO PROVIDED INPUT TO THE REMOTE SHUTDOWN PANEL. THE SURVEILLANCE PROMPTED INVESTIGATION AND THE REALIZATION THAT INOPERABILITY OF THIS INSTRUMENT PLACED THE PLANT IN TECH SPEC ACTION STATEMENT 3.3.3.5.B. THE TRANSMITTER WAS REPLACED, CALIBRATED, AND TESTED. THE LEVEL INSTRUMENT WAS BACK IN SERVICE AT 2030 ON 9/2/86. THE CAUSE OF THE LEVEL TRANSMITTER FAILURE IS DUE TO AMPLIFIER ASSEMBLY FAILURE. ACTUAL CAUSE OF THE AMPLIFIER ASSEMBLY FAILURE IS UNDETERMINED. THE ASSEMBLY IS A "POTTED" COMPONENT, THEREFORE, NON-DESTRUCTIVE FAILURE ANALYSIS IS NOT FEASIBLE. TO PROVIDE INFORMATION CONCERNING INSTRUMENTS THAT ALSO INPUT TO REMOTE OPERATING PANELS THE FOLLOWING ACTION WAS COMPLETED. INSTRUMENTS ON THE MAIN CONTROL BOARDS THAT PROVIDE INPUT TO THE REMOTE OPERATING PANELS HAVE BEEN IDENTIFIED WITH BLUE "HSD" OR "FIRE" TAGS TO SIGNIFY THE HOT SHUTDOWN PANEL OR FIRE PANEL. THERE ARE NO SIMILAR EVENTS.

[109] MILLSTONE 2 DOCKET 50-336 LER 87-003 REV 01
UPDATE ON DEFECTIVE STEAM GENERATOR TUBES NOT REPAIRED PRIOR TO STARTUP.
EVENT DATE: 012987 REPORT DATE: 092889 NSSS: CE TYPE: PWR
VENDOR: COMBUSTION ENGINEERING, INC.

(NSIC 215333) ON 1/29/87, AT 1715, WHILE OPERATING AT 100% POWER, A REANALYSIS OF STEAM GENERATOR TUBE EXAMINATION DATA, ORIGINALLY COLLECTED IN OCTOBER AND NOVEMBER 1986, IDENTIFIED THAT ONE TUBE WHICH HAD CONTAINED A REPAIRABLE DEFECT HAD NOT BEEN REPAIRED, AS REQUIRED BY TECH SPEC 4.4.5.1.4.B, PRIOR TO DECLARING THE STEAM GENERATOR OPERABLE. IN ACCORDANCE WITH TECH SPEC 3.0.3 A PLANT SHUTDOWN WAS INITIATED AND THE PLANT BROUGHT TO COLD SHUTDOWN APPROXIMATELY 25 HOURS LATER. CONTINUING REANALYSIS OF 1987 EDDY CURRENT DATA IDENTIFIED 37 OTHER TUBES WHICH ALSO CONTAINED REPAIRABLE DEFECTS WHICH HAD NOT BEEN REPAIRED. ALL TUBES WITH REPAIRABLE DEFECTS AND OTHERS CONSIDERED PRUDENT TO REPAIR WERE PLUGGED PRIOR TO INCREASING T(AVG) ABOVE 200F AS REQUIRED BY TECH SPEC 3.4.5. NO FURTHER CORRECTIVE ACTION IS REQUIRED.

[110] MILLSTONE 2 DOCKET 50-336 LER 87-007 REV 01
UPDATE ON REACTOR TRIP DUE TO A TURBINE TRIP CAUSED BY THE GENERATOR EXCITER FIELD BREAKER OPENING.
EVENT DATE: 041687 REPORT DATE: 092289 NSSS: CE TYPE: PWR

(NSIC 215290) ON 4/16/87 AT 0717 HOURS, AN AUTOMATIC REACTOR TRIP OCCURRED FROM 100% POWER OPERATION. THE TRIP WAS DUE TO A TURBINE TRIP WHICH WAS CAUSED BY THE GENERATOR EXCITER FIELD BREAKER OPENING. OPERATIONS PERSONNEL RESPONDED TO THE TRIP BY PERFORMING EMERGENCY OPERATIONS PROCEDURE 2525, "STANDARD POST TRIP ACTIONS" AND 2526, "REACTOR TRIP RECOVERY". PERFORMANCE OF THESE PROCEDURES PLACED THE UNIT IN A STABLE CONDITION. THE CAUSE OF THE EVENT WAS THE ACTUATION OF GENERATOR VOLTS/HERTZ RELAY (59X2) WHICH IS A COMMON TRIP FOR BOTH THE GENERATOR EXCITER FIELD BREAKER AND GENERATOR FIELD BREAKER. THE 59X2 RELAY IS

REMOVED FROM SERVICE AS IT ONLY PROVIDES BACKUP PROTECTION WHEN THE GENERATOR IS EXCITED AND NOT SYNCHRONIZED TO THE GRID. THE REASON FOR THE ACTUATION OF RELAY 59X2 IS UNKNOWN. THE PRECURSOR RELAY (59X1) WHICH IS INTERLOCKED WITH RELAY 59X2 WAS NOT ACTUATED, THEREFORE THE NORMAL SEQUENCE TO ACTUATE THE 59X2 RELAY WAS NOT USED. THE TWO REMAINING METHODS OF ACTUATION ARE ELECTROMAGNETIC INTERFERENCE (EMI) AND MANUAL ACTUATION OF THE RELAY. ALL EQUIPMENT RESPONDED AS REQUIRED FOR AN UNCOMPLICATED REACTOR TRIP.

[111] MILLSTONE 2 DOCKET 50-336 LER 87-010 REV 01
UPDATE ON FIRE PROTECTION FOR RACEWAY SUPPORTS NOT ADEQUATELY PROTECTED.
EVENT DATE: 073087 REPORT DATE: 092289 NSSS: CE TYPE: PWR

(NSIC 215291) ON 7/30/87, AT 1030 HOURS WITH THE PLANT AT 100% POWER, A TECHNICAL EVALUATION WAS RECEIVED WHICH IDENTIFIED INADEQUACIES IN THE FIRE PROTECTION COATING ON RACEWAY SUPPORT STRUCTURES LOCATED IN THE MAIN CABLE VAULT (AUXILIARY BUILDING 25'-6" AND THE HALLWAY LEADING TOE CHARGING PUMPS (AUXILIARY BUILDING -25'-6"). THE FIRE COATING PROTECTS THE CABLE TRAYS, CABLES AND SUPPORT STRUCTURES WERE NOT COVERED AND THUS A FAILURE OF THE SUPPORT SYSTEM COULD OCCUR DURING A SEVERE FIRE. THE IDENTIFIED INADEQUACIES WILL BE CORRECTED BEFORE THE END OF THE NEXT REFUELING OUTAGE. AN IMMEDIATE ROVING HOURLY FIRE WATCH WAS STARTED IN THE TWO AREAS IN ACCORDANCE WITH TECH SPEC 3.7.10.A.1. THERE WERE NO SAFETY IMPLICATIONS AS A RESULT OF THIS EVENT AS THE SUPPORT STRUCTURES WOULD ONLY FAIL DURING A SEVERE FIRE. THE MAIN CABLE VAULT HAS FIRE DETECTION, SPRINKLERS AND A MANUAL DELUGE SYSTEM. THE HALLWAY LEADING TO THE CHARGING PUMPS AND THE CHARGING PUMP AREA HAS FIRE DETECTION AND THE AREAS ARE SEPARATED FROM OTHER EQUIPMENT BY MEANS OF A WATER CURTAIN. THEREFORE THE POSSIBILITY OF A SEVERE FIRE IS REMOTE.

[112] MILLSTONE 2 DOCKET 50-336 LER 89-007 REV 01
UPDATE ON MISSED RADIATION MONITOR SOURCE CHECK DUE TO PERSONNEL ERROR.
EVENT DATE: 061389 REPORT DATE: 092089 NSSS: CE TYPE: PWR
VENDOR: NUCLEAR MEASUREMENTS CORP.

(NSIC 215314) ON 6/13/89, AT 1300 HOURS, WITH THE PLANT IN MODE 1, 100% POWER, IT WAS IDENTIFIED THAT THE CONDENSATE POLISHING FACILITY (CPF) RADIATION MONITOR, 2-CND-RE-245, HAD BEEN OPERATED WITHOUT CONDUCTING A SOURCE CHECK, WHICH IS IN VIOLATION OF THE PLANT'S TECH SPECS, FROM 2/1/86 UNTIL 3/22/89. WHEN THE ENVIRONMENTAL TECH SPECS (ETS) WERE CHANGED TO THE RADIOLOGICAL EFFLUENT TECH SPECS (RETS) IN 1986, THE SOURCE CHECK REQUIREMENT FOR THE CPF RADIATION MONITOR WAS ADDED IN REVISION 16 TO SP 2617A IN MARCH 1989. THE INFRACTION WAS REALIZED DURING A NUCLEAR REVIEW BOARD (NRB) AUDIT OF THE PLANT PROCEDURE. THE NRB AUDIT ALSO IDENTIFIED AN UNRESOLVED PLANT OPEN ITEM WITH THE WASTE GAS RADIATION MONITOR, RM9095, AND ITS ASSOCIATED PROCEDURAL REQUIREMENTS (OP 2617B) FOR A SOURCE CHECK. DURING THE TIME THAT THE SOURCE CHECKS WERE OMITTED, RADIOACTIVE LIQUID AND GASEOUS DISCHARGES WERE MADE THAT UTILIZED THESE RADIATION MONITORS. IT SHOULD BE NOTED THAT THESE RADIATION MONITORS WERE FUNCTIONALLY CHECKED ON A MONTHLY BASIS AND CALIBRATED QUARTERLY DURING THE TIME IN QUESTION WHICH DID NOT IDENTIFY ANY DEFICIENCIES WITH THE OPERATION OF THE RADIATION MONITORS. THE CORRECTIVE ACTIONS HAVE ALREADY BEEN TAKEN. THE CAUSE OF THE INFRACTIONS WAS PERSONNEL ERROR.

[113] MILLSTONE 3 DOCKET 50-423 LER 89-004 REV 01
UPDATE ON INOPERABLE FIRE DETECTION FOR EMERGENCY DIESEL GENERATOR FUEL OIL VAULT DUE TO OPERATOR ERROR.
EVENT DATE: 021689 REPORT DATE: 092989 NSSS: WE TYPE: PWR

(NSIC 215385) ON 2/16/89 AT APPROXIMATELY 2330 HOURS, AT 0% POWER, 133 DEGREES AND 14.7 PSIA, THE SHIFT SUPERVISOR (SS) OBSERVED THAT FIRE DETECTION IN THE 3

FOOT 8 INCH AND 24 FOOT 6 INCH ELEVATIONS OF THE CONTAINMENT BUILDING HAD BEEN INOPERABLE FOR APPROXIMATELY 10 HOURS. ON 2/27/89 AT 0950 HOURS, AT 100% POWER, THE SS WAS INFORMED THAT FIRE DETECTION FOR THE "B" TRAIN EMERGENCY DIESEL GENERATOR FUEL OIL VAULT HAD BEEN UNAVAILABLE FOR APPROXIMATELY 1 1/2 HOURS. THE CONTROL ROOM OPERATOR (CO) HAD ACKNOWLEDGED THE ALARM INDICATION AT APPROXIMATELY 0800 BUT DID NOT RESET THE FALSE ALARM SIGNAL. THE ROOT CAUSE OF BOTH EVENTS WAS PERSONNEL ERROR. PRESENCE OF AN INDICATED FIRE ALARM MASKED POSSIBLE FUTURE ALARMS RESULTING IN THE APPLICABLE FIRE ZONE BEING INOPERABLE. MISCOMMUNICATIONS BETWEEN OPERATIONS DEPARTMENT PERSONNEL RESULTED IN THE LOSS OF FIRE DETECTION FOR THE FIRST EVENT. IN THE SECOND EVENT, THE CO FAILED TO EVALUATE AN ALARM CONDITION. IMMEDIATE CORRECTIVE ACTION FOR BOTH EVENTS WAS TO ESTABLISH AN HOURLY FIRE WATCH. ALL ON-SHIFT PERSONNEL HAVE BEEN BRIEFED TO REINFORCE THE IMPORTANCE OF PROMPT EVALUATION AND RESPONSE TO ALL FIRE SYSTEM ALARMS.

[114] MILLSTONE 3 DOCKET 50-423 LER 89-018
INOPERABLE WASTE NEUTRALIZATION SUMP EFFLUENT RADIATION MONITOR DUE TO PERSONNEL ERROR.
EVENT DATE: 080289 REPORT DATE: 090189 NSSS: WE TYPE: PWR

(NSIC 215217) ON 8/2/89 AT 1430, IN MODE 1 AT 100% POWER, SHIFT PERSONNEL DECLARED THE CONDENSATE DEMINERALIZER WASTE NEUTRALIZATION SUMP EFFLUENT RADIATION MONITOR, 3CND-07 INOPERABLE DUE TO INADEQUATE TESTING OF THE HIGH ACTIVITY AUTOMATIC TERMINATION FEATURE, SINCE PLANT STARTUP. THE COMPENSATORY ACTIONS REQUIRED BY THE TECH SPECS HAD ALRE[illegible]EEN INCORPORATED INTO EXISTING PROCEDURES AND WERE BEING PERFORMED. HOWEVER, TECH SPECS ALSO REQUIRES AN EXPLANATION BE SUBMITTED IN THE NEXT SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT INDICATING WHY THE "INOPERABILITY" WAS NOT CORRECTED IN A TIMELY MANNER (30 DAYS). THIS ACTION WAS NEVER PERFORMED. THE ROOT CAUSE OF THE EVENT WAS PERSONNEL ERROR. PLANT PERSONNEL FAILED TO INCORPORATE ALL OF THE AUTOMATIC RELEASE TERMINATION CAPABILITY OF 3CND-07 INTO THE APPLICABLE SURVEILLANCE PROCEDURE. WITH THE RADITION MONITOR INOPERABLE AN EXPLANATION WAS NEVER PROVIDED IN THE SEMIANNUAL EFFLUENT RELEASE REPORT. NO IMMEDIATE CORRECTIVE ACTION WAS REQUIRED. THE SURVEILLANCE PROCEDURE WAS CHANGED TO INCOPORATE THE INTERLOCKS FOR ALL RELEASE PATHS AND WERE SATISFACTORILY TESTED. AN EXPLANATION WILL BE PROVIDED IN THE NEXT SEMIANNUAL RELEASE REPORT. NO OTHER DEFICIENCIES WERE IDENTIFIED IN OTHER RADIATION MONITORING SURVEILLANCES. LESSONS LEARNED WILL BE INCLUDED IN THE TRAINING FOR APPLICABLE PLANT PERSONNEL.

[115] MILLSTONE 3 DOCKET 50-423 LER 89-019
INOPERABLE FIRE PROTECTION HOSE HOUSE DUE TO MISPOSITIONED VALVE DUE TO SAFETY TAGGING DEFICIENCIES.
EVENT DATE: 080489 REPORT DATE: 090189 NSSS: WE TYPE: PWR

(NSIC 215218) ON 8/4/89 AT APPROXIMATELY 0300 HOURS, IN MODE 1, AT 100% POWER, SHIFT PERSONNEL DISCOVERED THAT THE ISOLATION VALVE FOR FIRE PROTECTION HOSE HOUSE NO. 3 WAS IN THE CLOSED POSITION AFTER REMOVING THE PLANT FROM TECH SPEC LIMITING CONDITION FOR OPERATION (LCO) 3.7.12.6.A, "YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES". THE CLOSED ISOLATION VALVE RENDERED THE HOSE HOUSE INOPERABLE. ROOT CAUSE OF THE EVENT WAS ADMINISTRATIVE DEFICIENCY. THE SHIFT TURNOVER REPORT DID NOT INDICATE ALL THE PERTINENT INFORMATION CONCERNING THE LIMITING CONDITION FOR OPERATION (LCO). IMMEDIATE CORRECTIVE ACTION WAS TO RESTORE THE ISOLATION VALVE FOR HOSE HOUSE NO. 3 TO ITS NORMALLY OPEN POSITION. AS INTERIM CORRECTIVE ACTION, A MEMORANDUM HAS BEEN ISSUED BY THE OPERATIONS DEPARTMENT SUPERVISOR TO ALL SHIFT SUPERVISORY PERSONNEL EMPHASIZING THAT ALL PERTINENT INFORMATION INVOLVING LCO'S SHALL BE DOCUMENTED IN THE SHIFT TURNOVER REPORT AS ACTION TO PREVENT RECURRENCE, THE APPLICABLE OPERATIONS DEPARTMENT INSTRUCTION WILL BE REVISED TO ENSURE THE ABOVE DESCRIBED INFORMATION WILL BE ENTERED INTO THE SHIFT TURNOVER REPORT. THIS REVISION WILL BE COMPLETED BY 10/31/89.

[116] MONTICELLO DOCKET 50-263 LER 89-015
SPURIOUS TRIP OF REACTOR BUILDING VENT WIDE RANGE GAS MONITOR INITIATES ISOLATION.
EVENT DATE: 082889 REPORT DATE: 092789 NSSS: GE TYPE: BWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 215344) REACTOR BUILDING VENTILATION ISOLATION AND STARTUP OF THE STANDBY GAS TREATMENT SYSTEM OCCURRED DURING ROUTINE REFUELING OPERATIONS AS A RESULT OF A SPURIOUS HIGH TRIP OF THE CHANNEL A REACTOR BUILDING VENT WIDE RANGE GAS MONITOR. THE ISOLATION WAS RESET, THE STANDBY GAS TREATMENT SYSTEM WAS SHUTDOWN AND THE WIDE RANGE GAS MONITOR RESET. DUE TO A PLANT PROCESS COMPUTER MAINTENANCE OUTAGE, AT THE TIME OF THE EVENT, A PLANT CHEMIST WAS RETRIEVING EFFLUENT RELEASE DATA FROM THE WIDE RANGE GAS MONITOR. THE PROBABLE CAUSE IS AN UNIDENTIFIED PROBLEM IN THE WIDE RANGE GAS MONITORS WHEN THE PLANT PROCESS COMPUTER IS UNAVAILABLE. INVESTIGATIONS OF FIRMWARE PROBLEMS WITH THE WIDE RANGE GAS MONITORS WILL BE CONTINUED WITH THE VENDOR AND THROUGH THE WIDE RANGE GAS MONITOR USERS GROUP.

[117] MONTICELLO DOCKET 50-263 LER 89-016
AUTO START OF #11 EMERGENCY DIESEL GENERATOR DUE TO INCORRECT JUMPER SPECIFIED.
EVENT DATE: 083089 REPORT DATE: 092989 NSSS: GE TYPE: BWR

(NSIC 215345) WHILE THE PLANT WAS SHUTDOWN IN THE REFUEL MODE, THE #11 EMERGENCY DIESEL GENERATOR STARTED AUTOMATICALLY WHEN A TEMPORARY JUMPER WAS INCORRECTLY SPECIFIED AND INSTALLED DUE TO PERSONNEL ERROR. ALL BYPASSES ASSOCIATED WITH THE WORK WERE REMOVED AND THE DIESEL GENERATOR WAS RETURNED TO EMERGENCY STANDBY CONDITION. THE ENGINEER INVOLVED WAS COUNSELED. PLANT PROCEDURES FOR BYPASS CONTROL WILL BE STRENGTHENED.

[118] MONTICELLO DOCKET 50-263 LER 89-017
WIDE RANGE GAS MONITOR TRIP DURING POWER SUPPLY TRANSFER.
EVENT DATE: 090189 REPORT DATE: 100189 NSSS: GE TYPE: BWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 215346) REACTOR BUILDING VENTILATION ISOLATION AND STARTUP OF THE STANDBY GAS TREATMENT SYSTEM OCCURRED DURING ROUTINE REFUELING OPERATIONS AS A RESULT OF A HIGH-HIGH TRIP OF THE CHANNEL A REACTOR BUILDING VENT WIDE RANGE GAS MONITOR. THE ISOLATION WAS RESET, THE STANDBY GAS TREATMENT SYSTEM SHUT DOWN AND THE WIDE RANGE GAS MONITOR RESET. THE EVENT OCCURRED DURING PERFORMANCE OF PROCEDURE OCD-4561-1 UNINTERRUPTIBLE POWER SUPPLY INVERTER (Y-71). WHILE TRANSFERRING THE UNINTERRUPTIBLE POWER SUPPLY INVERTER Y-71 TO ITS ALTERNATE POWER SUPPLY, THE WIDE RANGE GAS MONITOR SENSED A LOSS OF POWER. UPON A PERCEIVED RESTORATION OF POWER THE WIDE RANGE GAS MONITOR TRIPPED CAUSING THE EVENT. THE ROOT CAUSE IS A FIRMWARE PROBLEM WITH THE WIDE RANGE GAS MONITOR. INVESTIGATION OF THE FIRMWARE PROBLEM WITH THE WIDE RANGE GAS MONITORS WILL BE CONTINUED WITH THE VENDOR AND THROUGH THE WIDE RANGE GAS MONITOR USERS GROUP SO THAT THE GROUP CAN ASSIST THE VENDOR IN FINDING A SOLUTION. INVESTIGATION OF BUS TRANSFER CHARACTERISTICS AND INVERTER OUTPUT HAVE BEEN CONDUCTED AND ARE PRESENTLY BEING ANALYZED.

[119] NINE MILE POINT 1 DOCKET 50-220 LER 89-011
AUTOMATIC INITIATION OF REACTOR BUILDING EMERGENCY VENTILATION DUE TO POOR WORK PRACTICES.
EVENT DATE: 070389 REPORT DATE: 080189 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.
ROCKBESTOS COMPANY

(NSIC 214976) AT 1002 HOURS, ON 7/3/89, WITH THE MODE SWITCH IN "SHUTDOWN" AND NINE MILE POINT UNIT 1 (NMP1) IN AN EXTENDED REFUELING OUTAGE WITH THE CORE OFF-LOADED, AN INITIATION SIGNAL WAS RECEIVED FOR THE REACTOR BUILDING EMERGENCY

VENTILATION SYSTEM (RBEVS) DUE TO A LOSS OF POWER TO THE PROCESS RADIATION MONITORING SYSTEM (PRMS). THE LOSS OF POWER TO PRMS OCCURRED WHEN POWER SUPPLY FUSES WERE PULLED TO FACILITATE THE REPLACEMENT OF POWER CABLES. THE ROOT CAUSE OF THIS EVENT WAS POOR WORK PRACTICES. SPECIFICALLY, THERE WAS A LACK OF AN ADEQUATE REVIEW OF THE ELECTRICAL DRAWINGS AND THE PLANT IMPACT, BY BOTH OPERATIONS AND MAINTENANCE PERSONNEL, PRIOR TO PULLING THE PRMS POWER SUPPLY FUSES. IMMEDIATE CORRECTIVE ACTIONS WERE TO REINSTALL THE PRMS POWER SUPPLY FUSES, RESET RBEVS, RESTORE REACTOR BUILDING VENTILATION, AND MAKE A FOUR HOUR TELEPHONE NOTIFICATION TO THE NRC IN ACCORDANCE WITH 10CFR50.72(B)(2)(II). SUBSEQUENT CORRECTIVE ACTIONS WHICH HAVE BEEN TAKEN ARE: ADMONISHING THE PERSONNEL INVOLVED, ISSUING A NIGHT ORDER, AND REVISING AN OPERATIONS DEPARTMENT INSTRUCTION. ADDITIONAL CORRECTIVE ACTIONS WILL INCLUDE: REVISING ELECTRICAL DRAWINGS, COMPLETION OF A CROSS-REFERENCE BETWEEN PROCESS INSTRUMENTATION AND POWER SUPPLY FUSES, ADDITIONAL SYSTEMS TRAINING FOR ELECTRICAL MAINTENANCE PERSONNEL AND ISSUING A LESSONS LEARNED TRANSMITTAL.

[120] NINE MILE POINT 2 DOCKET 50-410 LER 89-021
ESF ACTUATION DUE TO POWER TRANSIENT TO GASEOUS RADIATION MONITOR CAUSED BY LIGHTNING STRIKE TO MAIN STACK.
EVENT DATE: 081589 REPORT DATE: 091289 NSSS: GE TYPE: BWR

(NSIC 215279) ON 8/15/89, AT 2154 HOURS, WITH THE REACTOR AT 96% OF RATED THERMAL POWER AND THE MODE SWITCH IN "RUN", NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION. THIS EVENT CONSISTED OF A PRIMARY CONTAINMENT VENT AND PURGE VALVE ISOLATION (GROUP 9 ISOLATION). IMMEDIATE INVESTIGATION INDICATED THAT THE ISOLATION WAS THE RESULT OF AN ISOLATION SIGNAL FROM THE STANDBY GAS TREATMENT (GTS) RADIATION MONITOR (2GTS-CAB105). THE ROOT CAUSE OF THE EVENT WAS DETERMINED TO BE A LIGHTNING STRIKE TO THE MAIN STACK TOWER RESULTING IN A POWER TRANSIENT/INTERRUPTION TO RADIATION MONITOR 2GTS-CAB105. CORRECTIVE ACTIONS INCLUDED: RETURNING THE SYSTEM TO NORMAL OPERATIONAL STATUS ONCE THE ISOLATION SIGNAL WAS DETERMINED NOT TO HAVE BEEN INITIATED BY A HIGH RADIATION CONDITION; AND ISSUING A WORK REQUEST TO TROUBLESHOOT THE SOURCE OF THE FALSE SIGNAL.

[121] NORTH ANNA 1 DOCKET 50-338 LER 88-022 REV 03
UPDATE ON AIR IN HIGH HEAD SAFETY INJECTION PUMP SUCTION LINES.
EVENT DATE: 101388 REPORT DATE: 092289 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)
VENDOR: PACIFIC PUMPS

(NSIC 215295) AT 1000 HOURS ON 10/13/88, WITH UNITS 1 AND 2 AT 100% POWER (MODE 1), IT WAS DETERMINED THAT VARIOUS PIPES OF THE SAFETY INJECTION SYSTEM (SI) AND CHEMICAL VOLUME AND CONTROL SYSTEM (CVCS) COLLECTED OR TRAPPED GAS WHICH MIGHT AFFECT THE FUNCTIONS OF THESE SYSTEMS. THERE WAS A CONCERN THAT THE GAS POCKETS MAY ADVERSELY EFFECT PUMP OPERATION. VOIDS WERE DETECTED IN SOME OF THE LOW HEAD SAFETY INJECTION (LHSI) TO HIGH HEAD SAFETY INJECTION (HHSI) PUMP SUCTION PIPING, AND IN THE REFUELING WATER STORAGE TANK (RWST) PIPING TO THE HHSI PUMPS. AFFECTED PIPING, WITH VENT PATHS, HAVE BEEN VENTED, ULTRASONIC INSPECTION OF THIS PIPING VERIFIED THE PIPING IS ADEQUATELY FILLED. A JUSTIFICATION FOR CONTINUED OPERATION WAS APPROVED BY THE STATION NUCLEAR SAFETY AND OPERATING COMMITTEE TO ALLOW CONTINUED OPERATION FOR PIPING WHICH COULD NOT BE VENTED. HIGH POINT VENTS HAVE BEEN INSTALLED ON SUSCEPTIBLE PIPING ON UNITS 1 AND 2. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS. MULTIPLE PUMP PERFORMANCE TESTS AT FULL FLOW HAVE NOT INDICATED ANY ADVERSE EFFECTS FROM VOIDING. AN ENGINEERING EVALUATION WAS PERFORMED WHICH DETERMINED THAT AN ACCEPTABLE LEVEL IN THE SUCTION PIPING TO EACH OF THE HHSI PUMPS IS 50% FULL. THIS 50% LEVEL IN A STATIC SYSTEM IS THEORETICALLY EQUIVALENT TO APPROXIMATELY 10 CUBIC FEET OF GAS IN THE SUCTION HEADER AND ASSOCIATED PUMP SUCTION PIPING.

[122] NORTH ANNA 1 DOCKET 50-338 LER 89-015
ERROR IN LBLOCA ANALYSIS FOR THE 18% STEAM GENERATOR TUBE PLUGGING CASE.
EVENT DATE: 080889 REPORT DATE: 091189 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)

(NSIC 215262) ON 8/8/89, WITH UNITS 1 AND 2 AT 100% POWER (MODE 1), ENGINEERING PERSONNEL DISCOVERED AN INPUT ERROR IN THE LBLOCA ANALYSIS FOR THE 18% STEAM GENERATOR TUBE PLUGGING LICENSING CASE. ON 8/12/89, RESULTS OF THE REANALYSIS DETERMINED THAT CORRECTION OF THE ERROR RESULTED IN PEAK CLAD TEMPERATURES (PCT) WHICH EXCEEDED THE CURRENT LICENSING BASIS AND THE 2200F LIMIT SPECIFIED IN 10CFR50.46. EFFORTS WERE INITIATED TO VALIDATE MARGINS IN THE SAFETY ANALYSES AND TO ISSUE A STATION DEVIATION REPORT. FOLLOWING RECEIPT OF A DEVIATION REPORT ON 8/14/89, THE STATION NUCLEAR SAFETY AND OPERATING COMMITTEE DETERMINED THIS EVENT TO BE REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(II)(B). A PCT ABOVE THE LIMIT SPECIFIED IN 10CFR50.46 RESULTED FROM AN ERROR IN THE INPUT VALUE FOR THE AVERAGE ENTHALPY OF THE PRESSURIZER STEAM AND WATER VOLUME IN THE LBLOCA ANALYSIS FOR THE 18% STEAM GENERATOR TUBE PLUGGING LICENSING CASE. AS A CORRECTIVE ACTION, ADMINISTRATIVE LIMITS WERE PLACED ON UNIT 1 OPERATING PARAMETERS TO REDUCE THE PCT BELOW THE 10CFR50.46 LIMIT DURING A LBLOCA. THE SAFETY CONSEQUENCES OF THIS EVENT ARE EXPECTED TO BE MINIMAL BASED ON THE FINAL RESULTS OF THE REANALYSIS OF THE LBLOCA ANALYSIS FOR THE 15% STEAM GENERATOR TUBE PLUGGING CASE. PRELIMINARY RESULTS OF THE REANALYSIS VERIFY THAT A LBLOCA DURING PREVIOUS OPERATION OF NORTH ANNA UNITS 1 AND 2 WOULD NOT HAVE RESULTED IN A PCT ABOVE THE 10CFR50.46 LIMIT.

[123] NORTH ANNA 1 DOCKET 50-338 LER 89-016
MOV-SW-101A AND B SIMULTANEOUSLY CLOSED RENDERING TWO RECIRCULATION SPRAY HEAT EXCHANGERS INOPERABLE.
EVENT DATE: 082689 REPORT DATE: 090889 NSSS: WE TYPE: PWR

(NSIC 215263) AT 2250 HOURS ON 8/26/89, WITH UNIT 1 AT 100% POWER (MODE 1), THE RECIRCULATION SPRAY HEAT EXCHANGER (RSHX) SERVICE WATER (SW) 'A' SUPPLY HEADER ISOLATION MOTOR OPERATED VALVES (MOVS), 1-SW-MOV-101B AND 1-SW-MOV-101A WERE SIMULTANEOUSLY DEENERGIZED IN THE CLOSED POSITION, DURING THE PERFORMANCE OF AN OPERATIONS PROCEDURE. THE CROSS TIE BETWEEN THE 'A' AND 'B' SW SUPPLY HEADERS WAS ALSO CLOSED AT THE TIME OF THIS EVENT. AS A RESULT, TWO RSHXS DID NOT HAVE SW FLOW AVAILABLE AS A HEAT TRANSFER MEDIUM AND WERE THEREFORE INOPERABLE. THIS IS OUTSIDE THE ACTION STATEMENT REQUIREMENTS OF TECH SPEC 3.6.2.2 AND IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(VII)(D). MOVS 1-SW-MOV-101A AND 1-SW-MOV-101B WERE SIMULTANEOUSLY DEENERGIZED IN THE CLOSED POSITION DUE TO MISCOMMUNICATION BETWEEN THE SHIFT SUPERVISOR AND PERSONNEL ASSIGNED TO REMOVE THE POWER FROM THE MOVS. AS A CORRECTIVE ACTION, POWER WAS REINSTATED TO 1-SW-MOV-101B. PERSONNEL INVOLVED IN THIS EVENT WERE REINSTRUCTED ON THE IMPORTANCE OF CORRECT COMMUNICATION AND ATTENTION TO DETAIL WHEN TAGGING COMPONENTS. NO SIGNIFICANT SAFETY CONSEQUENCES RESULTED FROM THIS EVENT BECAUSE AN OPERATOR WAS STATIONED IN THE QUENCH SPRAY PUMP HOUSE BASEMENT TO MANUALLY OPEN THE CROSS TIE BETWEEN THE 'A' AND 'B' SW HEADERS IN THE EVENT OF A DESIGN BASIS ACCIDENT.

[124] NORTH ANNA 2 DOCKET 50-339 LER 89-008
INADVERTENT ESF ACTUATION DURING SSPS TESTING.
EVENT DATE: 080689 REPORT DATE: 090589 NSSS: WE TYPE: PWR

(NSIC 215165) AT 1202 HOURS ON 8/6/89, WITH UNIT 2 AT 100% POWER (MODE 1), ENGINEERED SAFETY FEATURES VALVE 2-RS-MOV-201B WAS INADVERTENTLY CLOSED DURING THE INITIAL ON-LINE PERFORMANCE OF 2-PT-36.5.3A, "SOLID STATE PROTECTION SYSTEM OUTPUT SLAVE RELAY TEST (TRAIN A),". THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV) DUE TO THE INADVERTENT CLOSURE OF ENGINEERED SAFETY FEATURES VALVE 2-RS-MOV-201B. THE INADVERTENT CLOSURE OF 2-RS-MOV-201B RESULTED BECAUSE

AN INTERLOCK FOR 2-RS-MOV-201B, ENERGIZED BY TRAIN A SLAVE RELAY K645, WAS NOT CORRECTLY IDENTIFIED DURING THE DEVELOPMENT OF 2-PT-36.5.3A. 2-RS-MOV-201B WAS OPENED FOLLOWING TESTING OF SLAVE RELAY K645. 2-PT-36.5.3A/B WILL BE REVISED PRIOR TO THE NEXT SCHEDULED ON-LINE PERFORMANCE. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE 2-RS-MOV-201B RECEIVES AN "ASSURE" OPEN SIGNAL UPON INITIATION OF A CONTAINMENT DEPRESSURIZATION SIGNAL AND WOULD HAVE ALLOWED CASING COOLING FLOW TO 2-RS-P-2B.

[125] OCONEE 1 DOCKET 50-269 LER 89-012
KEOWEE HYDRO UNITS WERE EMERGENCY STARTED DUE TO UNKNOWN REASONS.
EVENT DATE: 062489 REPORT DATE: 090189 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 3 (PWR)

(NSIC 215209) ON 6/24/89, AT 1040 HOURS, WITH UNIT 2 AT COLD SHUTDOWN AND MAKING PREPARATION FOR A RESTART FROM A REFUELING OUTAGE, BOTH OF THE ON-SITE EMERGENCY POWER SOURCES (KEOWEE HYDRO UNITS 1 & 2) EMERGENCY STARTED. THE EMERGENCY START WAS INITIATED FROM THE UNIT 2 KEOWEE EMERGENCY START CHANNEL B CIRCUITRY DUE TO UNKNOWN REASONS. ALSO, SEVERAL VALVES OF ENGINEERED SAFEGUARDS (ES) SYSTEM CHANNEL 2 REPOSITIONED TO THEIR ES POSITION. RELATED ACTIVITIES AT THE TIME OF THE EVENT INITIATION WERE FUNCTIONAL TESTING OF ES CHANNELS 1 AND 2 AND THE STEAM GENERATOR LEVEL CONTROL SYSTEMS. STATION ENGINEERING EVALUATED THE INCIDENT BUT COULD NOT IDENTIFY ANY CAUSE FOR THE EMERGENCY START OF THE KEOWEE UNITS OR THE REPOSITIONING OF THE ES VALVES. THE KEOWEE UNITS WERE SECURED FROM THE EMERGENCY START AND ES CHANNELS 1 AND 2 WERE RESET FROM TEST TO NORMAL OPERATING POSITION. OCONEE UNITS 1 AND 3 WERE OPERATING AT 100% FULL POWER. THE ROOT CAUSE OF THIS EVENT IS CLASSIFIED AS UNKNOWN.

[126] OCONEE 1 DOCKET 50-269 LER 89-013
A REACTOR TRIP OCCURRED DUE TO AN INAPPROPRIATE ACTION WHILE RESETTING REACTOR PROTECTIVE SYSTEM TRIP SETPOINTS.
EVENT DATE: 081089 REPORT DATE: 091189 NSSS: BW TYPE: PWR

(NSIC 215241) ON 8/10/89, AT 1541 HOURS, UNIT 1 TRIPPED DUE TO AN INADVERTENTLY INDUCED REACTOR PROTECTIVE SYSTEM (RPS) ACTUATION. AT THE TIME OF THE TRIP, UNIT 1 WAS AT 40% FULL POWER DUE TO A PREVIOUS POWER REDUCTION TO ADD OIL TO THE 1A2 REACTOR COOLANT PUMP MOTOR. DURING THIS REDUCED POWER OPERATION WITH THREE PUMPS, THE STEADY STATE QUADRANT POWER TILT LIMIT WAS EXCEEDED IN ONE CORE QUADRANT. AS REQUIRED BY TECH SPECS, ACTIONS WERE INITIATED TO BOTH REDUCE THE QUADRANT POWER TILT WITHIN LIMITS AND TO REDUCE THE OVERPOWER TRIP SETPOINTS BASED ON FLUX AND FLUX/FLOW/IMBALANCE. WHILE PERFORMING THE PROCEDURE TO LOWER THE OVERPOWER TRIP SETPOINTS, AN INSTRUMENTATION AND ELECTRICAL TECHNICIAN INCORRECTLY POSITIONED A FLOW TEST CIRCUIT SELECTOR SWITCH. SINCE ANOTHER RPS CHANNEL WAS TRIPPED PER PROCEDURE, A REACTOR TRIP WAS INITIATED WHEN THE CHANNEL WAS RETURNED TO SERVICE WITH THE INCORRECTLY POSITIONED SWITCH. THE ROOT CAUSE OF THIS EVENT IS CLASSIFIED AS INAPPROPRIATE ACTION, FAILURE TO FOLLOW PROCEDURE. THE IMMEDIATE CORRECTIVE ACTION WAS TO STABILIZE THE UNIT AT HOT SHUTDOWN CONDITIONS.

[127] OCONEE 2 DOCKET 50-270 LER 89-006
IMPROPER RELAY SETTING DUE TO DESIGN DEFICIENCY COULD TRIP 2B REACTOR BUILDING SPRAY PUMP DURING LOCA/LOOP EVENT.
EVENT DATE: 080489 REPORT DATE: 090589 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 1 (PWR)
OCONEE 3 (PWR)
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215163) ON 8/4/89, AT 1045 HOURS, DUKE POWER DESIGN ENGINEERING DISCOVERED THAT THE OVERCURRENT PROTECTIVE RELAY SETTINGS FOR THE 2B REACTOR BLDG. SPRAY

PUMP MOTOR WERE INSUFFICIENT AND MAY TRIP THE MOTOR DURING STARTING UNDER VOLTAGE CONDITIONS ASSOCIATED WITH A LOCA/LOOP EVENT. THIS POTENTIAL TRIP CONDITION HAD EXISTED SINCE JUNE 1980 WHEN THE ORIGINAL MOTOR WAS REPLACED WITH A SPARE MOTOR. THE PROBLEM WAS IDENTIFIED WHILE PERFORMING A DESIGN BASIS DOCUMENTATION STUDY OF THE 4.16 KV ELECTRICAL SYSTEM. ALL THREE OCONEE UNITS WERE AT 100% FULL POWER WHEN DISCOVERY WAS MADE. A DESIGN DEFICIENCY DURING PURCHASE OF THE SPARE MOTOR IN 1979 WAS THE ROOT CAUSE OF THIS EVENT. IMMEDIATE AND SUBSEQUENT CORRECTIVE ACTIONS WERE TO DETERMINE THE APPROPRIATE RELAY SETTING AND MAKE THE NECESSARY SETTING ADJUSTMENT. A 7 DAY LIMITED CONDITION OF OPERATION FOR UNIT 2 WAS DECLARED ON 8/4/89, AND CLEARED ON 8/5/89.

[128] OCONEE 3 DOCKET 50-287 LER 89-003
POST MAINTENANCE LEAK RATE TEST OF CONTAINMENT ISOLATION VALVE MISSED DUE TO DEFICIENT COMMUNICATION.
EVENT DATE: 072789 REPORT DATE: 091189 NSSS: BW TYPE: PWR
VENDOR: GRINNELL CORP.
LIMITORQUE CORP.

(NSIC 215250) AT APPROXIMATELY 1530 HOURS ON 7/27/89, A REACTOR BUILDING ISOLATION VALVE, 3LWD-1, WAS RETURNED TO SERVICE WITHOUT BEING LEAK RATE TESTED FOLLOWING MAINTENANCE ON THE VALVE OPERATOR. THIS WAS DISCOVERED AT 1600 HOURS ON 8/11/89 DURING REVIEW AND ACCEPTANCE OF THE COMPLETED WORK REQUEST PACKAGE BY OPERATIONS PERSONNEL. INADEQUATE COMMUNICATION BETWEEN THE WORK SUPERVISOR AND TEST PERSONNEL RESULTED IN A MISUNDERSTANDING OF WORK SCOPE AND FAILURE TO IDENTIFY A REQUIRED TEST. 3LWD-1 WAS DECLARED INOPERABLE, WAS SUCCESSFULLY LEAK RATE TESTED, AND THEN WAS RETURNED TO SERVICE AT 1830 HOURS ON 8/11/89. UNIT 3 WAS AT 100% FULL POWER THROUGHOUT THIS EVENT. THE ROOT CAUSE IS INAPPROPRIATE ACTION DUE TO DEFICIENT COMMUNICATION.

[129] OCONEE 3 DOCKET 50-287 LER 89-004
UNIT 3 REACTOR TRIPPED DUE TO INAPPROPRIATE ACTION, POOR WORK PRACTICE.
EVENT DATE: 081889 REPORT DATE: 091889 NSSS: BW TYPE: PWR

(NSIC 215304) ON 8/18/89 AT 1233 HOURS, OCONEE UNIT 3 TRIPPED FROM 100% FULL POWER. THE REACTOR TRIP WAS AN ANTICIPATORY TRIP RESULTING FROM A FALSE ELECTROHYDRAULIC CONTROL (EHC) SYSTEM LOW HYDRAULIC PRESSURE TRIP SIGNAL. THE FALSE SIGNAL WAS GENERATED WHEN WATER DROPS MADE MOMENTARY CONTACT ACROSS THE TERMINAL STRIP ASSOCIATED WITH THE LOW HYDRAULIC PRESSURE TRIP CIRCUIT. THE STATION JANITORIAL SERVICE VENDOR AND OPERATIONS PERSONNEL HAD WASHED THE FLOOR AROUND THE EHC HYDRAULIC POWER UNIT CABINET PRIOR TO THE UNIT TRIP. THE CABINET DOOR WAS INADEQUATELY SHUT, POTENTIALLY ALLOWING MOISTURE TO ENTER THE HYDRAULIC POWER UNIT CABINET. PLANT RESPONSE TO THE TRIP WAS NORMAL WITH NO RADIOLOGICAL RELEASES OR ENGINEERED SAFEGUARD ACTUATIONS. THE ROOT CAUSE OF THIS INCIDENT IS CLASSIFIED AS AN INAPPROPRIATE ACTION, POOR WORK PRACTICE. IMMEDIATE CORRECTIVE ACTIONS WERE TO STABLILZE THE UNIT AT HOT SHUTDOWN CONDITIONS.

[130] PALISADES DOCKET 50-255 LER 88-013 REV 03
UPDATE ON INOPERABLE CONTROL ROOM VENTILATION SYSTEM.
EVENT DATE: 082388 REPORT DATE: 083189 NSSS: CE TYPE: PWR

(NSIC 215158) AT 1700 HOURS ON 8/22/88 THE CONTROL ROOM (CR) HEATING VENTILATION AND AIR CONDITIONING (HVAC) SYSTEM (VI) WAS DECLARED INOPERABLE FOR FAILING TO MEET ACCEPTANCE CRITERIA DEFINED IN TECH SPEC SURVEILLANCE PROCEDURE RO-28, "CONTROL ROOM/TECHNICAL SUPPORT CENTER VENTILATION." DURING THIS TEST THE CR HVAC SYSTEM FAILED TO MAINTAIN THE REQUIRED 0.125 INCHES WATER-GAUGE REQUIRED. ON 10/27/88 ERRORS WERE IDENTIFIED IN ANALYSES WHICH WERE UTILIZED TO SPECIFY CR HVAC IODINE REMOVAL CAPABILITIES. THESE ERRORS RESULTED IN THE CR HVAC BEING UNABLE TO MEET GENERAL DESIGN CRITERION (GDC) 19 SINCE ITS ORIGINAL OPERATION IN

1983. FAILURE OF THE CR HVAC TO MEET SURVEILLANCE REQUIREMENTS WAS DUE TO ELECTRICAL PENETRATIONS IN THE CR FLOOR BEING OPENED FOR OUTAGE MODIFICATIONS AND AN IMPROPER FAN LINE-UP SPECIFIED IN RO-28 FOR THE ADJACENT SWITCHGEAR AREA VENTILATION SYSTEM. THE FAILURE OF THE CR HVAC SYSTEM TO MMET GDC 19 IS DUE TO CALCULATION ERRORS MADE WHEN PRESCRIBING THE REQUIRED IODINE REMOVAL CAPACITY. ACCEPTABLE CR HABITABILITY ANALYSES HAVE BEEN PERFORMED FOR ALL DESIGN BASIS ACCIDENTS. RADIATION MONITORING INSTRUMENTATION WAS PLACED WITHIN THE CR HVAC ENVELOPE TO PROVIDE FOR EARLY DETECTION. AUTOMATIC SWITCHOVER OF THE CR HVAC INTO THE EMERGENCY MODE WILL BE PROVIDED FROM DETECTION OF PRESET RADIATION LEVELS WITHIN THE RADIOACTIVE GASEOUS MONITORING SYSTEM AND THE MAIN STEAM LINE GAMMA MONITOR.

[131] PALISADES DOCKET 50-255 LER 89-006 REV 01
UPDATE ON COMPONENT COOLING WATER AVAILABILITY FOLLOWING A HIGH ENERGY LINE BREAK.
EVENT DATE: 032389 REPORT DATE: 091989 NSSS: CE TYPE: PWR

(NSIC 215302) THROUGH THE EFFORTS ASSOCIATED WITH THE DESIGN BASIS RECONSTRUCTION PORTION OF THE CONFIGURATION CONTROL PROJECT (CCP), A CONCERN WAS IDENTIFIED REGARDING THE POTENTIAL FOR A COMPLETE LOSS OF COMPONENT COOLING WATER (CCW) (CC) DURING AN EVENT WHICH RESULTED IN A HIGH ENERGY LINE BREAK (HELB). THE CONCERN ASSUMES THAT A LOSS OF COOLANT ACCIDENT (LOCA) WITH THE LOSS OF OFFSITE POWER OCCURS. THE LOCA BECAUSE OF JET FORCES, PHYSICAL CONTACT OR MOVEMENT (I.E., HIGH ENERGY LINE) IS POSTULATED TO IMPACT THE CCW SYSTEM PIPING, RESULTING IN A SYSTEM BREACH. SUBSEQUENT TO THESE EVENTS, THE INSTRUMENT AIR SYSTEM IS ASSUMED TO FAIL. IF THESE EVENTS AND FAILURES WERE ALL TO OCCUR, THE CCW INVENTORY WOULD BE LOST TO THE CONTAINMENT BUILDING AND THE SYSTEM RENDERED INOPERABLE. IF, IN ADDITION, ONE OF THE TWO EMERGENCY DIESEL GENERATORS WERE TO FAIL, ALL CONTAINMENT HEAT REMOVAL SYSTEMS WOULD BE RENDERED INOPERABLE. SUBSEQUENTLY, ON 5/16/89 DURING ANOTHER TASK OF THE CCP, IT WAS IDENTIFIED THAT THE SINGLE FAILURE OF CV-0910 TO CLOSE DUE TO A MECHANICAL FAILURE OF THE VALVE COINCIDENT WITH THE MULTIPLE EVENTS IDENTIFIED ABOVE, EXCEPT THE FAILURE OF A DIESEL GENERATOR, WOULD ADDITIONALLY RESULT IN A LOSS OF CCW INVENTORY. AS THE NEWLY IDENTIFIED FAILURE IS NOT ASSOCIATED WITH A LOSS OF AIR, THE INITIAL PLANNED ACTION TO ADD A BACKUP NITROGEN SUPPLY WILL NOT BE PURSUED.

[132] PALISADES DOCKET 50-255 LER 89-020
REACTOR TRIP DUE TO BLOWN FUSE AND SUBSEQUENT AUXILIARY FEEDWATER PUMP START.
EVENT DATE: 080489 REPORT DATE: 090589 NSSS: CE TYPE: PWR

(NSIC 215208) ON 8/4, AT 1945, A REACTOR TRIP FROM 80% POWER OCCURRED DUE TO A BLOWN FUSE (JB;FU) IN A FLOW INDICATING CONTROLLER (SJ;FIC) THAT CAUSED THE FEEDWATER REGULATING VALVE (SJ;LCV) FOR THE B STEAM GENERATOR TO CLOSE, THEREBY DECREASING FEEDWATER FLOW TO THE STEAM GENERATOR AND RESULTING IN A REACTOR TRIP ON LOW STEAM GENERATOR LEVEL. THE PLANT RESPONSE TO THE TRIP WAS CONSIDERED NORMAL WITH NO SAFETY SIGNIFICANT DEVIATIONS OR ANOMALIES OBSERVED. ON 8/5/89 AT 0300, AN UNANTICIPATED START OF AUXILIARY FEEDWATER PUMP P-8B (BA;P) OCCURRED WHILE THE PLANT WAS IN THE HOT SHUTDOWN CONDITION. THE SPURIOUS SIGNAL WAS CAUSED BY A FAILURE BY THE OPERATORS TO RESET THE AFAS SIGNAL FOLLOWING THE REACTOR TRIP. CORRECTIVE ACTIONS TO BE TAKEN IN RESPONSE TO THESE INCIDENTS INCLUDE A REVIEW OF THE FUSE CONTROL PROGRAM AND A REVISION TO POST TRIP OPERATING PROCEDURES THAT WILL ADD INSTRUCTIONS TO THE OPERATORS FOR RESETTING ALARMS AND ACTIVATED ACTUATION SIGNALS.

[133] PALO VERDE 1 DOCKET 50-528 LER 88-006 REV 01
UPDATE ON SURVEILLANCE INTERVAL EXCEEDED FOR INCORE DETECTOR SYSTEM.
EVENT DATE: 032088 REPORT DATE: 092889 NSSS: CE TYPE: PWR

(NSIC 215335) ON 3/21/88, PALO VERDE UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT

100 PERCENT POWER WHEN ENGINEERING PERSONNEL (UTILITY, NON-LICENSED) DETERMINED THAT THE ALLOWABLE SURVEILLANCE TEST INTERVAL HAD BEEN EXCEEDED FOR THE INCORE DETECTOR SYSTEM (IG). THIS RESULTED IN THE INCORE DETECTOR SYSTEM AND THE CORE OPERATING LIMIT SUPERVISORY SYSTEM (COLSS) (ID) BECOMING ADMINISTRATIVELY INOPERABLE. THE ROOT CAUSE OF THE EVENT WAS A COGNITIVE PERSONNEL ERROR BY ENGINEERING PERSONNEL (UTILITY, NON-LICENSED) RESPONSIBLE FOR ASSIGNING THE PERFORMANCE OF THE SURVEILLANCE TEST. TO PREVENT RECURRENCE, THE INDIVIDUAL WILL RECEIVE APPROPRIATE COUNSELING AND/OR DISCIPLINARY ACTION. A PRELIMINARY EVALUATION HAS DETERMINED THAT A CONTRIBUTORY CAUSE WAS INADEQUATE PROGRAMMATIC CONTROLS TO ENSURE SURVEILLANCE TESTS ARE CONDUCTED WITHIN THE SPECIFIED INTERVALS. A REVIEW OF THE PROGRAMMATIC CONTROLS WAS PERFORMED AND HAS IDENTIFIED SEVERAL RECOMMENDED CORRECTIVE ACTIONS. AS A RESULT, UNIT 1, 2, AND 3 CROSS REFERENCE PROCEDURES FOR CONDITIONAL STS WERE DEVELOPED. HOWEVER, THIS REVIEW WAS PERFORMED BY AN INDEPENDENT GROUP, AND ADDITIONAL RECOMMENDED ACTIONS ARE BEING EVALUATED FOR IMPLEMENTATION BY THE APPROPRIATE ORGANIZATIONS. REVIEW AND DISPOSITION IS EXPECTED TO BE COMPLETED BY 10/30/89.

[134] PALO VERDE 1 DOCKET 50-528 LER 88-020 REV 01
UPDATE ON APPARENT GROUND CAUSES CONTROL ELEMENT ASSEMBLY SLIP.
EVENT DATE: 121088 REPORT DATE: 090789 NSSS: CE TYPE: PWR
VENDOR: COMBUSTION ENGINEERING, INC.

(NSIC 215229) AT APPROXIMATELY 1059 MST ON 12/10/88, UNIT 1 WAS IN MODE 1 AT 85% POWER WHEN CONTROL ELEMENT ASSEMBLY (CEA) 64 SLIPPED A TOTAL OF APPROXIMATELY 89 INCHES BELOW THE OTHER CEAS IN ITS GROUP. PRIOR TO THE EVENT, CONTROL ELEMENT ASSEMBLY CALCULATORS (CEACS) 1 AND 2 WERE DECLARED INOPERABLE. WITH BOTH CEACS INOPERABLE TECH SPEC 3.3.1, ACTION 6, REQUIRES THAT EACH CEA BE ALIGNED WITHIN 6.6 INCHES OF ALL OTHER CEAS IN ITS GROUP. AT APPROXIMATELY 1100 MST DURING RECOVERY OPERATIONS TO RESTORE CEA-64, CONTROL ROOM PERSONNEL IDENTIFIED THAT CEA-57 HAD SLIPPED TO APPROXIMATELY 105.7 INCHES WITHDRAWN. THE CAUSE OF CEA-64 SLIPPING WAS AN APPARENT GROUND ON THE LOWER LIFT COIL. AS IMMEDIATE CORRECTIVE ACTION, CEA-57 AND -64 WERE REALIGNED WITH THE OTHER CEAS IN ITS GROUP AT APPROXIMATELY 1126 MST ON 12/10/88 AND COMPLIANCE WITH THE ACTION REQUIREMENTS OF THE TECH SPECS WERE ACHIEVED. THE CAUSE OF THE UNIT BEING IN A CONDITION OUTSIDE THE ACTION STATEMENT WAS AN INAPPROPRIATE DELETION OF ACTION REQUIREMENTS MADE DURING A TECH SPEC REVISION. AN AMENDMENT IS BEING PURSUED TO PROVIDE ACTION REQUIREMENTS FOR A CEA MISALIGNMENT WITH CEACS INOPERABLE. THE CAUSE OF CEA-57 SLIPPING WAS A DESIGN DEFECT IN THE CONTROL ELEMENT DRIVE MECHANISM CONTROL SYSTEM (CEDMCS) THAT SEQUENCES THE OPERATION OF THE CEAS.

[135] PALO VERDE 1 DOCKET 50-528 LER 89-002 REV 01
UPDATE ON NON-QUALIFIED COMPONENTS INSTALLED ON ATMOSPHERIC DUMP VALVES.
EVENT DATE: 033089 REPORT DATE: 091989 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 2 (PWR)
PALO VERDE 3 (PWR)
VENDOR: CONTROL COMPONENTS

(NSIC 215325) ON 3/30/89 AT APPROXIMATELY 1715 MST PALO VERDE UNITS 1 AND 2 WERE IN MODE 3 (HOT STANDBY) AND UNIT 3 WAS IN MODE 5 (COLD SHUTDOWN) WHEN APS ENGINEERING PERSONNEL DETERMINED THAT THE ATMOSPHERIC DUMP VALVES (ADV'S) IN UNITS 1, 2, AND 3 DID NOT MEET THE REQUIRED ENVIRONMENTAL QUALIFICATIONS SINCE ORIGINALLY PLACED IN SERVICE DUE TO NON-ENVIRONMENTALLY QUALIFIED PRESSURE GAUGES BEING INSTALLED ON THE ADV POSITIONERS. THE INSTALLATION OF THE UNQUALIFIED PRESSURE GAUGES RESULTED IN PERIODIC NON-COMPLIANCE WITH TECH SPEC 3.7.1.6. THERE HAVE BEEN NO ADV FAILURES RESULTING FROM THE INSTALLATION OF THE UNQUALIFIED GAUGES. THE ADV'S WERE ORIGINALLY SUPPLIED TO PVNGS WITH THE PRESSURE GAUGES INSTALLED AND DID NOT GET REMOVED IN ACCORDANCE WITH MANUFACTURER'S INSTRUCTIONS PROVIDED WITH THE VALVES. THE CAUSE OF THIS EVENT WAS AN INADEQUATE VERIFICATION AND REVIEW OF DESIGN CHANGE CLOSEOUT DOCUMENTATION

DURING STARTUP. THE WORK AUTHORIZIATION DOCUMENTS HAD BEEN SIGNED OFF AS COMPLETE WITHOUT THE WORK BEING PERFORMED. AS IMMEDIATE CORRECTIVE ACTION, THE PRESSURE GAUGES WERE REMOVED AND QUALIFIED COMPONENTS WERE INSTALLED. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED PURSUANT TO 10CFR50.73.

[136] PALO VERDE 1 DOCKET 50-528 LER 89-005 REV 01
UPDATE ON ATMOSPHERIC DUMP VALVE DEFICIENCIES DUE TO INTERNAL PISTON RING NOT SEATING PROPERLY.
EVENT DATE: 041289 REPORT DATE: 091289 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 2 (PWR)
PALO VERDE 3 (PWR)
VENDOR: CONTROL COMPONENTS

(NSIC 215287) ON 4/12/89 APS COMPLETED AN EVALUATION OF A DEFICIENCY IDENTIFIED BY THE MANUFACTURER OF THE PVNGS UNITS 1, 2, AND 3 ATMOPSHERIC DUMP VALVES (ADV'S). THE ADV'S ARE MANUFACTURED BY CONTROL COMPONENTS INCORPORATED (CCI). BASED UPON APS'S EVALUATION, IT WAS DETERMINED THAT THE DEFICIENCIES REPORTED BY CCI CONSTITUTED A REPORTABLE CONDITION PURSUANT TO 10CFR21 AND CONSEQUENTLY 10CFR50.72 AND 73. ON 4/4/89, CCI NOTIFIED APS THAT AN EVALUATION HAD BEEN PERFORMED AND THAT EXCESSIVE INTERNAL VALVE LEAKAGE COULD RESULT IN THE INABILITY TO REMOTELY OR MANUALLY OPERATE THE PVNGS ADV'S. THE CAUSE OF THE EXCESSIVE LEAKAGE IS THE RESULT OF AN INTERNAL PISTON RING WHICH DOES NOT SEAT PROPERLY. EXCESSIVE LEAKAGE BY THE PISTON RING RESULTS IN HIGH INTERNAL PRESSURES WHICH WOULD PRECLUDE OPENING OF THE VALVE. AS CORRECTIVE ACTION, THE ADV INTERNALS HAVE BEEN REDESIGNED AND ARE BEING REPLACED IN UNITS 1, 2, AND 3. NO PREVIOUS SIMILAR EVENTS HAVE BEEN REPORTED PURSUANT TO 10CFR50.73.

[137] PALO VERDE 1 DOCKET 50-528 LER 89-012 REV 01
UPDATE ON EMERGENCY LIGHTING SYSTEM DEFICIENCIES FOR SAFE SHUTDOWN ACTIVITIES DUE TO INADEQUATE DESIGN.
EVENT DATE: 051089 REPORT DATE: 092289 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 2 (PWR)
PALO VERDE 3 (PWR)

(NSIC 215394) ON 5/10/89, PALO VERDE UNITS 1 AND 3 WERE IN REFUELING OUTAGES AND PALO VERDE UNIT 2 WAS IN MODE 5 (COLD SHUTDOWN) WHEN APS DETERMINED THAT EMERGENCY LIGHTING IN UNIT 2 DID NOT MEET PVNGS DESIGN BASES OR 10CFR50 APPENDIX R REQUIREMENTS. TWENTY-FOUR (24) AREAS WERE IDENTIFIED WHICH REQUIRE THE INSTALLATION OF OR MODIFICATION TO LIGHTING IN ORDER TO BE ABLE TO PERFORM REQUIRED SAFE SHUTDOWN ACTIVITIES ASSUMING A FIRE IN THE CONTROL ROOM CONCURRENT WITH A LOSS OF OFF-SITE POWER. A SIMILAR CONDITION WAS DETERMINED TO ALSO EXIST IN UNITS 1 AND 3. THE CAUSE OF THIS EVENT WAS THAT THE ORIGINAL EMERGENCY LIGHTING DESIGN WAS DEVELOPED IN 1984 USING OPERATING PROCEDURES AND STUDIES WHICH WERE NOT COMPLETE AND NOT APPROVED, INADEQUATE ENGINEERING GUIDANCE FOR CONDUCTING THE WALKDOWNS, AND INADEQUATE VERIFICATION THAT EMERGENCY LIGHTING REQUIREMENTS WERE MET. AS CORRECTIVE ACTION, DESIGN CHANGES UTILIZING APPROVED METHODOLOGIES WERE DEVELOPED TO UPGRADE THE EMERGENCY LIGHTING. UNIT 2 HAS COMPLETED INSTALLATION OF THE RE-DESIGNED EMERGENCY LIGHTING. UNITS 1 AND 3 WILL INSTALL THE RE-DESIGN LIGHTING PRIOR TO RE-START FROM THEIR CURRENT REFUELING OUTAGES. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED PURSUANT TO 10CFR50.73. THIS REPORT IS ALSO BEING PROVIDED PURSUANT TO TECH SPEC 6.9.3

[138] PALO VERDE 1 DOCKET 50-528 LER 89-006
INADVERTENT CONTAINMENT PURGE ISOLATION DUE TO LOSS OF POWER TO RAD MONITOR UNIT.
EVENT DATE: 073189 REPORT DATE: 083089 NSSS: CE TYPE: PWR

(NSIC 215178) ON 7/31/89, PALO VERDE UNIT 1 WAS IN A REFUELING OUTAGE WITH THE CORE OFF-LOADED TO THE SPENT FUEL POOL. ON THE NIGHT SHIFT OF 7/31/89, UNIT 1

PERSONNEL WERE MAKING PREPARATIONS FOR AN OUTAGE OF ALL THE TRAIN "B" CLASS 1E ELECTRICAL SWITCHGEAR. AUXILIARY OPERATORS WERE STRIPPING TRAIN "B" LOADS WHEN AT APPROXIMATELY 0115 MST ON 7/31/89, THERE WAS A LOSS OF POWER TO PANEL 1E-PNB-D26 WHICH CAUSED A LOSS OF POWER TO THE REMOTE INDICATING AND CONTROL (RIC) UNIT FOR RADIATION MONITOR RU-38 THUS INITIATING A TRAIN "B" CONTAINMENT PURGE ISOLATION ACTUATION SIGNAL (CPIAS). THE CPIAS CROSS-TRIPPED TRAIN "B" CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNAL (CREFAS) WHICH IN TURN CROSS-TRIPPED CREFAS "A", ALL IN ACCORDANCE WITH DESIGN. DUE TO THE PLANNED ELECTRICAL OUTAGE, ALL SAFETY EQUIPMENT FOR CPIAS AND CREFAS WAS IN ITS ACTUATED CONDITION PRIOR TO THE EVENT WITH THE EXCEPTION OF THE TRAIN "B" CONTROL ROOM ESSENTIAL AIR HANDLING UNIT WHICH STARTED AS DESIGNED. APPROXIMATELY ONE (1) MINUTE AFTER THE EVENT INITIATION, POWER WAS RESTORED TO 1E-PNB-D26 AND RU-38. RU-38 WAS PLACED BACK ON LINE AND AT APPROXIMATELY 0221 MST ON 7/31/89, THE CPIAS AND CREFAS WERE RESET. AN INVESTIGATION OF THIS EVENT IS BEING CONDUCTED IN ACCORDANCE WITH THE PVNGS INCIDENT INVESTIGATION PROGRAM.

[139] PALO VERDE 1 DOCKET 50-528 LER 89-009
TECH SPEC VIOLATION DUE TO PERSONNEL ERROR.
EVENT DATE: 081689 REPORT DATE: 091589 NSSS: CE TYPE: PWR

(NSIC 215288) ON 8/16/89 AT APPROXIMATELY 0232, A QC INSPECTOR ENTERED THE UNIT 1 STEAM GENERATOR #2 HOT LEG BOWL WITHOUT POSITION CONTROL OVER THE ACTIVITIES BEING PROVIDED BY A RADIATION PROTECTION QUALIFIED INDIVIDUAL AS REQURIED BY TECH SPEC 6.12.1.C. THE QC INSPECTOR HAD BEEN IN THE STEAM GENERATOR FOR APPROXIMATELY ONE MINUTE AND WAS EXITING THE STEAM GENERATOR WHEN RADIATION PROTECTION PERSONNEL DISCOVERED THE QC INSPECTOR HAD ENTERED THE STEAM GENERATOR. THE QC INSPECTOR RECEIVED LESS THAN 400 MREM WHOLE BODY DOSE AND MET ALL RADIATION ENTRY PERMIT REQUIREMENTS. THE CAUSE OF THIS EVENT WAS COGNITIVE PERSONNEL ERROR BY THE QC INSPECTOR (CONTRACTOR, NON-LICENSED). THE QC INSPECTOR FAILED TO USE FORMAL COMMUNICATION PROTOCOL TO VERIFY THAT RP WAS PERMITTING ENTRY INTO THE STEAM GENERATOR. A CONTRIBUTING FACTOR WAS THE RP TECHNICIANS ALLOWED THE QC INSPECTOR AND PLATFORM WORKER TO ACCESS THE PLATFORM WITHOUT RP BEING ON THE COMMUNICATION LINES. STEAM GENERATOR ENTRIES WERE STOPPED UNTIL RADIATION PROTECTION MODIFIED THE PRE-JOB BRIEFING ON COMMUNICATIONS AND IMPLEMENTED A POLICY OF MAINTAINING RP ON THE COMMUNICATION HEADSET ANY TIME PERSONNEL WERE ON THE SG PLATFORM. MOCKUP TRAINING FOR STEAM GENERATOR WORK WILL BE MODIFIED TO REINFORCE COMMUNICATIONS PROTOCAL.

[140] PALO VERDE 2 DOCKET 50-529 LER 89-006 REV 01
UPDATE ON REACTOR PROTECTION SYSTEM ACTUATION DUE TO DIP IN THE VOLTAGE SUFFICIENT TO INITIATE AN RPS ACTUATION.
EVENT DATE: 051089 REPORT DATE: 091289 NSSS: CE TYPE: PWR

(NSIC 215289) AT APPROXIMATELY 1109 MST ON 5/10/89, PALO VERDE UNIT 2 WAS IN MODE 5 (COLD SHUTDOWN) WITH THE REACTOR COOLANT SYSTEM (RCS) TEMPERATURE AT APPROXIMATELY 137F AND PRESSURIZER PRESSURE AT APPROXIMATELY 130 PSIA WHEN AN AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM (RPS) OCCURRED. IN ACCORDANCE WITH APPROVED PROCEDURES THE LOW STEAM GENERATOR PRESSURE AND LOW RCS FLOW INITIATION TRIP RELAYS WERE BEING ENERGIZED WITH A TEMPORARY ELECTRICAL POWER SUPPLY TO PERFORM CONTROL ELEMENT ASSEMBLY TESTING. THESE TRIPS ARE NOT REQUIRED TO BE OPERABLE IN MODE 5 AND WERE RESET WITH THE TEMPORARY POWER SUPPLY. THE REACTOR TRIP SWITCHGEAR BREAKERS WERE SHUT WHEN A COMPUTER TECHNICIAN PLUGGED IN A TEST CART INTO THE SAME WALL RECEPTABLE THAT WAS SUPPLYING TEMPORARY POWER TO THE RELAYS. PLUGGING IN THE TEST CART CAUSED A DIP IN THE VOLTAGE SUFFICIENT TO DEENERGIZE THE INITIATION RELAYS AND INITIATE AN RPS ACTUATION OPENING THE REACTOR TRIP SWITCHGEAR BREAKERS. AS CORRECTIVE ACTION, THE COMPUTER TEST CART WAS UNPLUGGED AND ALL OUTLETS FROM THE COMMON LIGHTING PANEL IN THE CONTROL ROOM WERE YELLOW CAUTION TAGGED WITH INSTRUCTIONS TO CONTACT THE SHIFT SUPERVISOR BEFORE USING THE RECEPTABLES.

[141] PALO VERDE 3 DOCKET 50-530 LER 88-008 REV 02
UPDATE ON RADIATION MONITORING SYSTEM INVALID SAMPLE RESULTS.
EVENT DATE: 121388 REPORT DATE: 092889 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 2 (PWR)

(NSIC 215336) ON 2/20/89, PVNGS ENGINEERING PERSONNEL COMPLETED AN EVALUATION OF THE EFFECTS OF EXCESSIVE MOISTURE CONDENSATION IN THE PARTICULATE AND IODINE FILTERS UTILIZED IN THE CONDENSER EVACUATION SYSTEM EFFLUENT LOW RANGE RADIATION MONITORS. IT WAS DETERMINED THAT EXCESSIVE MOISTURE CONDENSATION OCCASIONALLY DISCOVERED IN PALO VERDE UNIT 3 RESULTED IN INVALID SAMPLE RESULTS. THE INVALID SAMPLES RESULTED IN THE INABILITY TO SATISFY THE SAMPLING REQUIREMENTS OF TECH SPEC 3.3.3.8 ACTION 40 AND SURVEILLANCE REQUIREMENT 4.11.2.1.2. THE CAUSE OF THE EXCESSIVE MOISTURE BUILDUP WAS COOLING OF THE HIGH HUMIDITY SAMPLE STREAM WHICH RESULTED FROM INADEQUATE ELECTRICAL RESISTANCE HEATING APPLIED TO THE SAMPLE STREAM PIPING IN COMBINATION WITH WINTERTIME AMBIENT ENVIRONMENTAL CONDITIONS. PROPER HEAT TRACE WAS NOT INSTALLED IN ACCORDANCE WITH DESIGN REQUIREMENTS AT THE TIME OF INITIAL MONITOR INSTALLATION. AS INTERIM CORRECTIVE ACTION, TEMPORARY HEAT TRACING WAS INSTALLED IN UNITS 2 AND 3. AS CORRECTIVE ACTION TO PREVENT RECURRENCE, PERMANENT HEAT TRACING HAS BEEN INSTALLED IN UNIT 2 AND WILL BE INSTALLED IN UNITS 1 AND 3 PRIOR TO RESTART FROM THEIR CURRENT REFUELING OUTAGES. ADDITIONAL DESIGN ENHANCEMENTS TO PRECLUDE MOISTURE BUILDUP IN OTHER AREAS OF THE MONITOR WILL BE IMPLEMENTED DURING EACH UNIT'S NEXT REFUELING OUTAGE. A PREVIOUS SIMILAR EVENT WAS REPORTED IN UNIT 1 LER 85-37-01.

[142] PALO VERDE 3 DOCKET 50-530 LER 89-003 REV 01
UPDATE ON SPURIOUS ACTUATION OF CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNAL.
EVENT DATE: 021889 REPORT DATE: 092189 NSSS: CE TYPE: PWR
VENDOR: KAMAN SCIENCES CORP.

(NSIC 215327) AT APPROXIMATELY 1235 MST ON 2/18/89, PALO VERDE UNIT 3 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 100% POWER WHEN A CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNAL (CREFAS) TRAIN "B" WAS ACTUATED. THE TRAIN "B" CREFAS CROSSTRIPPED THE TRAIN "A" CREFAS AS DESIGNED. ALL EQUIPMENT ACTUATED AS REQUIRED. THE CREFAS WAS ACTUATED BY A SPURIOUS TRIP FROM THE CONTROL ROOM VENTILATION INTAKE RADIATION MONITOR RU-30. THE REDUNDANT MONITOR RU-29 WAS VERIFIED TO BE READING NORMALLY. AN OUTSIDE AIR SAMPLE WAS OBTAINED FROM ON TOP OF THE RADIOACTIVE WASTE BUILDING ROOF NEXT TO THE CONTROL BUILDING AND WAS ANALYZED WITH NO ABNORMAL RADIOACTIVITY DETECTED. THE CREFAS WAS RESET AT APPROXIMATELY 1253 MST AND CONTROL ROOM VENTILATION WAS RETURNED TO A NORMAL CONFIGURATION BY APPROXIMATELY 1340 MST ON 2/18/89. THE CAUSE OF THE EVENT WAS THE FAILURE OF THE DETECTOR IN RU-30. THE DETECTOR AND PRE-AMPLIFIER WERE REPLACED AND THE MONITOR WAS OBSERVED FOR SEVERAL DAYS WITH NO FURTHER MALFUNCTIONS IDENTIFIED. AFTER SATISFACTORY COMPLETION OF THE PROPER SURVEILLANCE TEST RU-30 WAS RETURNED TO OPERABLE STATUS AT APPROXIMATELY 1505 MST ON 2/24/89.

[143] PALO VERDE 3 DOCKET 50-530 LER 89-001 REV 01
UPDATE ON REACTOR TRIP DUE TO LOW STEAM GENERATOR LEVEL.
EVENT DATE: 030389 REPORT DATE: 092189 NSSS: CE TYPE: PWR
VENDOR: ALLEN-BRADLEY CO.
CONTROL COMPONENTS

(NSIC 215326) ON 3/3/89 AT APPROXIMATELY 0102 MST PALO VERDE UNIT 3 WAS OPERATING AT APPROXIMATELY 98% POWER WHEN AN ELECTRICAL GRID DISTURBANCE RESULTED IN THE MAIN GENERATOR OUTPUT BREAKERS OPENING. THIS RESULTED IN A REACTOR POWER CUTBACK (RPCB) AND STEAM BYPASS CONTROL SYSTEM (SBCS) ACTUATION. AN SBCS MALFUNCTION RESULTED IN A STEAM GENERATOR (S/G) NUMBER 2 LOW PRESSURE REACTOR TRIP, TURBINE TRIP, MAIN STEAM ISOLATION SIGNAL, AND CONTAINMENT ISOLATION ACTUATION SIGNAL AT

APPROXIMATELY 0103 MST. APPROXIMATELY SIX SECONDS LATER, A SAFETY INJECTION ACTUATION SIGNAL OCCURRED AS A RESULT OF LOW PRESSURIZER PRESSURE. CONTROL ROOM PERSONNEL ATTEMPTED TO REMOVE DECAY HEAT AND CONTROL S/G PRESSURE UTILIZING THE ATMOSPHERIC DUMP VALVES (ADV'S). CONTROL ROOM PERSONNEL COULD NOT REMOTELY OPERATE THE ADV'S FROM THE CONTROL ROOM OR REMOTE SHUTDOWN PANEL. HEAT REMOVAL WAS SUBSEQUENTLY ESTABLISHED BY MANUALLY OPENING THE ADV'S. IN THE INTERIM, ONE MAIN STEAM SAFETY VALVE CYCLED TO REMOVE DECAY HEAT AND CONTROL S/G PRESSURE. THE CAUSE OF THE REACTOR TRIP WAS A MALFUNCTION IN THE SBCS. AN INDEPENDENT INVESTIGATION HAS BEEN CONDUCTED TO DETERMINE THE CAUSES OF THE PROBLEMS OCCURRING DURING THE EVENT. BASED UPON THE INVESTIGATION, APPROPRIATE CORRECTIVE MEASURES HAVE BEEN DEVELOPED. THIS SUBMITTAL ALSO PROVIDES A SPECIAL REPORT IN ACCORDANCE WITH TECH SPEC 3.5.2 ACTION B.

[144]   PALO VERDE 3   DOCKET 50-530   LER 89-008 REV 01
UPDATE ON SURVEILLANCE REQUIREMENTS NOT PERFOR[illegible]D S[illegible]SFACTORILY.
EVENT DATE: 063089   REPORT DATE: 092[illegible]   [illegible]   TYPE: PWR

(NSIC 215395) ON 6/30/89, PAL[illegible] [illegible]NIT 3 WAS [illegible] REFUELING OU[illegible]GE WITH [illegible] CORE OFF-LOADED TO THE SPENT [illegible] WHEN MAI[illegible]ICE PERSONNEL IDENTIFIED A SIXTH TRIP LEVER ARM ON THE [illegible] [illegible]ING CRANE RAILS. THESE TRIP LEVER ARMS ROTATE A STAR WHEEL [illegible] [illegible] SWITCHES TO PREVENT LOADS IN EXCESS OF 2000 POUNDS FR[illegible] [illegible] [illegible]IES IN THE SPENT FUEL STORAGE POOL. THE SIXTH TR[illegible] [illegible] WHE[illegible] A S[illegible]XT[illegible] TIME ACTIVATING THE LIMIT SWITCHE[illegible] [illegible]/HO[illegible] [illegible] OVER THE SPENT FUEL STORAGE POOL [illegible] IN ACCORD[illegible] [illegible]TH A FIELD CHAN[illegible] REQUEST IN AN AUTH[illegible] [illegible] THERE SHO[illegible] BE FIVE (5) TRIP LEVER ARMS FOR THE [illegible] [illegible]ERLOCK[illegible] DES[illegible] CHA[illegible] WAS COMPLETED IN MARCH OF 1986. AN I[illegible]ATION OF [illegible]S EVEN[illegible] IS [illegible]NG [illegible] ACCORDANCE WITH THE PVNGS INCID[illegible] [illegible]STIGA[illegible] [illegible]GRAM. THE RES[illegible] OF [illegible] I[illegible]VESTIGATION WILL BE REPORTED IN A RE[illegible] LER ONLY IF IT WOULD SIG[illegible]TLY CHANGE THE READER'S PERCEPTION OF THE COURSE, SI[illegible] F[illegible]ANCE, IMPLICA[illegible]ONS, OR CONSEQUENCES OF THE EVENT; OR RESULTS IN SUBSTANTIAL [illegible] IN THE CORRE[illegible]IVE ACTION PLANNED BY THE LICENSEE. IMMEDIATE CORRECTIVE ACTION WAS TO MOVE THE CRANE TROL[illegible] FROM OVER THE SPENT FUEL STORAGE POOL AND DECLARE THE CRANE INOPERABLE. AS CORRECTIVE ACTION TO PREVENT RECURRENCE, THE SIXTH TRIP LEVER ARM WAS REMOVED.

[145]   PEACH BOTTOM 2   DOCKET 50-277   LER 89-015
MALFUNCTIONING ELECTRO-HYDRAULIC CONTROL SYSTEM COMPONENT CAUSES SCRAM WHEN REMOVED FROM SERVICE.
EVENT DATE: 072189   REPORT DATE: 081889   NSSS: GE   TYPE: BWR
VENDOR: GEN ELEC CO (STEAM TURB/ENGRD PROD)

(NSIC 215245) AT 2231 ON 7/21/89 WITH 2 AT 79% THERMAL POWER, AN ATTEMPT WAS MADE TO REMOVE A MALFUNCTIONING REACTOR PRESSURE VESSEL (RPV) PRESSURE REGULATOR SET FROM THE ELECTRONIC PORTION OF THE MAIN TURBINE (MT) ELECTRO-HYDRAULIC CONTROL (EHC) PRESSURE REGULATING SYSTEM. IMMEDIATELY, THE MT BYPASS AND CONTROL VALVES OPENED, CAUSING MAIN STEAM LINE PRESSURE TO DECREASE TO APPROXIMATELY 480 PSIG. AT 850 PSIG MAIN STEAM LINE PRESSURE A GROUP I ISOLATION OCCURRED CAUSING THE MAIN STEAM ISOLATION VALVES (MSIV) TO CLOSE. AS A RESULT, A FULL REACTOR SCRAM OCCURRED. RPV LEVEL DECREASE DUE TO SHRINK FOLLOWING MSIV CLOSURE RESULTED IN A GROUP II AND III ISOLATION AS LEVEL DECREASED BELOW 0 INCHES. TWO MAIN STEAM RELIEF VALVES (MSRV) LIFTED ONCE AUTOMATICALLY, FOLLOWED BY MANUAL OPERATOR CYCLING OF MSRVS TO CONTROL RPV PRESSURE BETWEEN 930 PSIG AND 1060 PSIG. THE HIGH PRESSURE COOLANT INJECTION (HPCI) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEMS WERE PLACED IN OPERATION TO CONTROL RPV PRESSURE AND LEVEL. THE ROOT CAUSE OF THIS EVENT APPEARS TO BE A MALFUNCTION OF THE ELECTRONIC PORTION OF THE "A" RPV PRESSURE REGULATOR SET. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. THE MAJORITY OF THE "A" REGULATORY ELECTRONIC COMPONENTS

WERE REPLACED. THE SUSPECT COMPONENT HAS BEEN RETURNED TO THE VENDOR FOR ANALYSIS.

[146] PEACH BOTTOM 2 DOCKET 50-277 LER 89-017
FAILURE TO ESTABLISH A CONTINUOUS FIRE WATCH AS REQUIRED BY TECH SPECS DUE TO INEFFECTIVE FIRE PROTECTION SYSTEM TRAINING.
EVENT DATE: 081589 REPORT DATE: 091489 NSSS: GE TYPE: BWR

(NSIC 215246) AT 1700 ON 8/15/89 WELDING ACTIVITIES REQUIRED SMOKE DETECTORS FOR THE UNIT 2 REACTOR RECIRCULATION MOTOR GENERATOR (MG) LUBE OIL (LO) ROOM TO BE BYPASSED. A ROVING HOURLY FIRE WATCH WAS ESTABLISHED TO COMPLY WITH THE TECH SPEC REQUIREMENT FOR INOPERABLE DETECTORS. A CONTINUOUS FIRE WATCH WAS POSTED FOR WELDING BEING PERFORMED IN THE MG SET LO ROOM AS REQUIRED BY IGNITION AND SOURCE CONTROL ADMINISTRATIVE PROCEDURE, A-12. THE CONTINUOUS FIRE WATCH TOOK TWO SHORT BREAKS (1900-1935 AND 2015-2035) WHEN WORK WAS TEMPORARILY SUSPENDED AND NOT REQUIRED BY A-12. AT 2035 THE FIRE PROTECTION ASSISTANT RECOGNIZED THAT BYPASSING THE SMOKE DETECTORS ALSO RENDERED AUTOMATIC ACTUATION CAPABILITY OF THE RECIRC MG SET LO ROOM SPRINKLER SYSTEM INOPERABLE. SHIFT MANAGEMENT WAS NOTIFIED AND A CONTINUOUS FIRE WATCH WAS ESTABLISHED AS REQUIRED BY TECH SPEC 3.14.E.2.A FOR THE INOPERABLE SPRINKLER SYSTEM. THE CAUSE OF THIS EVENT WAS THE TRAINING OF OPERATIONS PERSONNEL IN THE FUNCTIONAL RELATIONSHIP OF THE SMOKE DETECTORS AND ASSOCIATED PRE-ACTION SPRINKLER SYSTEMS WAS NOT EFFECTIVE. THIS EVENT WILL BE REVIEWED BY LICENSED PERSONNEL. THE FUNCTIONAL RELATIONSHIP OF DETECTORS AND PRE-ACTION SPRINKLER SYSTEMS WILL BE EMPHASIZED DURING LICENSED OPERATOR SYSTEM TRAINING. ONE PREVIOUS SIMILAR LER HAS BEEN IDENTIFIED.

[147] PEACH BOTTOM 2 DOCKET 50-277 LER 89-018
FAILURE TO TRIP THE ROD BLOCK LOGIC OF THE REACTOR MANUAL CONTROL SYSTEM DUE TO INADEQUATE COMMUNICATIONS.
EVENT DATE: 081889 REPORT DATE: 091889 NSSS: GE TYPE: BWR

(NSIC 215303) AT 1300 ON 8/18/89, IT WAS DETERMINED THAT FLOW VARIABLE AVERAGE POWER RANGE MONITOR (APRM) ROD BLOCK SETTINGS WERE NON-CONSERVATIVE DUE TO REACTOR RECIRCULATION DRIVE FLOW INDICATING HIGHER THAN ACTUAL. APRM GAIN ADJUSTMENTS CORRECTING THE ROD BLOCK SETTINGS WERE COMPLETED AT 1550. PERSONNEL FAILED TO TAKE THOSE ACTIONS REQUIRED BY TECH SPECS (TS) BETWEEN THE TIME OF IDENTIFYING THE SUSPECT INDICATORS AND THE TIME OF CORRECTING THE PROBLEM, APPROXIMATELY 2 HOURS AND 50 MINUTES. SUBSEQUENT REVIEW OF THE SETPOINT CALCULATIONS REVEALED THE FLOW VARIABLE APRM SCRAM SETTINGS HAD ALSO BEEN NON-CONSERVATIVE. HOWEVER, THESE HAD ALSO BEEN CORRECTED BY THE COMMON APRM GAIN ADJUSTMENT THAT WAS MADE FOR THE ROD BLOCK SETTINGS. THE ROOT CAUSE OF THE TS VIOLATION WAS FAILURE OF PERSONNEL TO COMMUNICATE KNOWLEDGE OF THE NON-CONSERVATIVE SETTINGS TO SHIFT SUPERVISION WHEN THEY BECAME SUSPECT. NO SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. THE FLOW VARIABLE APRM ROD BLOCK AND SCRAM TRIPS REMAINED FUNCTIONAL, BUT WITH REDUCED MARGIN. THE APRM SYSTEM APPLIES A CLAMP TO PREVENT THE SCRAM SETPOINT FROM EXCEEDING 120% REGARDLESS OF INDICATED FLOW. IF CONTROL ROD WITHDRAWAL HAD OCCURRED, THE POSSIBILITY OF AN APRM SCRAM TRIP EXISTED UNDER SPECIFIC TRANSIENT CONDITIONS. REACTOR RECIRCULATION DRIVEN FLOW INSTRUMENTATION AND APRM FLOW COMPARATOR CIRCUITRY WAS CALIBRATED.

[148] PEACH BOTTOM 2 DOCKET 50-277 LER 89-019
FIRE PROTECTION SURVEILLANCE TEST (ST) DID NOT MEET THE REQUIREMENTS OF TECH SPEC DUE TO A DEFICIENCY IN THE DEVELOPMENT OF THE ST.
EVENT DATE: 083089 REPORT DATE: 092989 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 215347) ON 8/30/89 WHILE REVIEWING SURVEILLANCE TEST (ST) 7.8.16 "FIRE DOOR

SUPERVISION SYSTEM FUNCTIONAL TEST," IT WAS DISCOVERED THAT FIVE FIRE DOORS REQUIRING MONTHLY FUNCTIONAL TESTS, IN ACCORDANCE WITH TECH SPEC 4.14.D.2.A, WERE NOT INCLUDED ON THE ST AND THEREFORE NOT FUNCTIONAL TESTED. THE ROOT CAUSE OF THIS EVENT WAS A DEFICIENCY IN THE METHODOLOGY AND REFERENCES USED DURING THE GENERATION OF THE FIRE PROTECTION ST. A TEMPORARY CHANGE TO ST 7.8.16 WAS COMPLETED AND THE SUPERVISION SYSTEM OPERABILITY FUNCTIONAL CHECKS WERE PERFORMED ON THE ADDITIONAL FIVE FIRE DOORS. THEY WERE COMPLETED SATISFACTORILY. ST 7.8.16 HAS BEEN REVISED TO INCLUDE THESE FIVE FIRE DOORS. NO SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. THERE WAS ONE PREVIOUS SIMILAR EVENT, LER 2-88-24.

[149] PEACH BOTTOM 3 DOCKET 50-278 LER 89-002
UNIT 3 REACTOR VESSEL LOLOLO LEVEL SIGNAL DURING TRANSMITTER ISOLATION RESULTS IN UNIT 2 RHR PUMP TRIP SIGNAL DUE TO A COMBINATION OF PERSONNEL ERROR AND DEGRADED EQUIPMENT.
EVENT DATE: 082989 REPORT DATE: 092889 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 2 (BWR)

(NSIC 215348) AT 1719 ON 8/29/89 WITH UNIT 3 IN THE REFUEL MODE WITH THE CORE OFFLOADED A REACTOR VESSEL LOLOLO LEVEL SIGNAL WAS GENERATED. THE SIGNAL WAS GENERATED INADVERTENTLY DURING REMOVAL OF AN "A" CHANNEL REACTOR LEVEL TRANSMITTER FROM SERVICE. THE "B" CHANNEL WAS PREVIOUSLY TRIPPED DUE TO MODIFICATION WORK. UNIT 3 REACTOR LOLOLO LEVEL ENGINEERED SAFETY FEATURE SYSTEM ACTUATIONS HAD BEEN BLOCKED PRIOR TO THE EVENT AND DID NOT ACTUATE. THE UNIT 2 RESIDUAL HEAT REMOVAL (RHR) PUMPS RECEIVED A TRIP SIGNAL DUE TO ENERGIZATION OF THE UNIT 3/UNIT 2 RHR PUMP TRIP INTERLOCK. HOWEVER, THE UNIT 2 RHR PUMPS WERE NOT RUNNING AT THE TIME. THE UNIT 3 "D" HIGH PRESSURE SERVICE WATER PUMP AND THE "A", "B" AND "E" COOLING TOWERS TRIPPED AS A RESULT OF THE SIGNAL. THE CAUSE OF THIS EVENT WAS A COMBINATION OF PERSONNEL ERROR AND DEGRADED EQUIPMENT CONDITION. THE REACTOR LOLOLO LEVEL SIGNAL OCCURRED AS A RESULT OF A PRESSURE SPIKE, GENERATED BY CLOSURE OF THE TRANSMITTER EQUALIZING VALVE, WHICH AFFECTED THE TRANSMITTER DUE TO AN INCOMPLETELY SEATED HI SIDE ISOLATION VALVE. REACTOR LEVEL DID NOT CHANGE AND NO SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. THE SIGNAL WAS CLEARED, THE PROBLEM ISOLATION VALVE WAS REPLACED AND THE TECHNICIAN WAS COUNSELED ON THIS ERROR. NO PREVIOUS SIMILAR LERS WERE IDENTIFIED.

[150] PERRY 1 DOCKET 50-440 LER 89-006 REV 01
UPDATE ON TYPE C LEAKAGE RATES EXCEED TECH SPEC LIMITS AND PROGRAM DEFICIENCY RESULTS IN NOT PERFORMING REQUIRED TYPE B TESTING.
EVENT DATE: 022489 REPORT DATE: 092289 NSSS: GE TYPE: BWR
VENDOR: ATWOOD & MORRILL CO., INC.
BORG-WARNER CORP.
ROCKWELL MANUFACTURING COMPANY

(NSIC 215324) DURING THE FIRST REFUELING OUTAGE (2/25-8/5/89), TYPE B AND C LEAKAGE RATES WERE IDENTIFIED TO EXCEED TECH SPEC (TS) ALLOWABLE VALUES. PENETRATIONS INVOLVED INCLUDE MAIN STEAM LINES (MSL) AND ASSOCIATED INBOARD MAIN STEAM ISOLATION VALVE (MSIV) UPSTREAM DRAIN LINE. ALSO, TS FOR PRIMARY CONTAINMENT INTEGRITY WAS VIOLATED DURING CYCLE 1 DUE TO FAILURE TO PERFORM REQUIRED SURVEILLANCE TESTING INVOLVING TWO RESIDUAL HEAT REMOVAL (RHR) SYSTEM FLANGES. MANY FACTORS CONTRIBUTED TO DEGRADED MATERIAL CONDITION OF THE MSIVS AND OTHER ISOLATION VALVES, RESULTING IN HIGH PENETRATION LEAKAGE AFTER CORRECTIVE MAINTENANCE, THE RESULTING LEAKAGE RATES OF THE TYPE C TESTING WERE WITHIN THE LIMITS DEFINED IN TECH SPEC. THE FAILURE TO PERFORM THE TYPE B TESTING OF THE RHR FLANGES IS A RESULT OF A DEFICIENCY IN THE LOCAL LEAK RATE TESTING (LLRT) PROGRAM; HOWEVER, NO OTHER SIMILAR DEFICIENCIES HAVE BEEN IDENTIFIED. THE LEAKAGE RATES OF THE FLANGES WERE DEMONSTRATED AS ACCEPTABLE DURING THE LASTEST TYPE A TEST. IN ORDER TO PREVENT RECURRENCE AND MINIMIZE MAIN STEAM LINE PENETRATION LEAKAGE, POTENTIAL DESIGN AND TESTING IMPROVEMENTS ARE

UNDER EVALUATION TO IMPROVE MSIV PERFORMANCE. POTENTIAL DESIGN CHANGES WILL BE EVALUATED TO ENHANCE THE ABILITY TO PERFORM TYPE B TESTING OF THE RHR FLANGES AND THE LLRT PROGRAM WILL BE REVISED TO INCLUDE THESE FLANGES.

[151] PERRY 1 DOCKET 50-440 LER 89-010 REV 01
UPDATE ON INSUFFICIENT REFUELING BELLOWS INSULATION RESULTS IN HEAT DAMAGE TO SAFETY RELATED CABLES AND SNUBBERS IN THE UPPER DRYWELL.
EVENT DATE: 032889 REPORT DATE: 092989 NSSS: GE TYPE: BWR

(NSIC 215387) ON 3/28/89 AN ELEVATED TEMPERATURE CONDITION WAS CONFIRMED TO HAVE EXISTED WITHIN THE UPPER REGION OF THE DRYWELL. SAFETY-RELATED CABLES SUPPLYING LOGIC TRAIN B OF THE AUTOMATIC DEPRESSURIZATION SYSTEM AND TWO HYDROGEN IGNITERS WERE VISIBLY DAMAGED AND PRESUMED TO MAKE THE ASSOCIATED EQUIPMENT INOPERABLE. AS THE REMAINING QUALIFIED LIFE OF THE CABLE WAS INDETERMINATE, THIS REPORT WAS PROVIDED FOR THE POTENTIAL SIGNIFICANCE OF A COMMON CAUSE FAILURE; HOWEVER, NO ACTUAL FAILURE OF SAFETY RELATED EQUIPMENT WAS IDENTIFIED. ALL AFFECTED SAFETY-RELATED AND VISIBLY DAMAGED NON-SAFETY RELATED CABLES WERE REPLACED. ALL SNUBBERS IN THE UPPER DRYWELL WERE STROKE TESTED AND TWO WERE REPLACED. THIS CONDITION WAS THE RESULT OF INSUFFICIENT INSULATION SURROUNDING THE REFUEL BELLOWS (RB) CREATING A RADIANT HEAT SOURCE ABOVE THE BIOSHIELD. CONTRIBUTING FACTORS ARE INSUFFICIENT MAIN STEAM PIPING INSULATION, ELEVATED AIR TEMPERATURES EXITING THE BIOSHIELD ANNULUS REGION AND INADEQUATE MIXING OF COOLER DRYWELL AIR WITH THE AIR UNDER THE BULKHEAD PLATE IN THE VICINITY OF THE RB. TO PREVENT RECURRENCE, REFLECTIVE INSULATION WAS INSTALLED ON THE DRYWELL SIDE OF THE RB AND ON MAIN STEAM PIPING TO PROVIDE SHIELDING OF THE AFFECTED ZONE. ADDITIONALLY, AIR FLOW THROUGH THE ANNULUS WAS IMPROVED AND A TEMPORARY TEMPERATURE MONITORING PROGRAM WAS DEVELOPED.

[152] PERRY 1 DOCKET 50-440 LER 89-022
DEFICIENT SURVEILLANCE INSTRUCTION RESULTS IN SCRAM DISCHARGE VOLUME HIGH LEVEL INSTRUMENT OUTSIDE TECH SPEC ALLOWABLE VALUES.
EVENT DATE: 062789 REPORT DATE: 072189 NSSS: GE TYPE: BWR

(NSIC 215401) ON 6/27/89, IT WAS DISCOVERED THAT A LEVEL TRANSMITTER FOR THE SCRAM DISCHARGE VOLUME (SDV), HAD BEEN BEYOND THE TECH SPEC ALLOWABLE VALUE SINCE ITS PREVIOUS CALIBRATION ON 11/20/87. THIS IS A TECH SPEC VIOLATION OF SECTION 3.3.1.A. SINCE THIS DEFICIENCY WAS NOT IDENTIFIED UNTIL THE NEXT SCHEDULED SURVEILLANCE, NO TECH SPEC ACTIONS WERE COMPLETED. THE CAUSE OF THE EVENT WAS A PROCEDURAL DEFICIENCY. DURING THE INITIAL PREPARATION OF SURVEILLANCE INSTRUCTIONS (SVI-C110T0045A THRU D), "SDV WATER. LEVEL HIGH CHANNEL CALIBRATION", INSTRUMENT CALIBRATION DATA WAS DEVELOPED UTILIZING INACCURATE CALIBRATION DATA SHEETS WHICH SUPPLIED ELEVATION DATA BASED ON THE DIFFERENCE BETWEEN THE SDV TAP AND THE INSTRUMENT HIGH PRESSURE TAP AS TAKEN FROM THE INSTRUMENT ISOMETRIC DRAWINGS. THIS WAS DETECTED BY OBSERVATION OF THE FOUR LEVEL TRANSMITTERS IN THE SDV TO BE READING SLIGHTLY DIFFERENT VALUES. THIS INACCURACY WAS CONFIRMED WHEN ACTUAL ELEVATION MEASUREMENTS WERE TAKEN FOR EACH INSTRUMENT. CORRECTIVE ACTIONS INCLUDED REVISING SVI-C11-T0045A THRU D AND THE INSTRUMENT CALIBRATION DATA SHEET TO INCLUDE THE ACTUAL ELEVATIONS. EACH SDV LEVEL TRANSMITTER WAS RE-CABLIRATED UTILIZING THE REVISED SVI'S. ALL OTHER RPS INSTRUMENTATION HAS BEEN REVIEWED FOR POTENTIAL HEAD CORRECTION FACTOR (HCF) ERRORS. NO ERRORS WERE IDENTIFIED.

[153] PERRY 1 DOCKET 50-440 LER 89-023
ENTRY INTO OPERATIONAL CONDITION 2 WITH AN INOPERABLE CONTROL ROD RESULTS IN TECH SPEC VIOLATION DUE TO PROCEDURE DEFICIENCY AND OPERATOR ERROR.
EVENT DATE: 072389 REPORT DATE: 082189 NSSS: GE TYPE: BWR

(NSIC 215078) ON 7/23/89 AT 0413 ENTRY INTO OPERATIONAL CONDITION 2, STARTUP, WAS

COMPLETED WITH A CONTROL ROD INOPERABLE DUE TO BEING UNTRIPPABLE AS A RESULT OF AN IMPROPER VALVE LINEUP. THIS IS A VIOLATION OF TECH SPECS 3.1.3.1, 3.1.3.3 AND 3.0.4. THE CONTROL ROD WAS RETURNED TO AN OPERABLE STATUS, AND THE VALVE LINEUPS OF THE REMAINING CONTROL RODS WERE VERIFIED TO BE CORRECT PRIOR TO CONTINUING REACTOR STARTUP. THE CAUSES OF THE EVENTWERE PROCEDURAL DEFICIENCY AND PERSONNEL ERROR. PLANT ADMINISTRATIVE PROCEDURE (PAP-0205), "OPERABILITY OF PLANT SYSTEM," DOES NOT REQUIRE INDEPENDENT VERIFICATION DOCUMENTATION TO BE COMPLETED AT THE SITE OF PERFORMANCE. NON-LICENSED PLANT OPERATORS PERFORMED A NITROGEN CHARGE OF A CONTROL ROD SCRAM ACCUMULATOR ON 7/19/89, APPARENTLY, WITHOUT COMPLETELY RESTORING THE HYDRAULIC CONTROL UNIT VALVE LINEUP. BELIEVING THE EVOLUTION TO HAVE BEEN COMPLETED PROPERLY, THE OPERATORS COMPLETED THE VERIFICATION CHECKLISTS AFTER LEAVING THE CONTAINMENT. THIS WAS IN AN EFFORT TO PREVENT UNNECESSARY CONTAMINATION OF VERIFICATION DOCUMENTS. IN ORDER TO PREVENT RECURRENCE, PAP-0205 WILL BE MODIFIED TO REQUIRE THAT DOCUMENTATION OF INDEPENDENT VERIFICATION BE COMPLETED AT THE SITE OF PERFORMANCE. ADDITIONALLY, THE OPERATORS INVOLVED IN THIS EVENT HAVE BEEN REMINDED OF THEIR RESPONSIBILITIES FOR INDEPENDENT VERIFICATION.

[154] PERRY 1 DOCKET 50-440 LER 89-024
PERSONNEL ERROR DURING VALVE LINE-UP AND INSTRUMENTATION DEFICIENCIES CAUSE TECH SPEC VIOLATION OF SUPPRESSION POOL MAKE-UP SYSTEM.
EVENT DATE: 072589 REPORT DATE: 082489 NSSS: GE TYPE: BWR

(NSIC 215150) ON JULY 23, 1989 AT 0413, ENTRY INTO OPERATIONAL CONDITION 2 WAS COMPLETED WITH THE SUPPRESSION POOL MAKEUP (SPMU) SYSTEM INOPERABLE, IN VIOLATION OF TECHNICAL SPECIFICATION (TS) 3.0.4. ON JULY 25, 1989 A VENT VALVE ON THE REFERENCE LEG OF A SP LEVEL INSTRUMENT WAS FOUND OPEN AND UNCAPPED. ON AUGUST 2, 1989 THE UPPER CONTAINMENT POOL (UCP) WAS FOUND TO BE BELOW THE WATER LEVEL ALLOWABLE LIMIT. THE INSTRUMENTS AND UCP LEVEL WERE RESTORED TO AN OPERABLE CONDITION. THE CAUSES OF THESE EVENTS ARE PERSONNEL ERROR AND INSTRUMENT APPLICATION DEFICIENCIES. DURING SPMU SYSTEM INSTRUMENT FILL AND VENT ON JULY 18, 1989 TECHNICIANS APPARENTLY FAILED TO PROPERLY RESTORE THE SYSTEM. ADDITIONALLY, FOLLOWING COMPLETION OF OUTAGE ACTIVITIES, THE UCP SKIMMER PLATES WERE NOT RETURNED TO THEIR CORRECT POSITION. THE UCP LEVEL INSTRUMENTS HAD BEEN REPLACED DURING THE REFUEL OUTAGE WITH A MORE RELIABLE DESIGN BUT TROUBLESHOOTING INDICATES THAT A PROBLEM EXISTS WITH THE NEW CONFIGURATION. TO PREVENT RECURRENCE, THE TECHNICIANS INVOLVED WITH THE FILL AND VENT ACTIVITY HAVE BEEN COUNSELED, WHILE ALL OTHER INSTRUMENT AND CONTROL FIELD PERSONNEL HAVE BEEN INSTRUCTED ON THE LESSONS LEARNED FROM THIS EVENT. ALSO, THE ASSOCIATED INSTRUMENT MAINTENANCE INSTRUCTION WILL BE REVISED TO INCLUDE STEP-BY-STEP SIGNOFFS FOR SYSTEM VERIFICATION.

[155] PERRY 1 DOCKET 50-440 LER 89-025
DESIGN LIMITATIONS AND OPERATIONAL CONSTRAINTS RESULTS IN INDICATED HIGH DIFFERENTIAL FLOW AND REACTOR WATER CLEANUP CONTAINMENT ISOLATIONS.
EVENT DATE: 072689 REPORT DATE: 082489 NSSS: GE TYPE: BWR

(NSIC 215222) ON 7/26/89 AT APPROX. 0040 AND ON 7/31 AT APPROX. 1132, REACTOR WATER CLEANUP (RWCU) SYSTEM CONTAINMENT ISOLATIONS OCCURRED DUE TO HIGH INDICATED DIFFERENTIAL FLOW DURING PLANNED, MANUAL SHUTDOWNS OF THE PLANT. IN RESPONSE TO THE ISOLATIONS, PLANT OPERATORS VERIFIED NO ACTUAL SYSTEM LEAKAGE EXISTED AND RETURNED THE RWCU SYSTEM TO SERVICE AT APPROXIMATELY 0110 ON JULY 26 AND AT 1200 ON JULY 31. THE CAUSE OF THE FIRST EVENT IS BELIEVED TO BE A PREVIOUSLY IDENTIFIED DESIGN LIMITATION OF THE INSTRUMENTATION AND OPERATIONAL CONSTRAINTS ASSOCIATED WITH THE REDUCED FEEDWATER TEMPERATURE MODE OF OPERATION. IN THE SECOND EVENT, THE CAUSE OF THE ISOLATION IS UNKNOWN, ALTHOUGH OPERATION IN THE REDUCED FEEDWATER TEMPERATURE MODE IS CONSIDERED TO BE A CONTRIBUTING FACTOR. FILLING AND VENTING OF THE INSTRUMENTATION WAS NECESSARY, HOWEVER, THIS IS NOT BELIEVED TO BE THE PRIMARY CAUSE OF THE EVENT. CORRECTIVE ACTIONS TAKEN WERE TO

ENSURE A FILL AND VENT OF THE INSTRUMENT LINES. AS A RESULT OF PREVIOUS SIMILAR EVENTS, AN INCREASE OF THE DIFFERENTIAL FLOW TRIP SETPOINT AND/OR TIME DELAY HAVE BEEN UNDER EVALUATION TO ALLOW ADDITIONAL OPERATING MARGIN FOR THE INDICATED RWCU DIFFERENTIAL FLOW. OTHER CORRECTIVE ACTIONS PREVIOUSLY COMPLETED AS A RESULT OF PRIOR EVENTS ARE DESCRIBED IN THEIR RESPECTIVE LERS.

[156] PILGRIM 1 DOCKET 50-293 LER 80-049 REV 01
UPDATE ON THERMAL OVERLOAD RELAY TRIPS RESULTING IN LOSS OF SALT SERVICE WATER PUMP.
EVENT DATE: 081080 REPORT DATE: 080489 NSSS: GE TYPE: BWR
VENDOR: GOULDS PUMPS INC.

(NSIC 215090) ON 8/10-11/80, THE THERMAL OVERLOAD RELAY ASSOCIATED WITH BREAKER B-1544 TRIPPED RESULTING IN A LOSS OF SALT SERVICE WATER (SSW) PUMP P208B. ON 8/11-17/80, THE THERMAL OVERLOAD RELAY ASSOCIATED WITH BREAKER B-1444 TRIPPED RESULTING IN A LOSS OF SSW PUMP P208E. ON EACH OCCASION, THE PUMPS WERE SUCCESSFULLY TESTED AND RETURNED TO SERVICE. THE ABILITY OF THE SSW SYSTEM TO PERFORM ITS DESIGN FUNCTION DURING ACCIDENT CONDITIONS WAS NOT DIMINISHED. THE CAUSE OF THESE TRIPS WAS ACTUATION OF THERMAL OVERLOAD DEVICES RESULTING FROM THE SSW PUMPS BEING OPERATED AT THE RUN-OUT CONDITION. LONG TERM CORRECTIVE ACTIONS TAKEN INCLUDED IMPROVED BIOFOULING CONTROL AND AN INCREASE IN THE OVERLOAD HEATER SIZE.

[157] PILGRIM 1 DOCKET 50-293 LER 82-006 REV 01
UPDATE ON CAP SCREW FAILURES DUE TO OVERLOAD CONDITIONS RESULTING FROM ERRATIC VALVE OPERATION.
EVENT DATE: 030982 REPORT DATE: 090689 NSSS: GE TYPE: BWR
VENDOR: SCHUTTE AND KOERING COMPANY

(NSIC 215199) ON 3/9/82, WHILE PERFORMING MAINTENANCE RELATED TO GENERAL ELECTRIC (GE) SERVICE INFORMATION LETTER (SIL) NO. 352 "HPCI TURBINE STOP VALVE STEAM BALANCE CHAMBER PRESSURE ADJUSTMENT", THREE (3) OF THE FOUR (4) CAP SCREWS THAT ATTACH THE MAIN DISK FLANGE TO THE MAIN DISK WERE FOUND TO BE BROKEN. SEPARATION OF THE FLANGE FROM THE MAIN DISK WOULD RESULT IN THE MAIN DISK REMAINING CLOSED, PREVENTING INITIATION OF THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM. THE CAP SCREW FAILURES WERE A RESULT OF OVERLOAD CONDITIONS RESULTING FROM ERRATIC VALVE OPERATION DUE TO THE TURBINE STOP VALVE STEAM BALANCE CHAMBER BEING OUT OF ADJUSTMENT. TO PRECLUDE RECURRENCE, A PROCEDURE WAS DEVELOPED FOR ADJUSTING THE TURBINE STOP VALVE STEAM BALANCE CHAMBER. THE BALANCE CHAMBER PRESSURE WAS ADJUSTED IN ACCORDANCE WITH GE SIL NO. 352.

[158] PILGRIM 1 DOCKET 50-293 LER 82-056 REV 01
UPDATE ON STEAM TUNNEL PRESSURE RELIEVING PARTITION INOPERABLE DUE TO INADEQUATE DESIGN DOCUMENTATION.
EVENT DATE: 121382 REPORT DATE: 092289 NSSS: GE TYPE: BWR

(NSIC 215223) ON 12/13/82, A REPORT WAS RECEIVED AT PILGRIM STATION THAT DETERMINED THE PRESSURE RELIEVING PARTITION LOCATED ON THE SOUTH WALL OF THE STEAM TUNNEL HAD A TWO HOUR FIRE RESISTIVE RATING. THE WALL HAD BEEN DESIGNATED AS A THREE HOUR RATED FIRE BARRIER. IN AUGUST, 1989, A REVIEW DETERMINED THAT THE INSTALLED CONFIGURATION OF THE PRESSURE RELIEVING PARTITION HAS A THREE HOUR FIRE RESISTIVE RATING, AS REQUIRED. THE CONDITION OF THE WALL HAD NO POTENTIAL TO ADVERSELY IMPACT THE PUBLIC HEALTH AND SAFETY. THE SPECIFICATION, TO WHICH THE STEAM TUNNEL PRESSURE RELIEVING PARTITION HAD BEEN DESIGNED, WAS NOT CLEARLY IDENTIFIED IN DESIGN DOCUMENTATION. A CONTINUOUS FIRE PATROL FOR THE STEAM TUNNEL WAS INITIALLY ESTABLISHED. THE FIRE PATROL FREQUENCE WAS CHANGED TO ONCE PER HOUR, THEN REMOVED, WHEN THE PARTITION WAS DETERMINED TO BE ADEQUATE TO PREVENT FIRE DAMAGE TO REDUNDANT TRAINS.

[159] PILGRIM 1 DOCKET 50-293 LER 83-061 REV 01
UPDATE ON LACK OF POSITION ANNUNCIATION DUE TO AN OUT OF ADJUSTMENT LIMIT SWITCH.
EVENT DATE: 120383 REPORT DATE: 092089 NSSS: GE TYPE: BWR

(NSIC 215224) ON 12/3/83, DURING STEADY STATE OPERATION, AND WHILE PERFORMING THE DRYWELL TO TORUS VACUUM BREAKER OPERABILITY TEST, THE POSITION INDICATION ON PANEL C-7 FAILED TO ANNUNCIATE FOR VACUUM BREAKER 5045H. THE REDUNDANT POSITION ALARM ON PANEL 905 WAS VERIFIED TO BE OPERABLE. A DIFFERENTIAL PRESSURE DECAY RATE TEST WAS INITIATED IMMEDIATELY AND SUCCESSFULLY COMPLETED. THE LACK OF POSITION ANNUNCIATION ON PANEL C-7 HAD NO POTENTIAL TO ADVERSELY IMPACT THE PUBLIC HEALTH AND SAFETY. THE CAUSE WAS AN OUT OF ADJUSTMENT LIMIT SWITCH. THE LIMIT SWITCHES ON ALL TORUS VACUUM BREAKERS WERE ADJUSTED AND POST WORK TESTING WAS SUCCESSFULLY PERFORMED DURING REFUELING OUTAGE #6. A PREVIOUS SIMILAR OCCURRENCE WAS REPORTED IN LER 82-017/03L-0. NO SUBSEQUENT SIMILAR OCCURRENCES WERE IDENTIFIED.

[160] PILGRIM 1 DOCKET 50-293 LER 84-010 REV 01
UPDATE ON CONTROL ROD DRIVE COLLET RETAINER TUBE WELD DEFECTS.
EVENT DATE: 082284 REPORT DATE: 091889 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 215329) ON 8/22/84, IT WAS DETERMINED THAT A CONTROL ROD DRIVE COLLET RETAINER TUBE HAD A LONGITUDINAL, THROUGH-WALL CRACK. THE CRACK WAS LOCATED WITHIN ONE INCH (BELOW) OF THE LOWER SHOULDER ATTACHMENT WELD. THE CAUSE OF THE CRACK HAS BEEN DETERMINED TO BE INTER-GRANULAR STRESS CORROSION CRACKING (IGSCC) IN COLD WORKED AND WELD SENSITIZED STAINLESS STEEL. CORRECTIVE ACTION TAKEN INCLUDED REPLACEMENT AND METALLURGICAL EXAMINATION OF THE CRACKED MATERIAL. NO EVIDENCE OF CIRCUMFERENTIAL CRACKING OR PROPAGATION WAS FOUND. ALL INDICATIONS WERE PARALLEL TO THE LONGITUDINAL AXIS OF THE COLLET RETAINER TUBE. BASED ON THESE RESULTS, IT WAS DETERMINED THAT THE CRACKING COULD NOT LEAD TO MECHANICAL FAILURE OF THE CONTROL ROD DRIVE. THE CRACK WAS DISCOVERED DURING A REFUELING OUTAGE WHILE IN COLD SHUTDOWN. THE REACTOR MODE SWITCH WAS IN THE SHUTDOWN POSITION. THE REACTOR VESSEL (RV) WATER TEMPERATURE WAS APPROXIMATELY 100F. THE RV PRESSURE WAS ZERO PSIG. THE REACTOR POWER LEVEL WAS 0%. THE REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(II).

[161] PILGRIM 1 DOCKET 50-293 LER 85-008 REV 01
UPDATE ON HIGH PRESSURE COOLANT INJECTION SYSTEM INOPERABLE DUE TO A FAULTY CONNECTOR IN THE CONTROL SYSTEM.
EVENT DATE: 033185 REPORT DATE: 091889 NSSS: GE TYPE: BWR
VENDOR: AMPHENOL
BERGEN-PATTERSON PIPE SUPPORT CORPORATION
BLACK, SIVALLS & BRYSON, INC.
TERRY STEAM TURBINE COMPANY

(NSIC 215330) ON 3/31/85 WHILE PERFORMING A ROUTINE HIGH PRESSURE COOLANT INJECTION (HPCI) OPERABILITY FLOW TEST, THE HPCI TURBINE TRIPPED ON OVERSPEED. SUBSEQUENTLY, A BLOWN RUPTURE DISC, BROKEN SNUBBER AND DAMAGED BASEPLATE ANCHOR BOLTS WERE IDENTIFIED ON THE HPCI TURBINE EXHAUST LINE. THE CAUSE OF THE TURBINE TRIP WAS A FAULTY CONNECTOR IN THE HPCI TURBINE CONTROL SYSTEM. THE CAUSE OF THE BLOWN RUPTURE DISC, BROKEN SNUBBER AND DEGRADED ANCHOR BOLTS WAS WATER HAMMER IN THE HPCI TURBINE EXHAUST PIPING. CORRECTIVE ACTION WAS TO REPLACE THE CONNECTOR, REPLACE THE RUPTURE DISC AND REBUILD THE SNUBBER AND BASEPLATES. THE FAULTY CONNECTOR IS CONSIDERED TO BE AN ISOLATED CASE. TO PREVENT THE POTENTIAL FOR OVERSPEED AND SUBSEQUENT WATER HAMMER, DESIGN CHANGES WERE IMPLEMENTED TO ADD A VACUUM RELIEF LINE TO THE HPCI SYSTEM TURBINE EXHAUST PIPE AND TO INSTALL AN OIL BYPASS AROUND THE ELECTRIC GOVERNOR ACTUATOR ASSEMBLY (EG/R). THE EVENT OCCURRED DURING POWER OPERATION WITH THE REACTOR MODE SELECTOR SWITCH IN THE RUN POSITION. THE REACTOR VESSEL (RV) WATER TEMPERATURE WAS APPROXIMATELY 540F AND THE RV

PRESSURE WAS 1030 PSIG. THE REACTOR POWER LEVEL WAS 100%. THIS REPORT IS SUBMITTED PURSUANT TO 10 CFR 50.73(A)(2)(V) AND THIS EVENT POSED NO THREAT TO THE HEALTH AND SAFETY OF THE PUBLIC.

[162] PILGRIM 1 DOCKET 50-293 LER 86-017 REV 01
UPDATE ON LOCAL LEAK RATE TEST RESULTS OF APPENDIX 'J' RELATED VALVES IN EXCESS OF LIMITS.
EVENT DATE: 070186 REPORT DATE: 092989 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 215331) WHILE SHUT DOWN FOR PLANT REPAIRS, 10CFR50 APPENDIX 'J', LOCAL LEAK RATE TESTING OF THE 18" ANCHOR DARLING FEEDWATER CHECK VALVES (FWCVS) WAS CONDUCTED FROM 6/26/86 TO 7/1/86. THE "AS-FOUND" TEST RESULTS SHOWED LEAKAGE IN EXCESS OF THE LIMITS OF PILGRIM NUCLEAR POWER STATION TECH SPEC SECTION 4.7.A.2.A. A ROOT CAUSE ANALYSIS WAS CONDUCTED BY THE FAILURE ANALYSIS TEAM AND THE CAUSE WAS DETERMINED TO BE THERMAL AGING AND EROSIVE WEAR OF THE SOFT SEAT MATERIAL. CORRECTIVE ACTION INCLUDED REPLACING THE SOFT SEAT MATERIAL WITH A NEW MATERIAL MORE RESISTIVE TO THERMAL AGING AND WEAR. FAILURE AND MALFUNCTION REPORTS (F&MRS) WERE GENERATED TO DOCUMENT THE LOCAL LEAK RATE TEST (L.L.R.T.) FAILURES. MAINTENANCE REQUESTS (MRS) WERE ISSUED TO CORRECT THE PROBLEMS IN ACCORDANCE WITH PLANT DESIGN CHANGE 86-46. ON 12/23/87, THE PERIODIC INTEGRATED LEAKAGE RATE TEST WAS SUCCESSFULLY COMPLETED AND THE PILGRIM NUCLEAR POWER STATION PERIODIC TEST REPORT WAS ISSUED ON 3/15/88. THIS EVENT OCCURRED WHILE THE PLANT WAS IN COLD SHUTDOWN WITH THE REACTOR MODE SELECTOR SWITCH IN THE SHUTDOWN POSITION. THE CONTROL RODS WERE IN THE INSERTED POSITION. THE REACTOR WATER TEMPERATURE WAS APPROXIMATELY 110F WITH NEGLIGIBLE CORE DECAY HEAT. THE REACTOR VESSEL PRESSURE WAS ATMOSPHERIC AND THE REACTOR POWER LEVEL 0%. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10CFR 50.73(A)(2)(I).

[163] PILGRIM 1 DOCKET 50-293 LER 89-024
AUTOMATIC CLOSING OF THE OUTBOARD PRIMARY CONTAINMENT SYSTEM GROUP 6 ISOLATION VALVES.
EVENT DATE: 072689 REPORT DATE: 082589 NSSS: GE TYPE: BWR

(NSIC 215126) ON 7/26/89 AT 0123 HOURS, AN AUTOMATIC ACTUATION OF THE OUTBOARD REACTOR WATER CLEANUP (RWCU) SYSTEM PORTION OF THE PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM (PCIS) OCCURRED. THE ACTUATION RESULTED IN THE AUTOMATIC CLOSING OF THE OUTBOARD PRIMARY CONTAINMENT SYSTEM GROUP 6 (SIX)/RWCU SYSTEM ISOLATION VALVES AND A TEMPORARY INTERRUPTION IN RWCU SYSTEM OPERATION. THE PCIS LOGIC CIRCUITRY WAS RESET AND THE RWCU SYSTEM WAS RETURNED TO SERVICE ON 7/26/89 AT APPROXIMATELY 0200 HRS. THE DIRECT CAUSE FOR THE ACTUATION WAS A TRIP SIGNAL FROM THE OUTBOARD RWCU SYSTEM FLOW SENSOR. A MULTI-DISCIPLINARY INVESTIGATION TEAM WAS FORMED TO DETERMINE THE CAUSE FOR THE TRIP SIGNAL. THE INVESTIGATION INCLUDED REVIEW OF RWCU SYSTEM OPERATIONAL DATA, THE PROCEDURE USED FOR CALIBRATING THE INBOARD AND OUTBOARD FLOW SENSORS, AND DOCUMENTS RELATED TO THE INSTRUMENTATION LINES OF THE FLOW SENSORS. CORRECTIVE ACTION PLANNED INCLUDES TESTING TO PROVIDE ADDITIONAL DATA. A SUPPLEMENT TO THIS REPORT WILL BE SUBMITTED FOLLOWING THE COMPLETION OF THE ROOT CAUSE INVESTIGATION. THIS EVENT OCCURRED DURING A STARTUP WITH THE REACTOR MODE SELECTOR SWITCH IN THE STARTUP POSITION. THE CONTROL RODS WERE IN A PARTIALLY WITHDRAWN POSITION. THE REACTOR VESSEL PRESSURE WAS APPROX. 10 PSIG AND THE RV WATER TEMPERATURE WAS APPROXIMATELY 221F. THE REACTOR POWER LEVEL WAS APPROXIMATELY 1%.

[164] PILGRIM 1 DOCKET 50-293 LER 89-026
AUTOMATIC SCRAM RESULTING FROM A TURBINE RUNBACK DUE TO FAILURE OF POTENTIAL TRANSFORMER AND VOLTAGE BALANCE RELAY WIRING ERROR.
EVENT DATE: 083089 REPORT DATE: 092989 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

WESTINGHOUSE ELECTRIC CORP.

(NSIC 215354) ON 8/30/89 AT 1917 HOURS, AN AUTOMATIC REACTOR PROTECTION SYSTEM (RPS) SCRAM SIGNAL AND REACTOR SCRAM OCCURRED WHILE AT 65% REACTOR POWER. AS EXPECTED, THE SCRAM SIGNAL RESULTED IN AN AUTOMATIC SEQUENCE OF DESIGNER RESPONSES THAT INCLUDED A TURBINE-GENERATOR TRIP, AUTOMATIC OPENING OF TWO 345 KV SWITCHYARD AIR CIRCUIT BREAKERS, AND AN AUTOMATIC TRANSFER OF STATION ELECTRICAL LOADS. THE DIRECT CAUSE FOR THE SCRAM SIGNAL WAS HIGH REACTOR VESSEL/MAIN STEAM SYSTEM PRESSURE (1069 PSIG) THAT OCCURRED AS A RESULT OF AN AUTOMATIC TURBINE RUNBACK. THE TURBINE RUNBACK WAS CAUSED BY THE FAILURE OF A NONSAFETY-RELATED 24 KV POTENTIAL TRANSFORMER (PT) IN CONJUNCTION WITH A NONSAFETY-RELATED VOLTAGE BALANCE RELAY THAT WAS WIRED IN ACCORDANCE WITH AN ARCHITECT-ENGINEER (BECHTEL) DRAWING THAT CONTAINED AN ERROR SINCE ORIGINAL CONSTRUCTION (C.1971). THE PT (STYLE EED2553, A SERIAL NUMBER 70F3376) WAS MANUFACTURED BY THE WESTINGHOUSE ELECTRIC CORPORATION. THE PT WAS REPLACED, THE DRAWING ERROR WAS CORRECTED AND THE VOLTAGE BALANCE RELAY WAS REWIRED IN ACCORDANCE WITH THE REVISED DRAWING. THE PROCEDURE USED TO FUNCTIONALLY TEST THE VOLTAGE BALANCE RELAY WAS STRENGTHENED. APPLICABLE PORTIONS OF THE STATION ELECTRICAL SYSTEM WERE TESTED AND/OR EVALUATED FOR IMPACT OF OVERVOLTAGE WITH SATISFACTORY RESULTS. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV).

[165] POINT BEACH 2 DOCKET 50-301 LER 89-004
REACTOR TRIP RESULTING FROM TRANSFORMERS SUDDEN PRESSURE SIGNAL.
EVENT DATE: 082089 REPORT DATE: 091939 NSSS: WE TYPE: PWR
VENDOR: MCGRAW EDISON CO., POWER SYSTEMS DIV

(NSIC 215305) ON 8/20/89, AT 1631, UNIT 2 EXPERIENCED A MAIN STEP-UP TRANSFORMER LOCKOUT, MAIN GENERATOR BREAKER TRIP, AND CONCURRENT TURBINE AND REACTOR TRIPS. THE UNIT HAD BEEN OPERATING AT 100% POWER. THE LOCKOUT OCCURRED SHORTLY AFTER WATER SPRAY HAD BEEN INITIATED TO THE TRANSFORMER OIL COOLERS TO REDUCE TEMPERATURES ON THE X01 TRANSFORMERS. THE UNIT TRANSFORMER LOCKOUT WAS INITIATED BY THE APPARENTLY RANDOM ACTUATION OF THE 2X01 "B" PHASE "SUDDEN PRESSURE" TRIP SYSTEM. THE UNIT RESPONDED IN A NORMAL MANNER TO THE TRIP WITH SEVERAL MINOR PROBLEMS. AN UNUSUAL EVENT WAS DECLARED AT 1644 DUE TO LOSS OF ELECTRICAL LOAD. THE UNUSUAL EVENT WAS TERMINATED AT 1856. AFTER EXTENSIVE INVESTIGATION AND THOROUGH CHECKOUT AND TESTING OF THE 2X02 "B" PHASE TRANSFORMER, THE UNIT WAS RETURNED TO SERVICE ON 8/21/89.

[166] PRAIRIE ISLAND 1 DOCKET 50-282 LER 89-012
AUTOMATIC CONTROL ROOM ISOLATION AND START OF CONTROL ROOM CLEANUP FAN DUE TO FAILURE OF A CHLORINE GAS MONITOR.
EVENT DATE: 080489 REPORT DATE: 090589 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)
VENDOR: M D A SCIENTIFIC, INC.

(NSIC 215211) THE EVENT WAS THE AUTOMATIC ACTUATION OF 121 CONTROL ROOM CLEAN UP FAN AND ISOLATION OF THE OUTSIDE AIR TO THE CONTROL ROOM. THE CONTROL ROOM VENTILATION SYSTEM WAS STARTED BY 111 CONTROL ROOM CHLORINE MONITOR. THE EVENT ACTUALLY OCCURRED SEVERAL TIMES FROM 8/4 THROUGH 8/15/89. BECAUSE A SINGLE CAUSE HAS BEEN IDENTIFIED FOR ALL OF THE EVENTS, THE EVENTS ARE BEING ADDRESSED WITH A SINGLE REPORT. THE CAUSE OF THE OCCURRENCES WAS DETERMINED TO BE THE FALSE HIGH CHLORINE SIGNALS GENERATED BY THE CHLORINE MONITOR WHEN AN OPTICS BLOCK WAS NOT BEING HELD FIRMLY IN PLACE AND, AS IT MOVED SLIGHTLY, THE INTENSITY OF REFLECTED LIGHT WOULD CHANGE, THE PHOTO CELLS DETECTED THIS CHANGE AND A SPIKE WOULD BE GENERATED. VARIOUS FACTORS CONTRIBUTED TO THE MOTION OF THE OPTICS BLOCK. THE FACTORS CONTRIBUTING TO THE MOVEMENT OF THE OPTICS BLOCK HAVE BEEN CORRECTED AND IMPROVED MAINTENANCE TECHNIQUES WILL DECREASE THE PROBABILITY OF RECURRENCES. THE CONTROL ROOM VENTILATION SYSTEM ENTERED ITS SAFEGUARDS MODE OF OPERATION AS DESIGNED.

[167] PRAIRIE ISLAND 1 DOCKET 50-282 LER 89-013
DETECTION INSTRUMENTATION FOR FIRE DETECTION ZONE FOUND IN BYPASS.
EVENT DATE: 083089 REPORT DATE: 092989 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)

(NSIC 215352) AT 1748 ON 8/29/89, WITH BOTH UNITS AT 100% POWER, A FIRE DRILL WAS INITIATED FOR A FIRE ZONE WHICH ENCOMPASSES THE LOWER LEVEL OF THE SCREEN HOUSE. AT 1752, AS PART OF THE FIRE DRILL AND IN ACCORDANCE WITH THE APPROVED FIRE FIGHTING PROCEDURE, A CONTROL ROOM OPERATOR SOUNDED THE FIRE ALARM, PLACED THE AFFECTED FIRE ZONE ALARM IN BYPASS ON THE FIRE PROTECTION PANEL AND MADE THE PROPER NOTIFICATIONS. THE FIRE DRILL WAS TERMINATED AT 1811. HOWEVER, THE BYPASS SWITCH FOR THE AFFECTED FIRE ZONE ALARM WAS NOT RETURNED TO NORMAL FOLLOWING THE DRILL. THE ALARM REMAINED IN BYPASS AND AS A RESULT THE CONTROL ROOM WOULD NOT HAVE RECEIVED FIRE ALARMS FROM AFFECTED FIRE ZONE. THE CONTROL ROOM OPERATOR PERFORMING THE SHIFT LOGGING AT 2200 ALSO FAILED TO NOTE THAT FIRE ZONE REMAINED IN BYPASS. THE CONTROL ROOM OPERATOR PERFORMING THE SHIFT LOGGING AT 0650 ON 8/30/89 FOUND THE FIRE ZONE ALARM IN BYPASS AND RETURNED THE BYPASS SWITCH TO THE NORMAL POSITION. PROCEDURE CHANGES HAVE BEEN IMPLEMENTED AND FURTHER CHANGES ARE PLANNED TO ENSURE THAT FIRE ZONE BYPASS SWITCHES ARE RETURNED TO THE NORMAL POSITION FOLLOWING FIRE DRILLS AND TO HELP IDENTIFY FIRE DETECTION PANEL BYPASS SWITCHES THAT ARE IN THE BYPASS POSITION.

[168] QUAD CITIES 1 DOCKET 50-254 LER 89-001 REV 01
UPDATE ON RCIC VALVE MO-1301-48 FAILURE TO OPEN DUE TO BINDING OF THE TORQUE SWITCH OF THE MOTOR OPERATOR.
EVENT DATE: 010689 REPORT DATE: 092989 NSSS: GE TYPE: BWR
VENDOR: CRANE VALVE CO.
LIMITORQUE CORP.

(NSIC 215341) AT 1720 HOURS ON 1/6/89, UNIT 1 WAS AT 96% POWER. AT THIS TIME, THE REACOR CORE ISOLATION COOLING (RCIC) PUMP DISCHARGE INBOARD ISOLATION VALVE, 1-1301-48, FAILED TO OPEN FROM THE FULL CLOSED POSITION WHEN GIVEN AN OPEN SIGNAL FROM CONTROL ROOM PANEL 901-3. THIS WOULD HAVE PREVENTED A RCIC INJECTION AND THEREFORE MADE THE SYSTEM INOPERABLE. THE CAUSE OF THIS EVENT IS EQUIPMENT FAILURE. THE CAM ON THE TORQUE SWITCH ON THE VALVE OPERATOR WAS BINDING, WHICH CAUSED THE SWITCH CONTACTS TO OPEN. THE TORQUE SWITCH WAS REPLACED AND THE VALVE SUCCESSFULLY TESTED AND RETURNED TO SERVICE AT 0035 HOURS ON 1/7/89. THE TORQUE SWITCH WILL BE TESTED BY PRODUCTION SERVICES DEPARTMENT (PSD) TO DETERMINE THE ROOT CAUSE OF THE EQUIPMENT FAILURE. BWR ENGINEERING WILL INVESTIGATE A PART 21 NOTIFICATION ISSUED BY LIMITORQUE CORPORATION ON THIS PARTICULAR TYPE OF TORQUE SWITCH. THIS REPORT IS SUBMITTED TO COMPLY WITH 10CFR50.73(A)(2)(V)(B).

[169] QUAD CITIES 1 DOCKET 50-254 LER 89-002 REV 01
UPDATE ON TOTAL COMBINED LOCAL LEAK RATE TESTS INTERVAL EXCEEDED DUE TO MISINTERPRETATION OF TECH SPECS.
EVENT DATE: 010989 REPORT DATE: 092989 NSSS: GE TYPE: BWR

(NSIC 215342) ON 1/9/89, UNIT ONE WAS IN THE RUN MODE AT 99% POWER. AT 1500 HOURS, TECHNICAL STAFF PERSONNEL DISCOVERED THAT THE TOTAL SURVEILLANCE INTERVAL FOR THREE LOCAL LEAK RATE TESTS (LLRT) ON THE MAIN STEAM ISOLATION VALVE (MSIV) AO 1-203-1A BETWEEN 9/6/82, AND 9/12/87, HAD EXCEEDED THE TECH SPEC LIMIT OF 3.25 TIMES THE SINGLE TECH SPEC SURVEILLANCE INTERVAL OF 18 MONTHS. THIS SITUATION CAME ABOUT BECAUSE THE ORIGINAL TECH SPEC LIMIT OF 18 MONTHS WAS BASED ON 12-MONTH CYCLES, AND BECAUSE OF THE INTERPRETATION THAT THE INTERVALS COULD BE MEASURED FROM END OF OUTAGE TO BEGINNING OF OUTAGE. WHEN THE CYCLE LENGTHS WENT TO 18 MONTHS AND THE INTERVALS WERE MEASURED TO TEST TO TEST, THE TECH SPEC SHOULD HAVE BEEN REEVALUTATED. THE SAFETY CONSEQUENCES OF THIS EVENT WERE MINIMAL SINCE THE VALVE PASSED EACH LLRT. A TECH SPEC CHANGE HAS BEEN APPROVED

TO EXTEND THE SURVEILLANCE INTERVAL TO ONCE PER CYCLE, NOT TO EXCEED 24 MONTHS. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)92)(I)(B).

[170] QUAD CITIES 1 DOCKET 50-254 LER 88-010 REV 01
UPDATE ON REACTOR SCRAM FROM AN INDUCED VOLTAGE DUE TO A LOOSE WIRE ON THE CONDENSER LOW VACUUM PRESSURE SWITCH INDICATING LAMP.
EVENT DATE: 062989 REPORT DATE: 082989 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215207) ON 6/29/89, QUAD CITIES UNIT ONE WAS IN THE RUN MODE AT 94% OF RATED CORE THERMAL POWER. AT 2239 HOURS, A REACTOR SCRAM OCCURRED DUE TO TURBINE STOP VALVE CLOSURE. THE STOP VALVE CLOSURE WAS THE RESULT OF A TURBINE TRIP. ALL SAFETY FEATURE ACTUATIONS OCCURRED AS DESIGNED. EMERGENCY NOTIFICATION SYSTEM (ENS) NOTIFICATION WAS COMPLETED AT 2330 HOURS ON 6/29/89, TO COMPLY WITH THE REQUIREMENTS OF 10 CFR 50.72(B)(2)(II). AN INVESTIGATION REVEALED THE CAUSE FOR THIS EVENT WAS A LOOSE CONNECTION ON THE 1-5500-PS-105B CONDENSER LOW VACUUM PRESSURE SWITCH INDICATING LAMP. WHEN THE LENS COVER FOR THE LAMP WAS PUT ON, THE LOOSE WIRE INDUCED A VOLTAGE IN THE K2D18 RELAY AND ENERGIZED THE MASTER TRIP BUS. THIS RESULTED IN A TURBINE TRIP.

[171] QUAD CITIES 1 DOCKET 50-254 LER 89-012
CONTROL VALVE FAST-ACTING SOLENOIDS INOPERABLE DUE TO A GROUND ON THE NO. 2 CONTROL VALVE TEST LIMIT SWITCH.
EVENT DATE: 082589 REPORT DATE: 092289 NSSS: GE TYPE: BWR
VENDOR: NAMCO CONTROLS

(NSIC 215301) ON 8/25/89, AT 1045 HOURS, UNIT 1 WAS IN THE RUN MODE AT 77% OF RATED CORE THERMAL POWER. INSTRUMENT MAINTENANCE (IM) DISCOVERED THE SUPPLY WIRE (CBL) TO THE ELECTROHYDRAULIC CONTROL (EHC) SYSTEM (TG) VALVE (V) TEST CIRCUITRY TO BE BURNT THROUGH. A LOAD REDUCTION TO LESS THAN 45% RATED STEAM FLOW WAS INITIATED IMMEDIATELY SINCE IT WAS INITIALLY BELIEVED THAT A REQUIRED SCRAM FUNCTION WAS INOPERABLE. AT 1100 HOURS, IN ACCORDANCE WITH THE GENERATING STATION EMERGENCY PLAN (GSEP), AN UNUSUAL EVENT WAS DECLARED AND A NUCLEAR ACCIDENT REPORTING SYSTEM (NARS) PHONE NOTIFICATION WAS MADE. AT 1122 HOURS, A NRC EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE NOTIFICATION WAS MADE TO SATISFY 10CFR50.72(A)(1)(I). AT 1220 HOURS, WITH LOAD BELOW 45%, THE GSEP UNUSUAL EVENT WAS TERMINATED. AT 1845 HOURS A GROUND ON THE #2 CONTROL VALVE (FCV) TEST LIMIT SWITCH (33) WAS FOUND AND WAS DETERMINED TO HAVE CAUSED THE SUPPLY WIRE TO BURN THROUGH. THE SUPPLY WIRE WAS REPLACED. THE LIMIT SWITCH WILL BE REPAIRED. AT 2120 HOURS, A TURBINE CONTROL VALVE FAST-CLOSURE FUNCTIONAL TEST WAS SUCCESSFULLY COMPLETED. THE UNIT RESUMED NORMAL POWER OPERATIONS. THIS REPORT IS BEING PROVIDED AS A VOLUNTARY REPORT.

[172] QUAD CITIES 2 DOCKET 50-265 LER 88-007 REV 01
UPDATE ON LEAK RATE FROM ALL VALVES AND PENETRATIONS IN EXCESS OF SPECIFICATION LIMITS.
EVENT DATE: 042088 REPORT DATE: 091889 NSSS: GE TYPE: BWR
VENDOR: BLAW-KNOX COMPANY
CRANE COMPANY

(NSIC 215293) ON 4/10/88, QUAD CITIES UNIT 2 WAS SHUT DOWN FOR THE END OF CYCLE 9 REFUELING AND MAINTENANCE OUTAGE. ON 4/13/88 AT 1630 HOURS, IT WAS DETERMINED THAT THE MEASURED COMBINED LEAKAGE RATE FROM ALL PENETRATIONS AND VALVES, EXCLUDING THE MAIN STEAM ISOLATION VALVES, EXCEEDED THE TECH SPECS (3.7.A.2) LIMIT OF 293.75 STANDARD CUBIC FEET PER HOUR (SCFH) (0.60LA). THIS WAS IDENTIFIED WHILE LOCAL LEAK RATE TESTING THE 2-220-58B AND 2-220-62B FEEDWATER CHECK VALVES. THE FAILURE MODE OF THE PENETRATIONS AND VALVES WAS FOUND TO BE GENERALLY DUE TO NORMAL WEAR. REPAIRS AND REPLACEMENTS WERE COMPLETED AS

NECESSARY AND THE RESULTS OF THE LOCAL LEAK RATE TESTING PROGRAM ARE PROVIDED. THIS REPORT IS SUBMITTED TO COMPLY WITH THE REQUIREMENTS OF 10CFR50.73(A)(2)(II).

[173] QUAD CITIES 2 DOCKET 50-265 LER 89-004
INABILITY OF ATMOSPHERIC CONTAINMENT ATMOSPHERE DILUTION SYSTEM TO PERFORM WITH LOSS OF UNIT TWO DIESEL GENERATOR.
EVENT DATE: 081189 REPORT DATE: 091189 NSSS: GE TYPE: BWR

(NSIC 215240) ON 8/11/89 AT 1155 HOURS, UNIT TWO WAS IN THE RUN MODE AT 99% OF RATED CORE THERMAL POWER. AT THIS TIME, SARGENT AND LUNDY ENGINEERS WERE CONDUCTING A STUDY OF THE EFFECTS OF D.C. POWER (EI) FAILURES ON SYSTEMS SHARED BETWEEN UNITS AT DRESDEN STATION. A DESIGN ERROR FOR BOTH DRESDEN AND QUAD CITIES UNIT TWO ATMOSPHERIC CONTAINMENT ATMOSPHERE DILUTION (ACAD) (BB) SYSTEM WAS DISCOVERED. THE ERROR FOUND WAS THAT THE ACAD SYSTEM WAS NOT DESIGNED AND CONSTRUCTED TO MEET SINGLE FAILURE CRITERION. THE APPARENT CAUSE OF THIS EVENT IS DESIGN ERROR MADE DURING ORIGINAL INSTALLATION. FOR CORRECTIVE ACTIONS, THE STATION HAS IMPLEMENTED PROCEDURE CHANGES TO ALERT OPERATORS TO THE ACAD DESIGN DEFICIENCY. THE SYSTEM STATUS IS UNDER CONSIDERATION IN ONGOING DISCUSSIONS WITH THE NRC REGARDING COMBUSTIBLE GAS CONTROL. THERE HAVE BEEN NO PREVIOUS EVENTS REPORTED SIMILAR TO THIS EVENT. SINCE ACAD IS NOT PART OF THE STATION DESIGN BASIS, THIS EVENT IS NOT REPORTABLE UNDER THE REQUIREMENTS OF 10CFR50.73. HOWEVER, THIS REPORT IS BEING SUBMITTED AS A VOLUNTARY LER.

[174] RANCHO SECO DOCKET 50-312 LER 89-010
FAILURE TO INPUT THE REQUIRED DEFAULT FLOW RATE VALUE FOR AUX. BLDG. STACK RADIATION MONITOR DUE TO PERSONNEL ERROR.
EVENT DATE: 082189 REPORT DATE: 092089 NSSS: BW TYPE: PWR

(NSIC 215382) AT APPROX. 1012 HOURS ON 8/21/89, INSTRUMENTATION & CONTROL (I&C) TECHNICIANS BEGAN PERFORMING SURVEILLANCE PROCEDURE SP.452. THIS SURVEILLANCE PROCEDURE INVOLVES REMOVING EFFLUENT FLOW RATE MEASURING DEVICES FROM THEIR RESPECTIVE VENTS AND REPLACING THEM WITH FLOW RATE MEASURING DEVICES CALIBRATED BY THE MANUFACTURER. REPLACING THE FLOW RATE MEASURING DEVICE REQUIRES THE DEVICE BE TAKEN OUT OF SERVICE. WITH THE FLOW RATE DEVICE OUT OF SERVICE, THE MEASURED PROCESS FLOW RATE DATA (ANALOG SIGNAL TO THE RM-11 RADIATION MONITOR DISPLAY CONSOLE) IS NOT AVAILABLE. THE SURVEILLANCE PROCEDURE THEREFORE REQUIRES THAT THE I&C TECHNICIAN SUBSTITUTE A DEFAULT VALUE (MAXIMUM DESIGN FLOW RATE) INTO THE RADIATION MONITOR DATABASE. WHEN PERFORMING THE SECTION OF THE SURVEILLANCE PROCEDURE THAT TOOK FLOW ELEMENT FE-15045 OUT OF SERVICE, THE I&C TECHNICIANS NEGLECTED TO INPUT THE DEFAULT FLOW RATE VALUE. THIS RESULTED IN FLOW ELEMENT FE-15045 BEING OUT OF SERVICE WITHOUT THE DEFAULT VALUE INPUT TO AUX. BLDG. STACK RADIATION MONITOR R-15045. THE COGNIZANT I&C SUPERVISOR CONDUCTED TRAINING FOR I&C TECHNICIANS REGARDING THE PROPER PERFORMANCE OF SURVEILLANCE PROCEDURES AND THE CONDUCT OF MAINTENANCE ACTIVITIES. MOREOVER, MAINTENANCE WILL ENSURE THAT INDIVIDUALS RESPONSIBLE FOR PERFORMING SURVEILLANCE PROCEDURES AS TRAINED ON THE PROPER PERFORMANCE OF THOSE PROCEDURES.

[175] RIVERBEND 1 DOCKET 50-458 LER 89-010 REV 02
UPDATE ON FIRE BARRIERS DECLARED INOPERABLE DUE TO MISSING OR INADEQUATE PENETRATION SEALS.
EVENT DATE: 031889 REPORT DATE: 083089 NSSS: GE TYPE: BWR

(NSIC 215177) AT 1300 ON 3/18/89 WITH THE UNIT IN OPERATIONAL CONDITION 5, AN UNSEALED PENETRATION WAS DISCOVERED IN A CONTROL BUILDING FIRE WALL ON THE 116 FOOT ELEVATION. ALSO, ON THE SAME ELEVATION OF THE CONTROL BUILDING FOUR CONDUITS LACKING INTERNAL SEALS WERE DISCOVERED AT 1400 ON 3/20/89. THESE DEFICIENCIES RENDERED THE FIRE BARRIERS INOPERABLE IN ACCORDANCE WITH RIVER BEND STATION TECH SPEC 3.7.7.A. UPON DISCOVERY OF THE NONCONFORMING CONDITIONS, THE

EXISTING HOURLY FIRE WATCH PATROLS FOR THESE CONDITIONS WERE ADDED TO LCO 87-078. ADDITIONAL CORRECTIVE ACTION WILL BE PROVIDED IN A SUPPLEMENT TO THIS LER. INSPECTIONS PERFORMED BY OPERATIONS QC WERE PART OF THEIR ONGOING ACTIVITIES. THE AREAS WHERE THE UNSEALED PENETRATIONS WERE DISCOVERED INVOLVED THE SAME SHUTDOWN DIVISION. THEREFORE, WITH THE ALTERNATE DIVISION AVAILABLE, PLANT SAFETY WAS ASSURED.

[176] ROBINSON 2 DOCKET 50-261 LER 89-010
INADEQUATE AUXILIARY FEEDWATER PUMP NET POSITIVE SUCTION HEAD.
EVENT DATE: 081689 REPORT DATE: 091589 NSSS: WE TYPE: PWR
VENDOR: PACIFIC PUMPS

(NSIC 215239) BASED ON CALCULATIONS PERFORMED AS PART OF THE ON-GOING DESIGN BASIS RECONSTITUTION, IT WAS DETERMINED THAT ADEQUATE NET POSITIVE SUCTION HEAD (NPSH) FOR THE AUXILIARY FEEDWATER (AFW) PUMPS COULD NOT BE ASSURED FOR ALL POSSIBLE COMBINATIONS OF RUNNING AFW PUMPS AND CONDENSATE STORAGE TANK (CST LEVELS. TO ASSURE ADEQUATE NPSH TO THE MD AFW PUMPS, THE SD AFW PUMP WAS ADMINISTRATIVELY REMOVED FROM SERVICE AT 1255 HOURS ON 8/16/89. THE NRC WAS NOTIFIED OF THIS DESIGN DEFICIENCY BY THE EMERGENCY NOTIFICATION SYSTEM (ENS) AT 1346 HOURS IN ACCORDANCE WITH 10CFR50.72(B)(1)(II)(B). REFINED CALCULATIONS SHOWED THAT SUFFICIENT NPSH COULD BE ASSURED FOR ONLY ONE MD AFW PUMP OPERATING AT RATED FLOW, WHICH RESULTED IN MD AFW PUMP "B" BEING ADMINISTRAIVELY REMOVED FROM SERVICE AT 2230 HOURS ON 8/21/89. PRACTICAL RESOLUTION OF THE PROBLEM COULD NOT BE ACHIEVED WITHIN THE 24 HOUR TECH SPEC TIME REQUIREMENT; A PLANT SHUTDOWN WAS INITIATED AT 0000 HOURS ON 8/22/89. PURSUANT TO THE REQUIREMENTS OF 10CFR50.72(B)(1)(I)(A), THE NRC WAS NOTIFIED OF THE UNIT SHUTDOWN VIA THE ENS AT 0011 HOURS ON 8/22/89. PRIOR TO RETURNING THE UNIT TO SERVICE, THE AFW SUCTION PIPING WILL BE REPLACED AND THE PUMPS INSPECTED AND REFURBISHED. THIS LICENSEE EVENT REPORT IS SUBMITTED TO FULFILL THE REQUIREMENTS OF 10CFR50.73(A)(2)(II)(B), 10CFR50.73(A)(2)(I)(A), AND TECH SPEC 3.4.4.A.

[177] SALEM 1 DOCKET 50-272 LER 89-028
TECH SPEC SURVEILLANCE NOT COMPLETED WITHIN THE REQUIRED TIME DUE TO INADEQUATE ADMIN. CONTROLS.
EVENT DATE: 062689 REPORT DATE: 090689 NSSS: WE TYPE: PWR

(NSIC 215243) ON 8/11/89, TECH DEPARTMENT SUPERVISION IDENTIFIED THAT THE REQUIREMENTS OF TECH SPEC SURV. 4.5.2B.2 HAD NOT BEEN COMPLETELY SATISFIED. TECH SPEC SURV. 4.5.2 ADDRESSES THE SURV. REQUIREMENTS FOR DEMONSTRATING OPERABILITY OF EACH EMERGENCY CORE COOLING SYSTEM. THIS SURV. IS REQUIRED TO BE PERFORMED EVERY 31 DAYS. PART 2 OF THE SURV. WAS RECENTLY INCORPORATED INTO UNIT 1 TECH SPECS VIA AMENDMENT NO. 94 ON 6/2/89. ITEM "2" OF THE SURV. WAS NOT PREVIOUSLY REQUIRED BY THE UNIT 1 TECH SPECS; ALTHOUGH, IT WAS A REQUIREMENT OF UNIT 2 TECH SPECS. UPON DISCOVERY OF THE MISSED SURVEILLANCE, TECH SPEC ACTION STATEMENT 3.0.3 WAS ENTERED (ON 8/11/89 AT 2310 HOURS). IT WAS EXITED ON 8/12/89 AT 0005 HOURS UPON SUCCESSFUL COMPLETION OF THE SURVEILLANCE. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INADEQUATE ADMINISTRATIVE CONTROLS ASSOCIATED WITH PROCESSING AND IMPLEMENTING AMENDMENTS. PERSONNEL ERROR DID CONTRIBUTE TO CAUSING THIS EVENT. THIS EVENT HAS BEEN REVIEWED BY TECH DEPARTMENT MANAGEMENT. THE TECH DEPT' ENGINEER INVOLVED IN THIS EVENT HAS BEEN COUNSELED. AMENDMENTS ISSUED AFTER JANUARY 1989 HAVE BEEN REVIEWED. NO ADDITIONAL TECH SPEC NON-COMPLIANCES WERE IDENTIFIED. THE TECH SPEC SURVEILLANCE AUDIT PROJECT, PERFORMED BETWEEN AUGUST 1988 AND FEBRUARY 1989, REVIEWED ALL AMENDMENTS ISSUED WAS REVISED TO INCLUDE THE PUMP CASING AND PIPING VENTING REQUIREMENTS.

[178] SALEM 2 DOCKET 50-311 LER 89-013 REV 01
UPDATE ON LOSS OF 5 OF 6 CIRCULATING WATER PUMPS DUE TO EXTERNAL CAUSES.
EVENT DATE: 061089 REPORT DATE: 090689 NSSS: WE TYPE: PWR
VENDOR: DRESSER INDUSTRIES, INC.
STRUTHERS DUNN, INC.

(NSIC 215254) ON 6/10/89, A REACTOR TRIP WAS MANUALLY INITIATED FROM 15% REACTOR POWER DUE TO THE LOSS OF THE MAIN CONDENSER ON HIGH BACK PRESSURE. AT 2306 HOURS, THE "SCREENWASH TROUBLE" OVERHEAD ANNUNCIATOR IN CONTROL ROOM ALARMED DUE TO HIGH DIFF. LEVEL (DL) ACROSS SEVERAL CIRCULATING WATER SYSTEM (CWS) SCREENS. A POWER RUNBACK WAS INITIATED. WITHIN 5 MIN. OF THE ALARM, 5 OF THE 6 CIRCULATING PUMPS TRIPPED ON HIGH DL ACROSS THEIR ASSOCIATED SCREENS. TURBINE WAS MANUALLY TRIPPED WITH THE REACTOR AT 48% POWER. THE FUNCTIONAL TURBINE BYPASS VALVES (STEAM DUMP) WERE ARMED AND OPENED. CONDENSER BACK PRESSURE INCREASED TO WHERE THE PERMISSIVE TO USE STEAM DUMPS WAS EXCEEDED. THE STEAM DUMP VALVES CLOSED AND THE FIRST MAIN STEAM SAFETY VALVE IN EACH STEAMLINE LIFTED. DIRECTION WAS GIVEN TO MANUALLY TRIP THE REACTOR. ROOT CAUSE HAS BEEN ATTRIBUTED TO EXTERNAL CAUSES AND INADEQUATE CORRECTIVE ACTION FROM A SIMILAR PRIOR EVENT. LARGE ACCUMULATIONS OF GRASS AND DEBRIS ON THE SCREENS CAUSED THE HIGH DL. ON 8/11/83, A SIMILAR EVENT OCCURRED. CORRECTIVE ACTION FROM THE PRIOR EVENT DID NOT REQUIRE ANY LONG TERM ACTIONS. A REVIEW OF PREVENTIVE MAINTENANCE (PM) HISTORY FOR CLEANING TRASH RACKS WAS CONDUCTED. IT WAS FOUND THAT NO SPECIFIC PM REQUIREMENT EXISTED. A PM TASK FOR SPECIAL CLEANING OF TRASH RACKS EVERY REFUELING OUTAGE HAS BEEN ESTABLISHED.

[179] SALEM 2 DOCKET 50-311 LER 89-014
ESF ACTUATION SIGNALS, CONTAINMENT VENTILATION ISOLATION DUE TO A DESIGN/EQUIPMENT CONCERN.
EVENT DATE: 081089 REPORT DATE: 090689 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 215212) ON 8/10/89 AT 1841 HOURS, THE RADIATION MONITORING SYSTEM (RMS) CONTAINMENT PARTICULATE MONITOR, 2R12B, CHANNEL FAILED. THIS RESULTED IN A CONTAINMENT PURGE/PRESSURE-VACUUM RELIEF SYSTEM (CP/P-VRS) ISOLATION SIGNAL. ON 8/23/89 AT 0222 HOURS, THE RMS PLANT VENT PARTICULATE MONITOR, 2R41A, CHANNEL FAILED. THIS ALSO RESULTED IN A CP/P-VRS ISOLATION SIGNAL. THE ROOT CAUSE OF THESE EVENTS HAVE BEEN ATTRIBUTED TO DESIGN/EQUIPMENT PROBLEMS. THE CHANNEL BACK PLANT WAS FOUND CORRODED; THE COLD CONTACTS WERE DISCOLORED CAUSING THE CHANNELS TO FAIL. THE ISOLATION FUNCTION ASSOCIATED WITH THESE CHANNELS ACTUATES UPON CHANNEL FAILURE. THE 2R12B AND 2R41A EQUIPMENT BACK PLANE CONTACTS ON ALL MODULES WERE CLEANED. THE CHANNELS WERE SUBSEQUENTLY RETURNED TO SERVICE. THE ELECTRICAL CONTACTS OF THE VICTOREEN MODEL EQUIPMENT HAS PROVEN TO BE A CONTINUING PROBLEM AS INDICATED IN PRIOR EVENTS (I.E., UNIT 2 LER 89-009-00). THE CORRECTIVE ACTION TO ADDRESS THIS CONCERN INCLUDES REPLACEMENT OF THE FIELD MONITORS, AS NEEDED, ONCE THE RMS CENTRAL PROCESSING UNIT (CPU) UPGRADE IS COMPLETE.

[180] SAN ONOFRE 1 DOCKET 50-206 LER 89-021
REACTOR TRIP ON LOW FLOW DUE TO INSTRUMENT CABLE DEGRADATION.
EVENT DATE: 080389 REPORT DATE: 090589 NSSS: WE TYPE: PWR

(NSIC 215202) AT 1800 ON 8/3/89, WITH UNIT 1 AT 91% POWER, A REACTOR TRIP OCCURRED DUE TO ACTUATION OF THE REACTOR PROTECTION SYSTEM (RPS) ON LOW REACTOR COOLANT SYSTEM (RCS) FLOW IN ONE LOOP. ALL SYSTEMS RESPONDED NORMALLY TO THE TRIP AND THE OPERATORS (UTILITY, LICENSED) STABILIZED THE PLANT IN MODE 3. THE RPS OPERATED IN ACCORDANCE WITH DESIGN, WITH NO MALFUNCTIONS NOTED. IT IS BELIEVED THAT THE BRIEF LOW RCS FLOW SIGNAL OCCURRED IN LOOP C AND WAS CAUSED BY A LOSS OF INSULATION RESISTANCE OF THE FLOW TRANSMITTER CABLE, WHICH IS INDICATIVE OF DEGRADATION CAUSED BY A MOIST SUBSTANCE. AT THIS TIME FAILURE DUE

TO A MANUFACTURING DEFECT (CONSIDERED TO BE LESS LIKELY HAS NOT BEEN DISCOUNTED. THE CABLE HAS BEEN SENT TO AN INDEPENDENT OFFSITE LABORATORY TO DETERMINE THE EXACT MECHANISM FOR THE LOSS OF INSULATION RESISTANCE. THE MOIST SUBSTANCE MOST LIKELY ENTERED THE CONDUIT DURING THE INSTALLATION OF A TEMPORARY RCS LEVEL TRANSMITTER (DURING THE RECENTLY COMPLETED CYCLE 10 REFUELING OUTAGE) WHICH UTILIZED THE CABLE. ALTHOUGH ROOT CAUSE CONCLUSIONS ARE PENDING LABORATORY ANALYSIS OF THE CABLE, THE MOST LIKELY ROOT CAUSE IS THE UNINTENTIONAL INTRODUCTION OF A CHEMICAL SUBSTANCE ONTO THE CABLE AND THE FAILURE TO EVALUATE AND/OR REPLACE THE CABLE FOLLOWING SUCH AN OCCURRENCE. A SUPPLEMENTAL LER WILL BE SUBMITTED UPON COMPLETION OF OUR ROOT CAUSE EVALUATION.

[181] SAN ONOFRE 2 DOCKET 50-361 LER 86-029 REV 03
UPDATE ON UNIT 2 TRIP DURING TRANSFER OF NON-1E POWER SUPPLY.
EVENT DATE: 121086 REPORT DATE: 090189 NSSS: CE TYPE: PWR
VENDOR: MOTOROLA
RCA ELECTRONIC COMPONENTS

(NSIC 215156) ON 12/10/86 AT 1037, WITH UNIT 2 AT 93% POWER, THE TURBINE TRIPPED DURING A POWER INTERRUPTION TO THE TURBINE GOVERNOR CONTROL SYSTEM (TGCS), CAUSING A REACTOR TRIP. THE STEAM BYPASS CONTROL SYSTEM (SBCS) DID NOT INITIALLY ACTUATE AND A MAIN STEAM SAFETY VALVE BRIEFLY ACTUATED. THE TRIP RECOVERY PROCEEDED NORMALLY, ALTHOUGH START-UP CHANNEL 'B' FAILED, AND PLANT PROTECTION SYSTEM (PPS) CHANNEL 'A' DID NOT TRIP. ALL OTHER REQUIRED SAFETY RELATED EQUIPMENT FUNCTIONED AS DESIGNED, AND THERE WERE NO SAFETY CONSEQUENCES. THE NON-1E 120 VAC LOAD WAS BEING TRANSFERRED FROM THE NON-1E UNINTERRUPTIBLE POWER SUPPLY (UPS) INVERTER TO THE ALTERNATE SOURCE. A PROCEDURAL STEP TO DEFEAT THE AUTOMATIC RETRANSFER CIRCUIT WAS NOT PERFORMED, CAUSING THE LOAD TO TRANSFER BACK TO THE PRIMARY SOURCE. WHEN THE UPS INVERTER WAS DISCONNECTED UNDER LOAD, THE AUTOMATIC TRANSFER TO THE ALTERNATE SOURCE DID NOT OCCUR IN TIME TO PREVENT THE TRIP. THE UPS IS EQUIPPED WITH AN AUTOMATIC TRANSFER SWITCH WHICH AUTOMATICALLY TRANSFERS THE LOAD TO THE ALTERNATE SOURCE ON LOSS OF INVERTER OUTPUT VOLTAGE. THE TRANSFER SWITCH WAS FOUND TO OPERATE CORRECTLY; HOWEVER, THE ENSUING TRANSIENT IS BELIEVED TO HAVE CAUSED THE TRIP. THE EVENT RESULTED FROM THE FAILURE TO FOLLOW THE PROCEDURE; ADDITIONALLY, THE JOB DID NOT RECEIVE THE CORRECT LEVEL OF ATTENTION BY OPERATIONS PERSONNEL.

[182] SAN ONOFRE 2 DOCKET 50-361 LER 89-012 REV 01
UPDATE ON FOXBORO TRANSMITTER MOUNTING CONFIGURATION DISCREPANCIES CAUSED BY ENVIRONMENTAL QUALIFICATION AND DESIGN CONTROL PROGRAM WEAKNESSES.
EVENT DATE: 070189 REPORT DATE: 081889 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 1 (PWR)
SAN ONOFRE 3 (PWR)
VENDOR: FOXBORO CO., THE

(NSIC 215028) ON 6/1/89, A NCR WAS ISSUED WHICH IDENTIFIED THAT FOXBORO TRANSMITTERS INSIDE UNIT 1 CONTAINMENT HAD BEEN INSTALLED WITH A MOUNTING CONFIGURATION DIFFERENT THAN THAT SPECIFIED BY THE ENVIRONMENTAL QUALIFICATION (EQ) TEST REPORT. ON 6/2/89, SINCE UNIT 2 WAS SHUTDOWN, AN INSPECTION WAS PERFORMED WHICH IDENTIFIED SIMILAR DISCREPANCIES, AND ANOTHER NCR WAS ISSUED. IN ORDER TO DETERMINE THE SIGNIFICANCE OF THESE DISCREPANCIES. DETAILED CALCULATIONS WERE PERFORMED WHICH DEMONSTRATED THE ACCEPTABILITY OF THE AS-FOUND INSTALLATIONS AND PROVIDED ACCEPTANCE CRITERIA FOR ADDITIONAL UNIT 2 AND 3 INSPECTIONS. ON 7/1/89, INSPECTIONS OF ALL EQ-RELATED FOXBORO TRANSMITTERS AT UNITS 2 AND 3 WERE COMPLETED WHICH RESULTED IN ADDITIONAL NCR'S FOR 52 FOXBORO TRANSMITTERS (26 IN EACH UNIT 2 AND UNIT 3 CONTAINMENT). BASED UPON THE OBSERVED CONFIGURATIONS AND NEW ANALYTICAL DATA, 16 TRANSMITTERS (5 IN UNIT 2 AND 11 IN UNIT 3) WERE DECLARED INOPERABLE, TECH SPEC ACTION STATEMENTS WERE FOLLOWED AND IMMEDIATE ACTIONS WERE TAKEN TO CORRECT THE TRANSMITTER CONFIGURATIONS. FOLLOW UP LAB TESTING ULTIMATELY CONFIRMED THAT THE CALCULATIONS WERE CONSERVATIVE AND THAT

ORIGINAL CONFIGURATIONS WERE ADEQUATE TO MEET ALL SAFETY FUNCTIONS. NONE OF THE EQ DISCREPANCIES RESULTED IN A REPORTABLE CONDITION.

[183] SAN ONOFRE 2 DOCKET 50-361 LER 89-013
PREMATURE REMOVAL OF FIRE WATCH POSTING WITH A DIESEL GENERATOR SPRAY/SPRINKLER SYSTEM INOPERABLE.
EVENT DATE: 072489 REPORT DATE: 082389 NSSS: CE TYPE: PWR

(NSIC 215139) AT 0045 ON 7/24/89, DURING A TOUR OF VARIOUS FIRE PROTECTION EQUIPMENT, A FIRE WATCH SUPERVISOR IDENTIFIED THAT THE UNIT 2 TRAIN 'B' DIESEL GENERATOR (2G003) LOCAL DELUGE CONTROL PANEL WAS IN ALARM. AT 0137, AFTER AN ATTEMPT TO RESET THE DELUGE ALARM WAS UNSUCCESSFUL, A FIRE WATCH WAS POSTED TO MEET THE REQUIREMENTS OF TECH SPECS (TS) 3.7.8.2. FURTHER INVESTIGATION DETERMINED THAT AT 0834 ON 7/22/89, THE HOURLY FIRE WATCH POSTED FOR MAINTENANCE ACTIVITIES HAD BEEN TERMINATED WITH THE 2G003 DELUGE SYSTEM INOPERABLE. THUS, THE REQUIREMENTS OF TS 3.7.8.2 WERE NOT MET FOR APPROX. 40 HOURS. THE ROOT CAUSE OF THIS EVENT IS THE FAILURE TO CAPTURE A CONTINUING DEFICIENCY ASSOCIATED WITH THE DELUGE VALUE WITHIN EXISTING PROGRAMS. ON 5/4/89, THE DELUGE VALVE WAS DECLARED INOPERABLE FOR THE PERFORMANCE OF MAINTENANCE. AT THAT TIME, INTERNAL VALVE LEAKAGE WAS DISCOVERED AND A MAINTENANCE ORDER (MO) GENERATED. PRIOR TO THE PERFORMANCE OF THE MO, ON 6/15/89, A FIRE ALARM ASSOCIATED WITH THE VALVE WAS RECEIVED AT THE LOCAL CONTROL PANEL AND THE EMERGENCY SERVICE OFFICER'S (ESO) OFFICE. ALTHOUGH WORK ON THE MO WAS STILL IN PROGRESS, THE ADDITIONAL DEFICIENCY ASSOCIATED WITH THE ALARM WAS NOT CAPTURED. CONSEQUENTLY, THE DELUGE SYSTEM WAS RETURNED TO SERVICE AND THE HOURLY FIRE WATCH INSPECTIONS WERE TERMINATED, WHEN IN FACT, THE 2G003 DELUGE SYSTEM REMAINED INOPERABLE.

[184] SAN ONOFRE 2 DOCKET 50-361 LER 89-015
MISSED CONTROL ELEMENT ASSEMBLY POSITION VERIFICATION DUE TO PERSONNEL ERROR.
EVENT DATE: 081089 REPORT DATE: 090889 NSSS: CE TYPE: PWR

(NSIC 215268) AT 0843 ON 8/10/89, WITH UNIT 2 AT 100% POWER, WORK WAS AUTHORIZED TO BE PERFORMED ON CORE PROTECTION CALCULATOR (CPC) CHANNEL B. CPC CHANNEL B WAS DECLARED INOPERABLE, AND IN ACCORDANCE WITH CPC/CEAC (CONTROL ELEMENT ASSEMBLY (CEA) CALCULATOR) OPERATING PROCEDURE, CEAC #1 INOP FLAGS WERE SET IN ALL FOUR CPC CHANNELS, EFFECTIVELY RENDERING CEAC #1 INOPERABLE. TECH SPEC (TS) 3.3.1 REQUIRES THAT WITH ONE CEAC INOPERABLE, OPERATION MAY CONTINUE PROVIDED THAT ATLEAST EVERY 4 HOURS, EACH CEA IS VERIFIED TO BE WITHIN 7 INCHES (INDICATED POSITION) OF ALL OTHER CEAS IN ITS GROUP. AT 1455, THE WORK WAS COMPLETED, AND IT WAS REALIZED THAT THE 4-HOUR CEA POSITION VERIFICATION (DUE AT 1243) REQUIRED BY TECH SPEC 3.3.1 HAD NOT BEEN PERFORMED; AT THAT TIME, THE POSITION VERIFICATION WAS INITIATED, AND THE PROPER POSITION OF ALL CEAS WAS VERIFIED AT 1520. DUE TO PERSONNEL ERROR, THE CONTROL ROOM OPERATORS FAILED TO IDENTIFY AND IMPLEMENT THE REQUIRED ACTIONS SPECIFIED BY THE TSS, EVEN THOUGH THESE ACTIONS ARE ADEQUATELY STATED IN THE CPC/CEAC OPERATING PROCEDURE. THIS EVENT WILL BE DISCUSSED WITH APPROPRIATE OPERATIONS PERSONNEL. ALTHOUGH CURRENT PROCEDURES ARE CONSIDERED ADEQUATE TO HAVE PREVENTED THIS EVENT, THE CPC ADDRESSABLE CONSTANT CHANGE LOG WILL BE AMENDED TO PROVIDE ADDITIONAL ASSURANCE THAT THE 4-HOUR CEA POSITION VERIFICATION IS INITIATED.

[185] SAN ONOFRE 3 DOCKET 50-362 LER 89-001 REV 02
UPDATE ON REACTOR TRIP ON LOW SG LEVEL DUE TO PARTIAL LOSS OF POWER TO FEEDWATER CONTROL SYSTEM.
EVENT DATE: 010689 REPORT DATE: 071489 NSSS: CE TYPE: PWR
VENDOR: SOLID STATE CONTROLS, INC.

(NSIC 214824) AT 2335 ON 1/6/89 WITH UNIT 3 AT 98% POWER, THE REACTOR TRIPPED ON LOW STEAM GENERATOR (SG) LEVEL AFTER A PARTIAL LOSS OF NON-1E UNINTERRUPTIBLE

POWER SUPPLY (UPS) POWER OCCURRED WHICH CAUSED FEEDWATER REGULATING VALVES TO REDUCE FLOW TO SG E089. THIS ALSO RESULTED IN ACTUATION OF EMERGENCY FEEDWATER TO SG E089. EMERGENCY FEEDWATER TO SG E088 ALSO ACTUATED DUE TO THE RESULTING LEVEL "SHRINK" IN SG E088, WHICH IS EXPECTED FOLLOWING A TRIP FROM HIGH POWER. SINCE THE STEAM BYPASS CONTROL SYSTEM WAS IN MANUAL TO PERFORM TURBINE VALVE TESTING, HEAT REMOVAL FROM THE SGS WAS GREATER THAN NORMAL. AT 2336, AS A RESULT OF THE LOWER SG TEMPERATURE, REACTOR COOLANT SYSTEM (RCS) PRESSURE DECREASED BELOW THE SAFETY INJECTION ACTUATION SIGNAL (SIAS) SETPOINT (1806 PSIA), RESULTING IN AN SIAS ACTUATION. THERE WAS NO SAFETY INJECTION FLOW INTO THE RCS SINCE RCS PRESSURE REMAINED ABOVE THE SHUTOFF HEAD OF THE INJECTION PUMPS. ALL SAFETY SYSTEMS OPERATED IN ACCORDANCE WITH DESIGN. AT 0025 ON 1/7/89, THE PLANT WAS STABILIZED AND ALL ESFS WERE RESET AND LINEUPS RETURNED TO NORMAL. 2 OF 3 NON-1E UPS PHASES WERE LOST BECAUSE OF A COMMON FAULT IN THE ASSOCIATED INVERTER'S CONSTANT VOLTAGE TRANSFORMER (CVT) OUTPUT WINDINGS. A TEMPORARY JUMPER BETWEEN UPS UNGROUNDED NEUTRAL AND GROUND, WHICH HAD NOT BEEN PROPERLY REMOVED DURING PREVIOUS MAINTENANCE, MAY HAVE CONTRIBUTED TO THE FAILURE.

[186] SEABROOK 1 DOCKET 50-443 LER 89-009
TECH SPEC SURVEILLANCE NOT PROPERLY PERFORMED.
EVENT DATE: 081089 REPORT DATE: 091189 NSSS: WE TYPE: PWR

(NSIC 215285) ON 8/10/89, THE REQUIREMENTS OF TECH SPEC 3.3.3.9, WERE NOT MET. AT THAT TIME, THE PRIMARY COMPONENT COOLING WATER (PCCW) HEAD TANK RATE OF CHANGE ALARM WAS OUT OF SERVICE, REQUIRING THAT SAMPLING OF THIS EFFLUENT PATHWAY OCCUR EVERY 12 HOURS. THE REQUIRED SAMPLES WERE OBTAINED. HOWEVER, THE SAMPLE TAKEN AT 5:30 PM ON 8/10/89 WAS NOT ANALYZED FOR THE 10,000 SECONDS REQUIRED TO ENSURE A LOWER LIMIT OF DETECTION OF 1.0E-08 MICROCURIE/ML. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO PERSONNEL ERROR. THE TECHNICIAN PERFORMING THE SAMPLE ANALYSIS INADVERTENTLY SELECTED THE INCORRECT ANALYSIS DURATION WHEN BEGINNING THE ANALYSIS. A REVIEW OF THE PROCEDURES INVOLVED IN THIS EVENT INDICATED THAT THE PROCEDURES ARE CLEAR AND ACCURATE, AND DIRECT THE TECHNICIAN TO THE PROPER COURSE OF ACTION. THE TECHNICIAN INVOLVED IN THIS EVENT HAS BEEN COUNSELED. ADDITIONALLY, THIS EVENT WAS REVIEWED AT A CHEMISTRY DEPARTMENT MEETING TO ENHANCE THE AWARENESS OF ALL TECHNICIANS PERFORMING THIS TYPE OF ANALYSIS. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THIS EVENT. ANALYSIS OF THIS SAMPLE, AS WELL AS THE PREVIOUS AND FOLLOWING SAMPLES, INDICATED NO DETECTABLE ACTIVITY. AT THE TIME OF THIS EVENT SEABROOK STATION WAS IN MODE 5 (COLD SHUTDOWN) WITH THE RCS AT A TEMPERATURE OF 133F AND VENTED TO ATMOSPHERE.

[187] SEABROOK 1 DOCKET 50-443 LER 89-010
ENGINEERED SAFETY FEATURE ACTUATION DUE TO DIESEL GENERATOR START CAUSED BY OPERATOR ERROR.
EVENT DATE: 081589 REPORT DATE: 091489 NSSS: WE TYPE: PWR

(NSIC 215286) AUGUST 15, 1989 AT 7:04 P.M. EDT, WHILE PERFORMING A TAGGING ORDER RESTORATION, AN AUXILIARY OPERATOR INADVERTENTLY OPENED THE POTENTIAL TRANSFORMER (PT) FUSE CUBICLE FOR EMERGENCY BUS E-6, CAUSING AN UNDERVOLTAGE SIGNAL. IN RESPONSE TO THE UNDERVOLTAGE SIGNAL FROM BUS E-6, EMERGENCY DIESEL GENERATOR 1B STARTED AND SEQUENCED ON ITS DESIGNED EMERGENCY LOADS. IT HAS BEEN DETERMINED THAT THE ROOT CAUSE OF THE INADVERTENT EMERGENCY DIESEL GENERATOR START WAS DUE TO OPERATOR ERROR. THE OPERATOR WAS UNAWARE THAT THE CONSTRUCTION OF THE PT FUSE COMPARTMENTS IS SUCH THAT WHEN THE COMPARTMENT DRAWER IS OPENED, IT AUTOMATICALLY DISCONNECTS THE PT FUSES FOR THAT COMPARTMENT, CAUSING THE BUS TO DEENERGIZE. A COMPLETE REVIEW OF THE TAGGING ORDER WAS PERFORMED TO DETERMINE IF THE EXISTING EQUIPMENT LABELING IS SUFFICIENT. IT HAS BEEN DETERMINED THAT ADDITIONAL EQUIPMENT IDENTIFICATION LABELING IS NOT REQUIRED. HOWEVER, CAUTION LABELING WILL BE PLACED ON THE PT FUSE COMPARTMENTS, WARNING OF THE CONSEQUENCES OF OPENING THE PT COMPARTMENT DOOR WITH THE BUS ENERGIZED. THE AUXILIARY OPERATOR INVOLVED IN THIS EVENT HAS BEEN COUNSELED ON THE OPERATION OF THE FUSE

COMPARTMENTS. ADDITIONAL TRAINING FOR ALL AUXILIARY OPERATORS WILL BE PERFORMED. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THIS EVENT AND ALL SAFETY SYSTEMS OPERATED AS DESIGNED.

[188] SEQUOYAH 1 DOCKET 50-327 LER 89-005 REV 01
UPDATE ON REACTOR TRIP SIGNAL RESULTING FROM CLOSURE OF MFW REGULATOR VALVES ON LOSS OF POWER TO VALVE CONTROLLERS DUE TO PERSONNEL ERROR.
EVENT DATE: 021089 REPORT DATE: 061589 NSSS: WE TYPE: PWR

(NSIC 214406) ON 2/10/89 WITH UNIT 1 IN MODE 1, A REACTOR TRIP OCCURRED AT 2036 EST. TRIP SIGNAL WAS A RESULT OF A STEAM GENERATOR (SG) STEAM FLOW TO FEEDWATER FLOW MISMATCH OF GREATER THAN 40% OF THE NOMINAL VALUE OF STEAM FLOW AT FULL POWER COINCIDENT WITH A LOW SG LEVEL SIGNAL ON SG LOOP 3. TWO INSTRUMENT MAINTENANCE (IM) TECHNICIANS WERE IMPLEMENTING WORK REQUEST (WR)-B238429 ON FLOW RECORDER (FR)-2-200/201." THE RECORDER PEN NEEDED TO BE RESTRUNG WHICH REQUIRED IT TO BE REMOVED FROM THE CASE. THE TECHNICIANS FULLY REMOVED THE RECORDER IN ITS CASE WHICH REQUIRED LIFTING THE POWER SUPPLY LEADS. AFTER REINSTALLATION, THE TECHNICIANS RETERMINATED THE POWER SUPPLY LEADS, AT WHICH TIME ONE TECHNICIAN DETERMINED THE TERMINATING LEADS WERE TOO CLOSE TO EACH OTHER. THE TECHNICIAN INADVERTENTLY SHORTED A SCREWDRIVER BETWEEN THE TERMINALS, TRIPPING OPEN BREAKER NO. 39 ON 120-VAC VITAL INSTRUMENT BOARD I-II WHICH IS THE POWER SUPPLY TO A PLUG MOLD SUPPLYING THE RECORDER. THE PLUG MOLD IS ALSO THE COMMON POWER SUPPLY TO THE FLOW INDICATING CONTROLLERS (FIC)-3-35, -90, AND -103 WHICH CONTROL MAIN FEEDWATER REGULATING VALVES (MFWRVS) FCV-3-35, -90, AND -103 FOR SG LOOPS 1, 3, AND 4. ROOT CAUSE OF THE REACTOR TRIP SIGNAL WAS PERSONNEL ERROR, IN THAT, APPROPRIATE PRECAUTIONS WERE NOT TAKEN IN PERFORMING TERMINATIONS OF ENERGIZED EQUIPMENT.

[189] SEQUOYAH 1 DOCKET 50-327 LER 89-007 REV 01
UPDATE ON MAIN CONTROL ROOM ISOLATION CAUSED BY INCOMPATIBILITY OF REPLACEMENT SMOKE DETECTOR UNIT RELAY CONTACT CONFIGURATION.
EVENT DATE: 031989 REPORT DATE: 090789 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
VENDOR: WALTER KIDDE & COMPANY, INC.

(NSIC 215213) ON 3/19/89, WITH UNIT 1 IN MODE 1 (100% POWER, 2235 PSIG, 578F) AND UNIT 2 IN MODE 5 (0% POWER, 3 PSIG, 130F), AN "A" TRAIN CONTROL ROOM ISOLATION (CRI) OCCURRED. AT APPROXIMATELY 1125 EDT, "A" TRAIN CRI SIGNAL, AS INDICATED ON MAIN CONTROL ROOM PANEL O-M-27B (WINDOW 20), WAS RECEIVED IN THE CONTROL ROOM. THE UNIT 1 ASSISTANT SHIFT OPERATIONS SUPERVISOR (ASOS) RESPONDED TO THE CRI ALARM. THE ASOS HAD KNOWLEDGE THAT MAINTENANCE ON CONTROL BUILDING (CB) FRESH AIR INTAKE DUCT SMOKE DETECTORS O-XS-31A-3 AND O-XS 31A-4 WAS BEING PERFORMED AND COULD HAVE INITIATED THE CRI. SUBSEQUENTLY, THE ASOS SUSPENDED THE MAINTENANCE ACTIVITY ON THE SMOKE DETECTORS. THE ASOS VERIFIED THAT NO OTHER CONDITIONS WERE PRESENT SUCH AS HIGH RADIATION OR A SAFETY INJECTION (SI) SIGNAL, AND INITIATED REALIGNMENT OF THE CONTROL ROOM VENTILATION SYSTEM TO NORMAL OPERATION IN ACCORDANCE WITH SYSTEM OPERATING INSTRUCTION (SOI)-30.1B, "CONTROL BUILDING AND CONTROL ROOM HEATING, AIR CONDITIONING AND VENTILATION SYSTEM." THE CB VENTILATION SYSTEM WAS RETURNED TO NORMAL OPERATION BY 1310 EST. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INCOMPATIBILITY OF REPLACEMENT SMOKE DETECTOR UNIT'S RELAY CONTACT CONFIGURATION WITH THE APPLICATION REQUIREMENTS. AS CORRECTIVE ACTION, A WORK REQUEST (WR) HAS BEEN INITIATED TO RECONFIGURE UNITS IN STOCK AT SQN.

[190] SEQUOYAH 1 DOCKET 50-327 LER 89-017 REV 01
UPDATE ON FAILURE TO PROPERLY CALIBRATE THE PRESSURIZER LEVEL CONTROL TRANSMITTERS RESULTED IN AN OPERATION PROHIBITED BY TECH SPECS.
EVENT DATE: 061389 REPORT DATE: 092989 NSSS: WE TYPE: PWR

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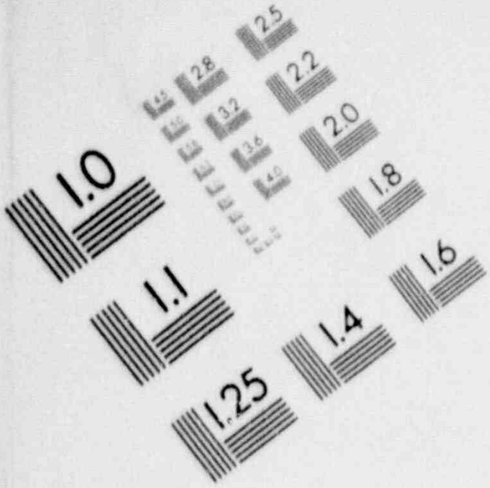
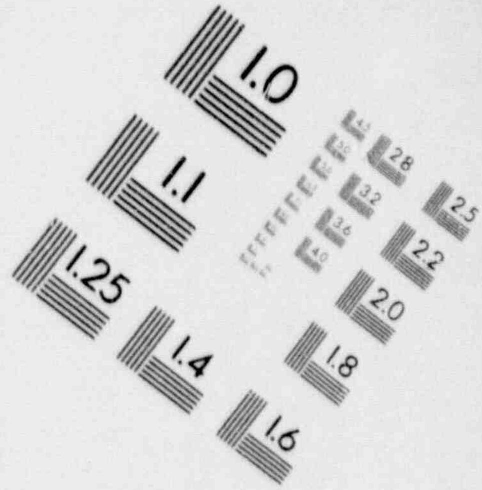
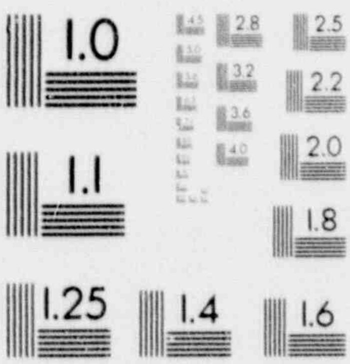
IMAGE EVALUATION TEST TARGET (MT-3)

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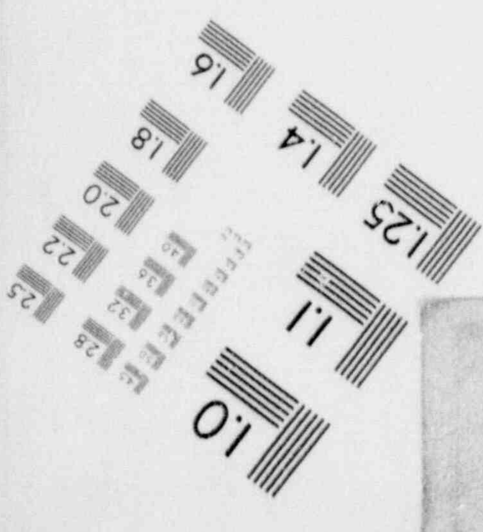
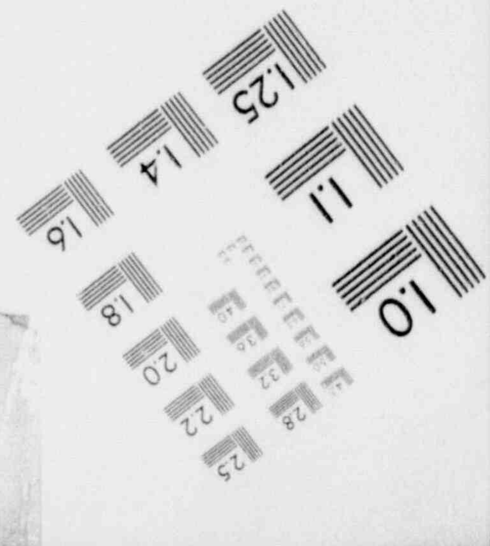

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IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

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IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

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IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
VENDOR: FOXBORO CO., THE

(NSIC 215365) ON 6/13/89, AT 1055 EDT WITH UNITS 1 AND 2 IN MODE 1 (100% POWER, 578F, AND 2,235 PSIG), THE ASSISTANT SHIFT OPERATIONS SUPERVISOR (ASOS) OBSERVED THAT THE AUXILIARY CONTROL ROOM (ACR) PRESSURIZER LEVEL INDICATORS WERE INDICATING HIGH AS COMPARED TO THE MAIN CONTROL ROOM (MCR) PRESSURIZER LEVEL INDICATORS. OPERATIONS PERSONNEL DECLARED BOTH CHANNELS OF THE ACR PRESSURIZER LEVEL INDICATORS INOPERABLE AND ENTERED TECH SPEC LIMITING CONDITION FOR OPERATION (LCO) 3.3.3.5. THIS EVENT WAS CAUSED FROM FAILURE TO PROPERLY CALIBRATE THE PRESSURIZER LEVEL TRANSMITTERS DURING THE LAST CALIBRATION CHECK; AN ADEQUATE SOAK TIME WAS NOT PROVIDED PRIOR TO CALIBRATION. THE LEVEL TRANSMITTERS WERE RECALIBRATED, AND LCO 3.3.3.5 WAS EXITED AT 0015 EDT ON 6/15/89. ANOTHER CONTRIBUTING FACTOR WAS THE PERFORMANCE OF AN INADEQUATE CHANNEL CHECK IN THAT ACR INSTRUMENTS WERE NOT REQUIRED TO BE COMPARED TO THE MCR INSTRUMENTS. A COMPARISON BETWEEN THE OTHER ACR AND MCR INSTRUMENTS IDENTIFIED NO OTHER CASES OF INADEQUATE CHANNEL CHECKS. ADDITIONAL CORRECTIVE ACTIONS INCLUDED REVISING SI-88, TO REQUIRE A SOAK TIME WHEN CALIBRATING THE PRESSURIZER LEVEL TRANSMITTERS. ALSO, FURTHER INVESTIGATION REVEALED THAT FOXBORO TRANSMITTERS, WHICH ARE EQUIPPED WITH A ZERO SUPPRESSION ARE SUSCEPTIBLE TO A ZERO SHIFT IF OVERRANGED.

[191] SEQUOYAH 1 DOCKET 50-327 LER 89-021 REV 01
UPDATE ON DIESEL GENERATOR BOARD ROOM FIRE PROTECTION SYSTEM INOPERABLE WHEN FIRE DOOR WAS INOPERABLE BECAUSE OF INADEQUATE TECHNICAL REVIEW OF A SURVEILLANCE TEST DEFICIENCY.
EVENT DATE: 071389 REPORT DATE: 083189 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 215162) THIS REPORT DESCRIBES AN EVENT CONCERNING THE OPERABILITY OF THE DIESEL GENERATOR (D/G) BOARD ROOM 1A-A CARBON DIOXIDE (CO2) FIRE PROTECTION SYSTEM WHEN FIRE DOOR 0-DOR-410-D24A FAILED TO CLOSE DURING PERFORMANCE OF SURVEILLANCE INSTRUCTION (SI) 237.2. THIS EVENT WAS DISCOVERED ON 7/13/89, WHEN THE SHIFT OPERATIONS SUPERVISOR (SOS) WOULD NOT ACCEPT WORK REQUEST (WR) B252362 AS COMPLETE BECAUSE IT REFERENCED A DEFICIENCY FOUND DURING THE PERFORMANCE OF SI-237.2, BUT THE SAME SI WAS NOT REPERFORMED AS A POSTMAINTENANCE TEST (PMT). FURTHER INVESTIGATION DETERMINED THE DEFICIENCY HAD BEEN EVALUATED BY AN SOS ON 6/1/89, AS NOT AFFECTING TECH SPEC OPERABILITY OF THE D/G BUILDING CO(2) SYSTEM SINCE THE FIRE DOOR WAS LEFT CLOSED. SURVEILLANCE REQUIREMENT (SR) 4.7.11.3.2.B.1 REQUIRES THE FIRE DOOR RELEASE MECHANISMS TO BE CAPABLE OF BOTH MANUAL AND AUTOMATIC ACTUATION FOR CO(2) SYSTEM OPERABILITY. THE FIRE DOOR WAS DECLARED INOPERABLE BY THE SOS ON 7/13/89, AND LIMITING CONDITION FOR OPERATION (LCO) 3.7.11.3 ACTION WAS ENTERED. THE FIRE DOOR WAS REPAIRED, APPLICABLE PORTIONS OF SI-237.2 WERE REPERFORMED, AND THE LCO ACTION WAS EXITED ON 7/14/89. THE ROOT CAUSE WAS AN INADEQUATE TECHNICAL EVALUATION OF THE SI-237.2 DEFICIENCY. CORRECTIVE ACTIONS TO PREVENT RECURRENCE INCLUDE REVISING SEVERAL PROCEDURES.

[192] SEQUOYAH 1 DOCKET 50-327 LER 89-023
FAILURE TO ENTER LIMITING CONDITION FOR OPERATION 3.4.11 AFTER DISCOVERING THE REACTOR COOLANT VENT SYSTEM WAS INOPERABLE.
EVENT DATE: 072889 REPORT DATE: 082889 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 215132) ON JULY 28, 1989, AT 1700 EDT, WITH BOTH UNITS IN MODE 1, LIMITING CONDITION FOR OPERATION (LCO) 3.4.11 WAS ENTERED FOR BOTH UNITS BECAUSE OF INOPERABLE REACTOR COOLANT SYSTEM (RCS) HEAD VENTS. THE SHIFT TECHNICAL ADVISOR (STA) ON DUTY DURING THE EVENING SHIFT ON JULY 27, 1989, WAS REVIEWING THE NEW RCS VENT TECHNICAL SPECIFICATION (TS). HE NOTED THAT THE UPSTREAM MANUAL ISOLATION VALVES WERE NOT IDENTIFIED AS LOCKED OPEN ON THE FLOW PRINTS. THE

CONFIGURATION CONTROL STATUS FILES WERE REVIEWED, AND THE VALVES WERE LISTED AS OPEN BUT NOT LOCKED OPEN. THE STA NOTIFIED THE SHIFT OPERATIONS SUPERVISOR BUT THE OPERATING SHIFT CREW FAILED TO DECLARE THE REACTOR COOLANT VENT SYSTEM INOPERABLE AND ENTER THE APPLICABLE LCO TS ACTION. AS A RESULT OF FURTHER DISCUSSIONS AND INVESTIGATIONS BETWEEN TVA AND NRC, THE REACTOR COOLANT VENT SYSTEM WAS DECLARED INOPERABLE, AND LCO 3.4.11 WAS ENTERED. AS IMMEDIATE CORRECTIVE ACTION, TVA REMOVED POWER FROM THE SOLENOID VALVES AND REQUESTED A WAIVER OF COMPLIANCE. LONG-TERM CORRECTIVE ACTIONS INCLUDE: ENSURING APPROPRIATE PLANS EXIST FOR IMPLEMENTATION OF NEW TS CHANGES AND A CLARIFICATION OF TS 3.4.11.

[193] SEQUOYAH 1 DOCKET 50-327 LER 89-024
INADVERTENT CONTAINMENT VENT ISOLATION EVENT RESULTING FROM A DIFFICULT MAN-MACHINE INTERFACE DURING SURVEILLANCE TESTING.
EVENT DATE: 080589 REPORT DATE: 090589 NSSS: WE TYPE: PWR

(NSIC 215164) AT 1020 EDT, ON 8/5/89, WITH BOTH UNITS IN MODE 1 AT 100% POWER, 2,235 LBS. PER SQUARE INCH GAUGE, 578F, AN INADVERTENT TRAIN B CONTAINMENT VENT ISOLATION (CVI) OCCURRED ON UNIT 1 WHILE PERFORMING SURVEILLANCE TESTING ON THE UNIT 2 UPPER CONTAINMENT RADIATION MONITOR. INSTRUMENT MECHANICS WERE CONNECTING A DIGITAL MULTIMETER TO A TERMINAL BLOCK WHEN AN INCORRECT TERMINAL WAS CONTACTED CAUSING THE CVI. THIS EVENT HAS TWO ROOT CAUSES RELATED TO A DIFFICULT MAN-MACHINE INTERFACE; FAILURE TO PROMPTLY REVISE SURVEILLANCE INSTRUCTIONS FOLLOWING THE INSTALLATIONOF NEW TEST POINTS TO FACILITATE THE SURVEILLANCE TESTING A ND FAILURE TO APPLY THE NECESSARY ATTENTION TO DETAIL FOR A HIGH HAZARD TASK. THE NEW TEST POINTS, INSTALLED 12 DAYS PRIOR TO THIS EVENT, WERE A CORRECTIVE ACTION FROM A SIMILAR EVENT REPORTED BY LER 50-328/88039. AS IMMEDIATE CORRECTIVE ACTION, UNIT 2 OPERATORS VERIFIED THAT NO ACTUAL HIGH-RADIATION CONDITION EXISTED AND INITIATED RECOVERY FROM THE CVI. TO PREVENT RECURRENCE, THE SURVEILLANCE INSTRUCTIONS HAVE BEEN REVISED TO SPECIFY USE OF THE NEW TEST POINTS, AND THE MODIFICATION PROCEDURE HAS BEEN REVISED TO REQUIRE AN EVALUATION OF A MODIFICATION'S IMPACT ON PROCEDURES AND THE PLANT BEYOND OPERABILITY CONSIDERATIONS. IN ADDITION, THE EVENTS IN THIS LER WILL BE REVIEWED FOR LESSONS LEARNED WITH THE APPROPRIATE INSTRUMENT MAINTENANCE PERSONNEL.

[194] SEQUOYAH 1 DOCKET 50-327 LER 89-025
INOPERABILITY OF ALL FOUR DIESEL GENERATORS RESULTING FROM AN INADEQUATE PROCEDURE.
EVENT DATE: 081589 REPORT DATE: 091489 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 215261) ON 8/15/89 AT 1109 EDT, WITH UNITS 1 AND 2 IN MODE 1 (100 PERCENT POWER, 2,235 PSIG, AND 578F), ALL FOUR EMERGENCY DIESEL GENERATORS (D/GS) WERE CLEARED INOPERABLE. THE D/GS WERE DECLARED INOPERABLE BECAUSE THE D/G FUEL OIL HAD NOT BEEN SAMPLED IN ACCORDANCE WITH TECH SPEC (TS) SURVEILLANCE REQUIREMENT (SR) 4.8.1.1.2 (I.E., A REPRESENTATIVE SAMPLE WAS NOT OBTAINED IN ACCORDANCE WITH AMERICAN SOCIETY OF TESTING MATERIALS (ASTM) STANDARD ASTM D270-1975). THIS CONDITION WAS IDENTIFIED DURING AN EVALUATION OF A NUCLEAR EXPERIENCE REVIEW REPORT CONCERNING A PROBLEM ANOTHER NUCLEAR POWER FACILITY ENCOUNTERED IN OBTAINING REPRESENTATIVE SAMPLES OF D/G FUEL OIL. DURING THE EVALUATION, IT WAS DETERMINED THAT THE SAMPLE METHOD BEING USED WAS INADEQUATE. THE ROOT CAUSE OF THE EVENT WAS AN INADEQUATE PROCEDURE RESULTING FROM THE FAILURE OF CHEMISTRY PERSONNEL TO INCORPORATE ASTM D270-1975 REQUIREMENTS INTO THE PROCEDURE. CORRECTIVE ACTIONS INCLUDED OBTAINING SAMPLES IN ACCORDANCE WITH TECH SPEC REQUIREMENTS. THE SAMPLES WERE ANALYZED AND DETERMINED TO MEET THE ACCEPTANCE CRITERIA EXCEPT FOR TANK 2A-A. AFTER THE INITAL SAMPLES WERE OBTAINED, WATER, SLUDGE, AND PARTICLES WERE REMOVED FROM THE TANKS, AND ADDITIONAL SAMPLES WERE OBTAINED. THE SECOND SAMPLE FROM THE 2A-A TANK WAS ANALYZED AND DETERMINED TO MEET THE ACCEPTANCE CRITERIA.

[195] SEQUOYAH 2 DOCKET 50-328 LER 88-019 REV 01
UPDATE ON INADEQUATE WORK CONTROL CAUSED TWO EMERGENCY CORE COOLING SYSTEM PUMPS TO BE INOPERABLE.
EVENT DATE: 040788 REPORT DATE: 092989 NSSS: WE TYPE: PWR

(NSIC 215334) AT 1215 EDT ON 4/7/88, WITH UNIT 2 IN MODE 3 (HOT STANDBY), IT WAS DETERMINED THAT TECH SPEC LIMITING CONDITION FOR OPERATION (LCO) 3.0.3 HAD BEEN IN EFFECT SINCE APPROX. 1156 EDT THAT MORNING. LCO 3.0.3 WAS APPLICABLE BECAUSE THE HANDSWITCHES FOR THE "B" TRAIN RESIDUAL HEAT REMOVAL (RHR) PUMP AND THE "A" TRAIN CENTRIFUGAL CHARGING PUMP (CCP) HAD BOTH BEEN PLACED IN THE "PULL-TO-LOCK" POSITION, RENDERING BOTH PUMPS INOPERABLE. AS A RESULT, 2 INDEPENDENT TRAINS OF EMERGENCY CORE COOLING SYSTEM (ECCS) SUBSYSTEMS WERE NOT AVAILABLE, AND UNIT 2 WAS NOT IN COMPLIANCE WITH EITHER THE LCO OR THE ACTION REQUIREMENTS OF TECH SPEC 3.5.2. AT 1226 EDT, CCP 2A-A WAS RETURNED TO SERVICE, AND LCO 3.0.3 WAS EXITED. IMMEDIATE CAUSE OF THIS EVENT WAS A PERSONNEL ERROR THAT RESULTED FROM THE HIGH LEVEL OF ACTIVITY IN THE MAIN CONTROL ROOM (MCR). ROOT CAUSE OF THIS EVENT WAS INADEQUATE WORK CONTROL. HANDSWITCHES FOR BOTH PUMPS WERE PLACED IN THE "PULL-TO-LOCK" POSITION IN ACCORDANCE WITH APPROVED PLANT PROCEDURES; THESE PROCEDURES SHOULD NOT HAVE BEEN ALLOWED TO BE PERFORMED CONCURRENTLY. TO ENSURE THAT THE ACTIVITY IN THE MCR IS MAINTAINED AT A REASONABLE LEVEL DURING THE RESTART OF UNIT 2, THE SHIFT OPERATIONS ADVISOR (SOA) WILL ASSIST THE SHIFT SUPERVISOR (SS) IN MONITORING ACTIVITY LEVEL IN THE MCR AND RECOMMEND ACTIVIES BE SLOWED DOWN.

[196] SEQUOYAH 2 DOCKET 50-328 LER 89-010
FAILURE TO PERFORM A SURVEILLANCE REQUIREMENT WITHIN THE SPECIFIED TIME INTERVAL BECAUSE OF PERSONNEL ERROR.
EVENT DATE: 081789 REPORT DATE: 091889 NSSS: WE TYPE: PWR

(NSIC 215311) ON 8/17/89, WITH UNITS 1 AND 2 IN MODE 1, AT 100% POWER, 2,235 POUNDS PER SQUARE INCH GAUGE, 578F, IT WAS DISCOVERED THAT A TECH SPEC SURVEILLANCE REQUIREMENT HAD NOT BEEN PERFORMED WITHIN THE REQUIRED SURVEILLANCE INTERVAL. THE SURVEILLANCE WAS TO PERFORM A REACTIVITY BALANCE AT LEAST EVERY 31 EFFECTIVE FULL POWER DAYS TO ENSURE THE OPERATING CHARACTERISTICS ARE CONSISTENT WITH DESIGN CHARACTERISTICS FOR BORON CONCENTRATION. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO PERSONNEL ERROR ON THE PART OF PERIODIC TEST PERSONNEL AND THE REACTOR ENGINEER INVOLVED. AS IMMEDIATE CORRECTIVE ACTION, THE SURVEILLANCE WAS SUBSEQUENTLY PERFORMED, AND THE RESULTS WERE VERIFIED ACCEPTABLE. AS ADDITIONAL CORRECTIVE ACTION, THE REACTOR ENGINEERING AND PERIODIC TEST PERSONNEL INVOLVED WITH THE EVENT HAVE BEEN DISCIPLINED AND COUNSELLED AS DESCRIBED IN THE TEXT.

[197] SEQUOYAH 2 DOCKET 50-328 LER 89-011
RE-90-119 RADIATION MONITORS INOPERABLE BECAUSE OF INADEQUATE SOURCE CHECK PERFORMANCE.
EVENT DATE: 082389 REPORT DATE: 092289 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 1 (PWR)

(NSIC 215312) ON 8/23/89, WITH UNITS 1 AND 2 IN MODE 1 AT 100% POWER, 2,235 POUNDS PER SQUARE INCH GAUGE, 578F, IT WAS DISCOVERED THAT A TECH SPEC (TS) SURVEILLANCE REQUIREMENT (SR) TO SOURCE CHECK THE RADIOACTIVE GASEOUS EFFLUENT MONITORS ON THE CONDENSER VACUUM PUMP EXHAUST WAS NOT BEING FULLY MET. A SOURCE CHECK IS DEFINED IN THE SQN TECH SPEC AS A QUALITATIVE ASSESSMENT OF CHANNEL RESPONSE WHEN THE CHANNEL SENSOR IS EXPOSED TO A RADIOACTIVE SOURCE. THE SUBJECT MONITORS USE A LIGHT-EMITTING DIODE (LED) LIGHT SOURCE TO SOURCE CHECK ALL COMPONENTS EXCEPT THE SCINTILLATION CRYSTAL. ADDITIONALLY, THE SOURCE CHECK ADEQUACY OF OTHER GASEOUS EFFLUENT RADIATION MONITORS THAT EXPOSE A SECOND, NONPROCESS SCINTILLATION CRYSTAL TO A RADIOACTIVE SOURCE DURING SOURCE CHECKING IS STILL BEING INVESTIGATED. THE ROOT CAUSE OF THIS EVENT IS STILL BEING

INVESTIGATED. AS INTERIM CORRECTIVE ACTION, THE TWO MONITORS WITH LEDS WERE SOURCE CHECKED WITH A RADIOACTIVE SOURCE TO DEMONSTRATE THEIR OPERABILITY. THE SURVEILLANCE INSTRUCTION HAS BEEN REVISED TO REQUIRE A RADIOACTIVE SOURCE TO BE USED FOR SOURCE CHECKING THESE TWO MONITORS. A SUPPLEMENTAL REPORT WILL BE MADE UPON COMPLETION OF THE INVESTIGATION.

[198] SHEARON HARRIS 1 DOCKET 50-400 LER 89-013
INADVERTENT ACTUATION OF SERVICE WATER BOOSTER PUMP DURING LOAD SEQUENCER TROUBLESHOOTING.
EVENT DATE: 081489 REPORT DATE: 091189 NSSS: WE TYPE: PWR

(NSIC 215278) ON 8/14/89 AT 1123, WITH THE PLANT AT 100% POWER, A TECHNICIAN TROUBLESHOOTING AN INDICATION PROBLEM IN THE EMERGENCY LOAD SEQUENCER PANEL B, INADVERTENTLY JUMPERED CONTACTS WHICH CAUSED THE EMERGENCY SERVICE WATER BOOSTER PUMP B TO START. THIS PUMP FUNCTIONS TO INCREASE PRESSURE IN THE CONTAINMENT BUILDING SERVICE WATER HEADER FOLLOWING AN ACCIDENT TO ENSURE SERVICE WATER PRESSURE EXCEEDS PEAK CONTAINMENT ACCIDENT PRESSURE, AND SO ELIMINATE THE POSSIBILITY OF LEAKAGE FROM CONTAINMENT. CONTROL ROOM OPERATORS DETECTED THE PUMP START BY FLOW ALARMS ON THE SERVICE WATER SYSTEM, AND THE PUMP WAS STOPPED FROM THE CONTROL ROOM WITHIN ONE MINUTE. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR IN THAT THE INCORRECT CONTACTS WERE JUMPERED. TROUBLESHOOTING WAS BEING CONDUCTED BY THE SYSTEM ENGINEER USING THE CONTROL WIRING DRAWINGS TO DETERMINE WHICH CONTACTS TO JUMPER. THE TECHNICIAN JUMPERED ACROSS CONTACTS DIRECTLY ABOVE THE PAIR CALLED FOR BY THE ENGINEER. THE CONGESTED ARRANGEMENT OF WIRE BUNDLES IN THE PANEL CONTRIBUTED TO THE ERROR. THE JUMPER WAS IMMEDIATELY REMOVED, AND TESTING WAS STOPPED UNTIL THE EVENT WAS INVESTIGATED AND UNDERSTOOD. PERSONNEL INVOLVED WERE COUNSELLED ON ATTENTION TO WORK ACTIVITIES AND THE CONSEQUENCES OF THEIR ACTIONS RESULTING IN AN UNNECESSARY CHALLENGE TO SAFETY SYSTEMS.

[199] SHEARON HARRIS 1 DOCKET 50-400 LER 89-014
TWO HIGH POINT EMERGENCY CORE COOLING SYSTEM VENTS WERE NOT BEING TESTED AS REQUIRED DUE TO PROCEDURAL DEFICIENCIES.
EVENT DATE: 082189 REPORT DATE: 092089 NSSS: WE TYPE: PWR

(NSIC 215323) THE PLANT WAS OPERATING IN MODE 1, POWER OPERATION, AT 100% REACTOR POWER ON 8/21/89. A WALKDOWN OF THE LOW PRESSURE EMERGENCY CORE COOLING SYSTEM (ECCS) PIPING AND ASSOCIATED PROCEDURES IDENTIFIED TWO ACCESSIBLE DISCHARGE PIPING HIGH POINT VENTS THAT WERE NOT BEING VERIFIED FULL OF WATER AT LEAST ONCE PER 31 DAYS AS REQUIRED BY TECH SPEC (TS) SURVEILLANCE REQUIREMENT 4.5.2.B.1. THIS WAS IDENTIFIED AT 1100 HOURS ON 8/21/89. THE SUBJECT VENTS WERE BEING TESTED ON A QUARTERLY BASIS IN CONTRAST TO THE REQUIRED MONTHLY FREQUENCY. THE TWO VENTS WERE TESTED, AND VERIFIED FREE OF AIR AT 1540 HOURS ON 8/21/89. THE CAUSE OF THE EVENT WAS PROCEDURAL DEFICIENCY AS PLANT PROCEDURES STATED TECH SPEC SURVEILLANCE REQUIREMENTS WERE BEING SATISFIED, WHEN IN FACT THEY WERE NOT. THE IMMEDIATE CORRECTIVE ACTION WAS TO VERIFY THE VENTS WERE FREE OF AIR, AND PLANT PROCEDURES ARE BEING REVISED TO CORRECT THE DEFICIENCIES. THERE WERE NO SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT. THERE ARE FIVE OTHER POINTS IN THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM WHICH HAVE BEEN VENTED MONTHLY. THE TWO SUBJECT VENTS (ONE PER RHR TRAIN) WERE BEING VENTED ON A QUARTERLY BASIS, AND WHEN CHECKED ON 8/21/89, NO AIR WAS DETECTED IN EITHER LINE. THIS EVENT IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(I)(B) AS A VIOLATION OF TECH SPECS ACTION 4.5.2.B.1.

[200] SHEARON HARRIS 1 DOCKET 50-400 LER 89-015
FUEL HANDLING BUILDING EQUIPMENT HATCH NOT INSTALLED AS REQUIRED DURING FUEL MOVEMENT DUE TO PROCEDURAL DEFICIENCIES.
EVENT DATE: 082789 REPORT DATE: 092689 NSSS: WE TYPE: PWR

(NSIC 215383) THE PLANT WAS OPERATING IN MODE 1, POWER OPERATION, AT 95% REACTOR POWER ON 8/27/89. PLANT PERSONNEL WERE IN THE PROCESS OF TRANSFERRING SPENT FUEL FROM THE SHIPPING CASK TO THE SPENT FUEL STORAGE POOL. FOLLOWING THE MOVEMENT OF THE FIFTH FUEL ELEMENT AT 1130 HOURS, IT WAS DISCOVERED THAT THE FUEL HANDLING BUILDING (FHB) OPERATING FLOOR EQUIPMENT HATCH WAS IN THE STORAGE LOCATION ON THE OPERATING DECK, AND NOT INSTALLED AS REQUIRED. FUEL MOVEMENT WAS IMMEDIATELY STOPPED. THE EQUIPMENT HATCH WAS THEN INSTALLED AND FUEL MOVEMENT RESUMED. SHNPP FINAL SAFETY ANALYSIS REPORT (FSAR), PAGE 9.1.4-7 ASSUMES THAT NO IRRADIATED FUEL, OUTSIDE OF SEALED CASKS, WILL BE HANDLED OR TRANSPORTED INSIDE THE FHB, UNLESS THE OEPRATING FLOOR EQUIPMENT HATCH TO THE UNLOADING AREA IS IN PLACE. THE REMOVAL OF THE HATCH COVER WOULD PREVENT THE FHB EMERGENCY EXHAUST SYSTEM FROM PERFORMING ITS INTENDED FUNCTION IN THE EVENT OF A POSTULATED FUEL HANDLING ACCIDENT. THE EVENT WAS CAUSED BY PROCEDURAL INADEQUACIES. CORRECTIVE ACTIONS WILL INCLUDE PROCEDURE REVISIONS AND PERSONNEL TRAINING. THE EVENT IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(V) AS IT RESULTED IN THE PLANT BEING IN AN UNANALYZED CONDITION.

[201] SHOREHAM DOCKET 50-322 LER 89-007
RPS ACTUATION WHEN SWITCHING POWER SUPPLY FROM "ALT" TO "NORM" FOR RPS "A" BUS.
EVENT DATE: 082389 REPORT DATE: 092289 NSSS: GE TYPE: BWR

(NSIC 215259) ON 8/23/89 AT 1503, A FULL REACTOR TRIP AND NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) ACTUATION OCCURRED WHILE AN OPERATOR WAS TRANSFERRING THE POWER SUPPLY FOR THE "A" RPS BUS FROM THE ALTERNATE FEED TRANSFORMER TO THE NORMAL SUPPLY FROM THE "A" RPS MG SET. THE REACTOR WAS DEFUELED, ALL FUEL ASSEMBLIES WERE IN THE SPENT FUEL POOL, WITH THE MODE SWITCH IN SHUTDOWN AND ALL RODS INSERTED IN THE CORE. WHILE MOVING THE POWER SOURCE SELECT SWITCH (S-1) FROM THE "ALT A" POSITION TO THE "NORM" POSITION, THE SWITCH INADVERTENTLY OVERTRAVELED BEYOND THE "NORM" POSITION, TO THE "ALT B" POSITION. THIS ACTION CAUSED THE TWO RPS BUSSES (A&B) TO BE MOMENTARILY DEENERGIZED AT THE SAME TIME, RESULTING IN A FULL REACTOR TRIP AND A FULL NSSSS ACTUATION. THE OPERATOR, IMMEDIATELY AWARE OF THE CAUSE OF THE EVENT, RESET THE TRIP AND THE TRIP LOGIC FOR THE NSSSS. PLANT MANAGEMENT WAS NOTIFIED OF THE EVENT AND THE NRC WAS NOTIFIED AT 1541 PER 10CFR50.72. THIS REPORT WILL BE DISCUSSED IN OPERATIONS NIGHT ORDERS AND WILL INSTRUCT OPERATIONS PERSONNEL TO REVIEW THE PROCEDURE, SP 23. 312.01-120 VAC RPS MG SETS, WHICH INCLUDES THE STEPS ON TRANSFERRING THE POWER SUPPLY WITH SWITCH S-1 AND APPLICABLE CAUTIONARY NOTES. IN ADDITION THIS LER WILL BE FORWARDED TO THE TRAINING DEPARTMENT TO BE INCLUDED FOR DISCUSSION DURING OPERATOR REQUALIFICATION CLASSES.

[202] SOUTH TEXAS 2 DOCKET 50-499 LER 89-018
CONTROL ROOM VENTILATION ACTUATION TO RECIRCULATION MODE DUE TO A FAILURE OF A TOXIC GAS ANALYZER.
EVENT DATE: 080289 REPORT DATE: 083189 NSSS: WE TYPE: PWR
VENDOR: FOXBORO CO., THE

(NSIC 215221) ON 8/2/89, UNIT 2 WAS IN MODE 1 AT 64% POWER. AT 0300 HOURS, THE CONTROL ROOM VENTILATION SYSTEM ACTUATED TO THE RECIRCULATION MODE AS A RESULT OF A HIGH LEVEL TRIP OF THE VINYL ACETATE CHANNEL ON A TOXIC GAS ANALYZER. THE REDUNDANT ANALYZER DID NOT ACTUATE. FURTHER INVESTIGATION DETERMINED THAT THE CAUSE OF THIS EVENT WAS A FAILURE OF AN ELECTRO-MECHANICAL POSITIONER WITHIN THE ANALYZER. THE ANALYZER HAS BEEN REPLACED.

[203] SOUTH TEXAS 2 DOCKET 50-499 LER 89-020
REACTOR TRIP DUE TO A SIMULTANEOUS TRIP OF THREE FEEDWATER PUMPS.
EVENT DATE: 082989 REPORT DATE: 092989 NSSS: WE TYPE: PWR

(NSIC 215393) ON 8/29/89, UNIT 2 WAS IN MODE 1 AT 100% POWER. AT 1400 HOURS ALL

THREE OPERATING TURBINE DRIVEN FEEDWATER PUMPS TRIPPED. THE LICENSED CONTROL ROOM OPERATOR IMMEDIATELY TRIPPED THE REACTOR IN ANTICIPATION OF LOW STEAM GENERATOR LEVEL. AN AUXILIARY FEEDWATER ACTUATION SUBSEQUENTLY OCCURRED ON LOW STEAM GENERATOR LEVEL. THE UNIT WAS STABILIZED IN MODE 3 WITH NO UNEXPECTED POST-TRIP TRANSIENTS. THE CAUSE OF THIS EVENT WAS A MOMENTARY INTERRUPTION OF CONTROL POWER TO THE FEEDWATER PUMP OVERSPEED PROTECTION CIRCUITS DUE TO THE FAILURE OF AN INVERTER. A CONTRIBUTING CAUSE WAS THE DESIGN OF THE FEEDWATER PUMP OVERSPEED PROTECTION CIRCUITS WHICH COULD NOT TOLERATE THE MOMENTARY LOSS OF CONTROL POWER WITHOUT TRIPPING THE PUMPS. THE INVERTER HAS BEEN REPAIRED AND RETURNED TO SERVICE. THE DESIGN OF THE FEEDWATER PUMP OVERSPEED PROTECTION HAS BEEN MODIFIED TO AN "ENERGIZE TO TRIP" SCHEME ON UNIT 2 AND WILL BE MODIFIED ON UNIT 1 PRIOR TO STARTUP FROM THE FIRST REFUELING OUTAGE.

[204] ST. LUCIE 2 DOCKET 50-389 LER 89-006
REMOVING DIESEL GENERATOR FROM SERVICE FOR PREVENTIVE MAINTENANCE RESULTED IN A CONDITION PROHIBITED BY TECH SPEC DUE TO PERSONNEL ERROR.
EVENT DATE: 081689 REPORT DATE: 091389 NSSS: CE TYPE: PWR
VENDOR: VALCOR ENGINEERING CORP.

(NSIC 215273) ON 8/16/89 WITH ST. LUCIE UNIT 2 IN MODE 1 AT 90% POWER, THE UNIT WAS FOUND TO BE OPERATING IN A CONDITION PROHIBITED BY PLANT'S TECH SPEC. THE 2B DIESEL GENERATOR HAD BEEN PLACED OUT OF SERVICE WITH THE 2A HYDROGEN ANALYZER ALSO OUT OF SERVICE. THIS CONDITION IS PROHIBITED BY TECH SPEC BY NOT HAVING ALL EQUIPMENT OPERABLE ON THE REMAINING OPERABLE DIESEL GENERATOR (2A). THE 2B DIESEL GENERATOR WAS PLACED OUT OF SERVICE TO PERFORM PREVENTATIVE MAINTENANCE ON THE AIR START SYSTEM. THE ROOT CAUSE OF THE EVENT WAS A COGNITIVE PERSONNEL ERROR BY A UTILITY-LICENSED OPERATOR BY NOT ADHERING TO PLANT PROCEDURE WHEN REMOVING THE 2B DIESEL GENERATOR FROM SERVICE FOR PREVENTATIVE MAINTENANCE. CORRECTIVE ACTIONS TAKEN WERE: 1) IMMEDIATELY RESTORED THE 2B DIESEL GENERATOR TO OPERABLE STATUS; 2) COUNSELED ALL INVOLVED PERSONNEL ON THE NEED TO PERFORM A MORE THROUGH REVIEW OF THE EQUIPMENT STATUS PRIOR TO TAKING SAFETY-RELATED EQUIPMENT OUT OF SERVICE; 3) RETURNED THE 2A HYDROGEN ANALYZER TO SERVICE.

[205] SUMMER 1 DOCKET 50-395 LER 89-014
WASTE GAS DECAY TANKS RELEASE WITH INADEQUATE SAMPLE.
EVENT DATE: 072389 REPORT DATE: 092189 NSSS: WE TYPE: PWR

(NSIC 215380) ON 8/23/89, SOUTH CAROLINA ELECTRIC & GAS COMPANY (SCE&G) IDENTIFIED A NONCOMPLIANCE WITH TECH SPEC 3.11.2.1, "GASEOUS EFFLUENTS," IN THAT TWO WASTE GAS STORAGE TANKS WERE RELEASED BETWEEN 7/23-24/89, WITH LOWER LIMIT OF DETECTION (LLD) VALUES GREATER THAN THAT ALLOWED BY SURVEILLANCE REQUIREMENT 4.11.2.1.2. FAILURE TO MEET THE LLD'S WAS IDENTIFIED BY HEALTH PHYSICS (HP) COUNT ROOM SUPERVISOR WHEN DISCUSSIONS WITH HIS PERSONNEL INDICATED THAT THEY WERE NOT OBTAINING AN ADEQUATE SAMPLE FOR THE ANALYSIS PRIOR TO AUTHORIZING RELEASES. ROOT CAUSE WAS PERSONNEL ERROR. HP TECHNICIAN DREW THE SAME SIZE SAMPLE AS NORMALLY USED TO DETERMINE CURIE CONTENT OF GAS STORAGE TANKS. A 1 ML SAMPLE IS DRAWN FOR THIS ANALYSIS IN ACCORDANCE WITH COUNTING PROCEDURE METHODOLOGY DUE TO EXPECTED PROBLEMS WITH DETECTOR DEAD TIME. TWO GAS TANKS HAD XE-133M ACTIVITIES OF 1.03E-2 AND 2.32E-2 MICROCI/M1 AND A XE-138 MINIMUM DETECTABLE ACTIVITY (MDA) OF 1.29E-4 AND 5.01E-4 MICROCI/ML. CALCULATIONS PERFORMED USING THE MDA OF 5.01E-4 MICROCI/ML DETERMINED THAT THERE WOULD HAVE BEEN A MAXIMUM IMPACT ON DOSES TO WHOLE BODY (3.32E-3 MREM/YR), SKIN (5.37E-3 MREM/YR), AIR DOSE (GAMMA)(4-13E-6 MRAD), AND AIR DOSE (BETA) (2.13E-6 MRAD). CORRECTIVE ACTIONS INVOLVE COMPUTER SOFTWARE CHANGES TO VERIFY CORRECT SAMPLE VOLUME AND TRAINING OF HP COUNT ROOM PERSON.

[206] SUMMER 1 DOCKET 50-395 LER 89-015
MANUAL REACTOR TRIP DUE TO PRESSURIZER SAFETY VALVE FAILURE.
EVENT DATE: 082589 REPORT DATE: 092089 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE & GAGE CO.

(NSIC 215322) AT 1000 HOURS ON 8/25/89, THE "A" PRESSURIZER SAFETY VALVE BODY INLET TEMPERATURE INCREASED TO GREATER THAN 450F AND A PLANT SHUT DOWN WAS INITIATED. SHORTLY AFTER THE LOAD REDUCTION WAS STARTED, THE "A" PRESSURIZER SAFETY VALVE OPENED AT A SYSTEM PRESSURE OF APPROXIMATELY 2260 PSIG. AT 1003 HOURS, THE ACOUSTIC LEAK MONITOR ALARM, THE REACTOR COOLANT SYSTEM (RCS) BEGAN TO RAPIDLY DEPRESSURIZE AND AT APPROXIMATELY 1004 HOURS THE SHIFT SUPERVISOR DIRECTED A MANUAL REACTOR TRIP. THE PRESSURIZER SAFETY VALVE RESEATED PRIOR TO REACHING THE SAFETY INJECTION SETPOINT OF 1850 PSIG. ALL PLANT PARAMETERS RECOVERED TO THEIR EXPECTED POST TRIP VALUES EXCEPT RCS PRESSURE WHICH WAS CONTROLLED AROUND 2000 PSIG TO AVOID LIFTING THE SAFETY VALVE AGAIN. THE PLANT WAS TAKEN TO COLD SHUTDOWN, THE "A" PRESSURIZER SAFETY VALVE REPLACED, AND THE REACTOR WAS RESTARTED AT 0635 HOURS ON 9/1/89. LER 89-011, DATED 6/27/89, DOCUMENTS A SIMILAR EVENT INVOLVING "C" PRESSURIZER SAFETY VALVE. IT WAS IDENTIFIED IN THE REPORT THAT THE LICENSEE IS STILL INVESTIGATING THE EVENT AND THE FINDINGS WILL BE DISCUSSED IN A SUPPLEMENT REPORT. AS OF THIS DATE, THE LICENSEE CONTINUES TO INVESTIGATE THESE EVENTS AND THE SUPPLEMENT REPORT WILL BE SUBMITTED DETAILING THE FINDINGS. NOTE: EXPECTED SUBMISSION DATE 12/20/89 AS IDENTIFIED IN LER 89-011.

[207] SURRY 1 DOCKET 50-280 LER 89-032
EMERGENCY CONDENSER STORAGE TANK BELOW TECH SPEC MINIMUM REQUIRED FOR AUX. FEEDWATER CROSS CONNECT AVAILABILITY DUE TO PERSONNEL ERROR.
EVENT DATE: 072789 REPORT DATE: 082589 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 215121) ON JULY 27, 1989 WITH UNIT 1 AT 100% POWER AND UNIT 2 AT COLD SHUTDOWN, AT 1822 HOURS AN OPERATOR DISCOVERED THAT THE UNIT 2 ABOVE GROUND EMERGENCY CONDENSATE STORAGE TANK (ECST) WAS BELOW THE MINIMUM LEVEL REQUIRED TO ENSURE AUXILIARY FEEDWATER (AFW) CROSS CONNECT CAPABILITY FOR UNIT 1. THIS OCCURRED AS THE RESULT OF AN EVOLUTION INVOLVING THE TRANSFER OF THE CONTENTS OF THE ECST TO ANOTHER TANK IN ORDER TO FILL THE SECOND TANK. THIS EVENT IS CONTRARY TO TECHNICAL SPECIFICATION (T.S.) 3.6.8.2. OPERATIONS PERSONNEL INVOLVED WITH THE TRANSFER OPERATION FAILED TO RECOGNIZE THAT A T.S. MINIMUM OF 60,000 GALLONS OF WATER WAS REQUIRED TO BE MAINTAINED IN THE UNIT 2 ECST FOR UNIT 1 OPERATION. THE PERSONNEL INVOLVED WERE DISCIPLINED. ALSO, DEFICIENCIES IN EXISTING PROCEDURES AND IN PROCEDURE USAGE WERE IDENTIFIED. CHANGES ARE BEING MADE TO ENHANCE CERTAIN OPERATING AND SURVEILLANCE PROCEDURES. IN ADDITION, RECOMMENDATIONS FOR ADDITIONAL PROCEDURE CHANGES AND CONTROL ROOM INDICATION AND ANNUNCIATION MODIFICATIONS ARE BEING EVALUATED FOR APPROPRIATE ACTION.

[208] SURRY 1 DOCKET 50-280 LER 89-035
AUTOMATIC START OF AN AUXILIARY VENTILATION SYSTEM FAN DUE TO AN INCORRECTLY LANDED LEAD.
EVENT DATE: 081389 REPORT DATE: 090889 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 215350) ON 8/13/89 AT 1532 HOURS, WITH UNIT 1 AT 100% POWER AND UNIT 2 IN COLD SHUTDOWN, ONE OF THE AUXILIARY VENTILATION SYSTEM FILTERED EXHAUST FANS AUTOMATICALLY RESTARTED AFTER IT HAD BEEN STOPPED. THE FAN IS DESIGNED TO AUTOMATICALLY START ON A SAFETY INJECTION (SI) SIGNAL. NO ACTUAL SI SIGNAL WAS PRESENT AT THE TIME. A FOUR HOUR NON-EMERGENCY REPORT WAS MADE TO THE NRC PER 10CFR50.72 OF AN UNPLANNED ENGINEERED SAFETY FEATURES COMPONENT ACTUATION. THE CAUSE OF THE EVENT WAS THE IMPROPER LANDING OF A PREVIOUSLY LIFTED LEAD ON A

PRESSURE SWITCH. THE INDIVIDUALS INVOLVED IN THE EVENT WERE REINSTRUCTED. A ROOT CAUSE EVALUATION WAS PERFORMED AND APPROPRIATE ACTIONS WILL BE IMPLEMENTED.

[209] SURRY 1 DOCKET 50-280 LER 89-027
CLOSURE OF CONTAINMENT ISOLATION VALVE DUE TO CONTAINMENT GASEOUS RADIATION MONITOR ALARM.
EVENT DATE: 081589 REPORT DATE: 090889 NSSS: WE TYPE: PWR

(NSIC 215349) ON 8/15/89 AT 0605 HOURS, WITH UNIT 1 AT 100% POWER, THE CONTAINMENT GASEOUS RADIATION MONITOR MOMENTARILY INCREASED ABOVE THE ALARM SETPOINT. THE ALARM RESULTED IN THE CLOSURE OF THE CONTAINMENT INSTRUMENT AIR (IA) COMPRESSOR'S NORMAL SUCTION VALVES FROM CONTAINMENT AND THE OPENING OF THE ALTERNATE SUCTION VALVE OUTSIDE CONTAINMENT. THE CONTAINMENT VALVES ALSO CLOSE UPON AN ENGINEERED SAFETY FEATURES (ESF) SIGNAL. THIS EVENT IS BEING REPORTED AS AN UNPLANNED ESF COMPONENT ACTUATION. A FOUR HOUR NON-EMERGENCY REPORT WAS MADE TO THE NUCLEAR REGULATORY COMMISION PER 10CFR50.72. THE ALARM WAS RESET, AND THE IA COMPRESSOR SUCTION VALVES WERE REALIGNED TO THEIR NORMAL POSITION. THE BASIS FOR THE ALERT AND ALARM SETPOINTS OF THE CONTAINMENT PARTICULATE AND GASEOUS MONITOR ARE BEING EVALUATED.

[210] SURRY 1 DOCKET 50-280 LER 89-036
REACTOR PROTECTION PERMISSIVE CIRCUIT P-10 SUSPECTED OF NOT BEING TESTED PER TECH SPEC SURVEILLANCE REQUIREMENTS.
EVENT DATE: 081589 REPORT DATE: 091489 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 215247) ON 8/15/89 WITH UNIT 1 AT 100% POWER AND UNIT 2 AT COLD SHUTDOWN (CSD), AN ONGOING SYSTEMATIC REVIEW OF TECH SPEC SURVEILLANCE REQUIREMENTS IDENTIFIED THAT THE TESTING OF THE REACTOR PROTECTION SYSTEM (RPS) P-10 PERMISSIVE CIRCUIT MAY NOT HAVE BEEN PERFORMED PRIOR TO EACH REACTOR STARTUP AS SPECIFIED. THE REACTOR PROTECTION FEATURES ASSOCIATED WITH THIS PERMISSIVE WERE DECLARED INOPERABLE AND A SIX HOUR CLOCK TO HOT SHUTDOWN WAS ENTERED. A FOUR HOUR NOTIFICATION PER 10CFR50.72 WAS MADE TO THE NUCLEAR REGULATORY COMMISSION. A JUSTIFICATION FOR CONTINUED OPERATION WAS WRITTEN AND APPROVED AND THE SIX HOUR CLOCK WAS EXITED. AN EXISTING PROCEDURE WAS DEVIATED AND PERFORMED TO VERIFY THE P-10 CIRCUIT WAS FULLY OPERABLE. SUBSEQUENT EXAMINATION OF THE EXISTING TESTING PROCEDURES DETERMINED THAT THE SURVEILLANCE TEST WAS ADEQUATELY PERFORMED. HOWEVER, THE PROCEDURES ARE BEING REVISED TO ENHANCE DOCUMENTATION OF THE REQUIRED TESTING.

[211] SURRY 2 DOCKET 50-281 LER 89-003
"B" CHARGING PUMP AUTOMATICALLY STARTED AFTER CONTROL SWITCH DISENGAGED FROM THE PULL-TO-LOCK POSITION.
EVENT DATE: 081389 REPORT DATE: 090889 NSSS: WE TYPE: PWR

(NSIC 215351) ON 8/13/89 AT 2208 HOURS, WITH UNIT 2 IN COLD SHUTDOWN (CSD), THE UNIT 2 "B" CHARGING PUMP AUTOMATICALLY STARTED WHEN AN OPERATOR TOUCHED ITS CONTROL SWITCH, AND IT DISENGAGED FROM THE PULL-TO-LOCK (PTL) POSITION. THE CAUSE OF THE EVENT WAS DUE TO THE OPERATOR NOT ADEQUATELY ENGAGING THE PUMPS CONTROL SWITCH IN THE PTL POSITION PRIOR TO THE EVENT. THE CONTROL SWITCH WAS IMMEDIATELY RETURNED TO THE PTL POSITION, THEREBY STOPPING THE PUMP. PROPER OPERATION OF THE CONTROL SWITCH WAS VERIFIED. THE SHIFT SUPERVISOR AND OPERATOR INVOLVED DISCUSSED THE EVENT. THE NEED TO VERIFY PROPER CONTROL SWITCH ENGAGEMENT WAS EMPHASIZED. THIS EVENT WAS REPORTED TO THE NRC ON 8/14/89 FOR INFORMATION ONLY.

[212] SURRY 2 DOCKET 50-281 LER 89-004
UNPLANNED ESF COMPONENT ACTUATION, SPURIOUS RECIRCULATION MODE TRANSFER INITIATION CAUSED BY INADVERTENTLY ENERGIZING RELAY WHILE PLACING ELECTRICAL JUMPER.
EVENT DATE: 081889 REPORT DATE: 091589 NSSS: WE TYPE: PWR

(NSIC 215248) ON 8/18/89 WITH UNIT 2 AT COLD SHUTDOWN (CSD) AT 1010 HOURS, 3 MOTOR OPERATED VALVES (MOVS) IN THE SAFETY INJECTION (SI) SYSTEM ACTUATED. THE VALVES ARE DESIGNED TO REPOSITION WHEN A RECIRCULATION MODE TRANSFER (RMT) SIGNAL IS GENERATED UPON A LOW LEVEL CONDITION IN THE RWST. HOWEVER, NO LOW LEVEL EXISTED AT THE TIME. THIS SPURIOUS RMT ACTUATION CONSTITUTES AN UNPLANNED ENGINEERED SAFETY FEATURES (ESF) COMPONENT ACTUATION AND WAS REPORTED TO THE NUCLEAR REGULATORY COMMISSION PER 10CFR50.72(B)(2)(II). AN ELECTRICIAN INADVERTENTLY ENERGIZED A RELAY THAT ACTUATED THE VALVES WHILE PLACING A JUMPER ON AN ADJACENT TERMINAL IN SUPPORT OF AN ENGINEERING WORK REQUEST. A HUMAN PERFORMANCE EVALUATION SYSTEM (HPES) INVESTIGATION WAS CONDUCTED AND A REPORT PREPARED. RECOMMENDATIONS MADE IN THE REPORT WILL BE EVALUATED AND APPROPRIATE ACTIONS TAKEN.

[213] SUSQUEHANNA 1 DOCKET 50-387 LER 89-021
MIS-LABELED DAMPER POSITION RESULTED IN ALIGNMENT PROHIBITED BY TECH SPECS.
EVENT DATE: 072689 REPORT DATE: 082589 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)

(NSIC 215143) ON JULY, 26, 1989, WITH BOTH UNIT 1 AND UNIT 2 IN CONDITION 1 AT 100% POWER, PLANT PERSONNEL, CONSISTING OF A SYSTEM ENGINEER AND CONTRACT TECHNICIANS, WERE IN PROCESS OF RECONFIRMING THE REACTOR BUILDING HEATING, VENTILATING AND AIR CONDITIONING (HVAC) SYSTEM AIR FLOW BALANCE. DURING THIS EVOLUTION, THEY DISCOVERED THAT A MANUAL ISOLATION DAMPER, XD-17513, HAD OPEN AND CLOSED POSITION LABELING REVERSED FROM ACTUAL DAMPER POSITION. THIS DAMPER IS REQUIRED TO BE CLOSED TO MAINTAIN ISOLATION OF THE RX BLDG ZONE III (COMMON REFUELING FLOOR) HVAC SYSTEM FROM OUTSIDE ATMOSPHERE, AND FROM ZONE 1 (UNIT 1 RX BLDG) DURING SEVERAL PLANT EVOLUTIONS, INCLUDING FUEL AND MATERIAL RECEIPT. DURING ITS INSTALLATION IN JANUARY 1984, THE DAMPER OPEN AND CLOSED POSITIONS WERE NOT ADEQUATELY LABELED. TEMPORARY LABELING AFFIXED, AS WELL AS PERMANENT LABELING APPLIED IN SEPTEMBER 1987, WAS OPPOSITE TO ACTUAL DAMPER POSITION. FOLLOWING DISCOVERY, ACTUAL DAMPER POSITION WAS VERIFIED BY OPENING THE DUCT AND VISUALLY INSPECTING THE DAMPER BLADES. THE POSITION LABELS WERE THEN CORRECTED. NUCLEAR PLANT ENGINEERING HAS EVALUATED THE POTENTIAL EFFECTS OF HAVING XD-17513 IN THE INCORRECT POSITION, ON BOTH MAIN CONTROL ROOM AND OFFSITE DOSE RATES CALCULATED DURING POSTULATED ACCIDENT SCENARIOS. THIS EVALUATION SHOWS THAT DOSE RATES WOULD REMAIN LESS THAN THE CONSERVATIVE LIMITS DEFINED IN 10CFR100 AND 10CFR50 APPENDIX A.

[214] SUSQUEHANNA 1 DOCKET 50-387 LER 89-022
ESF ACTUATION DUE TO SPURIOUS AUTO-TRANSFER OF HPCI SUCTION VALVES.
EVENT DATE: 080989 REPORT DATE: 090589 NSSS: GE TYPE: BWR

(NSIC 215173) ON 8/9/89 WITH UNIT 1 IN CONDITION 1 AT 100% POWER, AN AUTO-SWAP OF THE HPCI SUCTION VALVES OCCURRED DURING PERFORMANCE OF SURVEILLANCE TEST SI-152-308. THE AUTO-SWAP CONSISTED OF A REALIGNMENT OF THE HPCI SUCTION SOURCE FROM THE CONDENSATE STORAGE TANK TO THE SUPPRESSION POOL. NO HPCI INITIATION OR INJECTION OCCURRED AND NO OTHER SYSTEM COMPONENTS WERE AFFECTED. AN I&C TECHNICIAN INADVERTENTLY PROCEEDED FROM STEP 6.1.13 TO STEP 6.1.15, WITHOUT PERFORMING STEP 6.1.14 OF THE PROCEDURE. WHEN THE TECHNICIAN PERFORMED STEP 6.1.15, A SIGNAL FOR THE AUTO-SWAP OF THE HPCI SUCTION SOURCE WAS GENERATED. THE EVENT HAS BEEN DETERMINED TO BE REPORTABLE PER 10CFR50.73(A)(2)(IV) BECAUSE THE AUTO TRANSFER OF HPCI SUCTION VALVES IS AN ENGINEERED SAFETY FEATURE (ESF) OF THE HPCI SYSTEM. THE SPURIOUS ACTUATION OF THIS LOGIC AND REPOSITIONING OF THE

SUCTION VALVES CONSTITUTES AN UNPLANNED ESF ACTUATION. ALL ANNUNCIATION AND EQUIPMENT WORKED PER DESIGN. THE HPCI SYSTEM WAS RESTORED TO NORMAL STANDBY ALIGNMENT FOLLOWING THE INCIDENT. A DISCUSSION OF THIS EVENT WAS CONDUCTED FOR I&C PERSONNEL CENTERING ON THE FACTORS WHICH LED TO THE CAUSE OF THIS EVENT AND THE STEPS WHICH SHOULD BE TAKEN TO PREVENT RECURRENCE.

[215] SUSQUEHANNA 2 DOCKET 50-388 LER 89-007
PRIMARY CONTAINMENT ISOLATION VALVE CLOSURE DUE TO PERSONNEL ERROR.
EVENT DATE: 082989 REPORT DATE: 092889 NSSS: GE TYPE: BWR

(NSIC 215378) ON 8/29/89 AT 0410 HOURS WITH UNIT 2 OPERATING IN CONDITION 1 AT 100% POWER, AN ESF ACTUATION OCCURRED WHEN THE INBOARD AND OUTBOARD PRIMARY CONTAINMENT ISOLATION VALVES FOR THE "B" LOOP OF THE CONTAINMENT ATMOSPHERE CONTROL (CAC) SYSTEM ISOLATED. NUCLEAR PLANT OPERATORS WHO WERE SUPPORTING SYSTEM PREPARATION FOR LOCAL LEAK RATE TESTING OF PRIMARY CONTAINMENT PENETRATIONS ASSOCIATED WITH THE CONTAINMENT PURGE VALVES REMOVED FUSES IN THE WRONG PANEL RESULTING IN THE UNPLANNED CLOSURE OF THE CAC VALVES. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR CONSISTING OF INADEQUATE SELF CHECKING AND VERIFICATION. THE NPO'S REMOVED FUSES F14 AND F15 IN PANEL 2C661B3 INSTEAD OF IN PANEL 2C661A3. THE EVENT WAS REPORTABLE PER 10CFR50.73(A)(2)(IV) IN THAT THE CLOSURE OF THE CAC SYSTEM ISOLATION VALVES CONSTITUTED AN UNPLANNED ESF ACTUATION. THERE WERE NO SAFETY CONSEQUENCES OR COMPROMISE TO THE PUBLIC HEALTH OR SAFETY AS A RESULT OF THIS EVENT. THE ISOLATION VALVES ARE DESIGNED TO CLOSE FOLLOWING A DESIGN BASIS LOCA. SINCE THE VALVES ACTUATED TO THE CLOSED CONDITION, THEY WERE FULFILLING THEIR DESIGN ACCIDENT FUNCTION. OPERATIONS CONFIRMED PROPER PLANT RESPONSE AFTER WHICH FUSES F14 AND F15 WERE REINSTALLED IN PANEL 2C661B3. THE "B" LOOP OF THE CAC SYSTEM WAS RESTORED TO A PROPER LINEUP.

[216] THREE MILE ISLAND 2 DOCKET 50-320 LER 89-004
FAILURE OF A 4160/480V TRANSFORMER DUE TO GROUND IN PRIMARY SIDE OF TRANSFORMER.
EVENT DATE: 072689 REPORT DATE: 082489 NSSS: BW TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215131) AT 1658 ON TUESDAY, 7/25/89, A FAILURE OF THE TRANSFORMER SUPPLYING POWER TO UNIT SUBSTATION (USS) 2-46 RESULTED IN ENTERING THE ACTION STATEMENT OF TECH SPEC (TECH SPEC) 3.8.2.1.1. USS-2-46 WAS RE-ENERGIZED BY CROSS-TIEING TO USS-2-36 AT 1720. ELECTRICAL MAINTENANCE INVESTIGATED THE OCCURRENCE AND FOUND IT TO BE CAUSED BY A GROUND IN THE PRIMARY SIDE OF BUS 2-46 TRANSFORMER. DUE TO THIS FAULT, BUS 2-46 TRANSFORMER COULD NOT BE RE-ENERGIZED AND THE 8-HOUR TIMECLOCK OF THE TECH SPEC ACTION STATEMENT WAS EXCEEDED AT 0058 ON 7/26/89. THIS EVENT IS REPORTABLE PURSUANT TO 10 CFR 50.73(A)(2)(I)(B) DUE TO A CONDITION PROHIBITED BY THE PLANT'S TECH SPECS. THE ROOT CAUSE OF THIS EVENT WAS AN INTERNAL FAULT IN THE PRIMARY SIDE OF BUS 2-46 TRANSFORMER. THE APPARENT CAUSE WAS DEGRADATION OF THE INSULATION ON THE WINDINGS ON THE THE PRIMARY SIDE RESULTING IN GROUNDING OF THE TRANSFORMER RENDERING IT INOPERABLE. THE IMMEDIATE CORRECTIVE ACTION WAS TO RESTORE POWER TO BUS 2-46 BY EFFECTING A CROSS-TIE WITH BUS 2-36. IN THE LONG-TERM, GPU NUCLEAR IS SEEKING RELIEF FROM THE RECOVERY OPERATIONS PLAN REQUIREMENT OF SECTION 4.8.2.1.1 RELATING TO BUSSES 2-35, 2-36, 2-45, AND 2-46 TO ALLOW TIE BREAKERS TO REMAIN CLOSED IN THE EVENT THAT REDUNDANT BUS PAIRS CANNOT BE RE-ENERGIZED WITHIN THE 8-HOUR TIMECLOCK OF TECH SPEC 3.8.2.1.1.

[217] THREE MILE ISLAND 2 DOCKET 50-320 LER 89-005
FAILURE TO COMPLY WITH TECH SPEC 3.8.2.2.1 DUE TO LOW BATTERY VOLTAGE.
EVENT DATE: 083089 REPORT DATE: 092289 NSSS: BW TYPE: PWR
VENDOR: GOULDS INC. (INDUSTRIAL BATTERY DIV)

(NSIC 215258) ON WEDNESDAY, 8/30/89, OPERATIONS PERSONNEL OBSERVED THAT THE

VOLTAGE OF CELL #71 OF STATION STORAGE BATTERY 2-25B HAD DECREASED MORE THAN 0.1 VOLTS BELOW THE BASELINE VOLTAGE (I.E., 2.265 VOLTS) REQUIRED BY RECOVERY OPERATIONS PLAN SECTION 4.8.2.2.2.B.1. THIS CELL WAS OBSERVED TO HAVE HAD A SIMILAR VOLTAGE DROP ON 8/9/89, DURING THE PERFORMANCE OF QUARTERLY TECH SPECS SURVEILLANCE PROCEDURE 4223-SUR-3734.01; THE BATTERY WAS RECHARGED AT THAT TIME. THUS, OPERATIONS PERSONNEL DECLARED BATTERY BANK 2-25B OUT OF SERVICE AT 1100 HOURS ON 8/30/89, IN ORDER TO REPLACE CELL #71. THIS DECLARATION PLACED THE UNIT IN THE ACTION STATEMENT OF TECH SPEC 3.8.2.2.1, WHICH HAS A 2-HOUR TIMECLOCK. AT 1300 HOURS ON 8/30/89, THE TIMECLOCK WAS EXCEEDED. THUS, THIS EVENT IS REPORTABLE PURSUANT TO 10 CFR 50.73(A)(2)(I)(B). STATION STORAGE BATTERY BANK 2-25B WAS DECLARED OPERABLE AT 1530 HOURS ON 8/30/89, FOLLOWING THE REPLACEMENT OF CELL #71 WITH A SPARE CELL AND PERFORMANCE OF THE APPLICABLE TECH SPEC SURVEILLANCE PROCEDURES. THE ROOT CAUSE OF THIS EVENT WAS NORMAL DEGRADATION OF CELL #71 OVER THE LIFE CYCLE OF THE BATTERY CELL. TREND ANALYSIS OF THE VOLTAGE OF BATTERY CELLS WILL CONTINUE TO BE PERFORMED. NO FURTHER CORRECTIVE ACTIONS ARE CONSIDERED NECESSARY.

[218] TROJAN DOCKET 50-344 LER 89-006 REV 01
UPDATE ON STEAM DUMP VALVE FAILURE CAUSED ENGINEERED SAFETY FEATURE ACTUATION.
EVENT DATE: 040689 REPORT DATE: 083189 NSSS: WE TYPE: PWR
VENDOR: COPES-VULCAN, INC.

(NSIC 215166) AT 0024 HOURS ON 4/6/89, THE PLANT WAS AT 6% POWER IN MODE 1, IN THE PROCESS OF A PLANNED SHUTDOWN. WHEN THE STEAM DUMP VALVES WERE PARTIALLY OPENED AS A PART OF THE PLANNED EVOLUTION, ONE OF THE VALVES WENT TO THE FULL OPEN POSITION RATHER THAN MODULATING FLOW. SHORTLY AFTERWARD, THE SAME VALVE FULLY SHUT DUE TO LOSS OF INSTRUMENT AIR WHEN THE INSTRUMENT AIRLINE LEADING TO THE VALVE RUPTURED. THIS SEQUENCE OF EVENTS RESULTED IN AN INCREASE IN STEAM GENERATOR LEVEL IN THE "C" STEAM GENERATOR DUE TO EXCESSIVE STEAM FLOW. THE STEAM GENERATOR REACHED THE HIGH-HIGH ALARM SETPOINT, RESULTING IN FEEDWATER ISOLATION, AUXILIARY FEEDWATER PUMP START AND STEAM GENERATOR BLOWDOWN ISOLATION. THE FAILED VALVE WAS QUARANTINED AND PLANT SHUTDOWN RECOMMENCED. WHILE CONTINUING THE PLANT SHUTDOWN, BANK "A" SHUTDOWN RODS FAILED TO DRIVE IN PROPERLY. THERE WAS NO NEED TO WITHDRAW THE RODS, SO THE REACTOR WAS MANUALLY TRIPPED. THE CAUSE OF THE EVENT WAS A BROKEN WIRE THAT ENERGIZED THE "TRIP OPEN" SOLENOID IN THE VALVE CONTROL CIRCUITRY AND OPENED THE VALVE. A HIGH STRESS-LOW CYCLE FATIGUE FAILURE OF THE INSTRUMENT AIRLINE SHUT THE VALVE. THE RODS FAILED TO MOVE IN DUE TO A THERMAL CYCLE/AGE RELATED FAILURE OF A FUSE. CORRECTIVE ACTIONS WERE TO REPAIR THE BROKEN WIRE, OVERHAUL THE AFFECTED STEAM DUMP VALVES MECHANICALLY AND ELECTRICALLY, AND REPLACE APPLICABLE AIRLINES WITH A FLEXIBLE TUBING.

[219] TROJAN DOCKET 50-344 LER 89-012 REV 01
UPDATE ON CONTROL ROOM EMERGENCY VENTILATION HANGER DESIGN REQUIREMENT NOT MET.
EVENT DATE: 060289 REPORT DATE: 092089 NSSS: WE TYPE: PWR

(NSIC 215316) ON JUNE 2, 1989 THE PLANT WAS IN MODE 5 (COLD SHUTDOWN), WITH REACTOR COOLANT SYSTEM CONDITIONS OF 97F AND ATMOSPHERIC PRESSURE. TO ANSWER A NRC QUESTION ON THE CONTROL ROOM EMERGENCY VENTILATION (CB-1) SYSTEM, A COMPARISON OF ACTUAL INSTALLED CONDITIONS TO DESIGN DRAWINGS WAS DONE PRIOR TO PERFORMING A CALCULATION. IT WAS DETERMINED THAT SEVERAL HANGERS DID NOT MATCH THE DETAILS OF THE DESIGN DRAWINGS. IMMEDIATE CORRECTIVE ACTION WAS TO DECLARE BOTH TRAINS OF CB-1 INOPERABLE PER TROJAN TECH SPEC 3.7.6.1, "CONTROL ROOM EMERGENCY VENTILATION". AN EVALUATION DETERMINED THAT THE EXISTING HANGER CONFIGURATION WAS OUTSIDE THE ASSUMPTIONS USED IN THE SEISMIC ANALYSIS OF CB-1 DUCTS. INTERIM CORRECTIVE ACTIONS WERE TO PLACE RESTRICTIONS ON ENTRY INTO MODE 4 (HOT SHUTDOWN), AND THE PERFORMANCE OF CORE ALTERATIONS OR POSITIVE REACTIVITY ADDITIONS. AN INSPECTION OF OTHER SAFETY-RELATED VENTILATION SYSTEMS DETERMINED THAT THE SAME PROBLEM EXISTED IN THE EMERGENCY DIESEL GENERATOR ROOMS AND

HYDROGEN MIXING VENTILATION SYSTEMS. PERMANENT CORRECTIVE ACTION WAS TO RESTORE THE DESIGN MARGIN SPECIFIED IN THE FINAL SAFETY ANALYSIS REPORT FOR THE DUCT SUPPORTS BY REPAIR OR MODIFICATIONS. THE CAUSE OF THIS EVENT WAS INADEQUATE CONSTRUCTION WORK CONTROLS BY THE HEATING, VENTILATION, AND AIR-CONDITIONING DESIGN/INSTALLATION CONTRACTOR.

[220] TROJAN DOCKET 50-344 LER 89-017
REACTOR TRIP ON OVER TEMPERATURE DELTA TEMPERATURE SIGNAL.
EVENT DATE: 080989 REPORT DATE: 090889 NSSS: WE TYPE: PWR
VENDOR: CROUSE NUCLEAR SERVICE INC.

(NSIC 215267) ON 8/9/89 AT 1220 HOURS THE PLANT WAS OPERATING AT 50% POWER WHEN A REACTOR TRIP SIGNAL WAS GENERATED. THE REACTOR TRIPPED ON RECEIVING A ONE OUT OF THREE LOGIC FOR OVER TEMPERATURE DELTA TEMPERATURE (OT DELTA T). THE OTHER SIGNAL COMPLETING WHAT IS NORMALLY A TWO OUT OF FOUR LOGIC WAS ALREADY BEING GENERATED IN CHANNEL 3 OF THE OT DELTA T LOGIC DUE TO THE PERFORMANCE OF PERIODIC INSTRUMENTATION AND CONTROL TEST 11-1, "NUCLEAR INSTRUMENTATION, POWER RANGE". THE TRIP WAS CAUSED BY RECEIPT OF A SPURIOUS SIGNAL, APPARENTLY FROM CHANNEL 4 OF THE OT DELTA T CIRCUIT. THE SIGNAL WAS CONSIDERED SPURIOUS BECAUSE REACTOR COOLANT SYSTEM CONDITIONS WERE ACCEPTABLE WHEN THE TRIP OCCURRED. A COMPREHENSIVE INVESTIGATION WAS CONDUCTED WITH NO EXACT CAUSE IDENTIFIED. A CONTRIBUTING FACTOR WAS A LOWER OT DELTA T SETPOINT THAN IN PREVIOUS OPERATING CYCLES. IMMEDIATE CORRECTIVE ACTIONS PRIOR TO RETURN TO POWER INCLUDED REPLACEMENT OF THREE CHANNEL 4 OT DELTA T MODULES (EVEN THOUGH NO INDICATION OF FAILURE EXISTED) AND PERFORMANCE OF PICTS FOR INPUTS INTO ALL FOUR OF THE OT DELTA T CHANNELS. CORRECTIVE ACTION RELATED TO CONTINUING SPURIOUS ALARMS ON CHANNEL 4 OF OT DELTA T INCLUDES RECALIBRATING PROCESS INPUTS.

[221] TURKEY POINT 3 DOCKET 50-250 LER 88-021 REV 01
UPDATE ON PRESSURIZER POWER OPERATED RELIEF VALVE STROKE TIME EXCEEDS DESIGN BASIS AS SPECIFIED IN TECH SPEC BASIS SAFETY EVALUATION.
EVENT DATE: 091388 REPORT DATE: 090589 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)
VENDOR: COPES-VULCAN, INC.

(NSIC 215201) ON 9/13/88, WITH BOTH UNITS AT 100% POWER, IT WAS DETERMINED THAT THE PRESSURIZER POWER OPERATED RELIEF VALVES' (PORV) TIME TO OPEN EXCEEDED THE DESIGN BASIS VALUE FOR TECH SPEC 3.15 OF 2.0 SECONDS FOR THE MASS INPUT CASE AND 3.0 SECONDS FOR THE HEAT INPUT CASE. THE COLD OVERPRESSURE MITIGATING SYSTEM WAS DESIGNED TO PROVIDE PROTECTION FOR THE INADVERTENT START OF TWO CHARGING PUMPS WITH A LOSS OF LETDOWN, OR THE START OF A SAFETY INJECTION PUMP (SI) AND ITS INJECTION INTO A WATER SOLID RCS, OR THE START OF AN IDLE REACTOR COOLANT PUMP (RCP) WITH THE SECONDARY WATER TEMPERATURE 50F ABOVE THE RCS COLD LEG TEMPERATURE. TEST DATA SINCE THE INITIATION OF IST OF THE PORVS IN 1984 INDICATES OPENING TIMES BETWEEN 2 AND APPROXIMATELY 6 SECONDS. THE REASON FOR THESE OPENING TIMES IS ATTRIBUTED TO UNDERSIZED AIR AND NITROGEN BACKUP SUPPLY LINES TO THE PORV ACTUATORS. THE CAUSE OF THIS EVENT WAS AN INADEQUATE DESIGN PROCESS WHICH DID NOT ASSURE THAT THE DESIGN BASIS OPENING TIME FOR THE PORVS COULD BE MET. UNTIL THE SITUATION IS RESOLVED, THE SI PUMPS WILL BE ISOLATED AT RCS TEMPERATURES LESS THAN 380F BY MEANS OF TWO CLOSED VALVES. PROCEDURES WERE REVISED TO PREVENT A THIRD CHARGING PUMP FROM OPERATING WITH RCS TEMPERATURES LESS THAN 285F, AND TO MAINTAIN A PRESSURIZER BUBBLE WITH RCS TEMPERATURES GREATER THAN 200F.

[222] TURKEY POINT 3 DOCKET 50-250 LER 88-026 REV 02
UPDATE ON UNITS 3 AND 4 OUTSIDE THE FSAR DESIGN BASIS WITH REGARD TO HURRICANE FLOOD PROTECTION.
EVENT DATE: 110788 REPORT DATE: 090889 NSSS: WE TYPE: PWR

OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)

(NSIC 215089) ON 11/7/88, AT APPROXIMATELY 1900, WITH UNIT 3 IN COLD SHUTDOWN AND UNIT 4 DEFUELED, IT WAS DETERMINED THAT UNITS 3 AND 4 WERE OUTSIDE THEIR DESIGN BASIS WITH REGARD TO HURRICANE FLOOD PROTECTION. THE FINAL SAFETY ANALYSIS REPORT STATES, "THE UNIT IS DESIGNED FOR A HURRICANE TIDE TO AN ELEVATION OF +20 FEET WITH WAVE RUN-UP TO AN ELEVATION OF 22.5 FEET ON THE EAST SIDE OF THE UNIT." THE FOLLOWING CONDITIONS WERE IDENTIFIED BY A QUALITY ASSURANCE AUDITOR: 1) THE DIESEL OIL TRANSFER PUMPS ARE MOUNTED AT ELEVATION 19.0 FEET WITHOUT FLOOD PROTECTION. 2) A SECTION OF A FLOOD PROTECTION WALL, APPROXIMATELY 8 FEET IN LENGTH, BETWEEN THE EMERGENCY DIESEL GENERATOR (EDG) BUILDING AND THE UNIT 3 SWITCHGEAR ENCLOSURE HAS BEEN TEMPORARILY REMOVED AS A PORTION OF A PLANT MODIFICATION. 3) THE STOPLOGS ON THE EAST FACE OF THE AUX. BLDG PROVIDE PROTECTION ONLY TO ELEVATION 20 FEET. ITEMS 1 AND 2 ABOVE WERE CAUSED BY DESIGN ERROR. ITEM 3 WAS CAUSED BY THE LACK OF PLANT DRAWINGS THAT CLEARLY IDENTIFY STOP LOG DETAILS. AS AN INTERIM CORRECTIVE ACTION IN THE EVENT OF A HURRICANE WARNING, A TEMPORARY FLOOD PROTECTION DIKE WILL BE ERECTED USING SANDBAGS AND POLYETHYLENE SHEET. AS LONG TERM CORRECTIVE ACTIONS, PLANT MODIFICATIONS WILL BE PERFORMED TO PROVIDE FLOOD PROTECTION FOR THE ABOVE CONCERNS.

[223] TURKEY POINT 3 DOCKET 50-250 LER 89-012
SOURCE SURVEILLANCES REQUIRED BY TECH SPECS 3.11 AND 4.13 MISSED DUE TO INADEQUATE ADMINISTRATIVE CONTROLS.
EVENT DATE: 070789 REPORT DATE: 091389 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)

(NSIC 215233) ON 7/7/89, WITH UNIT 3 AND UNIT 4 IN MODE 1 AT 100%, HEALTH PHYSICS PERSONNEL (NON-LICENSED UTILITY PERSONNEL) DISCOVERED A LIQUID SCINTILLATION COUNTER WITH AN INTERNAL CALIBRATION SOURCE THAT HAD NOT BEEN LEAK TESTED AS REQUIRED BY TECH SPECS (TS) 3.11 AND 4.13. THE SUBJECT CESIUM-137 CALIBRATION SOURCE HAS AN ACTIVITY OF 30 MICROCURIES. TECH SPEC 3.11 REQUIRES THOSE QUANTITIES OF BY-PRODUCT MATERIAL THAT EXCEED THE QUANTITIES LISTED IN 10CFR30.71 ARE TO BE LEAK TESTED IN ACCORDANCE WITH THE SCHEDULE DESCRIBED IN TECH SPEC 4.13. THE REQUIRED LEAKAGE TESTS HAVE NOT BEEN PERFORMED, SINCE RECEIPT OF THE SUBJECT LIQUID SCINTILLATION COUNTER IN JUNE, 1987. THIS LER IS BEING SUBMITTED LATE, SINCE THE APPROPRIATE PERSONNEL WERE NOT NOTIFIED OF THE INCIDENT UNTIL 8/16/89. THE SUBJECT CESIUM-137 CALIBRATION SOURCE PASSED A LEAK TEST CONDUCTED ON 7/9/89. ON 6/30/89, ADMINISTRATIVE PROCEDURE, O-ADM-022, "PROCURING RADIOACTIVE SOURCES," WAS ISSUED. THIS PROCEDURE ADMINISTRATIVELY CONTROLS THE PROCURING OF ALL RADIOACTIVE SOURCES (EXCEPT FUEL), AND REQUIRES ALL SHIPPERS TO INDICATE ON THE OUTSIDE OF THE PACKAGE IF THE PACKAGE CONTAINS RADIOACTIVE MATERIAL. THE INDIVIDUAL RESPONSIBLE FOR THE TARDINESS OF THIS LER HAS BEEN COUNSELED.

[224] TURKEY POINT 4 DOCKET 50-251 LER 89-008
CONTAINMENT SPRAY PUMP OUT OF SERVICE FOR MAINTENANCE FOR LONGER THAN TECH SPEC ALLOWED PERIOD DUE TO UNEXPECTED INCREASE IN MOTOR VIBRATION.
EVENT DATE: 080989 REPORT DATE: 090789 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215206) AT 0930 ON 8/9/89, THE 4A CONTAINMENT SPRAY PUMP (CSP) EXCEEDED THE TECH SPEC (TS) LIMITING CONDITION FOR OPERATION BY REMAINING OUT OF SERVICE FOR MORE THAN 24 HOURS. THE CSP WAS REMOVED FROM SERVICE AT 0930 ON 8/8/89 TO PERFORM MAINTENANCE AIMED AT REDUCING PUMP VIBRATION LEVELS FROM THE "ALERT" RANGE TO THE ACCEPTABLE RANGE. AFTER REALIGNING THE PUMP AND MOTOR, THE MOTOR AND PUMP VIBRATION LEVELS INCREASED INTO THE "REQUIRED ACTION" RANGE. ADDING WEIGHTS TO THE PUMP AND MOTOR COUPLINGS REDUCED THE PUMP VIBRATION TO WITHIN THE "ALERT" RANGE, HOWEVER, THE MOTOR VIBRATION REMAINED IN THE "REQUIRED ACTION" RANGE. A DECISION TO REPLACE THE CSP MOTOR WITH A SPARE MOTOR REQUIRED MORE TIME

THAN ALLOWED BY THE TECH SPEC. DISCUSSIONS WERE HELD WITH THE NRC AND DISCRETIONARY ENFORCEMENT WAS OBTAINED TO ALLOW ONE CSP TO BE INOPERABLE FOR 72 HOURS. WITH VENDOR ASSISTANCE, THE 4A CSP WAS RETURNED TO SERVICE AT 2322 ON 8/10/89. THE CAUSE FOR THIS EVENT IS ATTRIBUTABLE TO INSTALLATION WHICH LED TO STRESSES INDUCED INTO THE MOTOR CASING. FPL BELIEVES UNEVENESS IN EITHER THE MOTOR FEET AND/OR THE MOUNTING BASE PLATE LED TO THE INABILITY TO REDUCE MOTOR VIBRATION LEVELS AFTER REALIGNING THE PUMP AND MOTOR. THE MOUNTING BASE PLATE AND MOTOR FEET WILL BE EXAMINED AND WORKED AS NECESSARY DURING THE NEXT REFUELING OUTAGE.

[225] TURKEY POINT 4 DOCKET 50-251 LER 89-009
BORIC ACID TRANSFER PUMP SEAL POT FOUND WITH NO VISIBLE LEVEL RESULTING IN NO FLOW PATH FROM BORIC ACID TANK TO UNIT 4 REACTOR COOLANT SYSTEM.
EVENT DATE: 082389 REPORT DATE: 092289 NSSS: WE TYPE: PWR
VENDOR: DURAMETALLIC CORP.
GOULDS PUMPS INC.

(NSIC 215300) ON 8/23/89, AT 1715, UNIT 4 ENTERED TECH SPEC 3.0.1 FOR 33 MINUTES WHEN THE 4B BORIC ACID TRANSFER PUMP (BATP) WAS DECLARED OUT OF SERVICE. AT THE TIME OF THE EVENT, THE 4A BATP WAS OUT OF SERVICE TO REPAIR A FLANGE LEAK. AT 1715, WHILE MAKING NORMAL ROUNDS, OPERATIONS PERSONNEL DISCOVERED THAT THE SEAL POT, WHICH PROVIDES SEAL COOLING WATER TO BE THE 4B BATP, HAD NO VISIBLE WATER LEVEL. FOLLOWING THE DISCOVERY, THE 4B BATP SEAL POT WAS REFILLED AND PUMP FLOW WAS VERIFIED. THE SEAL POT LEVEL WAS OBSERVED TO REMAIN STABLE. THIS WAS COMPLETED AFTER 33 MINUTES, AND UNIT 4 EXITED TECH SPEC 3.0.1. THE CAUSE OF THIS EVENT IS MOST LIKELY ATTRIBUTABLE TO THE DESIGN OF THE BATP DOUBLE SEALS. FPL BELIEVES THAT FREQUENT PUMP STARTS AND STPS MAY CAUSE AXIAL SHAFT MOVEMENT WHICH RESULTS IN SEPARATION OF THE INNER SEAL FACES. WHEN THE 4B BATP STOPPED AUTOMATIC MAKE-UP TO THE VOLUME CONTROL TANK AT APPROXIMATELY 1700, FPL BELIEVES THE INNER SEAL FACES TEMPORARILY "COCKED", RESULTING IN A DRAIN OF SEAL WATER INTO THE PUMP CASING. FPL ENGINEERING IS CONTINUING TO REVIEW ALTERNATE SEAL DESIGNS FOR THE BATPS WHICH WILL NOT REQUIRE QUENCH WATER TO THE SEAL.

[226] VERMONT YANKEE DOCKET 50-271 LER 89-021
FAILURE OF RM-16-19-1B PRIMARY CONTAINMENT HIGH RANGE RADIATION MONITOR DUE TO AN OPEN CIRCUIT ON THE DETECTOR SIGNAL CABLE.
EVENT DATE: 072589 REPORT DATE: 090689 NSSS: GE TYPE: BWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 215210) THIS SPECIAL REPORT IS BEING SUBMITTED TO COMPLY WITH TECH SPEC TABLE 3.2.6, WHICH REQUIRES A SPECIAL REPORT TO THE COMMISSION WHENEVER ONE OF THE TWO HIGH RANGE CONTAINMENT RADIATION MONITORS IS OUT OF SERVICE FOR 30 DAYS. ON 7/25/89 AT 1830 HOURS, WITH THE PLANT OPERATING AT 100% POWER, THE "B" CONTAINMENT HIGH RANGE RADIATION MONITOR (RM-16-19-1B) (EIIS-IL), WAS DECLARED INOPERABLE DUE TO FAILURE TO RESPOND. TROUBLESHOOTING HAS DETERMINED THAT THE CAUSE OF THE PROBLEM IS AN OPEN CIRCUIT ON THE DETECTOR SIGNAL CABLE INSIDE THE PRIMARY CONTAINMENT. REPAIR OF THE INSTRUMENT WILL REQUIRE A PLANT SHUTDOWN AND PRIMARY CONTAINMENT ENTRY. THE PRESENT PLANS ARE TO MAKE THIS REPAIR AT THE NEXT AVAILABLE OPPORTUNITY WHEN THE PRIMARY CONTAINMENT IS ACCESSIBLE.

[227] VERMONT YANKEE DOCKET 50-271 LER 89-020
REMOVAL OF A TECH SPEC SURVEILLANCE REQUIREMENT FROM A PROCEDURE DUE TO AN INADEQUATE TECH SPEC REVIEW.
EVENT DATE: 081189 REPORT DATE: 090889 NSSS: GE TYPE: BWR

(NSIC 215242) ON 8/11/89, WITH THE PLANT AT 100% POWER, VERMONT YANKEE DISCOVERED THAT THE PROCEDURE CONTROLLING BATTERY MAINTENANCE AND TESTING WAS NOT CONSISTENT WITH TECH SPEC REQUIREMENTS. TECH SPEC SECTION 4.10.2.A REQUIRES THAT A

TEMPERATURE READING BE OBTAINED FROM THE CELLS ADJACENT TO THE PILOT CELL. INVESTIGATION REVEALED THAT THE REQUIREMENT TO OBTAIN THE TEMPERATURE FROM THE ADJACENT CELLS WAS REMOVED FROM THE PROCEDURE IN MARCH OF 1987 DURING AN EFFORT TO UPGRADE MAINTENANCE AND TESTING METHODS TO BE CONSISTENT WITH IEEE STD. 450-1980 "IEEE RECOMMENDED PRACTICE FOR MAINTENANCE, TESTING AND REPLACEMENT OF LARGE LEAD STORAGE BATTERIES FOR GENERATING STATIONS AND SUBSTATIONS". THIS STANDARD RECOMMENDS THAT THE TEMPERATURE BE OBTAINED ONLY FROM THE PILOT CELL. THE ROOT CAUSE OF THIS EVENT IS AN INADEQUATE TECH SPEC REVIEW OF A PROCEDURE REVISION. THE INDIVIDUAL THAT PREPARED THE PROCEDURE REVISION, AND REVIEWED IT FOR TECH SPEC COMPLIANCE, DID NOT IDENTIFY THAT A DIFFERENCE EXISTED BETWEEN SPECIFIC TESTING REQUIREMENTS OF THE TECH SPECS AND THE REVISED PROCEDURE. A SIMILAR EVENT WAS REPORTED TO THE COMMISSION, IN THE LAST FIVE YEARS, AS LER 87-04.

[228] VOGTLE 1 DOCKET 50-424 LER 89-016 REV 01
UPDATE ON MANUAL REACTOR TRIP DUE TO FAILURE OF MAIN FEEDWATER ISOLATION VALVE.
EVENT DATE: 070889 REPORT DATE: 083189 NSSS: WE TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 215175) ON 7/8/89, AT 0327 CDT, A MANUALLY INITIATED REACTOR TRIP OCCURRED ON UNIT 1 WITH THE REACTOR AT 100% OF RATED THERMAL POWER. THE MANUAL TRIP WAS INITIATED BECAUSE THE LOOP 4 MAIN FEEDWATER ISOLATION VALVE (MFIV) FAILED CLOSED RESULTING IN A DECREASE IN THE NO. 4 STEAM GENERATOR LEVEL. THE PLANT WAS STABILIZED IN MODE 3 FOLLOWING THE REACTOR TRIP. TROUBLESHOOTING FOLLOWING THE MANUAL TRIP FAILED TO IDENTIFY THE EXACT CAUSE OF THE SPURIOUS CLOSURE; HOWEVER, 2 POTENTIAL FAILURE MECHANISMS WERE EVALUATED; SOLENOID FAILURE AND AUXILIARY RELAY FAILURE. CORRECTIVE ACTIONS FOR THE 7/8/89 EVENT INCLUDED TROUBLESHOOTING OF THE VALVE CONTROL CIRCUITRY, REPAIR OF THE ASSOCIATED HANDSWITCH FOR A PROBLEM NOTED DURING TROUBLESHOOTING BUT UNRELATED TO THE FAILURE, AND CONTINUED CONTROL LOOP MONITORING WITH A MULTI-CHANNEL RECORDER. ON 8/3/89, THE LOOP 4 MFIV SPURIOUSLY CLOSED AND THE RESULTING DECREASE IN FEEDWATER LED THE OPERATOR TO TRIP THE REACTOR AT 1445 CDT. CONTROL ROOM OPERATORS ACHIEVED STABLE PLANT CONDITIONS BY 1505 CDT. CORRECTIVE ACTION FOR THIS EVENT CONSISTED OF REPLACING A FAILED SOLENOID VALVE AND RETURNING THE VALVE TO THE VENDOR FOR FAILURE ANALYSIS.

[229] VOGTLE 2 DOCKET 50-425 LER 89-025
ENTRY INTO LCO 3.0.3 DUE TO TRIPPING OF ESF ROOM COOLERS ON THERMAL OVERLOAD.
EVENT DATE: 080189 REPORT DATE: 083189 NSSS: WE TYPE: PWR
VENDOR: CUTLER-HAMMER

(NSIC 215176) ON 8/1/89, AT 1808 CDT, CONDITIONS FOR ENTERING LIMITING CONDITION FOR OPERATION (LCO) 3.0.3 OCCURRED WHEN A TRAIN B ENGINEERED SAFETY FEATURE (ESF) ROOM COOLER TRIPPED ON THERMAL OVERLOAD. SHORTLY BEFORE THIS, A TRAIN A ESF ROOM COOLER HAD TRIPPED ON THERMAL OVERLOAD. WITH A ROOM COOLER IN EACH TRAIN OF ESF ROOM COOLERS OUT OF SERVICE, A CONDITION NOT PROVIDED FOR IN THE ACTION REQUIREMENTS OF TECH SPEC 3.7.11 EXISTED AND ENTRY INTO LCO 3.0.3 WAS REQUIRED. SEVERAL TRIPS OCCURRED FOR THESE ROOM COOLERS ON 8/1/89. IN EACH CASE, THE TRIP WAS ON THERMAL OVERLOAD, AND AFTER EACH TRIP THE ROOM COOLER WAS RESTORED TO OPERABLE STATUS BY RESETTING THE OVERLOAD DEVICE AND VERIFYING THE COOLER FAN WAS RUNNING. SUBSEQUENT TO 8/1/89, TWO OTHER ESF ROOM COOLERS EXPERIENCED TRIPS ON THERMAL OVERLOAD. CORRECTIVE ACTION HAS BEEN TAKEN TO INCREASE THE THERMAL OVERLOAD TRIP SETPOINTS FOR ALL UNIT 2 AUXILIARY BUILDING ESF ROOM COOLERS. THIS ACTION IS ALSO BEING IMPLEMENTED FOR SIMILAR UNIT 1 ESF ROOM COOLERS.

[230] VOGTLE 2 DOCKET 50-425 LER 89-026
INCOMPLETE COMMUNICATIONS LEAD TO MISSED ASME SECTION XI VALVE TESTING.
EVENT DATE: 082989 REPORT DATE: 100289 NSSS: WE TYPE: PWR
VENDOR: VALCOR ENGINEERING CORP.

(NSIC 215386) ON 9/3/89, THE SHIFT SUPERVISOR NOTICED THAT A HIGH-ENERGY LINE BREAK VALVE FOR STEAM GENERATOR BLOWDOWN ISOLATION HAD NOT BEEN TESTED PER ASME SECTION XI WITHIN THE SPECIFIED TIME INTERVAL. TESTING WAS IMMEDIATELY INITIATED AND SUCCESSFULLY COMPLETED. ALTHOUGH TESTING WAS NOT PERFORMED AS SCHEDULED, TECH SPEC TABLE 3.3-11 OPERABILITY REQUIREMENTS WERE SATISFIED AT ALL TIMES SINCE ONE BLOWDOWN ISOLATION VALVE WAS ALWAYS OPERABLE. THE CAUSE OF THIS EVENT WAS COGNITIVE PERSONNEL ERROR RESULTING FROM INCOMPLETE COMMUNICATIONS. THE SHIFT SUPERVISOR RESPONSIBLE FOR COMPLETING THE TESTING ON 7/30/89 NOTED ON THE TEST COMPLETION DOCUMENTATION THAT THE VALVE WAS NOT TESTED WHEN ORIGINALLY SCHEDULED. HOWEVER, THE SURVEILLANCE TRACKING COORDINATOR DID NOT FULLY UNDERSTAND THE NOTE AND DID NOT RESCHEDULE THE TEST PRIOR TO ITS DUE DATE. IN ADDITION, THE SHIFT SUPERVISOR DID NOT APPROPRIATELY NOTIFY THE NECESSARY PERSONNEL OF INCOMPLETE TESTING AS REQUIRED BY PLANT PROCEDURES. AS CORRECTIVE ACTIONS, THE INDIVIDUALS INVOLVED WERE COUNSELED AND A MEMO WAS SENT TO APPROPRIATE PLANT PERSONNEL EMPHASIZING PROCEDURAL REQUIREMENTS TO BE FOLLOWED WHEN TESTING IS NOT COMPLETED.

[231] WATERFORD 3 DOCKET 50-382 LER 89-015
CONTAINMENT ISOLATION VALVE INOPERABLE DUE TO INADEQUATE DESIGN AND INADEQUATE PROCEDURE.
EVENT DATE: 072889 REPORT DATE: 082889 NSSS: CE TYPE: PWR

(NSIC 215142) A TECHNICAL EVALUATION PERFORMED FOR STEAM GENERATOR (SG) #2 BLOWDOWN PIPING CONCLUDED THAT THE STRUCTURAL INTEGRITY OF THE PIPING SYSTEM (INCLUDING THE OUTSIDE CONTAINMENT ISOLATION VALVE AND SHIELD BUILDING PENETRATION) COULD NOT BE ASSURED DURING A SEISMIC EVENT DUE TO A DAMAGED PIPING SUPPORT. THIS CONDITION EXISTED IN EXCESS OF THE TIME LIMIT ALLOWED BY TECHNICAL SPECIFICATION (TS) 3.6.3 ACTION REQUIREMENT A., AND ON AUGUST 8, 1989, WAS DETERMINED TO BE REPORTABLE AS A CONDITION PROHIBITED BY TSS. THE ROOT CAUSE OF THIS EVENT IS BELIEVED TO BE A COMBINATION OF AN INADEQUATE PROCEDURE AND AN INADEQUATE SYSTEM DESIGN. WHILE PERFORMING THE ENGINEERED SAFETY FEATURE ACTUATION SIGNAL (ESFAS) SUBGROUP RELAY TEST PROCEDURE, THE POTENTIAL EXISTED FOR WATER HAMMER TO OCCUR IN THE BLOWDOWN LINES. A WATER HAMMER TRANSIENT IS BELIEVED TO HAVE DAMAGED THE PIPE SUPPORT. THE ESFAS SUBGROUP RELAY TEST PROCEDURE IS BEING REVISED AND SYSTEM DESIGN IS BEING EVALUATED TO MINIMIZE WATER HAMMER EFFECTS. DUE TO THE LOW PROBABILITY OF A SEISMIC EVENT AND BECAUSE THE INSIDE CONTAINMENT ISOLATION VALVE WAS OPERABLE AT ALL TIMES; THIS EVENT DID NOT THREATEN THE HEALTH AND SAFETY OF THE GENERAL PUBLIC OR PLANT PERSONNEL.

[232] WATERFORD 3 DOCKET 50-382 LER 89-016
INADVERTENT ENGINEERED SAFETY FEATURE ACTUATIONS DUE TO PERSONNEL ERROR.
EVENT DATE: 080289 REPORT DATE: 090189 NSSS: CE TYPE: PWR

(NSIC 215172) AT 1734 HOURS ON 8/2/89, AND AT 0138 HOURS ON 8/4/89, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% POWER WHEN A LOSS OF ELECTRICAL POWER TO A FUEL HANDLING BUILDING (FHB) ISOLATION RADIATION MONITOR RESULTED IN AUTOMATIC STARTS OF THE 'B' TRAIN FHB EMERGENCY FILTRATION UNIT. THESE EVENTS ARE REPORTABLE AS AUTOMATIC ENGINEERED SAFETY FEATURE (ESF) ACTUATIONS. THE ACTUATION WHICH OCCURRED 8/4/89, WAS ATTRIBUTED TO PERSONNEL ERROR. A TECHNICIAN WAS INVESTIGATING AN ABNORMALITY WITH THE MONITOR DURING WHICH TIME HIS HAND INADVERTENTLY TRIPPED THE MONITOR'S POWER SWITCH CAUSING A LOSS OF POWER TO THE MONITOR AND SUBSEQUENT ESF ACTUATION. THE AUGUST 2 ACTUATION IS BELIEVED TO HAVE A SIMILAR CAUSE, BUT HAS NOT BEEN CONFIRMED. ELECTRICAL POWER WAS IMMEDIATELY RESTORED TO THE MONITOR AND ALL EQUIPMENT FUNCTIONED AS DESIGNED. IF THE EXACT CAUSE IS LATER DETERMINED, IT WILL BE

DESCRIBED IN A REVISION TO THIS REPORT. BECAUSE THE FHB EMERGENCY FILTRATION UNIT OPERATED AS REQUIRED FOR THE LOSS OF POWER TO THE MONITOR, THESE EVENTS DID NOT THREATEN THE HEALTH AND SAFETY OF THE GENERAL PUBLIC OR PLANT PERSONNEL.

[233] WATERFORD 3 DOCKET 50-382 LER 89-017
REACTOR TRIP DUE TO COMPLICATIONS ASSOCIATED WITH CONTROL ELEMENT ASSEMBLY MALFUNCTION.
EVENT DATE: 081989 REPORT DATE: 091889 NSSS: CE TYPE: PWR
VENDOR: ELECTRO - MOTIVE DIV. OF GM

(NSIC 215272) AT 1319 HOURS ON 8/19/89, AN AUTOMATIC REACTOR TRIP OF WATERFORD STEAM ELECTRIC STATION UNIT 3 OCCURRED WHILE OPERATING AT 23% POWER. THE TRIP WAS INITIATED BY THE PLANT PROTECTION SYSTEM (PPS) IN RESPONSE TO VARIATIONS IN CORE AXIAL SHAPE INDEX (ASI), A MEASURE OF CORE POWER DISTRIBUTION, INDUCED BY THE DOWN POWER REQUIRED FOR AN ABNORMAL CONTROL ELEMENT ASSEMBLY (CEA) CONFIGURATION. THIS EVENT IS REPORTABLE AS AN AUTOMATIC REACTOR PROTECTION SYSTEM ACTUATION. THE ROOT CAUSE OF THIS EVENT IS EQUIPMENT MALFUNCTION. DURING ROUTINE CEA OPERABILITY ESTING, CEA 18 WOULD NOT MOVE IN EITHER DIRECTION. AFTER REPAIRS WERE MADE TO CEA CONTROL CIRCUITRY, CEA 18 WAS INSERTED BELOW THE TECH SPEC (TS) LIMIT OF 145 INCHES WHILE VERIFYING RESPONSE. CEA 18 WOULD NOT WITHDRAW, NECESSITATING A REACTOR POWER REDUCTION PER TSS. WHILE ATTEMPTING TO CONTROL ASI SUBSEUQENT TO THE POWER REDUCTION, THE REACTOR TRIPPED. ALL DEFECTIVE EQUIPMENT HAS BEEN REPLACED AND TESTED SATISFACTORILY. BECAUSE ALL PROTECTIVE FEATURES FUNCTIONED AS DESIGNED, THE HEALTH AND SAFETY OF THE GENERAL PUBLIC OR PLANT PERSONNEL WAS NOT ADVERSELY AFFECTED BY THIS EVENT.

[234] WOLF CREEK 1 DOCKET 50-482 LER 89-016
INATTENTION TO DETAIL LEADS TO ERROR IN SCHEDULE CAUSING FAILURE TO MEET TECH SPEC SURVEILLANCE REQUIREMENT.
EVENT DATE: 070389 REPORT DATE: 090889 NSSS: WE TYPE: PWR

(NSIC 215220) ON 8/9/89, AT APPROXIMATELY 0950 CDT, DURING THE PREPARATION OF SURVEILLANCE SCHEDULES, IT WAS DISCOVERED THAT THE SURVEILLANCE PROCEDURE FOR THE TRAIN 'B' REACTOR TRIP BREAKER (RTB) TRIP ACTUATING DEVICE OPERATIONAL TEST HAD NOT BEEN PERFORMED BY 7/3/89, AS REQUIRED BY TECH SPEC (T/S) SURVEILLANCE REQUIREMENT 4.3.1.1. UPON DISCOVERY, CONTROL ROOM PERSONNEL WERE NOTIFIED, THE TRAIN 'B' RTB WAS DECLARED INOPERABLE, AND PERFORMANCE OF THE SURVEILLANCE PROCEDURE WAS INITIATED. THE SURVEILLANCE PROCEDURE WAS SATISFACTORILY COMPLETED ON 8/9/89, AT APPROXIMATELY 1201 CDT, AND THE TRAIN 'B' RTB WAS DECLARED OPERABLE. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INATTENTION TO DETAIL BY NON-LICESNED OPERATIONS PERSONNEL WHO ERRED DURING THE SCHEDULING OF THE SURVEILLANCE TEST WHEN MANUALLY ALTERING THE COMPUTER GENERATED SCHEDULES. AN ENHANCEMENT HAS BEEN MADE TO THE COMPUTERIZED SCHEDULING PROGRAM TO ALLOW PRE-SCHEDULING OF SURVEILLANCE PROCEDURES WITH 62-DAY INTERVALS. THIS PROGRAM ENHANCEMENT WILL MINIMIZE THE NEED FOR MANUALLY ALTERING THE COMPUTER GENERATED SCHEDULES, THEREBY LESSENING THE POTENTIAL FOR RECURRENCE OF THE SURVEILLANCE SCHEDULING ERROR.

[235] WOLF CREEK 1 DOCKET 50-482 LER 89-017
SPURIOUS SPIKE ON CHLORINE MONITOR RESULTS IN ENGINEERED SAFETY FEATURES ACTUATION.
EVENT DATE: 083089 REPORT DATE: 092589 NSSS: WE TYPE: PWR

(NSIC 215390) ON 8/30/89, AT APPROXIMATELY 1707 CDT, A CONTROL ROOM VENTILATION ISOLATION SIGNAL (CRVIS) WAS INITIATED WHEN CHLORINE MONITOR GK-AITS-3 INDICATED HIGH CHLORINE LEVEL IN THE OUTSIDE AIR MAKEUP TO THE CONTROL BUILDING HEATING, VENTILATION, AND AIR CONDITIONING SYSTEM. UPON RECEIPT OF THE CRVIS, ALL REQUIRED ENGINEERED SAFETY FEATURES EQUIPMENT RESPONDED PROPERLY. AT THE TIME OF

THIS EVENT, THE PLANT WAS OPERATING IN MODE 1, POWER OPERATION, AT APPROXIMATELY 100% RATED THERMAL POWER. NO CHLORINE WAS PRESENT AS EVIDENCED BY NORMAL READINGS ON THE REDUNDANT CHLORINE MONITOR. AT APPROXIMATELY 1715 CDT, CHLORINE MONITOR GK-AITS-3 WAS PLACED IN BYPASS FOR TROUBLESHOOTING AND THE AFFECTED SYSTEMS WERE RESTORED TO THEIR NORMAL CONFIGURATION. CONTROL ROOM PERSONNEL CONFIRMED THAT THE CRVIS WAS THE RESULT OF A SPURIOUS SPIKE AND NOT THE RESULT OF AN ACTUAL HIGH CHLORINE CONDITION. SUBSEQUENT TROUBLESHOOTING BY INSTRUMENTATION AND CONTROL PERSONNEL COULD NOT DETERMINE A ROOT CAUSE FOR THE SPURIOUS SPIKE. THIS IS THE FIRST CRVIS INITIATED BY A CHLORINE MONITOR SINCE INSTALLATION OF A NEW TYPE OF MONITOR IN LATE 1988.

[236] WPPSS 2 DOCKET 50-397 LER 89-023
ENGINEERED SAFETY FEATURE ISOLATIONS AND ACTUATIONS DUE TO LOSS OF REACTOR PROTECTION SYSTEM BUS DURING TESTING DUE TO PERSONNEL ERROR/PROCEDURAL INADEQUACY.
EVENT DATE: 053189 REPORT DATE: 063089 NSSS: GE TYPE: BWR

(NSIC 214570) ON 5/31/89 AT 1406 HRS AN ELECTRICAL PROTECTION ASSEMBLY (EPA) BREAKER TRIPPED CAUSING A LOSS OF POWER TO RPS BUS B. LOSS OF POWER TO RPS BUS B CAUSED A HALF-SCRAM IN RPS DIVISION B AND MULTIPLE PRIMARY AND SECONDARY CONTAINMENT ISOLATIONS AND ESF ACTUATIONS OF VENTILATION SYSTEMS. PLANT WAS SHUTDOWN FOR ANNUAL MAINTENANCE AND REFUELING OUTAGE. LOSS OF RPS B POWER CAUSES NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM CONTAINMENT INBOARD AND OUTBOARD ISOLATIONS FOR GROUPS 1,2,5,6 AND 7; AND A REACTOR BLDG EXHAUST PLENUM RADIATION MONITOR "Z" SIGNAL WHICH INITIATES SEVERAL ESF ACTUATIONS INCLUDING STANDBY GAS TREATMENT (SGT) SYSTEM, THE CONTROL ROOM EMERGENCY FILTRATION SYSTEM, AND A REACTOR BLDG VENTILATION SYSTEM ISOLATION. PLANT OPERATORS RESPONDED BY RESTORING ALL SYSTEMS, INCLUDING RESIDUAL HEAT REMOVAL (RHR) SHUTDOWN COOLING, TO PRE-EVENT LINEUP STATUS BY 1430 HRS. THE CAUSES OF THIS EVENT ARE PERSONNEL ERROR IN THAT A PLANT TEST ENGINEER AND PLANT OPERATORS DID NOT ADEQUATELY REVIEW THE CONSEQUENCES OF STARTING A REACTOR RECIRCULATION PUMP WHILE THE PLANT WAS ALIGNED FOR THE PERFORMANCE OF LOGIC SYSTEM FUNCTIONAL TESTING (LSFT) OF THE ATWS RECIRCULATION PUMP "A" TRIP SYSTEM, AND 2) INADEQUATE PROCEDURE IN THAT THE LSFT PROCEDURE DID NOT SPECIFICALLY CAUTION AGAINST STARTING AN RRC PUMP DURING TEST PERFORMANCE.

[237] WPPSS 2 DOCKET 50-397 LER 89-025
ESF ACTUATIONS DURING EXCESS FLOW CHECK VALVE TESTING DUE TO PROCEDURAL INADEQUACIES.
EVENT DATE: 061789 REPORT DATE: 071789 NSSS: GE TYPE: BWR

(NSIC 214826) DURING THE PERFORMANCE OF THE PLANT PROCEDURE FOR EXCESS FLOW CHECK VALVE TESTING, 3 SEPARATE BUT RELATED EVENTS OCCURRED WHICH CAUSED ESF ISOLATIONS AND ACTUATIONS. THESE EVENTS ARE COMBINED INTO 1 LER IN ACCORDANCE WITH THE GUIDANCE PROVIDED IN NUREG 1022 (SUPPLEMENT NO. 1). AT THE TIME OF THE EVENTS THE PLANT WAS IN A SHUTDOWN CONDITION FOR THE ANNUAL MAINTENANCE AND REFUELING OUTAGE. (1) ON JUNE 17, 1989 AT 1222 HOURS, THE INBOARD RESIDUAL HEAT REMOVAL (RHR) SHUTDOWN COOLING SUPPLY VALVE (RHR-V-9) AUTOMATICALLY ISOLATED DURING EXCESS FLOW CHECK VALVE TESTING. THE ROOT CAUSE OF THIS EVENT IS PROCEDURAL INADEQUACY IN THAT THE PROCEDURE DID NOT CAUTION AGAINST INCREASING PRESSURE TO A VALUE THAT WOULD CAUSE THE RPV HIGH PRESSURE ISOLATION ACTUATION. PLANT OPERATORS WERE IN THE PROCESS OF INCREASING REACTOR PRESSURE BY RAISING VESSEL LEVEL WITH WATER FROM THE CONTROL ROD DRIVE (CRD) SYSTEM. WHEN RPV PRESSURE REACHED 122 PSIG, A REACTOR RECIRCULATION (RRC) SYSTEM PRESSURE SWITCH ACTUATED AND, BY DESIGN, RHR-V-9 CLOSED WHICH ISOLATED RHR SHUTDOWN COOLING. (2) ON 6/18/89, AT 1045 HOURS, DURING THE PERFORMANCE OF EXCESS FLOW CHECK VALVE TESTING, A REACTOR PROTECTION SYSTEM (RPS) "A" HALF-SCRAM AND A HPCS DIESEL GENERATOR START OCCURRED DUE TO A PRESSURE TRANSIENT CAUSED WHEN AN I&C TECH. OPENED THE WRONG VALVE DUE TO PROCEDURAL INADEQUACY.

[238] WPPSS 2 DOCKET 50-397 LER 89-027 REV 01
UPDATE ON INADEQUATE SEISMIC RESTRAINT OF ISOLATION VALVES COULD RESULT IN UNISOLATABLE BREACH OF PRIMARY CONTAINMENT CAUSED BY INADEQUATE WORK PRACTICES.
EVENT DATE: 063089 REPORT DATE: 072889 NSSS: GE TYPE: BWR

(NSIC 215144) ON JUNE 30, 1969 A PRELIMINARY ENGINEERING EVALUATION DETERMINED THAT TWO SEISMIC SUPPORTS MISSING ON EACH OF TWO POST ACCIDENT SAMPLING SYSTEM (PASS) CONTAINMENT ISOLATION VALVES, FOUND BY A DESIGN ENGINEER ON JUNE 27, 1989, WOULD PROBABLY RESULT FAILURE OF THE PIPE AT ITS PRIMARY CONTAINMENT PENETRATION DURING A DESIGN BASIS EARTHQUAKE (DBE). THIS WOULD CREATE AN UNISOLATABLE BREACH OF PRIMARY CONTAINMENT. "AS FOUND" CONDITION WAS DISCOVERED WHILE THE DESIGN ENGINEER WAS PERFORMING A VISUAL INSPECTION OF PLANT SUPPORTS AND WHILE THE PLANT WAS AT 3% POWER AND IN MODE 2 (STARTUP). AT 1650 HOURS ON JUNE 30, 1989 THE PRIMARY CONTAINMENT TECHNICAL SPECIFICATION ACTION STATEMENT 3.6.1.1 WAS ENTERED AND PREPARATIONS WERE MADE TO RESTORE RESTRAINTS TO REQUIRED PLANT CONFIGURATION. AT 1745 HOURS WHEN WORK WAS NOT COMPLETED ON RESTRAINTS, A PLANT SHUTDOWN WAS INITIATED. PRIMARY CONTAINMENT TECHNICAL SPECIFICATION ACTION STATEMENT WAS EXITED AT 1843 HOURS WHEN THE RESTRAINTS WERE RESTORED. THE ROOT CAUSES OF THE EVENT ARE 1) LESS THAN ADEQUATE WORK PRACTICES TO ENSURE THE PLANT CONFIGURATION REMAINS WITHIN DESIGN REQUIREMENTS, AND 2) LESS THAN ADEQUATE TRAINING OF PROJECT PERSONNEL TO IMPLEMENT PLANT MODIFICATIONS. NO FURTHER CORRECTIVE ACTIONS WERE IDENTIFIED THAT WOULD SIGNIFICANTLY MINIMIZE RECURRENCE OF THIS CONDITION.

[239] WPPSS 2 DOCKET 50-397 LER 89-031
REACTOR SCRAM DUE TO LOW RPV LEVEL AS A RESULT OF LOSS OF REACTOR FEEDWATER PUMP DURING LUBE OIL PUMP SURVEILLANCE TESTING.
EVENT DATE: 080689 REPORT DATE: 090589 NSSS: GE TYPE: BWR

(NSIC 215274) AT 2027 HOURS ON 8/6/89, A LOW REACTOR PRESSURE VESSEL (RPV) LEVEL REACTOR SCRAM WAS INITIATED BY THE REACTOR PROTECTIVE SYSTEM IN RESPONSE TO AN ACTUAL LOW WATER LEVEL CONDITION CAUSED BY AN UNPLANNED TRIP OF REACTOR FEEDWATER PUMP 1B (RFW-P-1B). SCRAM OCCURRED DURING SURVEILLANCE TESTING OF THE AUXILIARY AND EMERGENCY LUBE OIL PUMPS FOR REACTOR FEEDWATER TURBINE 1B (RFW-DT-1B) WITH THE PLANT AT 100% POWER. WHEN A SOLENOID OPERATED DRAIN VALVE WAS ACTUATED TO DEPRESSURIZE ONLY THE AUTO START PRESSURE SWITCH FOR THE AUXILIARY LUBE OIL PUMP, THE ENTIRE "B" FEEDWATER PUMP LUBE OIL SYSTEM WAS SUBJECTED TO A LOW LUBE OIL PRESSURE TRANSIENT SUFFICIENT TO CAUSE A LOW LUBE OIL TRIP OF RFW-P-1B. REMAINING FEEDWATER PUMP WAS UNABLE TO SUPPLY ENOUGH CAPACITY TO MAINTAIN RPV LEVEL ABOVE THE REACTOR SCRAM SETPOINT. ROOT CAUSE INVESTIGATION IS STILL IN PROGRESS. TWO MAJOR AREAS HAVE BEEN IDENTIFIED: 1) INAPPROPRIATE RRC FCV RUNBACK SETPOINT COUPLED WITH THE CHANGE IN FEEDWATER PUMP GOVERNOR MAXIMUM SPEED CAPABILITY; 2) INADVERTENT TRIP OF RFW-P-1B ON LOW LUBE PRESSURE DURING TESTING OF THE AUXILIARY AND EMERGENCY OIL PUMPS. CORRECTIVE ACTION CONSISTS OF REVISION OF THE OPERATING PROCEDURE FOR REACTOR FEEDWATER PUMPS TO INCORPORATE INSTRUCTION TO ENSURE THAT TEST PUSHBUTTON IS DEPRESSED FOR A SUFFICIENT LENGTH OF TIME DURING FEEDWATER TURBINE STARTUP.

[240] WPPSS 2 DOCKET 50-397 LER 89-032
VIOLATION OF ELECTRICAL SEPARATION CRITERIA FOUND DURING TECHNICAL EVALUATION CAUSED BY DESIGN DEFICIENCY.
EVENT DATE: 081189 REPORT DATE: 091189 NSSS: GE TYPE: BWR

(NSIC 215275) ON 8/10/89, A TECHNICAL EVALUATION BEING CONDUCTED AS FOLLOW UP FOR AN OPERATING EXPERIENCE REPORT RESULTED IN THE DISCOVERY OF A NON-CLASS 1E 120 VOLT AC ELECTRICAL POWER SUPPLY BRANCH CIRCUIT (CIRCUIT NUMBER 40 POWER PANEL PP-8A-A) THAT VIOLATED THE WNP-2 ELECTRICAL SYSTEM SEPARATION CRITERIA. ON 8/11/89, FURTHER INVESTIGATION IDENTIFIED THIS CONDITION AS REQUIRING CORRECTIVE ACTION TO ALLEVIATE ITS AFFECT ON THE SAFETY RELATED PORTION OF THE POWER PLANT ELECTRICAL SYSTEM. BEFORE CORRECTIVE ACTION COULD BE FINALIZED AND IMPLEMENTED

ON 8/11/89, THE PLANT WAS SHUT DOWN IN RESPONSE TO ANOTHER UNRELATED ELECTRICAL SYSTEM CONDITION WHICH AFFECTED THE OPERABILITY OF A MAJOR PORTION OF THE CLASS 1E 480 VOLT AC ELECTRICAL DISTRIBUTION SYSTEM (SEE LER 89-34). DURING SHUTDOWN, INVESTIGATION WAS CONTINUED AND RESULTED IN THE DISCOVERY OF TWO MORE CIRCUITS WHICH CONTAINED SEPARATION CRITERIA VIOLATIONS. PP-8A-A CIRCUIT NUMBER 7, WHICH SUPPLIES PROCESS RADIATION MONITORING INSTRUMENTS, CONTAINED ONLY A SINGLE SERIES FUSE. PP-8A-E CIRCUIT NUMBER 9, WHICH SUPPLIES THE LOOSE PARTS DETECTION SYSTEM, ALSO CONTAINED ONLY A SINGLE SERIES FUSE. ALL 3 CIRCUIT DESIGN DEFICIENCIES WERE CORRECTED DURING THE OUTAGE. ROOT CAUSE OF THIS EVENT WAS EVALUATED AS BEING: EQUIPMENT/DESIGN DEFICIENCY/SPECIFICATION LESS THAN ADEQUATE AND EQUIPMENT/DESIGN DEFICIENCY/REVIEW DID NOT DETECT ERROR.

[241] WPPSS 2 DOCKET 50-397 LER 89-033
REACTOR WATER CLEANUP DELTA FLOW ISOLATION DUE TO BLOWN FUSE.
EVENT DATE: 081189 REPORT DATE: 091189 NSSS: GE TYPE: BWR
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)

(NSIC 215328) ON 8/11/89 AT 0254 HOURS, WITH THE PLANT AT 80% POWER, A REACTOR WATER CLEANUP SYSTEM (RWCU) ISOLATION OCCURRED. THE ISOLATION OCCURRED WHEN A FUSE IN THE POWER SUPPLY TO THE LEAK DETECTION MONITOR BLEW, GIVING A FALSE HIGH FLOW SIGNAL TO THE ISOLATION LOGIC. THERE WAS NO APPARENT CAUSE FOR THE BLOWN FUSE. AS AN IMMEDIATE CORRECTIVE ACTION, THE FUSE WAS REPLACED AND THE RWCU SYSTEM WAS RESTORED TO OPERATION. PLANT CONDITIONS: A) POWER LEVEL - 80%; B) PLANT MODE - 1. ON 8/11/89 AT 0254 HOURS, WITH THE PLANT AT 80% POWER, A REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATION OCCURRED. THIS ISOLATION IS PART OF A NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NS4) GROUP 7 ISOLATION. THE ISOLATION WAS CAUSED BY A BLOWN FUSE (E31A-F18) IN THE POWER SUPPLY TO THE LEAK DETECTION FLOW SWITCH (LD-FS-605B) WHICH PROVIDES INPUT TO THE ISOLATION LOGIC.

[242] WPPSS 2 DOCKET 50-397 LER 89-034
TECH SPEC REQUIRED SHUTDOWN COMPLETED AS A RESULT OF INOPERABILITY OF CLASS 1E 480 VOLT AC POWER DISTRIBUTION SYSTEM CAUSED BY DESIGN DEFICIENCY.
EVENT DATE: 081189 REPORT DATE: 091189 NSSS: GE TYPE: BWR

(NSIC 215276) ON 8/11/89 AT 1608 HOURS, WNP-2 TECH SPEC LCO 3.0.3 WAS ENTERED, AN UNUSUAL EVENT WAS DECLARED, AND A NORMAL REACTOR SHUTDOWN WAS COMMENCED AS A RESULT OF DECLARING SIX CLASS 1E 480 VOLT A.C. MOTOR CONTROL CENTERS (MCCS) TO BE INOPERABLE DUE TO DISCOVERY OF A DESIGN DEFICIENCY. THE REACTOR PLANT WAS SHUT DOWN AND THE CONTROL RODS WERE FULLY INSERTED BY 1804 HOURS. THE REACTOR PLANT WAS PLACED IN COLD SHUTDOWN, THE TECH SPEC 3.0.3 LCO WAS EXITED, AND THE UNUSUAL EVENT SECURED AT 0220 HOURS ON 8/12/89. AN URGENT PLANT MODIFICATION REQUEST WAS PROCESSED TO IMPLEMENT A CORRECTION TO THE DESIGN OF THE CLASS 1E 480 VOLT AC POWER DISTRIBUTION SYSTEM. ALL OF THE AFFECTED MCC POWER SUPPLY CIRCUIT BREAKERS WERE REPLACED WITH JUMPER CABLES OF EQUAL CAPACITY WITH THE EXCEPTION OF ONE (MC-8B-A) WHICH WAS REPLACED WITH A FUSED DISCONNECT OF EQUAL CAPACITY. THE CAUSE OF THIS EVENT WAS EVALUATED AS BEING DESIGN DEFICIENCY IN THAT THE DESIGN OF THE 480 VOLT AC POWER DISTRIBUTION SYSTEM DID NOT INCLUDE FAULT TRIPPING COORDINATION ALL THE WAY DOWN TO THE INDIVIDUAL LOADS AT THE SUBFED CLASS 1E MOTOR CONTROL CENTERS. AN ENGINEERING EVALUATION OF THE WNP-2 CLASS 1E 480 VOLT POWER DISTRIBUTION SYSTEM WILL BE PERFORMED TO DETERMINE THE NECESSITY FOR FUTHER MODIFICATIONS TO THE SYSTEM DESIGN.

[243] WPPSS 2 DOCKET 50-397 LER 89-035
REACTOR SCRAM DURING AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) SURVEILLANCE TESTING DUE TO PERSONNEL ERROR.
EVENT DATE: 081789 REPORT DATE: 091389 NSSS: GE TYPE: BWR

(NSIC 215277) AT 0819 HOURS, ON 8/17/89, A REACTOR SCRAM OCCURRED DURING A

SURVEILLANCE BEING PERFORMED ON A REACTOR LEVEL INSTRUMENT ASSOCIATED WITH THE AUTOMATIC DEPRESSURIZATION SYSTEM (ADS). THE SCRAM OCCURRED WHEN AN I&C TECHNICIAN PREMATURELY OPENED THE ISOLATION VALVE FROM THE REFERENCE INSTRUMENT LEG OF THE DEVICE BEING TESTED. THIS CAUSED A PRESSURE TRANSIENT IN THE REFERENCE AND VARIABLE INSTRUMENT LINES WHICH INITIATED A REACTOR SCRAM BY THE REACTOR PROTECTION SYSTEM (RPS) ON REACTOR WATER LEVEL LOW - LEVEL 3. THE ROOT CAUSE OF THIS EVENT WAS A PERSONNEL ERROR AND EQUIPMENT DESIGN DEFICIENCY. IMMEDIATE CORRECTIVE ACTION INCLUDED PLANT SHUT DOWN TO HOT STANDBY AND A MEMO FROM THE PLANT MAINTENANCE MANAGER SUMMARIZING IMMEDIATE CORRECTIVE ACTIONS. FURTHER CORRECTIVE ACTION INCLUDES IMPROVED TRAINING AND INCREASED VISIBILITY FOR "CRITICAL" SURVEILLANCES. A DESIGN STUDY WILL ALSO BE INITIATED TO EVALUATE THE DESIGN OF THE LEVEL TRIP SYSTEM AND ITS INTERFACING INSTRUMENTATION. SINCE ALL SAFETY SYSTEMS OPERATED AS DESIGNED AND PLANT OPERATORS ACTED PROMPTLY TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION, THIS EVENT POSED NO THREAT TO THE HEALTH AND SAFETY OF THE PLANT PERSONNEL OR THE PUBLIC.

[244] WPPSS 2 DOCKET 50-397 LER 89-037
RESIDUAL HEAT REMOVAL SYSTEM DIFFERENTIAL PRESSURE INDICATING SWITCH (RHR-DPIS-12B) DISCOVERED TO BE ISOLATED AND EQUALIZED DUE TO UNKNOWN CAUSE.
EVENT DATE: 083089 REPORT DATE: 092889 NSSS: GE TYPE: BWR

(NSIC 215381) ON 8/30/89, AT 0214 HOURS, DURING THE PERFORMANCE OF A SURVEILLANCE, PLANT INSTRUMENT AND CONTROL (I&C) TECHNICIANS DISCOVERED DIFFERENTIAL PRESSURE INDICATING SWITCH RHR-DPIS-12B TO BE ISOLATED AND EQUALIZED. THE FUNCTION OF RHR-DPIS-12B IS TO PROVIDE AN AUTO-CLOSE SIGNAL TO RHR-V-9 (RESIDUAL HEAT REMOVAL SHUTDOWN COOLING INBOARD ISOLATION VALVE) UPON RECEIPT OF A HIGH FLOW (DIFFERENTIAL PRESSURE) INDICATION IN THE SHUTDOWN COOLING LINE. BECAUSE RHR-DPIS-12B WAS INOPERABLE, IT WAS INCAPABLE OF PERFORMING ITS INTENDED FUNCTION. FOLLOWING DISCOVERY, THE PROBLEM WAS BROUGHT TO THE ATTENTION OF THE SHIFT MANAGER AND THE DECISION WAS MADE TO COMPLETE THE PROCEDURE. IMMEDIATE CORRECTIVE ACTION CONSISTED OF SUCCESSFULLY COMPLETING THE SURVEILLANCE AND RESTORING RHR-DPIS-12B TO SERVICE. THE ROOT CAUSE OF THIS EVENT IS INDETERMINATE. A REVIEW OF PREVIOUS SURVIELLANCES AND WORK HISTORY ASSOCIATED WITH THE INSTRUMENT DID NOT REVEAL A REASON FOR RHR-DPIS-12B TO BE ISOLATED. FURTHER CORRECTIVE ACTION CONSISTS OF 1) DEVELOPING AN I&C WORK PRACTICES MANUAL WHICH DESCRIBES THE INDEPENDENT VERIFICATION PROCESS AND 2) PROVIDING TRAINING TO PLANT I&C TECHNICIANS ON THAT PROCESS. THIS EVENT DID NOT AFFECT THE HEALTH AND SAFETY OF EITHER THE PUBLIC OR PLANT PERSONNEL.

[245] YANKEE ROWE DOCKET 50-029 LER 88-002 REV 01
UPDATE ON LOSS OF POWER TO NUCLEAR INSTRUMENT CABINET "A".
EVENT DATE: 032288 REPORT DATE: 090189 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.
SOLA ELECTRIC COMPANY

(NSIC 215157) ON 3/22/88, AT 0042, WITH THE PLANT OPERATING IN MODE 1 AT 100% POWER, AN AUTOMATIC REACTOR SCRAM OCCURRED WHEN POWER TO THE SCRAM AMPLIFIERS IN THE NUCLEAR INSTRUMENT (NI) CABINET "A" WAS LOST. THE LOSS OF POWER TO NI CABINET "A" WAS A RESULT OF A FAILED SOLA TRANSFORMER (TYPE CV) WITHIN THE NI CABINET. THE LOSS OF POWER PRODUCES A LOSS OF INDICATION OF NI CHANNELS 1, 3 & 6. AN ENS PHONE CALL WAS MADE AT 0141 ON 3/22/88. THE PLANT EMERGENCY DIESEL GENERATOR NO. 2 STARTED AS REQUIRED WITH THE EXPECTED LOSS OF POWER TO THE #1-2400 VOLT AND #4-480 VOLT BUSSES FOLLOWING THE TURBINE GENERATOR TRIP. THE ROOT CAUSE OF THIS EVENT IS THE FAILURE OF ONE OF TWO CAPACITORS IN THE SOLA TRANSFORMER. BOTH CAPACITORS WERE REPLACED WITH DIRECT REPLACEMENTS (IN KIND) WHICH WERE DEDICATED IN ACCORDANCE WITH THE PLANT QUALITY ASSURANCE PROGRAM. THE NI CHANNELS WERE SATISFACTORILY TESTED TO THE APPROVED PLANT SURVEILLANCE PROCEDURE. THERE WAS NO ADVERSE AFFECT ON THE HEALTH OR SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT. SIMILARLY, BUT NOT REPORTABLE, A CAPACITOR IN THE SOLA

TRANSFORMER FAILED IN 1985 AND SUBSEQUENTLY WAS REPLACED DURING TESTING IN 1985. THE REPLACEMENT CAPACITORS WERE OF A DIFFERENT MANUFACTURER, SINCE THE ORIGINAL CAPACITOR IS NO LONGER PRODUCED. THIS IS THE FIRST REPORTED EVENT OF THIS NATURE. NO OTHER CORRECTIVE ACTIONS ARE DEEMED NECESSARY AT THIS TIME.

[246] YANKEE ROWE DOCKET 50-029 LER 89-003 REV 01
UPDATE NO. 2 STEAM GENERATOR BLOWDOWN MONITOR INOPERATIVE.
EVENT DATE: 021089 REPORT DATE: 091589 NSSS: WE TYPE: PWR
VENDOR: CONVAL INC.

(NSIC 215297) ON 2/9/89, AT 0445 HOURS WITH THE PLANT IN MODE 2, CHEMISTRY ANALYSIS OF NO. 2 STEAM GENERATOR BLOWDOWN EFFLUENT INDICATED ABNORMAL RESULTS FOR STEAM GENERATOR BLOWDOWN EFFLUENT. AFTER FURTHER SAMPLING AND ANALYSIS THE NO. 2 BLOWDOWN MONITOR WAS DECLARED INOPERABLE AT 1100 HR. ON 2/10/89 SINCE IT WAS MONITORING FEEDWATER AND NOT THE STEAM GENERATOR BLOWDOWN. THE SOURCE OF THE FEEDWATER WAS BACK LEAKAGE FROM THE STEAM GENERATOR FEEDLINE THROUGH THE CROSS-CONNECT CHECK VALVE BF-V-892. TECH SPEC 3.3.3.1 ACTION STATEMENT REQUIRES AN OPERABLE, CONTINUOUS MONITOR BE PLACED IN SERVICE WITHIN 8 HOURS. THIS ACTION WAS NOT MET BECAUSE OF THE FAILURE TO CONFIRM THE PRESENCE OF FEEDWATER IN THE BLOWDOWN, AND TAKE CORRECTIVE ACTION TO RESEAT THE CROSS CONNECT CHECK VALVE IN THE 8 HOURS. A SIMILAR EVENT WAS REPORTED IN LER 85-08. CHECK VALVE BF-V-892 WAS EXERCISED, USING AN EMERGENCY BOILER FEED PUMP, TO SEAT THE VALVE. CHEMISTRY ANALYSIS VERIFIED EXPECTED BLOWDOWN RESULTS ON 2/10/89 AT 1230 HOURS. THERE WAS NO DETECTABLE PRIMARY TO SECONDARY LEAKAGE BEFORE OR AFTER THE EVENT AND ANY SIGNIFICANT LEAKAGE, HAD IT OCCURRED DURING THE EVENT, WOULD HAVE BEEN DETECTED BY TURBINE CONDENSER AIR EJECTOR.

[247] YANKEE ROWE DOCKET 50-029 LER 89-012
DEGRADATION OF A MAIN COOLANT SYSTEM BOUNDARY RESULTS IN SYSTEM LEAK.
EVENT DATE: 082489 REPORT DATE: 092589 NSSS: WE TYPE: PWR

(NSIC 215337) ON 8/24/89, AT 2338 HOURS, DURING A PLANT START-UP FOLLOWING A MAINTENANCE OUTAGE, (MODE 4 - MAIN COOLANT SYSTEM (MCS) PRESSURE 305 PSIG AND TEMPERATURE 310F) A REPORT OF A STEAM LEAK IN LOOP 2 WAS INVESTIGATED. AT 0035 HOURS, ON 8/25/89, A LEAK WAS FOUND IN A WELD ATTACHING THE EVENT LINE TO THE LOOP BYPASS LINE. AT 0105 HOURS THE ACTION STATEMENT OF TECH SPEC 3.4.5.2, REQUIRING THE PLANT TO BE COLD SHUTDOWN WITHIN 30 HOURS, WAS ENTERED. AT 0115 HOURS AN UNUSUAL EVENT (UE) WAS DECLARED BASED ON MCS LEAKAGE WITHIN THE CAPACITY OF A CHARGING PUMP. NOTIFICATION WAS MADE TO THE STATES OF VERMONT AND MASSACHUSETTS AND THE NRC. FOLLOWING THE COOLDOWN TO MODE 5, THE UE WAS TERMINATED AT 0825 HOURS. SIMILAR VENT LINES IN THE REMAINING LOOPS WERE INSPECTED AND FOUND IN SATISFACTORY CONDITION. THE VENT LINE WAS REPAIRED AND THE PLANT RETURNED TO POWER AT 0525 HOURS ON 8/29/89. AN INITIAL ENGINEERING EVALUATION ATTRIBUTED THE FAILURE TO FATIGUE. A LABORATORY EVALUATION WILL BE CONDUCTED TO CONFIRM THE RESULTS OF THE INITIAL EVALUATION. IF SIGNIFICANT NEW INFORMATION RESULTS FROM THIS EVALUATION A SUPPLEMENTAL REPORT WILL BE SUBMITTED. NO OTHER CORRECTIVE ACTIONS ARE PLANNED AT THIS TIME. A SIMILAR EVENT WAS REPORTED IN LER 83-25. THERE WAS NO ADVERSE EFFECT TO THE PUBLIC HEALTH OR SAFETY AS A RESULT OF THIS EVENT.

[248] YANKEE ROWE DOCKET 50-029 LER 89-013
REACTOR SCRAM DUE TO INADVERTENT ACTUATION OF REACTOR PROTECTION SYSTEM.
EVENT DATE: 082989 REPORT DATE: 092889 NSSS: WE TYPE: PWR

(NSIC 215338) ON 8/29/89, AT 1738 HOURS, FOLLOWING A MAINTENANCE OUTAGE, WHILE IN MODE 2 AT APPROXIMATELY 1% POWER WITH MAIN COOLANT SYSTEM PRESSURE AT 2000 PSIG, A REACTOR SCRAM RESULTED WHEN THE TRAIN B NONRETURN VALVE (NRV) TRIP SWITCH WAS INADVERTENTLY PLACED IN THE TRIP POSITION. PLANT RESPONSE FOLLOWING RECEIPT OF

THE TRIP SIGNAL WAS NORMAL. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO PERSONNEL ERROR. A CONTRIBUTING CAUSE WAS A PROCEDURAL DEFICIENCY. THE CONTROL ROOM OPERATOR INADVERTENTLY TURNED THE TRAIN B NRV TRIP SWITCH TO THE TRIP POSITION WHILE PERFORMING PROCEDURE STEPS THAT WERE NOT REQUIRED FOR THE PLANT OPERATING CONDITIONS. THE PROCEDURE HAS BEEN REVISED TO ELMINATE THE SWITCH OPERATION AND A CAUTION SIGNAL AFFIXED TO THE CONTROL SWITCH CABINET. AN EVALUATION OF THE CONTROL PANEL SWITCH DESIGN WILL BE INITIATED TO DETERMINE IF HUMAN FACTORS ENGINEERING IS APPROPRIATE. THIS IS THE SECOND EVENT OF THIS NATURE. A PREVIOUS REACTOR SCRAM DUE TO INADVERTENT MISPOSITIONING OF THE NRV TRIP SWITCH WAS REPORTED AS LER 86-13. THERE WAS NO ADVERSE EFFECT ON THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT.

[249] ZION 1 DOCKET 50-295 LER 89-013
MAIN STEAM SAFETY VALVE INOPERABILITY DUE TO TESTING METHOD ERRORS.
EVENT DATE: 082689 REPORT DATE: 092589 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE

(NSIC 215355) ON 8/23/89 AT 1230 ZION STATION UNIT 1 WAS AT 39% POWER. MAINTENANCE DEPARTMENT PERSONNEL ASSISTED BY FURMANITE COMPANY TECHNICIANS BEGAN TESTING UNIT 1 MAIN STEAM SAFETY VALVES (MSSV'S) (SB) AS REQURIED BY TECH SPEC 4.7.1. AS TESTING PROCEEDED, THE MAJORITY OF VALVES WERE FOUND TO HAVE SETPOINTS LOW OUTSIDE OF THE +/- 1% TOLERANCE PERMITTED BY TECH SPECS. AN INVESTIGATION REVEALED THAT THE TWO DIGITAL PRESSURE GAGES USED TO MEASURE MS HEADER PRESSURE GAVE DIFFERENT READINGS WHEN LOCATED AT DIFFERENT TEST TAPS. PENDING CONFIRMATION OF THEIR ACTUAL SETPOINTS, ALL SIXTEEN SAFETY VALVES WHICH HAD BEEN TESTED AND/OR RESET WERE DECLARED INOPERABLE. UNIT 1 WAS SHUT DOWN TO HOT SHUTDOWN MODE 3 IN ACCORDANCE WITH TECH SPECS. SEVERAL FACTORS CONTRIBUTED TO THE CAUSE OF THE EVENT, INCLUDING PERSONNEL ERROR, EXTREMELY HIGH TEMPERATURES AT THE WORK SITE AFFECTING INSTRUMENT ACCURACY, AND INCONSISTENCIES IN THE TYPE OF TESTING USED PREVIOUS TO THIS TEST. SAFETY ANALYSIS INDICATES THAT THE SETPOINTS AS FOUND HAD NO ADVERSE AFFECTS ON THE RESULTS OF ACCIDENT ANALYSES DONE FOR THE PLANT. CORRECTIVE ACTIONS INCLUDED A PLANT SHUTDOWN, ANALYSIS AND EVALUATION OF ALL INSTRUMENTATION USED FOR THE TEST, A PENDING TECH SPEC CHANGE, AND DEVELOPMENT OF STATION PROCEDURES TO PROVIDE STANDARDS TO WHICH VENDOR PROCEDURES MUCH ADHERE.

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

COMPONENT INDEX

SYSTEM INDEX

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

SYSTEM INDEX

SYSTEM INDEX

KEYWORD INDEX

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

KEYWORD INDEX

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SEE INSTRUCTIONS ON THE REVERSE

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For month of October 1989

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11a. TYPE OF REPORT
Monthly Report

b. PERIOD COVERED (Inclusive dates)
October 1989

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

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

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS
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