NUREG/CR-2000
ORNL/NSIC-200
Vol. 7, No. 4

Licensee Event Report (LER) Compilation

For month of April 1988

Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory
Commission

Available from

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

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

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

NUREG/CR-2000
ORNL/NSIC-200
Vol. 7, No. 4

Licensee Event Report (LER) Compilation

For month of April 1988

Manuscript Completed: April 1988
Date Published: May 1988

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

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

Abstract

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

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

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

Questions regarding LER searches should be directed to

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

[1] ARNOLD DOCKET 50-331 LER 88-001 REV 01
UPDATE ON HIGH PRESSURE COOLANT INJECTION INOPERABILITY DUE TO TEST CIRCUIT PROBLEM WITH SUBSEQUENT REACTOR CORE ISOLATION COOLING INOPERABILITY DUE TO MOTOR-OPERATOR BEARING FAILURE.
EVENT DATE: 011188 REPORT DATE: 020588 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 208190) ON JANUARY 11, 1988 AT 0405 HOURS, THE PLANT WAS OPERATING AT 99.9% POWER WITH A SCHEDULED HIGH PRESSURE COOLANT INJECTION (HPCI) OPERABILITY SURVEILLANCE TEST PROCEDURE (STP) IN PROGRESS. AT 0406 HOURS, THE HPCI TURBINE TRIPPED DUE TO STEAM SUPPLY HIGH FLOW VIA AN ISOLATION SIGNAL TO THE STEAM INLET ISOLATION VALVE. SUBSEQUENT INVESTIGATION REVEALED LOOSE CONTROL ROOM PANEL TERMINAL CONNECTIONS FOR THE SPEED CONTROL POTENTIOMETER, USED TO CONTROL TURBINE SPEED DURING TESTING, WHICH ULTIMATELY CAUSED THE TURBINE TRIP. ON THE SAME DAY AT 1540 HOURS, WITH THE PLANT OPERATING AT 99.8% POWER, THE REACTOR CORE ISOLATION COOLING (RCIC) OPERABILITY STP WAS BEING PERFORMED VIA THE HPCI INOPERABILITY TEST. AT 1543 HOURS, A RCIC STEAM SUPPLY ISOLATION VALVE WAS SHUT PER THE STP. COINCIDENTALLY THE VALVE MOTOR- OPERATOR BREAKER TRIPPED. SUBSEQUENT INVESTIGATION REVEALED A BROKEN MOTOR-OPERATOR TORQUE SWITCH WHICH CAUSED THE BREAKER TO TRIP ON OVERCURRENT DUE TO THE TORQUE SWITCH'S INABILITY TO TRIP AND DE-ACTIVATE THE VALVE CLOSING CIRCUIT. THE APPROPRIATE LIMITING CONDITIONS FOR OPERATION (LCOS) AND EMERGENCY ACTION LEVEL (EAL) WERE RECOGNIZED AND, UPON RESTORATION, EXITED. THERE WAS NO EFFECT ON THE SAFE OPERATION OF THE PLANT.

[2] BEAVER VALLEY 1 DOCKET 50-334 LER 88-001
STEAM GENERATOR TUBE PLUGGING.
EVENT DATE: 020888 REPORT DATE: 022388 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELEC CORP.-NUCLEAR ENERGY SYS

(NSIC 208329) DURING THE SIXTH REFUELING OUTAGE, BEAVER VALLEY CONDUCTED INSPECTIONS TO OBTAIN U-TUBE WALL THICKNESSES ON ALL THREE STEAM GENERATORS (SG). ALL INSERVICE TUBES ON ALL THREE SGS WERE INSPECTED BY MULTI-FREQUENCY EDDY CURRENT TESTING. A TOTAL OF 72 TUBES WERE PLUGGED (37 TUBES ON THE 1A SG, 20 TUBES ON THE 1B SG AND 15 TUBES ON THE 1C SG). FORTY-SEVEN (47) TUBES WERE FOUND TO HAVE DEGRADATION IN EXCESS OF 40%, 19 TUBES IN ROWS 1 AND 2 WERE FOUND WITH DEGRADATION LESS THAN 40%, BUT LOCATED IN THE U-BEND RADIUS PORTION OF THE TUBE AND WERE PLUGGED PREVENTIVELY AND 6 TUBES WERE IDENTIFIED AS SUSCEPTIBLE TO THE NORTH ANNA FAILURE MECHANISM. OF THE PLUGGED TUBES ON THE 1A SG, 29 EXHIBITED DEGRADATION IN THE U-BEND PORTION, 4 ON THE 1ST SUPPORT BAR ON THE COLD LEG SIDE (SBCL), 2 ON THE 2ND SBCL, 1 ON THE 3RD SBCL AND 1 ON THE 6TH SBCL. OF THE PLUGGED TUBES ON THE 1B SG 10 EXHIBITED DEGRADATION IN THE U-BEND PORTION, 3 ON THE 1ST SBCL, 5 ON THE 1ST SUPPORT BAR, HOT LEG SIDE (SBHL), 1 ON THE 1ST, 2ND, 3RD SBHL AND 1 NEAR THE TUBE SHEET. OF THE PLUGGED TUBES ON THE 1C SG, 9 EXHIBITED DEGRADATION IN THE U-BEND PORTION, 3 ON THE 1ST SBCL, 2 ON THE 1ST SBHL AND 1 ON THE 2ND SBCL. THIS INFORMATION IS BEING REPORTED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 4.4.5.5.A, FOLLOWING THE INSERVICE INSPECTIONS OF ALL THREE SGS CONDUCTED THIS SIXTH REFUELING OUTAGE.

[3] BEAVER VALLEY 2 DOCKET 50-412 LER 88-001
REFUELING WATER STORAGE TANK LOW LEVEL CHANNELS INOPERABLE.
EVENT DATE: 010588 REPORT DATE: 020488 NSSS: WE TYPE: PWR
VENDOR: ROSEMOUNT, INC.

(NSIC 208201) ON 1/5/88, AT 0746 HOURS, 2 OF THE 4 REFUELING WATER STORAGE TANK'S LEVEL INDICATORS WERE DECLARED INOPERABLE WHEN THE INSTRUMENTS FAILED TO THEIR LOW-LOW POSITION AS A RESULT OF THEIR SENSING LINES FREEZING. THE ACTION STATEMENT OF TECHNICAL SPECIFICATION 3.3.2.1, TABLE 3.3-3 WAS EXCEEDED AND TECHNICAL SPECIFICATION 3.03 WAS ENTERED. AT 1238 HOURS, (2QSS-LT104D) WAS

DECLARED OPERABLE AND AT 1249 HOURS, (2QSS-LT104B) WAS DECLARED OPERABLE AFTER APPLYING HEAT TO THE INSTRUMENTS' SENSING LINES. DESIGN CHANGES WILL BE IMPLEMENTED TO PREVENT A RECURRENCE AND IN THE INTERIM, WHILE THE POTENTIAL FOR FREEZING EXISTS, TEMPORARY HEATING IS BEING PROVIDED. THERE WERE NO SAFETY IMPLICATIONS RESULTING FROM THIS EVENT, SINCE THE TRANSMITTERS WERE RESTORED TO OPERABLE STATUS WITHIN THE TIME PERMITTED BY TECHNICAL SPECIFICATIONS.

[4] BEAVER VALLEY 2 DOCKET 50-412 LER 88-002
REACTOR TRIP AND CONTROL ROOM EMERGENCY BOTTLED AIR PRESSURIZATION SYSTEM ACTUATION.
EVENT DATE: 012788 REPORT DATE: 022688 NSSS: WE TYPE: PWR
VENDOR: I-T-E CIRCUIT BREAKER

(NSIC 208412) IN SUPPORT OF NORMAL PLANT TESTING THE 21C SERVICE WATER PUMP WAS SHUTDOWN WHICH CAUSED THE 21A SERVICE WATER PUMP DISCHARGE PRESSURE TRANSMITTER TO FAIL DOWNSCALE RESULTING IN THE AUTO START OF THE 21A EMERGENCY SERVICE WATER PUMP. AT 0152 HOURS, THE 21A EMERGENCY SERVICE WATER PUMP WAS SHUTDOWN AND 62 MILLISECONDS LATER, THE 4KV BUS 2A SUPPLY BREAKER (ACB-42C) TRIPPED ON PHASE OVERCURRENT. THE LOSS OF THE 2A BUS CAUSED THE 21A REACTOR COOLANT PUMP TO TRIP ON UNDERVOLTAGE RESULTING IN A PLANT TRIP. THE AUTO-TRANSFER OF THE 2A BUS DID NOT OCCUR DUE TO THE OVERCURRENT TRIP. DE-ENERGIZING THE 2AE BUS CAUSED THE NO. 1 EMERGENCY DIESEL GENERATOR TO AUTOMATICALLY START AND ENERGIZE THE 4KV EMERGENCY BUS 2AE. DURING THIS TRANSIENT, THE RADIATION MONITORS FOR THE UNIT II CONTROL ROOM DE-ENERGIZED MOMENTARILY RESULTING IN THE ACTUATION OF THE CONTROL ROOM EMERGENCY BOTTLED AIR PRESSURIZATION SYSTEM (CREBAPS). THE SERVICE WATER PUMP DISCHARGE PRESSURE TRANSMITTER WAS CALIBRATED SATISFACTORILY AND IS SCHEDULED FOR REPLACEMENT. A MODIFICATION TO PREVENT A CREBAPS ACTUATION ON A LOSS OF POWER TO THE UNIT 2 CONTROL ROOM RADIATION MONITORS IS BEING EVALUATED. THE CAUSE OF THE ACB-42C TRIP IS UNDER INVESTIGATION. THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC AS A RESULT OF THIS EVENT AS ALL PROTECTIVE SYSTEMS ACTUATION AS DESIGNED.

[5] BEAVER VALLEY 2 DOCKET 50-412 LER 88-003
IMPROPER CLEARANCE RESULTS IN ESF ACTUATION.
EVENT DATE: 012988 REPORT DATE: 022988 NSSS: WE TYPE: PWR

(NSIC 208413) ON 1/29/88, AN EQUIPMENT CLEARANCE PERMIT WAS GENERATED IN ORDER TO PERFORM CORRECTIVE MAINTENANCE ON THE 1H9 4160VAC BREAKER (4160VAC SUPPLY BREAKER FROM THE 2H BUS TO THE 480VAC 2K BUS). A SWITCHING ORDER WAS PREPARED DETAILING THE ELECTRICAL SWITCHING REQUIRED TO MAINTAIN THE 2K BUS ENERGIZED AFTER BREAKER 1H9 WAS OPENED. THIS WAS TO BE ACCOMPLISHED BY CLOSING THE 480VAC TIE-BREAKER BETWEEN THE 2J AND 2K BUSSES AND THEN OPENING THE 2K 480VAC SUPPLY BREAKER. DUE TO AN ERROR IN THE SWITCHING ORDER, THE 2J 480VAC SUPPLY BREAKER WAS OPENED INSTEAD OF THE 2K 480VAC SUPPLY BREAKER. WHEN BREAKER 1H9 WAS OPENED, BOTH THE 2J AND 2K BUSSES WERE DE-ENERGIZED. THIS CAUSED A LOSS OF POWER TO THE LEAK COLLECTION VENT RADIATION MONITOR CAUSING A VENTILATION REALIGNMENT TO THE MAIN FILTER BANK (ESF ACTUATION). THE CAUSE FOR THIS EVENT WAS PERSONNEL ERROR DURING THE APPROVAL OF THE SWITCHING ORDER. TO PREVENT FUTURE OCCURRENCES OF THIS TYPE, THE INDIVIDUALS INVOLVED WERE COUNSELED REGARDING THE PROPER SELECTION OF CLEARANCE POINTS. ADDITIONALLY, THE FEASIBILITY OF CHANGING THE REALIGNMENT FEATURE ON A RADIATION MONITOR LOSS OF POWER IS BEING INVESTIGATED. THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC DUE TO THIS EVENT BECAUSE THERE WAS NO ACTUAL RADIATION RELEASE TO INITIATE A VENTILATION REALIGNMENT.

[6] BIG ROCK POINT DOCKET 50-155 LER 88-001
UPSCALE/DOWNSCALE REACTOR TRIP DUE TO PERSONNEL ERROR.
EVENT DATE: 020288 REPORT DATE: 022388 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 208392) ON FEBRUARY 2, 1988, WITH PLANT SHUTDOWN IN PROGRESS FOR A PLANNED MAINTENANCE OUTAGE, A REACTOR PROTECTION SYSTEM (JC) ACTUATION OCCURRED. POWER LEVEL AT THE TIME OF THE TRIP WAS LESS THAN 25E-5 PERCENT AND ALL CONTROL RODS (AA) WITHDRAWN SUCCESSFULLY INSERTED. CAUSE OF THE TRIP WAS A COMBINATION OF OPERATOR ERROR DUE TO INCORRECT RANGE SWITCH MOVEMENT AND SUSCEPTIBILITY OF THE PICOAMMETERS TO ELECTRICAL NOISE. FOLLOWING THE TRIP PLANT COOLDOWN CONTINUED. NO OTHER ENGINEERED SAFETY FEATURES WERE CHALLENGED BY THE TRIP. ALTHOUGH NOT AS A DIRECT RESULT OF THIS EVENT, PLANS ARE BEING MADE TO CHANGE OUT THE POWER RANGE MONITORING INSTRUMENTATION TO THE NEW GENERAL ELECTRIC "NUMAC" LINE WHICH DO NOT UTILIZE RANGE SWITCHES AND ARE BELIEVED TO BE MUCH LESS SUSCEPTIBLE TO ELECTRICAL NOISE. THIS IS NOT CONSIDERED A REACTOR TRIP WHILE CRITICAL PER INPO REPORTING CRITERIA.

[7] BIG ROCK POINT DOCKET 50-155 LER 88-002
MANUAL REACTOR TRIP DUE TO LOSS OF RECIRCULATION FLOW.
EVENT DATE: 020588 REPORT DATE: 030788 NSSS: GE TYPE: BWR

(NSIC 208449) ON FEBRUARY 5, 1988 AT 0816 HOURS, BOTH (2) RECIRCULATION PUMPS TRIPPED AS DESIGNED, DURING A MOMENTARY LOSS OF STATION POWER (THREE SECONDS). PER PROCEDURE, PLANT OPERATORS TRIPPED THE REACTOR MANUALLY IN RESPONSE TO THE LOSS OF RECIRCULATION FLOW AND ALL CONTROL RODS WITHDRAWN SUCCESSFULLY INSERTED. NO OTHER ENGINEERED SAFETY FEATURES WERE CHALLENGED BY THE EVENT. REACTOR POWER WAS APPROXIMATELY 10 PERCENT AT THE TIME OF THE TRIP. THE MOMENTARY LOSS OF STATION POWER OCCURRED WHEN THE 138KV TRANSMISSION TIE BREAKER TRIPPED DURING MAIN GENERATOR SYNCHRONIZATION. THE BREAKER TRIP WAS CAUSED WHEN THE PLANT OPERATOR ATTEMPTED TO CLOSE THE MAIN GENERATOR BREAKER SLIGHTLY OUT OF PHASE WITH THE LOAD. FOLLOWING CHECK-OUT OF ALL ASSOCIATED EQUIPMENT (RELAYS, BREAKERS, TRANSFORMER), NO EQUIPMENT PROBLEMS WERE IDENTIFIED AND RESTART COMMENCED ON FEBRUARY 7, 1988. TO PREVENT RECURRENCE A TRAINING MEMO WAS DEVELOPED TO DISCUSS THE EVENT, SYSTEM OPERATION, AND PROPER METHOD OF SYNCHRONIZATION WITH ALL PLANT OPERATORS.

[8] BRAIDWOOD 1 DOCKET 50-456 LER 87-063 REV 01
UPDATE ON INOPERABLE NON-ACCESSIBLE VA FILTER PLENUM DUE TO MISALIGNMENT.
EVENT DATE: 121887 REPORT DATE: 012088 NSSS: WE TYPE: PWR

(NSIC 208066) AT 0810 ON DECEMBER 18, 1987, LIMITING CONDITION FOR OPERATION ACTION REQUIREMENT (LCOAR) WAS ENTERED IN SUPPORT OF A CHARCOAL SAMPLE SURVEILLANCE ON THE 'OC' AUXILIARY BUILDING NON-ACCESSIBLE FILTER PLENUM. AT 2330 IT WAS DISCOVERED THAT THE INLET ISOLATION DAMPER FOR PLENUM "OB" WAS CLOSED BUT OPERABLE. THE 'OB' INLET ISOLATION DAMPER WAS IMMEDIATELY OPENED CHANGING THE 'OB' PLENUM STATUS FROM STANDBY TO IN-SERVICE. THE CAUSE WAS FAILURE TO ADEQUATELY VERIFY SYSTEM STATUS PRIOR TO AUTHORIZING THE OUT-OF-SERVICE IN THAT THE IN-SERVICE PLENUM 'OC' WAS REMOVED FROM SERVICE PRIOR TO PLACING THE 'OB' PLENUM IN-SERVICE. CONTRIBUTING TO THIS WAS A MISAPPLICATION OF SUPPLEMENTARY GUIDANCE RELATIVE TO THE SUBJECT TECHNICAL SPECIFICATION. THIS EVENT HAS BEEN REVIEWED WITH THE APPROPRIATE SHIFT OPERATING PERSONNEL INVOLVED IN THIS EVENT. DOCUMENTED TRAINING OF STATION CONTROL ROOM ENGINEERS WITH REGARD TO THE BASIS FOR THE SUPPLEMENTARY GUIDANCE FOR THE SUBJECT TECHNICAL SPECIFICATION AS WELL AS SIMILAR EVENTS EXPERIENCED AT BYRON STATION WILL BE PROVIDED. THIS REPORT WILL BE INCLUDED IN THE LICENSED OPERATOR REQUALIFICATION PROGRAM. NO PREVIOUS OCCURRENCES.

[9] BRAIDWOOD 1 DOCKET 50-456 LER 88-001
TRAIN A CONTROL ROOM RADIATION MONITORING INOPERABLE DUE TO NOISY PRESSURE SWITCHES.
EVENT DATE: 010888 REPORT DATE: 020288 NSSS: WE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.

ITT-BARTON

(NSIC 208241) AT 0950 ON JANUARY 8, 1988, AND AT 1024 ON JANUARY 12, 1988, SPURIOUS SPIKES WERE RECEIVED ON TRAIN A CONTROL ROOM RADIATION MONITOR OPR32J, WHICH CAUSED AN AUTOMATIC ACTUATION OF THE CONTROL ROOM VENTILATION SYSTEM WHICH SHIFTED THE VENTILATION SYSTEM TO ITS EMERGENCY MAKEUP MODE. AT 1230 ON JANUARY 11, 1988, A SPURIOUS SPIKE WAS RECEIVED ON TRAIN A CONTROL ROOM RADIATION MONITOR OPR31J, CAUSING AN AUTOMATIC ACTUATION OF THE CONTROL ROOM VENTILATION SYSTEM TO SHIFT TO ITS EMERGENCY MAKEUP MODE. IN ALL THREE OCCURRENCES, CHARCOAL ADSORBER BYPASS DAMPER OVC43Y FAILED TO CLOSE. AFTER THE OCCURRENCES ON JANUARY 8, AND JANUARY 11, IT WAS DETERMINED THAT THE ACTUATIONS WERE SPURIOUS, AND THE MONITORS WERE IMMEDIATELY RETURNED TO SERVICE. AFTER THE OCCURRENCE ON JANUARY 12, TRAIN A OF CONTROL ROOM VENTILATION WAS MAINTAINED IN ITS EMERGENCY MAKEUP MODE. A WORK REQUEST WAS WRITTEN TO INVESTIGATE THE FAILURE OF THE DAMPER TO CLOSE. ADDITIONALLY, NOISE ATTENUATING FILTERS ARE BEING INSTALLED, AS NECESSARY, TO THE PRESSURE SWITCH CIRCUITRY TO ELIMINATE ANY FUTURE SPIKING PROBLEMS. THERE HAVE BEEN NO PREVIOUS OCCURRENCES.

[10] BRAIDWOOD 1 DOCKET 50-456 LER 88-003
LOSS OF PULSES TO FUEL HANDLING INCIDENT MONITOR QRT-AR056 FOR UNKNOWN REASONS.
EVENT DATE: 011388 REPORT DATE: 012988 NSSS: WE TYPE: PWR

(NSIC 208243) AT 1910 ON JANUARY 13, 1988, FUEL HANDLING BUILDING INCIDENT MONITOR ORT-AR056 WENT INTO AN ALARM CONDITION DUE TO A LOSS OF PULSES. THE ROOT CAUSE OF THE EVENT IS UNKNOWN. AN IMMEDIATE INVESTIGATION REVEALED NO WORK ACTIVITIES IN THE VICINITY OF MONITOR ORT-AR056. THE DETECTOR WAS INSPECTED AND NO PHYSICAL DAMAGE WAS FOUND. THE DETECTOR CABLE WAS CHECKED FOR TIGHTNESS AND WAS ABLE TO BE TIGHTENED TWO TURNS. THE CABLE SLACKNESS IS NOT CONSIDERED TO HAVE CAUSED THE LOSS OF PULSES WHICH RESULTED IN THE FUEL HANDLING BUILDING TO SHIFT TO ITS EMERGENCY MAKEUP MODE OF OPERATION. THE LOSS OF PULSES IMMEDIATELY CLEARED AND HAS NOT RECURRED. THERE HAS BEEN ONE PREVIOUS OCCURRENCE.

[11] BRAIDWOOD 1 DOCKET 50-456 LER 88-002
REACTOR TRIP AND SAFETY INJECTION DUE TO COGNITIVE PERSONNEL ERROR.
EVENT DATE: 012588 REPORT DATE: 021788 NSSS: WE TYPE: PWR

(NSIC 208363) AT 1449 ON JANUARY 25, 1988, A REACTOR TRIP SIGNAL AND A SAFETY INJECTION WERE GENERATED AS A RESULT OF A COGNITIVE PERSONNEL ERROR BY AN INSTRUMENT MECHANIC DURING THE PERFORMANCE OF INSTRUMENT SURVEILLANCE ANALOG OPERATIONAL TEST AND CHANNEL VERIFICATION/CALIBRATION FOR LOOP P-0935, CONTAINMENT PRESSURE III. INSTEAD OF MAKING A CONNECTION TO CHANNEL 935, HE MADE THE CONNECTION TO CHANNEL 936, WHICH WAS AN ACTIVE ENGINEERED SAFEGUARD FEATURE CHANNEL. THIS RESULTED IN A REACTOR TRIP SIGNAL AND A SAFETY INJECTION, WHICH INJECTED APPROXIMATELY 1250 GALLONS OF WATER INTO THE REACTOR COOLANT SYSTEM. ALL SYSTEMS PERFORMED AS DESIGNED. IMMEDIATE CORRECTIVE ACTIONS INCLUDED TERMINATING THE SAFETY INJECTION, SECURING THE RUNNING EQUIPMENT, AND RESTORING LOOP 936 TO ITS ORIGINAL CONFIGURATION. LONG TERM CORRECTIVE ACTION IS CURRENTLY BEING FINALIZED. A SUPPLEMENTAL REPORT WILL BE ISSUED. THERE HAVE BEEN NO PREVIOUS OCCURRENCES.

[12] BRAIDWOOD 1 DOCKET 50-456 LER 88-005
DIESEL GENERATOR START ON A SAFETY INJECTION SIGNAL INSTEAD OF AN UNDERVOLTAGE SIGNAL DURING TESTING DUE TO OPERATOR MISCOMMUNICATION.
EVENT DATE: 020488 REPORT DATE: 022388 NSSS: WE TYPE: PWR

(NSIC 208475) AT 1352 ON FEBRUARY 4, 1988, DURING THE PERFORMANCE OF SURVEILLANCE 1BWVS 8.1.1.2.F-13, THE 1A DIESEL GENERATOR AUTO-STARTED ON A SAFETY INJECTION SIGNAL INSTEAD OF AN UNDERVOLTAGE AS A RESULT OF A MISCOMMUNICATION BETWEEN THE

TWO LICENSED OPERATORS CONDUCTING THE SURVEILLANCE. THE MISCOMMUNICATION OCCURRED WHILE THE OPERATORS WERE PRACTICING THE SEQUENCE THAT WOULD BE USED FOR THE TEST. CORRECTIVE ACTION INCLUDED SECURING THE DIESEL GENERATOR, AND COUNSELING THE INDIVIDUALS INVOLVED IN THE TEST STRESSING THE IMPORTANCE OF EFFECTIVE COMMUNICATIONS. THERE HAVE BEEN NO PREVIOUS OCCURRENCES OF INADVERTENT DIESEL GENERATOR ACTUATIONS DURING SURVEILLANCE TESTING.

[13] BRAIDWOOD 2 DOCKET 50-457 LER 88-002
CONTAINMENT VENTILATION ISOLATION FROM LOSS OF PULSES TO RADIATION MONITOR 2RT-AR011 DUE TO LOW BACKGROUND RADIATION.
EVENT DATE: 012488 REPORT DATE: 021688 NSSS: WE TYPE: PWR

(NSIC 208300) AT 0727 ON JANUARY 24, 1988, THE CONTAINMENT BUILDING FUEL HANDLING INCIDENT AREA RADIATION MONITOR 2RT-AR011 MOMENTARILY WENT INTO AN ALERT ALARM AND INTERLOCK ACTUATION DUE TO A LOSS OF PULSES. THE ASSOCIATED CONTAINMENT ISOLATION VALVES WERE ALREADY CLOSED. THE ALARM WAS ACKNOWLEDGED AND THE ALARM STATUS CLEARED. AN IMMEDIATE INVESTIGATION DID NOT REVEAL ANY WORK ACTIVITY IN THE AREA THAT WOULD HAVE CONTRIBUTED TO THE EVENT. THEREFORE, THE CONTAINMENT ISOLATION SIGNAL WAS IMMEDIATELY RESET. THE EVENT WAS CAUSED BY PULSES NOT BEING RECEIVED BY THE DETECTOR WITHIN A FIVE MINUTE TIME PERIOD DUE TO LOW BACKGROUND RADIATION. THE PULSE TIME INTERVAL HAS BEEN INCREASED FROM FIVE MINUTES TO 10 MINUTES AS NOTED IN THE VENDOR MANUAL. ONE PREVIOUS OCCURRENCE OF A LOSS OF PULSES. THAT EVENT WAS DUE TO CONTRACTOR ACTIVITY WHICH DAMAGED THE DETECTOR. REFERENCE LER 87-003 DOCKET 456.

[14] BRAIDWOOD 2 DOCKET 50-457 LER 88-004
UNDERVOLTAGE START OF 2A DIESEL GENERATOR DUE TO OPERATOR ERROR.
EVENT DATE: 012988 REPORT DATE: 022588 NSSS: WE TYPE: PWR

(NSIC 208364) AT 1027 ON JANUARY 29, 1988, THE UNIT 2 EMERGENCY DIESEL GENERATOR (DG) (EK) AUTO-STARTED. THIS WAS THE RESULT OF AN UNDERVOLTAGE SIGNAL GENERATED BY THE INADVERTENT REMOVAL OF THE BUS 241 POTENTIAL TRANSFORMER FUSES BY AN EQUIPMENT OPERATOR, (EO), INSTEAD OF THE BUS 24 SYSTEM AUXILIARY TRANSFORMER FEED POTENTIAL TRANSFORMER FUSES. THE DG CLOSED TO THE BUS, BUT THE TRAIN A EQUIPMENT DID NOT SEQUENCE ON BECAUSE THE SEQUENCER DID NOT SEE VOLTAGE ON THE BUS. IMMEDIATE CORRECTIVE ACTION INCLUDED STARTING THE 2B ESSENTIAL SERVICE WATER PUMP TO PROVIDE COOLING TO THE DG, SHUTTING DOWN OF THE 2A AND 2B REACTOR COOLANT PUMPS WHICH WERE RUNNING AT THE TIME OF THE EVENT, AND REINSTALLING THE BUS FUSES AND RESTORING POWER TO BUS 241. THE EVENT WAS REVIEWED WITH THE INDIVIDUAL INVOLVED, WHO HAS RECEIVED ADDITIONAL TRAINING, WHICH INCLUDED A WALK-THROUGH WITH A TRAINING INSTRUCTOR AND A LICENSED FOREMAN TO VERIFY HIS TECHNICAL KNOWLEDGE. ALSO, THE INFORMATION SHEET WHICH WAS IN USE AT THE TIME IS BEING REVISED TO REFLECT THE ACTUAL EQUIPMENT NOMENCLATURE IN THE FIELD. THERE HAVE BEEN NO PREVIOUS OCCURRENCES OF AN AUTO-START OF A DG AS A RESULT OF REMOVING THE WRONG POTENTIAL FUSES.

[15] BROWNS FERRY 1 DOCKET 50-259 LER 85-026 REV 03
UPDATE ON CABLE TRAY NOT BEING SEISMICALLY QUALIFIED.
EVENT DATE: 070185 REPORT DATE: 012688 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 208248) ON JULY 1, 1985, FIELD INSPECTIONS AND SUBSEQUENT STRUCTURAL EVALUATIONS DETERMINED VARIOUS CABLE TRAY SECTIONS AND THEIR SUPPORTS NOT SEISMICALLY QUALIFIED IN ACCORDANCE WITH THE PLANT'S ORIGINAL DESIGN CRITERIA. IN ADDITION, A NUMBER OF CABLE TRAY SECTIONS WERE DETERMINED TO HAVE EXCESSIVE FIRE RETARDANT COATING APPLIED, RAISING A CONCERN ON CABLE AMPACITY. THE ESSENTIAL CABLE TRAYS AND THEIR SUPPORTS IN UNIT 2 AND IN OTHER AREAS ESSENTIAL

TO OPERATION OF UNIT 2, WERE SUBJECTED TO AN EVALUATION BY UNITED ENGINEERS AND CONSTRUCTORS (UE&C). THIS EVALUATION SPECIFIED MODIFICATIONS TO ENSURE STRUCTURAL INTEGRITY OF THE CABLE TRAY SUPPORTS. THESE MODIFICATIONS WILL BE COMPLETED PRIOR TO UNIT 2 STARTUP. A DESIGN STANDARD HAS BEEN WRITTEN TO DERATE CABLE BASED UPON EXCESSIVE DEPTH. CABLES DETERMINED TO BE UNSATISFACTORY WILL BE REPLACED.

[16] BROWNS FERRY 1 DOCKET 50-259 LER 88-002
ENGINEERED SAFETY FEATURE ACTUATION DUE TO PERSONNEL ERROR DURING SWITCH CALIBRATION.
EVENT DATE: 010688 REPORT DATE: 020588 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 208115) ON JANUARY 6, 1988, AT 1954 HOURS, WITH ALL THREE UNITS DEFUELED, FOUR EMERGENCY EQUIPMENT COOLING WATER (EECW) PUMPS WERE INADVERTENTLY STARTED DUE TO A PERSONNEL ERROR ASSOCIATED WITH THE CALIBRATION OF A RAW COOLING WATER (RCW) PRESSURE SWITCH. LOW RCW PRESSURE IS A DESIGNED START SIGNAL FOR THE EECW PUMPS. WHILE RETURNING A WIRE TO ITS NORMAL POSITION IT WAS ALLOWED TO MOMENTARILY CONTACT AN ADJACENT TERMINAL POSITION. THIS SIMULATED A LOW RCW PRESSURE AND THEREBY COMPLETED THE START LOGIC FOR THE EECW PUMPS. THIS WAS AN UNPLANNED ACTUATION OF AN ENGINEERED SAFETY FEATURE. THE CRAFT PERSONNEL PERFORMING THE CALIBRATION RETURNED THE CIRCUIT TO THE NORMAL CONFIGURATION IN ACCORDANCE WITH THE PROCEDURE. OPERATIONS PERSONNEL SURVEYED THE PANEL FOR BLOWN FUSES OR DAMAGE AND FOUND NONE. FOUR EECW PUMPS WERE RETURNED TO STANDBY READINESS BY 2100 HOURS. THE CRAFT PERSONNEL INVOLVED WERE CAUTIONED TO EXERCISE EXTREME CARE WHEN WORKING WITH ENERGIZED CIRCUITS. ALL INSTRUMENT MAINTENANCE PERSONNEL WILL BE PROVIDED WITH A DESCRIPTION OF THIS EVENT.

[17] BROWNS FERRY 1 DOCKET 50-259 LER 88-003
INADEQUATE PROCEDURE CAUSES INADVERTENT START OF EMERGENCY EQUIPMENT COOLING WATER PUMPS.
EVENT DATE: 011288 REPORT DATE: 020588 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 208261) ON JANUARY 12, 1988, AT 0043 HOURS, WITH ALL THREE UNITS DEFUELED, FOUR EMERGENCY EQUIPMENT COOLING WATER (EECW) PUMPS WERE INADVERTENTLY STARTED DUE TO A PROCEDURE DEFICIENCY. WHILE ATTEMPTING TO PUT A RAW COOLING WATER (RCW) PUMP INTO SERVICE AND TAKE ANOTHER PUMP OUT OF SERVICE, THE RCW HEADER PRESSURE DROPPED BELOW THE LOW PRESSURE SETPOINT. THIS IS A DESIGNED START SIGNAL FOR THE EECW PUMPS. THIS WAS AN UNPLANNED ACTUATION OF AN ENGINEERED SAFETY FEATURE. THE ASSISTANT UNIT OPERATOR AND THE ASSISTANT SHIFT ENGINEER RESTARTED THE REQUIRED RCW PUMPS AND RETURNED FOUR EECW PUMPS TO STANDBY READINESS. THE OPERATING INSTRUCTIONS FOR THE RCW SYSTEM WILL BE REVISED TO PROVIDE INSTRUCTIONS FOR ALTERNATING PUMPS IN AND OUT OF SERVICE. A REVIEW OF THIS EVENT WILL BE PROVIDED TO CURRENT OPERATIONS PERSONNEL.

[18] BROWNS FERRY 1 DOCKET 50-259 LER 88-001
UNPLANNED REACTOR WATER CLEANUP ISOLATION DUE TO LOOSE CONNECTION.
EVENT DATE: 011788 REPORT DATE: 021288 NSSS: GE TYPE: BWR

(NSIC 208320) ON JANUARY 17, 1988, AT 0810 HOURS, WITH THE UNIT DEFUELED, THE TEMPERATURE SWITCH MONITORING THE DISCHARGE OF THE UNIT 1 REACTOR WATER CLEANUP (RWCU) HEAT EXCHANGER INITIATED A SPURIOUS ISOLATION OF THE RWCU SYSTEM THROUGH THE PRIMARY CONTAINMENT ISOLATION SYSTEM. THIS WAS AN UNPLANNED ACTUATION OF AN ENGINEERED SAFETY FEATURE. AN INVESTIGATION WAS INITIATED TO DETERMINE THE CAUSE OF THE ISOLATION. THE WATER TEMPERATURE WAS VERIFIED TO BE WELL BELOW THE

ISOLATION SETPOINT. DURING TROUBLESHOOTING THE RELAY CONTROLLED BY THE TEMPERATURE SWITCH RESET TO THE NORMAL POSITION ALLOWING THE OPERATOR TO RESET THE ISOLATION LOGIC. A LOOSE SOLDER CONNECTION THAT INITIATED THE ISOLATION WAS FOUND IN THE TEMPERATURE SWITCH. THE CONNECTION WAS REPAIRED AND THE SWITCH RECALIBRATED. THE SYSTEM WAS RETURNED TO SERVICE THE FOLLOWING DAY AT 0945 HOURS. A BRUSH RECORDER WAS TEMPORARILY CONNECTED TO MONITOR SWITCH BEHAVIOR. NO ABNORMALITIES WERE OBSERVED DURING THE THIRTY HOURS THE RECORDER WAS ATTACHED. NO FURTHER ACTIONS ARE PLANNED.

[19]   BROWNS FERRY 1   DOCKET 50-259   LER 88-008
STANDBY GAS TREATMENT RELATIVE HUMIDITY HEATERS HAVE NOT BEEN TESTED IN ACCORDANCE WITH TECH SPECS DUE TO INADEQUATE PROCEDURES.
EVENT DATE: 012088   REPORT DATE: 021988   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 208388) ON JANUARY 20, 1988, DURING A PROGRAMMATIC UPGRADE OF THE BROWNS FERRY SURVEILLANCE INSTRUCTIONS IT WAS DISCOVERED THAT THE SURVEILLANCE INSTRUCTION WHICH TESTS THE STANDBY GAS TREATMENT (SBGT) RELATIVE HUMIDITY HEATERS DID NOT FULLY IMPLEMENT THE TESTING DESCRIBED IN THE AMERICAN NATIONAL STANDARDS INSTITUTE (ANSI) STANDARD N510-1975 AS REQUIRED BY TECH SPEC 4.7.B.1.B. ALL THREE TRAINS OF SBGT WERE DECLARED INOPERABLE. ANSI N510-1975 HAD BEEN INCORPORATED INTO TECH SPECS BY AUGUST 20, 1976. SINCE THAT TIME THE REACTORS HAVE BEEN OPERATED AT POWER AND FUEL HANDLING OPERATIONS HAVE BEEN CARRIED OUT. FAILURE TO IMPLEMENT A CHANGE TO THE TECH SPEC BY REVISING THE APPLICABLE SURVEILLANCE INSTRUCTION ALLOWED OPERATION OUTSIDE OF THE TECH SPECS. AT THE TIME OF DISCOVERY ALL THREE UNITS WERE DEFUELED. ALL THREE SBGT TRAINS HAD BEEN PREVIOUSLY DECLARED INOPERABLE FOR OTHER REASONS. ALL FUEL HANDLING AND OPERATIONS OVER THE SPENT FUEL POOLS HAD BEEN HALTED. THE SURVEILLANCE INSTRUCTION HAS BEEN REVISED TO FULLY INCORPORATE THE TESTING DESCRIBED IN ANSI N510-1975. THE REVISED SURVEILLANCE INSTRUCTION IS CURRENTLY BEING RUN ON ALL THREE TRAINS OF SBGT.

[20]   BROWNS FERRY 1   DOCKET 50-259   LER 88-005
DIESEL GENERATOR AND EMERGENCY EQUIPMENT COOLING WATER PUMP ACTUATIONS DUE TO PERSONNEL ERROR.
EVENT DATE: 012288   REPORT DATE: 021988   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)

(NSIC 2083..) ON JANUARY 22, 1988, AT 1019 HOURS, A POTENTIAL TRANSFORMER FUSE COMPARTMENT WAS OPENED FOR INSPECTION. THIS ACTION RESULTED IN THE DEENERGIZATION OF A 4KV SHUTDOWN BOARD AND THE START OF A DIESEL GENERATOR (DG) AND AN EMERGENCY EQUIPMENT COOLING WATER (EECW) PUMP. UNITS 1, 2, AND 3 WERE DEFUELED WHEN THIS EVENT OCCURRED. OPERATIONS RESTORED NORMAL POWER TO THE SHUTDOWN BOARD AND RETURNED THE DG AND EECW PUMP TO STANDBY READINESS BY 1045 HOURS. THESE ENGINEERED SAFETY FEATURE ACTUATIONS WERE A RESULT OF PERSONNEL ERROR CAUSED BY A FAILURE TO OBSERVE POSTED WARNINGS, INADEQUATE COMMUNICATION, AND INEXPERIENCE. THE PERSONNEL INVOLVED HAVE BEEN COUNSELED ON THE EVENT. ADDITIONALLY THE WALKDOWN COORDINATOR HAS REVIEWED THE EVENT WITH ENGINEERING AND CONTRACT PERSONNEL TO ADDRESS THE EXTENT TO WHICH THEY CAN DIRECT CRAFT EFFORT DURING A WALKDOWN. A DESCRIPTION OF THIS EVENT WILL BE PROVIDED TO MAINTENANCE, MODIFICATIONS, OPERATIONS, AND ENGINEERING GROUPS. THE WALKDOWN PROCEDURE WILL BE UPGRADED TO INCLUDE A PLANNING REVIEW BY A PERSON WITH INTEGRATED PLANT KNOWLEDGE WHEN PHYSICAL WORK IS REQUIRED.

[21] BROWNS FERRY 1 DOCKET 50-259 LER 88-010
INADEQUATE PROCEDURES CAUSE TWO CASES OF MISSED SAMPLES THAT WERE REQUIRED TO COMPENSATE FOR INOPERABLE EFFLUENT RADIATION MONITORS.
EVENT DATE: 012288 REPORT DATE: 030488 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 3 (BWR)

(NSIC 208473) ON JANUARY 22, 1988 AT 2100 HOURS, AND ON JANUARY 29, 1988, AT 1500 HOURS, WITH ALL THREE UNITS DEFUELED, TWO SIMILAR EVENTS INVOLVING MISSED COMPENSATORY SAMPLING FOR INOPERABLE EFFLUENT RADIATION MONITORS OCCURRED. THE CAUSE OF BOTH EVENTS WAS INADEQUATE PROCEDURES. THE FIRST EVENT OCCURRED WHEN COMPENSATORY SAMPLING OF THE RAW COOLING WATER (RCW) DISCHARGE FROM A REACTOR BUILDING CLOSED COOLING WATER HEAT EXCHANGER WAS STOPPED. THE HEAT EXCHANGER WAS REMOVED FROM SERVICE BUT THE RCW FLOW WAS NOT ISOLATED. THE SECOND EVENT WAS DECLARING THE OFF-GAS POSTTREATMENT MONITORS OPERABLE WITHOUT COMPLETING ALL SURVEILLANCE REQUIREMENTS. COMPENSATORY SAMPLING WAS STOPPED. THE IMMEDIATE CORRECTIVE ACTION FOR BOTH EVENTS WAS THE RESUMPTION OF THE COMPENSATORY SAMPLING. THE CORRECTIVE ACTION FOR BOTH EVENTS IS THE REVISION OF APPLICABLE PROCEDURES.

[22] BROWNS FERRY 1 DOCKET 50-259 LER 88-007
RESIDUAL HEAT REMOVAL SERVICE WATER PUMP FLOW NOT DEMONSTRATED IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS DUE TO CALIBRATION ERROR.
EVENT DATE: 012788 REPORT DATE: 022688 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 208403) ON JANUARY 27, 1988, AT 1300 HOURS, WITH ALL UNITS DEFUELED, A CALIBRATION ERROR WAS DISCOVERED IN THE FOUR RESIDUAL HEAT REMOVAL SERVICE WATER (RHRSW) PUMP FLOW INSTRUMENTS. AN INDICATED FLOW OF 4500 GPM CORRESPONDS TO AN ACTUAL FLOW OF APPROXIMATELY 3400 GPM. IT WAS CAUSED BY AN ERROR ON THE FLOW TRANSMITTER DATA SHEET AND FAILURE TO VERIFY THE NSSS SUPPLIED DATA SHEET AGAINST THE VENDOR FLOW ORIFICE CALCULATIONS FOR THE FLOW ELEMENT. ALL RHRSW PUMPS WILL BE DEMONSTRATED OPERABLE WITH CORRECTLY CALIBRATED FLOW INSTRUMENTS PRIOR TO REBURNING TO SERVICE. VENDOR FLOW CALCULATION DATA SHEETS WILL BE OBTAINED AND WILL BE VERIFIED FOR ALL FLOW LOOPS WITH GENERAL ELECTRIC SUPPLIED DATA SHEETS.

[23] BROWNS FERRY 1 DOCKET 50-259 LER 88-009
UNPLANNED STANDBY GAS TREATMENT TRAIN ACTUATIONS DUE TO INADVERTENT RELAY OPERATIONS.
EVENT DATE: 020288 REPORT DATE: 030388 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 208472) ON FEBRUARY 2, 1988, WITH ALL THREE UNITS DEFUELED, THE C TRAIN OF STANDBY GAS TREATMENT (SBGT) WAS BEING RETURNED TO SERVICE FOLLOWING EXTENSIVE MAINTENANCE TO THE TRAIN'S SUPPLY BREAKER. AT 1902 HOURS, WITH NO AUTOMATIC INITIATION SIGNALS PRESENT, THE SUPPLY BREAKER WAS MANUALLY CLOSED AND THE TRAIN SPURIOUSLY ACTUATED. THE BREAKER WAS MANUALLY OPENED, THE MAIN CONTROL ROOM HAND SWITCH WAS PLACED IN STOP (SPRING RETURN TO AUTO) AND THE ISOLATION LOGIC WAS RESET. THE BREAKER WAS MANUALLY CYCLED AGAIN AT 1917 HOURS CAUSING A SECOND SPURIOUS ACTUATION. THE BREAKER WAS MANUALLY CLOSED A THIRD TIME, AT 1943 HOURS, AGAIN ACTUATING THE TRAIN. THE TRAIN WAS SECURED THIS TIME BY PLACING THE MAIN CONTROL ROOM HAND SWITCH IN STOP. THE FIRST TWO STARTS WERE UNPLANNED ACTUATIONS OF AN ENGINEERED SAFETY FEATURE. THE CONTROL LOGIC WAS INSPECTED AND FOUND TO BE IN THE PROPER CONFIGURATION AND OPERATING PROPERLY. THE SUSPECTED CAUSE OF THE ACTUATIONS WAS A LATCHING RELAY (LABELED MCX) IN THE TRAIN'S CONTROL LOGIC. THE MCX RELAY WAS INADVERTENTLY LATCHED IN THE ACTUATION STATE DURING THE REFERENCED

MAINTENANCE ACTIVITES. FUNCTIONAL TESTING COULD ONLY BE PERFORMED FOLLOWING MAINTENANCE WHEN THE BREAKER WAS REPLACED AND CLOSED.

[24] BRUNSWICK 1 DOCKET 50-325 LER 88-001
UNPLANNED PRIMARY CONTAINMENT GROUP 6 ISOLATION, REACTOR BUILDING VENTILATION ISOLATION, AND AUTO START OF STANDBY GAS TREATMENT SYSTEM DURING PLANT TESTING.
EVENT DATE: 010488 REPORT DATE: 020388 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)

(NSIC 208230) AT 1843 HOURS ON 1/4/88, AN UNEXPECTED UNIT 1 PRIMARY C NTAINMENT GROUP 6 ISOLATION, REACTOR BUILDING VENTILATION ISOLATION, AND AUTO S.ART OF THE STANDBY GAS TREATMENT SYSTEM OCCURRED ALONG WITH A UNIT 1 PROCESS OFF-GAS VENT WAS IN A REFUEL/MAINTENANCE OUTAGE. THE UNIT 2 LOADING (OPERABILITY) TEST OF THE UNITS' COMMON 4160 VAC EMERGENCY DIESEL GENERATOR (DG) NO. 3, MAINTENANCE SURVEILLANCE TEST (MST) 2MST-DG13R, WAS IN PROGRESS. WITH EXCEPTION OF THE SUBJECT UNIT 1 RESPONSES, THE RESPONSES OF BOTH UNITS TO THE TEST WERE AS EXPECTED. THE AFFECTED UNIT 1 SYSTEMS WERE RETURNED TO THEIR NORMAL CONFIGURATION SHORTLY AFTER THE EVENT OCCURRENCE. DUE TO AN INADEQUATE REVIEW OF A PLANT MODIFICATION AND THE ASSOCIATED PROCEDURE REVISION REQUIREMENTS, 2MST-DG13R DID NOT PROPERLY REFLECT REVISED LOGIC CIRCUITRY OF THE MAIN STACK RADIATION MONITOR. THIS LOGIC HAD BEEN REVISED THROUGH PLANT MODIFICATIONS (PM) TO MEET NUREG-0737 REQUIREMENTS. BY 2/22/88, APPROPRIATE PROCEDURE REVISIONS WILL BE IMPLEMENTED TO IDENTIFY AND/OR PREVENT FUTURE SIMILAR OCCURRENCES. THE CIRCUMSTANCES OF THIS EVENT WILL BE REVIEWED BY MAINTENANCE PERSONNEL INVOLVED WITH REVIEWS OF PMS FOR PROCEDURAL APPLICABILITY.

[25] BRUNSWICK 1 DOCKET 50-325 LER 88-003
MISSED TECH SPEC SURVEILLANCE DUE TO FAILURE TO COMPLETE THE SURVEILLANCE.
EVENT DATE: 010888 REPORT DATE: 020588 NSSS: GE TYPE: BWR
VENDOR: NORDBERG

(NSIC 208193) DURING UNIT 1 POWER OPERATION AT 100%, CONCURRENT WITH A UNIT 2 REFUEL/ MAINTENANCE OUTAGE, A TECHNICAL SPECIFICATIONS (TS) SURVEILLANCE REQUIRED BY 0045 HOURS ON 1/8/88 TO DEMONSTRATE THE OPERABILITY OF OFF-SITE ALTERNATING CURRENT (AC) POWER SOURCES WAS NOT COMPLETED UNTIL 0250 HOURS ON 1/8/88. THIS SURVEILLANCE WAS REQUIRED DUE TO THE INOPERABILITY OF THE UNITS' COMMON AC DIESEL GENERATOR (DG) NO. 2. AT THE TIME, THE UNITS' OTHER COMMON DGS (NOS. 1, 3, AND 4) WERE OPERABLE. THE SUBJECT SURVEILLANCE IS REQUIRED BY TS 4.8.1.1.1.A AND IS ACCOMPLISHED THROUGH PERFORMANCE OF PERIODIC TEST (PT)-12.8. AT 0205 HOURS ON 1/8/88, THE SUBJECT SURVEILLANCE WAS COMPLETED AND AT 0330 HOURS, DG NO. 2 WAS RETURNED TO STANDBY READINESS. THIS EVENT RESULTED FROM FAILURE OF THE UNIT 1 SHIFT FOREMAN TO ADEQUATELY TRACK AND VERIFY COMPLETION OF THE SURVEILLANCE. A CONTRIBUTING FACTOR TO THIS FAILURE WAS DISTRACTION OF THE SHIFT FOREMAN'S ATTENTION DUE TO PROBLEMS WITH THE REACTOR BUILDING VENTILATION SYSTEM. THE INVOLVED SHIFT FOREMAN WAS DISCIPLINED AND FACTORS RELATIVE TO THIS EVENT WERE EMPHASIZED WITH THE SHIFT OPERATING SUPERVISORS (SOSS) BY THE PLANT OPERATIONS MANAGER. A MEMORANDUM WAS DISTRIBUTED WHICH CALLED ATTENTION OF THE SOSS TO THIS EVENT AND POINTED OUT THE VULNERABILITY TO SUCH TYPE ERRORS AND STRESSED THE IMPORTANCE OF STRICT TS VERBATIM COMPLIANCE. THIS EVENT WILL BE REVIEWED DURING OP

[26] BRUNSWICK 1 DOCKET 50-325 LER 88-004 REV 01
UPDATE ON PRIMARY CONTAINMENT GROUP 3 ISOLATION WHILE PLACING REACTOR WATER CLEANUP SYSTEM FILTER/DEMINERALIZER A INTO SERVICE.
EVENT DATE: 011588 REPORT DATE: 021888 NSSS: GE TYPE: BWR
VENDOR: BERGEN-PATTERSON PIPE SUPPORT CORPORATION
LAMONS METAL GASKET CO.

(NSIC 208328) SHORTLY BEFORE 1553 HOURS ON 1/15/88, WHILE PLACING THE REACTOR WATER CLEANUP (RWCU) SYSTEM (G31) A FILTER DEMINERALIZER (F/D) INTO SERVICE, AN RWCU HIGH DIFFERENTIAL FLOW ALARM ANNUNCIATION WAS RECEIVED. THE CONTROL OPERATOR BEGAN ISOLATING THE RWCU SYSTEM; HOWEVER, A PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) GROUP 3 ISOLATION OCCURRED (AUTO CLOSURE OF RWCU SYSTEM SUCTION PCIS INBOARD AND OUTBOARD ISOLATION VALVES, G31-F001 AND F004). AT 0245 HOURS, THE RWCU SYSTEM B F/D WAS PLACED INTO SERVICE. UNIT 1 WAS OPERATING AT 100% POWER. THE EVENT RESULTED FROM EXCEEDING THE </= 53 GALLONS PER MINUTE PCIS LEAKAGE ACTUATION SETPOINT DUE TO LEAKAGE FROM THE A F/D HEAD FLANGE GASKET AND THE FLANGE GASKETS OF THE A F/D VESSEL PIPING. LEAKAGE FROM THE HEAD FLANGE GASKET IS ATTRIBUTED TO EXTRUSION OF THE HEAD FLANGE GASKET WHILE LEAKAGE OF THE F/D PIPING FLANGE GASKETS IS ATTRIBUTED TO MECHANICAL SHOCK OF THE PIPING FLANGE GASKETS FROM THE RESULTANT DEPRESSURIZATION OF THE F/D CAUSED BY FAILURE OF THE HEAD FLANGE GASKET. THE SUBJECT GASKETS WERE REPLACED AND ON 1/21/88, THE F/D WAS RETURNED TO SERVICE.

[27]   BRUNSWICK 1   DOCKET 50-325   LER 88-005
LIMITING CONDITION FOR OPERATION FOR PRIMARY CONTAINMENT OXYGEN CONCENTRATION EXCEEDED AND TECHNICAL SPECIFICATION 3.0.3 ENTERED DURING PLANNED UNIT SHUTDOWN.
EVENT DATE: 012488   REPORT DATE: 022388   NSSS: GE   TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 208409) AT 1737 HOURS ON JANUARY 22, 1988, DEINERTING OF THE UNIT 1 PRIMARY CONTAINMENT ATMOSPHERE WAS INITIATED IN PREPARATION FOR A SCHEDULED OUTAGE. AT 1741 HOURS, PRIMARY CONTAINMENT OXYGEN CONCENTRATION REACHED 4%. A REACTOR SHUTDOWN WAS INITIATED AT 2143 HOURS FROM A POWER OF 100%; HOWEVER, AT APPROXIMATELY 23% POWER AT 0612 HOURS ON JANUARY 23, THE ROD SEQUENCE CONTROL SYSTEM (RSCS) WAS DETERMINED TO BE INOPERABLE. TECHNICAL SPECIFICATION (TS) 3.6.6.3 REQUIRES THAT THE PLANT BE IN STARTUP WITHIN 30 HOURS OF REACHING 4% OXYGEN CONCENTRATION. AT 0141 HOURS ON JANUARY 24, THE TS WAS EXCEEDED AND TS 3.0.3 WAS ENTERED AT A POWER OF APPROXIMATELY 14%. AN INVESTIGATION INTO WHY THE TS 3.6.6.3 WAS EXCEEDED DETERMINED THAT PLANT MANAGEMENT MADE A COGNITIVE DECISION TO REMAIN AT POWER UNTIL RSCS WAS REPAIRED. TROUBLESHOOTING OF RSCS (C11) DETERMINED THAT CONTROL SWITCH (CS) 1.C11.CS-Z8-S1 WAS INCORRECTLY CONFIGURED. THE SWITCH WAS REPLACED WITH A CORRECTLY CONFIGURED SWITCH AND THE REACTOR WAS PLACED IN SHUTDOWN AT 0215 ON JANUARY 24, 34 MINUTES INTO TS 3.0.3.

[28]   BRUNSWICK 2   DOCKET 50-324   LER 88-001
MANUAL REACTOR SCRAM DUE TO DECREASING MAIN CONDENSER VACUUM AND FAILURE OF PRIMARY CONTAINMENT GROUP 2 VALVES TO CLOSE ON ISOLATION SIGNAL.
EVENT DATE: 010288   REPORT DATE: 020188   NSSS: GE   TYPE: BWR
VENDOR: ASCO VALVES
        GENERAL ELECTRIC CO.
        GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 208228) WHILE PERFORMING A ROUTINE REACTOR SHUTDOWN IN PREPARATION FOR A SCHEDULED REFUELING/MAINTENANCE OUTAGE, A MANUAL REACTOR PROTECTION SYSTEM TRIP (SCRAM) WAS INITIATED AT 0017 HOURS ON 1/2/88, DUE TO DECREASING CONDENSER VACUUM. AT THE TIME, REACTOR POWER WAS APPROXIMATELY 55% AND VACUUM HAD DECREASED TO APPROXIMATELY -22 INCHES MERCURY. FOLLOWING THE SCRAM, VESSEL LEVEL DECREASED, AS EXPECTED, TO APPROXIMATELY 153 INCHES, THUS INITIATING AN AUTOMATIC CLOSURE (APPROX. 162.5"). OPERATOR VERIFICATION OF THIS AUTOMATIC ACTION DETERMINED THAT THE GROUP 2 PCIVS 2-G16-F003, F004, F019, AND F020 FAILED TO CLOSE. THESE ARE THE INBOARD AND OUTBOARD ISOLATION VALVES FOR THE DRYWELL FLOOR DRAIN SUMP (F003, F004) AND THE DRYWELL EQUIPMENT DRAIN SUMP (F019, F020). THE REMAINING SAFETY SYSTEMS OPERATED AS DESIGNED DURING THIS EVENT. INVESTIGATION OF THE DECREASING VACUUM CONDITION WILL BE CONDUCTED DURING THE CURRENT OUTAGE AND IDENTIFIED PROBLEMS CORRECTED. INVESTIGATION OF THE FAILURE OF THE GROUP 2 PCIVS TO CLOSE HAS BEEN UNABLE TO DETERMINE A SPECIFIC FAILURE. BY 6/3/88 A

SUPPLEMENT TO THIS REPORT WILL BE ISSUED TO ADDRESS FINAL CAUSE AND CORRECTIVE ACTIONS.

[29] BRUNSWICK 2 DOCKET 50-324 LER 88-002
FAILURE OF DRYWELL HEAT OUTER SEAL REVEALED THROUGH LOCAL LEAK RATE TESTING.
EVENT DATE: 010488 REPORT DATE: 020188 NSSS: GE TYPE: BWR
VENDOR: MOORE, I. B. COMPANY

(NSIC 208229) DURING THE UNIT 2 1988 REFUEL/MAINTENANCE OUTAGE, TYPE B LOCAL LEAK RATE TESTING (LLRT) OF THE DRYWELL HEAD FLANGE SEAL ON 1/4/88 REVEALED A NONQUANTIFIABLE LEAKAGE RATE. AS SUCH, A CALCULATED PRIMARY CONTAINMENT LEAKAGE RATE OF GREATER THAN 0.60L(A), AS REQUIRED BY TECH SPEC 3.6.1.2B, COULD NOT BE ACHIEVED. THE DRYWELL HEAD FLANGE SEAL IS OF DOUBLE SEAL RING CONSTRUCTION AND THE LLRT IS PERFORMED BY PRESSURIZING BETWEEN THE INNER AND OUTER SEAL RINGS. THE FAILURE TO OBTAIN A QUANTIFIABLE LEAK RATE IS ATTRIBUTED TO DEFORMATION/EXTRUSION OF THE DRYWELL HEAD FLANGE OUTER SEAL RING (MANUFACTURED BY IRVING B. MOORE CORPORATION) IN THREE LOCATIONS. AN EXAMINATION OF THE DRYWELL HEAD INNER SEAL INDICATED THE SEAL WAS INTACT. PORTIONS OF THE FAILED OUTER SEAL WERE SENT TO THE CAROLINA POWER & LIGHT COMPANY TESTING AND ANALYSIS LABORATORY FOR FAILURE ANALYSIS OF THE SEAL MATERIAL. A FOLLOW-UP TO THIS REPORT APPROPRIATELY ADDRESSING CORRECTIVE ACTIONS TO THE DRYWELL SEAL FAILURE AND ANY OTHER LLRT REPORTABLE PROBLEMS WILL BE SUBMITTED BY 5/31/88.

[30] BRUNSWICK 2 DOCKET 50-324 LER 88-005
SAFETY RELIEF VALVES' SETPOINTS EXCEEDED DURING TESTING.
EVENT DATE: 020488 REPORT DATE: 030488 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 208459) DURING THE UNIT 2 1988 REFUEL/MAINTENANCE OUTAGE, TESTING OF 10 OF THE UNIT'S 11 SAFETY RELIEF VALVES (SRVS) IN ACCORDANCE WITH TECH SPEC 3.4.2 REVEALED SRVS F013A, B, C, G, J, AND K LIFTED AT PRESSURES OUTSIDE THE TS TOLERANCE OF APPROXIMATELY 1%. THE TEST RESULTS FOR THE FOUR VALVES WHICH FAILED HIGH ARE ATTRIBUTED TO CORROSION INDUCED BONDING OF THE SRVS' PILOT VALVE DISC TO SEAT SURFACES. TWO OF THE SRVS TESTED WERE INADVERTENTLY LIFTED DURING TESTING AS THE RESULT OF PERSONNEL ERROR BY THE TEST FACILITY, WYLE LABORATORIES. SRV F013L WAS NOT TESTED DUE TO IMPROPER DISASSEMBLY BY PLANT MAINTENANCE PERSONNEL WHO DID NOT ADHERE TO THE INVOLVED MAINTENANCE INSTRUCTION (MI). THE AUTOMATIC DEPRESSURIZATION SYSTEM AND MANUAL INITIATION FUNCTIONS OF THE SRVS WERE NOT AFFECTED BY THE OBSERVED SETPOINT DRIFT. THE RESULTING AVERAGE SETPOINT DRIFT PROVIDES A SUFFICIENT MARGIN FOR REACTOR OVERPRESSURE PROTECTION AND IS BOUNDED BY PRIOR SRV FAILURE RATE EXPERIENCED ON UNIT 2 (LER 2.86.001). THE SRVS WERE REFURBISHED AND RECERTIFIED BY WYLE AND WILL BE REINSTALLED PRIOR TO COMPLETION OF THE CURRENT UNIT OUTAGE. WYLE PLANS TO CONDUCT BRIEFINGS RELATIVE TO THIS EVENT AND WILL IMPLEMENT APPROPRIATE PROCEDURE REVISIONS. APPROPRIATE COUNSELING WAS CONDUCTED CONCERNING THE DISASSEMBLY OF SRV F013L AND THE INVOLVED MI WILL BE REVISED ACCORDINGLY.

[31] BYRON 1 DOCKET 50-454 LER 08-001
CONTROL ROOM VENTILATION ACTUATION DUE TO MAIN CONTROL ROOM OUTSIDE AIR INTAKE B PROCESS RADIATION MONITOR GAS CHANNEL SPIKE FROM UNKNOWN CAUSE.
EVENT DATE: 020988 REPORT DATE: 030388 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 2 (PWR)

(NSIC 208444) ON FEBRUARY 9, 1988, AT 0103 WITH BOTH BYRON UNITS IN POWER OPERATION (MODE 1), THERE WAS A SPIKE ON THE PROCESS RADIATION MONITOR FOR THE MAIN CONTROL ROOM OUTSIDE AIR INTAKE. THIS SPIKE CAUSED AN AUTOMATIC TRANSFER OF THE MAIN CONTROL ROOM VENTILATION SYSTEM TO ITS ENGINEERED SAFETY FEATURES (ESF) CONFIGURATION. SAMPLES WERE TAKEN AND NO ACTUAL RADIOACTIVITY WAS FOUND.

FOLLOWING TROUBLESHOOTING AND MONITORING OF THE SYSTEM, THE CAUSE OF THE SPIKE WAS DEEMED INDETERMINABLE AND THE VENTILATION SYSTEM WAS RETURNED TO THE NORMAL OPERATING MODE. THERE HAVE BEEN OTHER NOISE SPIKE INDUCED VENTILATION SYSTEM ACTUATIONS AT BYRON STATION.

[32] CALLAWAY 1 DOCKET 50-483 LER 88-002
TWO RHR TRAINS RENDERED INOPERABLE WHEN OPERATOR INADVERTENTLY CLOSED CROSS-CONNECT VALVE DURING SYSTEM VENTING EVOLUTION.
EVENT DATE: 010788 REPORT DATE: 020888 NSSS: WE TYPE: PWR
VENDOR: LIMITORQUE CORP.

(NSIC 208213) FROM 1136 TO 1151 CST ON 1/7/88, BOTH TRAINS OF THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM WERE RENDERED INOPERABLE AND TECHNICAL SPECIFICATION (T/S) 3.0.3 WAS ENTERED. A LICENSED UTILITY REACTOR OPERATOR (RO) INADVERTENTLY CLOSED THE TRAIN 'B' HOT LEG RECIRCULATION ISOLATION VALVE, EJ-HV-8716B, WHILE INTENDING TO CLOSE ANOTHER VALVE THAT WOULD HAVE SECURED A RHR VENTING LINEUP. THIS VALVE MUST BE OPENED FOR EITHER TRAIN TO INJECT FLOW INTO ALL FOUR REACTOR COOLANT SYSTEM (RCS) COLD LEGS DURING AN ACCIDENT. 8716B WAS RE-OPENED AT 1151. THE PLANT WAS IN MODE 1 - POWER OPERATION AT 100% POWER. THE FOLLOWING WERE IDENTIFIED AS ROOT CAUSES. THE SWITCHES FOR THE TWO VALVES WERE IN CLOSE PROXIMITY TO EACH OTHER. FREQUENT VENTING WAS BEING PERFORMED IN ORDER TO PREVENT RCS BACKLEAKAGE THROUGH SAFETY INJECTION ACCUMULATOR CHECK VALVES FROM AFFECTING THE ACCUMULATOR LEVELS. THE LATTER LED TO AN INCOMPLETE SELF-CHECK BY THE RO. OPERATIONS PERSONNEL HAVE REVIEWED THIS EVENT. PLASTIC "STOP/THINK" COVERS WERE PLACED ON THE SWITCHES FOR 8716A/B AND ARE BEING CONSIDERED FOR OTHER T/S 3.0.3 VALVES. OPERATORS WILL REVIEW T/S 3.0.3 VALVES IN TRAINING. SELF-CHECKING METHODS WILL BE RE-EMPHASIZED. OPTIONS TO FREQUENT VENTING ARE UNDER REVIEW FOR UTILIZATION UNTIL THE CHECK VALVES ARE REPAIRED.

[33] CALLAWAY 1 DOCKET 50-483 LER 88-003
CONTAINMENT HIGH RANGE RADIATION MONITORS DECLARED INOPERABLE AS A RESULT OF MISSING HEAT SHRINK MOISTURE SEALS.
EVENT DATE: 020288 REPORT DATE: 030388 NSSS: WE TYPE: PWR
VENDOR: CONAX CORP.
GENERAL ATOMIC CO.
RAYCHEM CORP.

(NSIC 208417) AT 1840 CST ON 2/2/88, IT WAS DETERMINED THAT THE ELECTRICAL PENETRATION ASSEMBLIES CONNECTED VIA COAX CABLING TO THE CONTAINMENT HIGH RANGE RADIATION MONITORS, GT-RE-59 AND GT-RE-60, WERE NOT INSTALLED WITH PROTECTIVE HEAT SHRINK MOISTURE SEALS, A CONFIGURATION THAT WOULD NOT ENSURE POST ACCIDENT OPERABILITY. THIS RESULTED IN TWO INOPERABLE CONTAINMENT HIGH RANGE RADIATION MONITORS, A CONDITION PROHIBITED BY TECHNICAL SPECIFICATION 3.3.3.6 FOR ACCIDENT MONITORING INSTRUMENTATION. THE ROOT CAUSE OF THE EVENT IS THE FAILURE OF THE PLANT'S ARCHITECTURAL ENGINEER TO IDENTIFY THE NEED FOR MOISTURE SEALS ON AN ELECTRICAL PENETRATION ASSEMBLY COAX CONNECTOR DUE TO AN INADEQUATE VENDOR MANUAL. THE VENDOR MANUAL PROVIDED A BASIS FOR THE DEVELOPMENT OF AN ELECTRICAL TERMINATION LIST, THE REFERENCE DOCUMENT USED DURING CONSTRUCTION OF THE PLANT. THE ELECTRICAL TERMINATION LIST DID NOT REQUIRE HEAT SHRINK MOISTURE SEALS ON THE CONTAINMENT PENETRATION ASSEMBLY COAX CONNECTORS. CORRECTIVE ACTION TAKEN INCLUDED RESTORING HIGH RANGE MONITOR OPERABILITY BY INSTALLING HEAT SHRINK MOISTURE SEALS, REVISING INADEQUATE DESIGN DOCUMENTATION, AND REVIEWING COAX CABLING CONNECTIONS FOR SIMILAR CONDITIONS.

[34] CALVERT CLIFFS 1 DOCKET 50-317 LER 88-002
ATTEMPTED INTRODUCTION OF CONTRABAND INTO A PROTECTED AREA.
EVENT DATE: 022388 REPORT DATE: 030788 NSSS: CE TYPE: PWR

(NSIC 208474) AT APPROXIMATELY 2055, AN INTERSTATE CONTRACTOR VEHICLE OPERATOR, PRIOR TO THE VEHICLE SEARCH, RELINQUISHED A 12 GAUGE WINCHESTER PUMP SHOTGUN (UNLOADED) TO ONE OF THE VEHICLE SEARCHES. WHEN ASKED IF HE HAD ANY OTHER WEAPONS IN THE VEHICLE, THE DRIVER REPLIED NO. DURING A ROUTINE SEARCH OF THE SLEEPING COMPARTMENT, A .357 DAN WESSON LOADED REVOLVER WAS DISCOVERED UNDER THE MATTRESS. THE VEHICLE OPERATOR STATED THAT HE THOUGHT HIS WIFE HAD REMOVED THE WEAPON PRIOR TO LEAVING ON THIS TRIP. MARYLAND STATE POLICE WERE CALLED AND WERE ABLE TO VERIFY THROUGH THE FBINCIC THAT THE WEAPON WAS NOT STOLEN. THE VEHICLE OPERATOR WAS DENIED ACCESS TO THE PROTECTED AREA. THE NRC WAS NOTIFIED AT 2125. LICENSEE PERSONNEL EFFECTED MOVEMENT OF THE LOAD INTO THE PROTECTED AREA AND RETURNED THE TRAILER TO THE OPERATOR. THE TWO WEAPONS WERE RETURNED TO THE VEHICLE OPERATOR UPON DEPARTING THE PROPERTY. THERE WERE BOTH PROPRIETARY AND CONTRACTOR SECURITY FORCES ON-SITE AT THE TIME OF THE INCIDENT. THE NEWS MEDIA WAS NOT NOTIFIED AND THERE WAS A SIMILAR EVENT ON 2/15/88 (LER 88-01). NO SAFETY SYSTEMS WERE AFFECTED.

[35]   CALVERT CLIFFS 2   DOCKET 50-318   LER 88-001
SAFETY INJECTION TANK LOW WATER VOLUME.
EVENT DATE: 011288   REPORT DATE: 020588   NSSS: CE   TYPE: PWR
VENDOR: FISCHER & PORTER CO.

(NSIC 208189) ON 12 JANUARY 88, AT 100% POWER, WHILE TROUBLE SHOOTING AN OSCILLATING WATER LEVEL INDICATION OF THE 21B SAFETY INJECTION TANK (SIT), CALIBRATION REVEALED THAT THE ZERO LEVEL REFERENCE HAD SHIFTED. THE LEVEL WAS 182 INCHES. THE MINIMUM TECHNICAL SPECIFICATION LIMIT IS 187 INCHES. OPERATORS FILLED 21B SIT TO PROPER WATER LEVEL. THE CAUSE OF THE ZERO LEVEL SHIFT WAS TRANSMITTER CALIBRATION DRIFT. THE LEVEL TRANSMITTER WILL BE REPLACED WHEN A REPLACEMENT TRANSMITTER IS SATISFACTORILY REBUILT AND RADIOLOGICAL CONDITIONS PERMIT.

[36]   CALVERT CLIFFS 2   DOCKET 50-318   LER 88-002
LOSS OF FEED TRIP DUE TO THE LOSS OF INSTRUMENT BUS 22 (2Y10).
EVENT DATE: 012288   REPORT DATE: 021988   NSSS: CE   TYPE: PWR

(NSIC 208327) DURING TROUBLESHOOTING OF THE THREE PHASE UNIT 2 COMPUTER INVERTER, POWER WAS LOST TO THE NON-VITAL 208/120V A.C. INSTRUMENT BUS 22 (2Y10). THE DE-ENERGIZATION OF 2Y10 RESULTED IN REDUCED FEEDWATER FLOW AND CAUSED THE REACTOR TO TRIP ON LOW STEAM GENERATOR LEVEL. THE LOSS OF INSTRUMENT BUS 22 (2Y10) WAS CAUSED BY PERSONNEL ERROR WHEN A PLANT ELECTRICIAN'S MISINTERPRETATION OF ELECTRICAL PRINTS AND UNCLEAR COMMUNICATIONS WITH THE INVERTER VENDOR LED TO THE PLACEMENT OF TEMPORARY JUMPERS IN THE INVERTER, CAUSING A DIRECT SHORT CIRCUIT THAT WAS REFLECTED BACK TO THE MAIN POWER FEED CIRCUIT BREAKER CAUSING IT TO TRIP THUS DE-ENERGIZING INSTRUMENT BUS 22 (2Y10). CORRECTIVE ACTIONS: 1. A STUDY OF THE PROPER SIZE FUSING AND FUSE TO CIRCUIT BREAKER COORDINATION IS IN PROGRESS. 2. CLARIFY THE INVERTER MANUFACTURERS ELECTRICAL PRINTS TO SHOW THE ACTUAL PLACEMENT, IN THE CIRCUIT, OF THE POWER FACTOR CORRECTION CAPACITORS. 3. INVESTIGATE THE USE OF "SPECIAL" PROCEDURES IN TROUBLESHOOTING COMPLEX EQUIPMENT. 4. ALL MAINTENANCE ELECTRICIANS WILL BE TRAINED ON THE DETAILS OF THIS EVENT AS PART OF THE CONTINUING TRAINING PROGRAM.

[37]   CATAWBA 1   DOCKET 50-413   LER 88-001
MISSED HOURLY FIRE WATCHES RESULTING IN TECHNICAL SPECIFICATION VIOLATIONS DUE TO PERSONNEL ERROR AND MANAGEMENT DEFICIENCIES.
EVENT DATE: 011288   REPORT DATE: 021188   NSSS: WE   TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 208275) ON JANUARY 12, 1988, AND ON JANUARY 18, 1988, HOURLY FIRE WATCHES WERE NOT PERFORMED AT 0800 HOURS AND 0900 HOURS FOR ROOM 107, CONTAINMENT SPRAY

PUMP 2A. THE HOURLY FIRE WATCH WAS ALSO NOT PERFORMED ON JANUARY 26, 1988, AT 0700 HOURS. THE HOURLY FIRE WATCH WAS REQUIRED DUE TO CONCRETE HATCH COVERS BEING REMOVED TO ALLOW MAINTENANCE WORK. FOLLOWING EACH DISCOVERY, THE REQUIRED HOURLY FIRE WATCHES WERE RESUMED. UNIT 2 WAS IN MODE 6, REFUELING, THROUGHOUT THIS EVENT. UNIT 1 WAS IN MODE 1, POWER OPERATION, AT 100% POWER ON JANUARY 12, AND JANUARY 26, AND MODE 5, COLD SHUTDOWN, ON JANUARY 18, 1988. THE JANUARY 12, 1988 OCCURRENCE IS ATTRIBUTED TO A MANAGEMENT DEFICIENCY. THE APPROPRIATE SUPERVISOR RESPONSIBLE FOR ASSIGNMENT OF THE HOURLY FIRE WATCH DID NOT ASSIGN ANYONE TO PERFORM THE TASK UNTIL APPROXIMATELY 0945 HOURS AFTER TWO HOURLY FIRE WATCHES HAD BEEN MISSED. THE OCCURRENCE ON JANUARY 18, 1988, IS ATTRIBUTED TO A PERSONNEL ERROR. THE TECHNICIAN RESPONSIBLE FOR ENSURING THE HOURLY FIRE WATCH WAS CONDUCTED DID NOT ASSIGN ANYONE TO PERFORM THE TASK UNTIL HE WAS INFORMED THAT TWO FIRE WATCHES HAD ALREADY BEEN MISSED. THE OCCURRENCE ON JANUARY 26, 1988 IS ATTRIBUTED TO A DEFICIENCY. THERE WAS APPARENTLY A FAILURE TO ADEQUATELY COMMUNICATE THE FIRE WATCH REQUIREMENT DURING THE MORNING SHIFT TURNOVER BETWEEN TWO SUPERVISORS.

[38] CATAWBA 1 DOCKET 50-413 LER 88-003
WIDE RANGE TEMPERATURE MONITORING INSTRUMENTATION TECHNICALLY INOPERABLE DURING CERTAIN ACCIDENT CONDITIONS DUE TO INSTALLATION AND DESIGN DEFICIENCIES.
EVENT DATE: 011688 REPORT DATE: 021588 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 208354) ON 1/15/88 AT 1030 HOURS, WITH UNIT 2 IN MODE 6, REFUELING, 6 OF THE 8 UNIT 2 REACTOR COOLANT (NC) SYSTEM WIDE RANGE (W/R) HOT AND COLD LEG RESISTANCE TEMPERATURE DETECTOR (RTD) CABLES WERE DISCOVERED TO HAVE BEEN IMPROPERLY INSTALLED. THE RTD CABLES WERE ORIGINALLY SUPPLIED BY WESTINGHOUSE WITH A SEALED STAINLESS STEEL BELLOWS HOSE TO MAINTAIN ENVIRONMENTAL QUALIFICATION. HOWEVER, THE AFFECTED UNIT 2 RTD CABLES AND BELLOWS HOSES HAD BEEN CUT DURING INITIAL INSTALLATION. AS A RESULT, THE RTD'S ENVIRONMENTAL QUALIFICATION WAS INVALIDATED. SINCE THE CORRESPONDING UNIT 1 RTD'S WERE SUSPECTED TO BE SIMILARLY INSTALLED, THEY WERE DECLARED INOPERABLE AT 1400 HOURS, AND UNIT 1 SHUTDOWN WAS COMMENCED AT 2250 HOURS. UNIT 1 ENTERED MODE 3, HOT STANDBY, AT 0541 HOURS, AND MODE 4, HOT SHUTDOWN, AT 2230 HOURS, ON 1/16/88. SUBSEQUENTLY, THE CORRESPONDING UNIT 1 W/R HOT AND COLD LEG RTD CABLES WERE ALSO FOUND TO BE IMPROPERLY INSTALLED. UNIT 1 WAS IN MODE 1, POWER OPERATION, AT 100% POWER AT THE TIME OF DISCOVERY. BOTH UNITS HAVE OPERATED IN ALL MODES WITH THE AFFECTED RTD'S TECHNICALLY INOPERABLE. THIS INCIDENT IS ATTRIBUTED TO AN INSTALLATION DEFICIENCY PRIOR TO STARTUP OF EACH UNIT DUE TO A MISINTERPRETATION OF NOTES ON THE CONNECTION DIAGRAMS.

[39] CATAWBA 1 DOCKET 50-413 LER 88-004
REACTOR TRIP BREAKERS OPEN DURING UNIT SHUTDOWN DUE TO A FUSE FAILURE.
EVENT DATE: 011688 REPORT DATE: 021588 NSSS: WE TYPE: PWR
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)

(NSIC 208355) ON 1/16/88, THE UNIT WAS IN THE PROCESS OF BEING SHUTDOWN. WHEN REACTOR POWER REACHED 10E-10 AMPS ON THE INTERMEDIATE RANGE (I/R), THE CONTROL POWER FUSE FAILED IN I/R CHANNEL N36, CAUSING THE LOW POWER HIGH FLUX REACTOR TRIP BISTABLE TO ACTUATE. THE REACTOR TRIP BREAKERS OPENED AND ALL OF THE UNINSERTED RODS DROPPED INTO THE CORE. (KEFF WAS <.99 AT THE TIME OF THE ACTUATION.) THE OPEN REACTOR TRIP BREAKERS COINCIDENT WITH AVERAGE REACTOR COOLANT TEMPERATURE BELOW 564 DEGREES F RESULTED IN A FEEDWATER ISOLATION. THE UNIT WAS IN MODE 3, HOT STANDBY, AT THE TIME OF THIS INCIDENT. SUBSEQUENT TO THIS EVENT, AN OPERATIONS NUCLEAR EQUIPMENT OPERATOR (NEO) WAS SENT TO THE UNIT 1 COOLING TOWERS TO INSPECT THEM FOR ICE. DURING THE INSPECTION, HE FOUND THAT A FRACTURE OF THE CONDENSER CIRCULATING WATER (RC) PIPING HAD OCCURRED. THE FEEDWATER ISOLATION SIGNAL WAS RESET AND THE AFFECTED VALVES WERE RETURNED TO THEIR PREVIOUS ALIGNMENT. WORK REQUESTS WERE ISSUED TO INVESTIGATE AND REPAIR

I/R CHANNEL N36 AND PERFORM TESTING ON CHANNEL N36. THE RC PIPING WAS REPAIRED, AND THE UNIT WAS RETURNED TO POWER. ALSO THE SHUTDOWN PROCEDURE WAS REVISED TO RAISE THE LOW LIMIT RC TEMPERATURE TO PREVENT FURTHER BRITTLE FRACTURE PROBLEMS WITH THE RC SYSTEM.

[40] CATAWBA 1 DOCKET 50-413 LER 88-006
UNPLANNED NUCLEAR SERVICE WATER SWAP TO THE STANDBY NUCLEAR SERVICE WATER POND DUE TO A MANAGEMENT DEFICIENCY.
EVENT DATE: 012288 REPORT DATE: 021988 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 208414) ON JANUARY 22, 1988, AT APPROXIMATELY 0800 HOURS, AN UNPLANNED SWAP OF THE NUCLEAR SERVICE WATER (RN) SYSTEM TO THE STANDBY NUCLEAR SERVICE WATER POND (SNSWP) OCCURRED. A CONSTRUCTION AND MAINTENANCE DEPARTMENT (CMD) ELECTRICIAN WAS IN THE PROCESS OF TERMINATING SPARE CONDUCTORS IN AN ESSENTIAL AREA TERMINAL CABINET (EATC) UNDER A NUCLEAR STATION MODIFICATION (NSM) WORK REQUEST WHEN THE BREAKER SUPPLYING POWER TO UNIT 2 TRAIN A RN PIT LEVEL TRANSMITTER TRIPPED. THE TRANSMITTER FAILING LOW UPON LOSS OF POWER SATISFIED THE LOGIC FOR THE SWAP. UNIT 1 WAS IN MODE 4, HOT SHUTDOWN, AND UNIT 2 WAS IN NO MODE, DEFUELED, AT THE TIME OF THE EVENT. THIS EVENT HAS BEEN ATTRIBUTED TO A MANAGEMENT DEFICIENCY DUE TO AN INADEQUACY IN THE WORK CONTROL SYSTEM. THE CMD ELECTRICIAN WAS NOT REQUIRED BY THE PRESENT NSM WORK CONTROL SYSTEM TO FOLLOW THE JOB SEQUENCE ON THE WORK REQUEST. THE GROUNDS ON THE AFFECTED TRANSMITTER WERE SUBSEQUENTLY CLEARED AND THE RN SYSTEM WAS REALIGNED TO LAKE WYLIE. A PROGRAM IS UNDER DEVELOPMENT REQUIRING THAT AN APPROVED STATION TEMPORARY PROCEDURE BE DEVELOPED AND USED ON ALL NSM IMPLEMENTATION WORK. PRE-IMPLEMENTATION MEETINGS WILL STRESS THE LIMITS AND PRECAUTIONS TO BE USED. ADEQUATE FLOW WAS AVAILABLE TO BOTH UNITS AT ALL TIMES. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS EVENT.

[41] CATAWBA 1 DOCKET 50-413 LER 88-007
SAFETY INJECTION ACTUATED DURING UNIT HEAT-UP DUE TO A PERSONNEL ERROR.
EVENT DATE: 012388 REPORT DATE: 022288 NSSS: WE TYPE: PWR

(NSIC 208415) ON 1/23/88, AT 1039 HOURS, SAFETY INJECTION WAS ACTUATED DUE TO LOW STEAMLINE PRESSURE. UNIT HEATUP WAS IN PROGRESS. WHEN REACTOR COOLANT SYSTEM PRESSURE REACHED 1955 PSIG SAFETY INJECTION WAS AUTOMATICALLY INITIATED BECAUSE MAIN STEAM PRESSURE WAS LESS THAN 725 PSIG. THE UNIT WAS IN MODE 3, HOT STANDBY, AT THE TIME OF THIS INCIDENT. THIS INCIDENT IS ATTRIBUTED TO A PERSONNEL ERROR. THE CONTROL ROOM OPERATOR DID NOT ENSURE THAT MAIN STEAM PRESSURE WAS NOT INCREASED SUFFICIENTLY TO GREATER THAN 725 PSIG, PRIOR TO REACTOR COOLANT SYSTEM PRESSURE REACHING 1955 PSIG AS REQUIRED BY PROCEDURE. CONTROL ROOM PERSONNEL UTILIZED THE APPROPRIATE EMERGENCY PROCEDURES, RESET SAFETY INJECTION, LOAD SEQUENCER, AND CONTAINMENT ISOLATION SIGNALS, AND SECURED THE EMERGENCY CORE COOLING SYSTEM. TO ENSURE ADEQUATE OVERSIGHT OF UNIT EVOLUTIONS, OPERATIONS WILL REVIEW THE FUNCTIONS OF THE CONTROL ROOM SENIOR REACTOR OPERATOR AND REVISE IF APPROPRIATE. ALSO, A COMPUTER ALARM WILL BE REQUESTED TO INDICATE THAT REACTOR COOLANT PRESSURE IS INCREASING TO 1955 PSIG, AND AN OPERATOR UPDATE WILL BE ISSUED CONCERNING THIS INCIDENT. WITH THE UNIT IN MODE 3, THE EFFECTS OF INJECTION FLOW ON PRIMARY SYSTEM PARAMETERS WERE MINIMAL. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS EVENT.

[42] CLINTON 1 DOCKET 50-461 LER 88-002
AUTO-START OF STANDBY GAS TREATMENT SYSTEM RESULTS FROM SPURIOUS ELECTRICAL SPIKE OF PROCESS RADIATION MONITOR OUTPUT DUE TO DETECTOR TUBE FAILURE.
EVENT DATE: 010688 REPORT DATE: 012988 NSSS: GE TYPE: BWR
VENDOR: L N D, INC.

(NSIC 208212) ON JANUARY 6, 1988, WITH THE PLANT IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 100% REACTOR POWER, A SPURIOUS ELECTRICAL SPIKE ON A DIVISION 1 CONTINUOUS CONTAINMENT PURGE SYSTEM (CCP) EXHAUST PROCESS RADIATION MONITOR (PRM) OUTPUT CAUSED THE CCP SYSTEM TO ISOLATE, RESULTING IN AN AUTO-START OF THE STANDBY GAS TREATMENT SYSTEM (SGTS) A AND B TRAINS AND ISOLATION OF THE FUEL BUILDING VENTILATION SYSTEM. A DIVISION 2 CCP EXHAUST PRM WAS IN A TRIPPED CONDITION AT THE TIME OF THE EVENT DUE TO CHANNEL FUNCTIONAL TEST FAILURE. WITH A DIVISION 2 PRM TRIPPED, THE SPURIOUS ELECTRICAL SPIKE ON THIS DIVISION 1 PRM OUTPUT RESULTED IN CCP SYSTEM ISOLATION SINCE THE LOGIC IS ONE-OUT-OF-TWO-TWICE FOR THE HIGH CCP EXHAUST RADIATION TRIP. OPERATORS SECURED SGTS TRAIN A AND THE B TRAIN WAS ALLOWED TO RUN. OPERATORS PERFORMED THE AUTO-ISOLATION CHECKLIST AND FOUND NO DISCREPANCIES. THE CAUSE OF THE EVENT IS ATTRIBUTED TO RANDOM RADIATION DETECTOR TUBE FAILURE ON THE DIVISION 1 MONITOR. THE DETECTOR TUBE WAS REPLACED. THE EVENT WAS NOT SAFETY SIGNIFICANT SINCE THE SYSTEMS RESPONDED TO THE HIGH EXHAUST RADIATION OUTPUT AS DESIGNED. THIS EVENT IS REPORTABLE UNDER THE PROVISIONS OF 10CFR50.73(A)(2)(IV) DUE TO AN AUTOMATIC ACTUATION OF AN ENGINEERED SAFETY FEATURE.

[43] CLINTON 1 DOCKET 50-461 LER 88-003
MALFUNCTION OF PROCESS RADIATION MONITOR DURING CHECK SOURCE FUNCTION RESULTS IN PREMATURE RE-LANDING LEAD WIRES AND ISOLATION OF HYDROGEN/OXYGEN MONITOR.
EVENT DATE: 011488 REPORT DATE: 021088 NSSS: GE TYPE: BWR
VENDOR: EBERLINE INSTRUMENT CORP.

(NSIC 208280) ON JANUARY 14, 1988, WITH THE PLANT IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 100% REACTOR POWER, THE DIVISION 1 CONTAINMENT MONITORING SYSTEM CONTAINMENT ISOLATION VALVES FOR THE HYDROGEN/OXYGEN MONITOR ISOLATED. THE ISOLATION OCCURRED DURING CHANNEL FUNCTIONAL TESTING (CFT) OF THE CHANNEL A CONTAINMENT VENTILATION EXHAUST PROCESS RADIATION MONITOR (PRM). THE EVENT WAS IDENTIFIED BY OPERATORS WHILE WALKING DOWN BACKPANELS DURING SHIFT TURNOVER. FOLLOWING IDENTIFICATION OF THE ISOLATION, THE RADIATION LEVEL WAS VERIFIED TO BE NORMAL, THE ISOLATION LOGIC WAS RESET AND THE VALVES WERE REOPENED. THE CAUSE OF THE EVENT IS ATTRIBUTED TO A MALFUNCTION OF THE PRM THAT RESULTED IN ILLUMINATION OF THE GREEN "NORMAL" STATUS LIGHT. ILLUMINATION OF THIS LIGHT PROVIDES INDICATION THAT THE CHECK SOURCE FUNCTION OF THE MONITOR INCLUDING A MOMENTARY ACTIVATION OF THE OUTPUT TRIP RELAY IS COMPLETE. BASED ON THIS LIGHT ILLUMINATING, LEAD WIRES, PREVIOUSLY LIFTED TO PREVENT SEAL- IN OF THE TRIP SIGNAL, WERE RE-LANDED ALLOWING THE MOMENTARY ACTIVATION OF THE OUTPUT TRIP RELAY TO CAUSE THE ISOLATION. THE CFT PROCEDURE HAS BEEN REVISED TO REQUIRE ADDITIONAL CONFIRMATION OF CHECK SOURCE CYCLE COMPLETION. A PLANT MODIFICATION WILL REMOVE THE MOMENTARY ACTIVATION OF THE OUTPUT TRIP RELAY AT THE END OF THE CHECK SOURCE FUNCTION.

[44] CLINTON 1 DOCKET 50-461 LER 88-004
MANIPULATION OF INCORRECT TEMPERATURE CONTROLLER RESULTS IN INOPERABLE DIVISION I PRIMARY CONTAINMENT HYDROGEN RECOMBINER WHILE DIVISION II DIESEL GENERATOR WAS INOPERABLE.
EVENT DATE: 012988 REPORT DATE: 022388 NSSS: GE TYPE: BWR

(NSIC 208365) ON 2/5/88 WITH THE PLANT IN MODE 1 (POWER OPERATION), AT 100% REACTOR POWER, A WALKDOWN OF THE DIVISION I (A) PRIMARY CONTAINMENT HYDROGEN RECOMBINER ELECTRICAL LINE-UP IDENTIFIED THAT THE "A" RECOMBINER WAS INOPERABLE DUE TO INCORRECT TEMPERATURE SETTINGS ON THE REACTION CHAMBER GAS TEMPERATURE CONTROLLER AND THE GAS RETURN TEMPERATURE CONTROLLER. FURTHER INVESTIGATION IDENTIFIED THAT THE "A" RECOMBINER WAS INOPERABLE SINCE JANUARY 14, 1988. OPERATORS RESTORED THE "A" RECOMBINER TO OPERABLE STATUS ON 2/5 BY PERFORMING AN ELECTRICAL LINE-UP, AND VERIFIED THE DIVISION II (B) RECOMBINER TO BE SATISFACTORILY ALIGNED. THE DIVISION II DIESEL GENERATOR (DG) WAS OUT OF SERVICE FOR MAINTENANCE FROM 1/29 THROUGH 1/31. WHEN THE DIVISION II DG IS INOPERABLE,

TECHNICAL SPECIFICATIONS REQUIRE ALL EQUIPMENT DEPENDING ON THE DIVISION I DG FOR EMERGENCY POWER TO BE OPERABLE, THEREFORE TECHNICAL SPECIFICATIONS WERE VIOLATED. THE CAUSE OF THE EVENT IS ATTRIBUTED TO UTILITY NON-LICENSED OPERATOR ERROR DUE TO RESETTING AND INDEPENDENTLY VERIFYING AN INCORRECT TEMPERATURE CONTROLLER DURING THE RESTORATION STEP OF SURVEILLANCE PROCEDURE, PRIMARY CONTAINMENT RECOMBINER AND VALVE OPERABILITY. CORRECTIVE ACTION FOR THE EVENT IS TO COUNSEL THE OPERATORS INVOLVED AND TO INFORM ALL OTHER OPERATORS OF THE EVENT.

[45] CONNECTICUT YANKEE DOCKET 50-213 LER 88-002
FIRE DETECTION SUBSYSTEM DECLARED INOPERABLE DUE TO DAMAGED HEAT DETECTORS.
EVENT DATE: 011088 REPORT DATE: 021988 NSSS: WE TYPE: PWR

(NSIC 208315) AT 0500, ON 1/10/88, WITH THE PLANT IN MODE 6 WITH THE REACTOR CORE OFFLOADED, FIRE DETECTION SUBSYSTEM PANEL FDS-2 WAS DECLARED INOPERABLE DUE TO A SYSTEM FAILURE INDICATED BY BOTH "SYSTEM TROUBLE" AND GROUND ALARMS. THE AFFECTED PANEL PROVIDES FIRE DETECTION FOR THE REACTOR CONTAINMENT BUILDING, AUXILIARY FEEDWATER PUMP AREA, AND CONTAINMENT CABLE VAULT. IN ACCORDANCE WITH TECH SPEC 3.22.E.2.A, A FIRE WATCH PATROL WAS ESTABLISHED TO INSPECT THE ZONES WITH THE INOPERABLE INSTRUMENTS. THE CAUSE OF THE INOPERABILITY WAS PHYSICAL DAMAGE TO TWO HEAT DETECTORS LOCATED NEAR THE REACTOR COOLANT PUMPS AND THE DAMAGE OF THE CONDUIT FEED TO A THIRD. IN ADDITION, COLD AIR INTRUSION MAY HAVE AFFECTED THE BACK-UP POWER SUPPLY (INTERNAL BATTERY) IN THE PANEL. THE HEAT DETECTORS WERE REPLACED, THE CONDUIT WAS REPAIRED, THE BATTERY WAS REPLACED, AND THE SYSTEM WAS RETURNED TO SERVICE ON 2/11/88. THIS REPORT IS SUBMITTED AS REQUIRED BY TECH SPEC 3.22.E.2.B WHICH STATES THAT IF THE FIRE DETECTION SYSTEM CANNOT BE RESTORED TO OPERABLE STATUS WITHIN 14 DAYS, A SPECIAL REPORT WILL BE SUBMITTED WITHIN THE NEXT 30 DAYS.

[46] CONNECTICUT YANKEE DOCKET 50-213 LER 88-001
DESIGN ERROR FOUND IN STEAM GENERATOR BLOWDOWN ISOLATION CIRCUIT.
EVENT DATE: 011488 REPORT DATE: 021188 NSSS: WE TYPE: PWR

(NSIC 208259) DURING A COMPONENT QUALIFICATION REVIEW, AN ENGINEER NOTED A DESIGN DEFICIENCY IN THE CONTROL CIRCUIT COMMON TO FOUR CONTAINMENT ISOLATION TRIP VALVES. THE REVIEW OF THE WIRING DIAGRAM INDICATED THAT THE STEAM GENERATOR BLOWDOWN ISOLATION VALVES (BD-TV-1312-1, 2, 3, 4) WOULD NOT TRIP IN RESPONSE TO A HIGH CONTAINMENT PRESSURE (HCP) SIGNAL IF PRECEDED BY A LOSS OF VOLTAGE TO 4160V BUSES 1-2 AND 1-3 OR 480V BUSES 4, 6, OR 7. IN ADDITION, THE BLOWDOWN VALVES WOULD NOT HAVE CLOSED UPON RECEIPT OF AN UNDERVOLTAGE SIGNAL. THE SYSTEM DESIGN CALLED FOR CLOSURE OF THE BLOWDOWN VALVES IF EITHER AN HCP OR LOW VOLTAGE SIGNAL WAS PRESENT. THE EXISTING CONTROL LOGIC WILL BE MODIFIED TO MEET THE REQUIRED SYSTEM DESIGN. THE DESIGN ERROR WAS MADE DURING THE IMPLEMENTATION OF A POST-TMI MODIFICATION TO THE CONTAINMENT ISOLATION VALVE CONTROL CIRCUITS. THIS EVENT IS REPORTABLE PER 10CFR50.73(A)(2)(V) AS IT COULD HAVE PREVENTED FULFILLMENT OF A SAFETY FUNCTION. THIS EVENT IS ALSO REPORTABLE PER 10CFR50.73(A)(2)(I)(B) SINCE IT INVOLVES A CONDITION PROHIBITED BY THE TECHNICAL SPECIFICATIONS.

[47] CONNECTICUT YANKEE DOCKET 50-213 LER 88-003
ELECTRIC FIRE PUMP DECLARED INOPERABLE DUE TO HIGH AMPERAGE.
EVENT DATE: 020488 REPORT DATE: 030388 NSSS: WE TYPE: PWR
VENDOR: FINCOR

(NSIC 208401) AT 0933, ON FEBRUARY 4, 1988 WITH THE PLANT IN MODE 6 AND THE REACTOR CORE OFF LOADED, THE ELECTRIC DRIVEN FIRE PUMP WAS DECLARED INOPERABLE DUE TO HIGH AMPERAGE MEASURED AFTER A MANUAL START. BACKUP FIRE WATER SUPPLY IS PROVIDED BY THE DIESEL DRIVEN FIRE PUMP. THE CAUSE OF THE INOPERABILITY WAS PHYSICAL DAMAGE TO THE STUFFING BOX BRASS BUSHING LOCATED IN THE UPPER SHAFT AREA OF THE ELECTRIC DRIVEN FIRE PUMP. THIS CAUSED THE BRASS BEARING TO SHEAR,

RESULTING IN A LOCKED ROTOR CONDITION. BASED ON OUR EVALUATION AND DISCUSSIONS WITH THE MANUFACTURER, IT APPEARS THAT OPERATING THE PUMP AT LOW FLOW CONDITIONS MAY HAVE CAUSED THIS FAILURE. REPAIRS TO THE ELECTRIC FIRE PUMP HAVE BEEN INITIATED AND IT IS EXPECTED TO BE RETURNED TO SERVICE BY MARCH 15, 1988. THIS REPORT IS SUBMITTED AS REQUIRED BY TECHNICAL SPECIFICATION 3.22.A.2 WHICH STATES THAT IF THE PUMP CANNOT BE RESTORED TO OPERABLE STATUS WITHIN SEVEN DAYS, A SPECIAL REPORT WILL BE SUBMITTED WITHIN 30 DAYS OF THE OCCURRENCE.

[48] COOK 1 DOCKET 50-315 LER 87-022 REV 01
UPDATE ON VIOLATION OF ESF INSTRUMENTATION LIMITING CONDITIONS FOR OPERATION TOLERANCES DUE TO TRANSMITTERS CALIBRATION SHIFT.
EVENT DATE: 100787 REPORT DATE: 021188 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)
VENDOR: FOXBORO CO., THE

(NSIC 208252) DURING THE PAST UNIT ONE REFUELING OUTAGE PERFORMED BETWEEN JUNE AND OCTOBER 1987, THE AS FOUND DATA OF TWO PRESSURIZER LEVEL CHANNELS EXCEEDED THE TECH SPEC LIMITING CONDITION FOR OPERATION (LCO) VALUES, ALTHOUGH THEY REMAINED WITHIN THE SAFETY ANALYSIS. THE AS FOUND DATA WAS OBTAINED DURING ROUTINE CALIBRATION EVOLUTIONS, AND THE ASSOCIATED COMPONENTS (FOXBORO TRANSMITTERS) WERE ADJUSTED AND RESTORED TO ACCEPTABLE AS LEFT CONDITIONS. THE CAUSE OF THE DRIFT IS CONSIDERED ATTRIBUTABLE TO TWO FACTORS. FIRST, THE TWO TRANSMITTERS HAD BEEN INSTALLED DURING THE PREVIOUS REFUELING OUTAGE AS PART OF AN ENVIRONMENTAL QUALIFICATION UPGRADE. IT IS NOT CONSIDERED UNUSUAL FOR NEW FORCE BALANCE TRANSMITTERS TO EXHIBIT SLIGHTLY HIGHER DRIFT DURING THEIR FIRST CALIBRATION CYCLE FOLLOWING INSTALLATION. SECONDLY, THE ELAPSED TIME BETWEEN INSTALLATION AND FIRST RECALIBRATION WAS LONGER THAN NORMAL DUE TO AN EXTENSION OF THE REFUELING OUTAGE START DATE. THIS MAY HAVE CONTRIBUTED TO THE OBSERVED MAGNITUDE OF THE DRIFT. CORRECTIVE ACTION TO PREVENT RECURRENCE WILL BE TO PERFORM CALIBRATION CHECKS FOR NEW TRANSMITTER INSTALLATIONS BETWEEN OUTAGES AS PLANT CONDITIONS ALLOW. THE "WEAR IN" PERIOD HAS ELAPSED FOR TRANSMITTERS INSTALLED SINCE THE PREVIOUS OUTAGE, AND FURTHER DRIFTING IS NOT EXPECTED.

[49] COOK 1 DOCKET 50-315 LER 87-023 REV 01
UPDATE ON DEFICIENT DESIGN RESULTS IN FAILURE TO PROVIDE ELECTRICAL ISOLATION BETWEEN LOCAL SHUTDOWN AND INDICATION PANELS.
EVENT DATE: 110987 REPORT DATE: 021888 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)

(NSIC 208307) THE PRIMARY PURPOSE OF THIS SUPPLEMENTAL REPORT IS TO PROVIDE ADDITIONAL INFORMATION REGARDING THE SEQUENCE OF EVENTS. ON NOVEMBER 9, 1987, DURING A REVIEW OF AN INVESTIGATION CONCERNING REGULATORY GUIDE 1.97 COMPLIANCE, IT WAS DISCOVERED THAT A POTENTIAL PROBLEM COULD EXIST INVOLVING THE FUSES REQUIRED FOR ISOLATION BETWEEN THE VARIOUS LOCAL SHUTDOWN AND INDICATION (LSI) PANELS (IMPROPERLY LOCATED ON UNIT 2 AND NOT INCORPORATED INTO THE EXISTING DESIGN ON UNIT 1). ON DECEMBER 22, 1987 IT WAS CONFIRMED THAT A CONDITION EXISTED THAT, IN THE EVENT OF A FIRE LOCAL TO A LSI PANEL, POWER (BOTH NORMAL AND ALTERNATE) TO SOME OR ALL OF THE SAME UNITS' REMAINING PANELS COULD HAVE BEEN LOST. THE CAUSE OF THE EVENT WAS AN OVERSIGHT BY DESIGN ENGINEERS IN THE DESIGN AND VERIFICATION PROCESS ASSOCIATED WITH THE INITIAL APPENDIX R MODIFICATIONS. FIRE WATCHES WERE ASSIGNED TO TOUR THE AFFECTED AREAS. DESIGN CHANGES HAVE BEEN IMPLEMENTED WHICH PROVIDE THE NECESSARY ISOLATION IN THE EVENT OF A FIRE. TO PREVENT RECURRENCE THE APPROPRIATE ENGINEERING PROCEDURES HAVE BEEN PREFACED TO ADDRESS THIS SPECIFIC ELECTRICAL ISOLATION CONCERN.

[50] COOK 1 DOCKET 50-315 LER 88-001
ENGINEERED SAFETY FEATURES ACTUATION DUE TO THE INADVERTENT OPENING OF A REACTOR TRIP BREAKER AS A RESULT OF PERSONNEL ERROR.
EVENT DATE: 011388 REPORT DATE: 020488 NSSS: WE TYPE: PWR

(NSIC 208245) ON JANUARY 13, 1988, AT 0821 HOURS, UNIT ONE EXPERIENCED AN ENGINEERED SAFETY FEATURES ACTUATION (REACTOR TRIP) AS A RESULT OF THE INADVERTANT OPENING OF THE TRAIN B REACTOR TRIP BREAKER. AT THE TIME OF THIS EVENT THE UNIT WAS OPERATING IN MODE 1 (POWER OPERATION) AT 90 PERCENT RATED THERMAL POWER. THIS EVENT WAS THE RESULT OF COGNITIVE PERSONNEL ERROR ON THE PART OF THE LICENSED OPERATOR INVOLVED, WHO ATTEMPTED TO MANIPULATE THE TRAIN B REACTOR TRIP BREAKER INSTEAD OF REACTOR TRIP BYPASS BREAKER B DURING THE MONTHLY SOLID STATE PROTECTION SYSTEM (SSPS) SURVEILLANCE. FOLLOWING THE REACTOR TRIP ALL ACTIONS AND VERIFICATIONS REQUIRED BY THE FACILITY EMERGENCY OPERATING PROCEDURES WERE COMPLIED WITH AND THE UNIT WAS STABILIZED IN MODE 3 (HOT STANDBY). ADMINISTRATIVE CONTROLS HAVE BEEN ESTABLISHED TO ENSURE PROPER EQUIPMENT CONFIGURATIONS AND ADEQUATE JOB BRIEFINGS DURING SURVEILLANCE TESTING. PROCEDURAL ENHANCEMENTS WILL BE MADE TO IMPROVE COMMUNICATIONS AND COMPONENT STATUS VERIFICATIONS, AND ARE TENTATIVELY SCHEDULED FOR COMPLETION BY APRIL 7, 1988. LASTLY, APPROPRIATE ADMINISTRATIVE ACTION AND COUNSELING HAS BEEN EFFECTED WITH THE PERSONNEL INVOLVED IN THIS OCCURRENCE.

[51] COOK 2 DOCKET 50-316 LER 88-001
MISSED TECHNICAL SPECIFICATION REQUIRED VENT STACK SAMPLING AS A RESULT OF INCORRECT SAMPLE VALVE ALIGNMENT DUE TO PERSONNEL ERROR.
EVENT DATE: 123087 REPORT DATE: 012888 NSSS: WE TYPE: PWR

(NSIC 208188) ON JANUARY 6, 1988, AT 1310 HOURS, IT WAS DISCOVERED THAT THE SAMPLE VALVE LINEUP FOR VRS-2500, UNIT 2 AUXILIARY BUILDING VENT RADIATION MONITOR, WAS INCORRECT. AS A RESULT OF THE INCORRECT LINEUP TECHNICAL SPECIFICATION 3.3.3.10 TABLE 3.3-13 SECTION 3A WAS NOT MET FOR CONTINUOUS NOBLE GAS MONITORING FROM DECEMBER 30, 1987 TO JANUARY 6, 1988. THE APPARENT CAUSE OF THIS EVENT WAS COGNITIVE PERSONNEL ERROR. THE CHEMISTRY TECHNICIAN INVOLVED DID NOT RESTORE THE VALVES TO THEIR NORMAL ALIGNMENT AFTER OBTAINING WEEKLY GRAB SAMPLE OFF VRS-2500 ON DECEMBER 30, 1987 AS PER PROCEDURE. AFTER DISCOVERY OF INCORRECT VALVE LINEUP ON JANUARY 6, 1988, PROPER FLOW WAS RESTORED TO VRS-2500 AND THE VALVE LINEUP WAS VERIFIED CORRECT (COMPLETED AT 1320 HOURS). THE CHEMISTRY TECHNICIAN INVOLVED WAS COUNSELLED AND THE SAMPLING PROCEDURE WAS CHANGED TO REQUIRED INDEPENDENT VERIFICATION OF THE PROPER VALVE ALIGNMENT AT VRS-2500.

[52] COOPER DOCKET 50-298 LER 88-002
REACTOR SCRAM DUE TO APRM HIGH FLUX RESULTING FROM AN EQUIPMENT PROBLEM WHEN RESTORING A REACTOR RECIRCULATION PUMP TO SERVICE.
EVENT DATE: 012888 REPORT DATE: 022688 NSSS: GE TYPE: BWR

(NSIC 208408) ON 1/28/88 AT 7:12 P.M., A REACTOR SCRAM OCCURRED DURING STARTUP OF THE 1B REACTOR RECIRCULATION (RR) PUMP DUE TO AVERAGE POWER RANGE MONITOR (APRM) HIGH FLUX. DURING THE PUMP STARTUP SEQUENCE, THE 1B REACTOR RECIRCULATION MOTOR GENERATOR (RRMG) SPEED CONTROL SYSTEM MALFUNCTIONED, RESULTING IN A CORE FLOW INCREASE GREATER THAN EXPECTED WHEN THE 1B RECIRCULATION LOOP DISCHARGE VALVE AUTOMATICALLY JOGGED OPEN. CONSEQUENTLY, REACTOR POWER INCREASED TO THE APRM HIGH FLUX SCRAM SETTING (APPROXIMATELY 66% POWER) FOR THE EXISTING REACTOR FLOW CONDITION. DURING THE ENSUING REACTOR VESSEL WATER LEVEL TRANSIENT, WATER LEVEL DECREASED TO THE POINT WHERE ACTUATION OF GROUPS 2, 3, AND 6 ISOLATIONS OCCURRED. ADDITIONALLY, UPON TRANSFER OF 4160V AC POWER, BOTH DIESEL GENERATORS AUTOMATICALLY STARTED. AT THE TIME OF THIS EVENT, THE PLANT WAS IN OPERATION AT APPROXIMATELY 35% POWER (275 MWE). THE FLOW CONTROL SYSTEM MALFUNCTION OCCURRED AS A RESULT OF A LOCKING SCREW, USED TO SECURE A DISC THAT ACTUATES HIGH AND LOW

TRAVEL STOPS, WHICH HAD CHOOSEN AND VIBRATED OFF. THE DISC DROPPED INTO THE LOW LIMIT SWITCH AND, DUE TO ITS ACTUATION, SCOOP TUBE DRIVE IN THE LOWER DIRECTION WAS ELECTRICALLY STOPPED. THEREFORE, THE EXPECTED RUNBACK OF THE RRMG TO MINIMUM SPEED FOLLOWING PUMP BREAKAWAY WAS PREVENTED.

[53] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-018 REV 01
UPDATE ON INADEQUATE ADMINISTRATIVE CONTROL ALLOWS PERSONNEL ERRORS TO RESULT IN INADEQUATE ISI PROGRAM.
EVENT DATE: 090287 REPORT DATE: 020888 NSSS: BW TYPE: PWR

(NSIC 208069) ON SEPTEMBER 2, 1987 AT 1700, CRYSTAL RIVER UNIT 3 WAS OPERATING AT 64% RATED POWER AND WAS GENERATING 533 MWE WITH REACTOR POWER LIMITED BY HAVING ONE REACTOR COOLANT PUMP INOPERABLE. DURING A FOUR MONTH REVIEW OF THE INSERVICE INSPECTION PROGRAM CONDUCTED JUNE TO SPETEMBER 1987 IT WAS DETERMINED THAT CERTAIN SYSTEM HYDROSTATIC TESTS (HYDROS) PERFORMED IN 1980 THROUGH 1986 WERE NOT PERFORMED IN ACCORDANCE WITH THE TECH SPEC REQUIRED EDITION OF THE ASME SECTION XI CODE. A TOTAL OF EIGHTY-FIVE DEFICIENCIES OCCURRING IN FIFTY-FIVE HYDROS ARE BEING REPORTED IN THIS LER SUPPLEMENT. FIFTY-SIX OF THE EVENTS RESULTED FROM PERSONNEL ERRORS. THE ROOT CAUSE OF THESE EVENTS IS INADEQUATE ADMINISTRATIVE CONTROLS FOR ISI AND MODIFICATION PROGRAM HYDROSTATIC TESTS. TWENTY-NINE OF THE EVENTS RESULTED FROM LACK OF UNDERSTANDING OF THE DOCUMENTATION REQUIRED FOR ASME PROGRAMS. FOR THOSE HYDROS EXHIBITING A DEFICIENCY, THE NECESSITY OF REPERFORMANCE OF THE TEST HAS BEEN EVALUATED. ALL RETESTS WERE COMPLETED BEFORE ASCENDING INTO MODE 2 FOLLOWING THE FALL 1987 OUTAGE.

[54] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-020 REV 01
UPDATE ON PERSONNEL ERROR DURING ORIGINAL PLANT DESIGN SPECIFICATION DEVELOPMENT LEADS TO ULTIMATE HEAT SINK TEMPERATURE EXCEEDING LIMIT AND TO OPERATION OUTSIDE DESIGN BASIS.
EVENT DATE: 090387 REPORT DATE: 012288 NSSS: BW TYPE: PWR

(NSIC 208372) ON SEPTEMBER 3, 1987, CRYSTAL RIVER UNIT 3 (CR3) WAS OPERATING AT APPROXIMATELY 63% RATED THERMAL POWER. AN NRC AUDIT OF PLANT COOLING WATER SYSTEMS REVEALED THAT THE ULTIMATE HEAT SINK (UHS) TEMPERATURE EXCEEDED THE MAXIMUM VALUE ASSUMED IN THE PLANT DESIGN BASIS. ALSO, THE PLANT TECH SPEC LIMIT FOR UHS TEMPERATURE WAS HIGHER THAN THE DESIGN BASIS. THIS EVENT WAS THE RESULT OF AN INADEQUATE PLANT DESIGN SPECIFICATION. THE MAXIMUM SEAWATER TEMPERATURE SPECIFIED FOR PLANT DESIGN WAS 85 DEGREES F., WHILE ACTUAL TEMPERATURES EXCEED THIS VALUE DURING THE SUMMER MONTHS. THE TECH SPEC ERROR APPEARS TO HAVE BEEN CAUSED BY INADVERTENTLY SELECTING A TEMPERATURE LIMIT FROM A CLOSED CYCLE COOLING LOOP RATHER THAN THE UHS DESIGN SPECIFICATION. ANALYSES INDICATE THAT THE NUCLEAR SERVICES CLOSED CYCLE COOLING SYSTEM 105 DEGREE F. TEMPERATURE LIMIT CAN BE MET WITH SEAWATER TEMPERATURES AS HIGH AS 92.3 DEGREES F IF ONLY TWO REACTOR BUILDING (RB) FAN COOLERS ARE USED. CR3 OPERATION CONTINUED BASED ON THIS ANALYSIS UNTIL SEPTEMBER 19, 1987 WHEN IT SHUT DOWN FOR A REFUELING OUTAGE. CR3 RESTARTED BASED ON A REVISED TECH SPECS LIMIT OF 85 F. ALL THREE RB FAN COOLERS CAN BE USED. FPC CONTINUES TO EVALUATE THE MAXIMUM TEMPERATURES PERMITTED IN THE CLOSED CYCLE COOLING SYSTEMS AND THE ULTIMATE HEAT SINK.

[55] CRYSTAL RIVER 3 DOCKET 50-302 LER 88-001
IMPROPER POSITIONING OF TURBINE BYPASS VALVES AND SLUGGISH RESPONSE OF A STARTUP FEEDWATER CONTROL VALVE RESULTS IN A LOW LEVEL IN "B" STEAM GENERATOR AND SUBSEQUENT EF ACTUATION.
EVENT DATE: 010788 REPORT DATE: 020888 NSSS: BW TYPE: PWR

(NSIC 208222) ON JANUARY 7, 1988, CRYSTAL RIVER UNIT 3 WAS IN THE HOT STANDBY CONDITION (MODE 3). AN AUTOMATIC ACTUATION OF EMERGENCY FEEDWATER OCCURRED DUE TO 2 OUT OF 4 EFIC CHANNELS DETECTING A LOW LEVEL IN THE "B" STEAM GENERATOR. A

SIMILAR ACTUATION ALSO OCCURRED ON JANUARY 9, 1988. IN BOTH CASES THE EMERGENCY FEEDWATER SYSTEM RESPONDED AS DESIGNED TO THE CONDITIONS IN THE "B" STEAM GENERATOR. THERE WERE SEVERAL FACTORS CONTRIBUTING TO THE EVENT OF JANUARY 7, INCLUDING A SLIGHT STEAM GENERATOR PRESSURE INCREASE AND SLUGGISH CONTROL RESPONSE OF THE "B" STARTUP FEEDWATER CONTROL VALVE. THE JANUARY 9 EVENT WAS DUE TO AN IMPROPER INTEGRAL SETTING IN A MODULE IN THE "B" STARTUP CONTROL VALVE CONTROL CIRCUIT. IN BOTH CASES THE EF ACTUATIONS WERE RESET ONCE NORMAL LEVEL CONTROL WAS ESTABLISHED WITH A MAIN FEEDWATER PUMP. AS A RESULT OF BOTH ACTUATIONS, WORK HAS BEEN PERFORMED ON THE "B" STARTUP FEEDWATER CONTROL VALVE AND ITS CONTROL CIRCUIT. IN ADDITION, NEW PROCEDURES ARE BEING WRITTEN FOR PREVENTIVE MAINTENANCE OF THE INTEGRATED CONTROL SYSTEM, AND OPERATORS WILL BE REMINDED OF THE INCREASED MONITORING REQUIRED AS A RESULT OF HAVING MULTIPLE CONTROL STATIONS IN MANUAL.

[56] CRYSTAL RIVER 3 DOCKET 50-302 LER 88-002
TECHNICIAN ERROR CAUSES TRIP OF OPERATING FEEDWATER PUMP WHICH RESULTS IN EMERGENCY FEEDWATER ACTUATION AND SUBSEQUENT OVERSPEED OF STEAM DRIVEN EMERGENCY FEEDWATER PUMP.
EVENT DATE: 010788 REPORT DATE: 020888 NSSS: BW TYPE: PWR

(NSIC 208223) ON JANUARY 7, 1988, CRYSTAL RIVER UNIT 3 WAS IN THE HOT STANDBY MODE (MODE 3). AN EMERGENCY FEEDWATER ACTUATION OCCURRED ON A LOSS OF BOTH MAIN FEEDWATER PUMPS. THE OPERATING FW PUMP TRIPPED WHEN CONTROL POWER WAS LOST TO ITS GOVERNOR. THE MOTOR DRIVEN EF PUMP STARTED AS DESIGNED, BUT THE STEAM DRIVEN EF PUMP STARTED AND THEN TRIPPED ON AN OVERSPEED CONDITION. THE CAUSE OF THE LOSS OF POWER TO THE FW PUMP GOVERNOR WAS AN ERROR BY AN I&C TECHNICIAN WORKING ON A RADIATION MONITOR WHICH IS POWERED FROM THE SAME BREAKER AS THE FW PUMP GOVERNOR. THE CAUSE OF THE TRIP OF THE STEAM DRIVEN EF PUMP WAS THE IMPROPER POSITIONING OF THE BYPASS VALVE AROUND THE STEAM SUPPLY VALVES TO THE PUMP. THE IDLE FW PUMP WAS STARTED, AND ONCE IT WAS VERIFIED THAT THE MAIN FEEDWATER SYSTEM WAS CONTROLLING LEVEL PROPERLY, THE EF ACTUATION WAS RESET. THE TECHNICIAN INVOLVED HAS BEEN COUNSELLED IN ACCORDANCE WITH APPROVED PLANT POLICIES. A MECHANICAL LOCKING DEVICE HAS BEEN INSTALLED ON THE STEAM SUPPLY BYPASS VALVE, AND A PROCEDURE CHANGE HAS BEEN MADE TO IMPROVE THE MONITORING OF THIS VALVE'S POSITION.

[57] DAVIS-BESSE 1 DOCKET 50-346 LER 88-001
SEISMIC TRIGGER FAILED TO MEET TECHNICAL SPECIFICATION FREQUENCY RANGE.
EVENT DATE: 010688 REPORT DATE: 020588 NSSS: BW TYPE: PWR
VENDOR: TELEDYNE-GEOTECH

(NSIC 208195) ON JANUARY 6, 1988 WHILE PREPARING TO DO THE REQUIRED CALIBRATION, IT WAS DETERMINED THAT THE SEISMIC TRIGGER THAT HAS BEEN INSTALLED SINCE JUNE 1977, DOES NOT HAVE THE FREQUENCY RANGE RESPONSE THAT IS LISTED IN TECHNICAL SPECIFICATION 3.3.3.3. IT DOES MEET REGULATOR GUIDE 1.12 REFERENCED IN THE BASIS SECTION OF THIS SPECIFICATION. THE ORIGINAL TRIGGER INSTALLED PRIOR TO JUNE 1977 DID MEET THE FREQUENCY RANGE RESPONSE BUT WAS REPLACED WHEN IT FAILED A SURVEILLANCE TEST (LER 77.13). THE REPLACEMENT MODEL WAS A TELEDYNE GEOTECH MODEL SP215C VERSION 36180.01.01 AS OPPOSED TO THE 36180.01.26 VERSION THAT HAD FAILED. THE TRIGGER HAS BEEN REMOVED, CALIBRATED OFF-SITE, AND REINSTALLED. IT IS CONSIDERED FUNCTIONAL BUT NOT TECHNICAL SPECIFICATION OPERABLE. SINCE THE ORIGINAL TELEDYNE GEOTECH MODEL IS NO LONGER MADE, A NEW TRIGGER WILL BE OBTAINED WHICH WILL MEET THE TECHNICAL SPECIFICATION (T.S.) RANGE SO THAT THE SYSTEM CAN BE DECLARED T.S. OPERABLE. PROCEDURES ARE NOW IN PLACE WHICH WOULD REQUIRE SIMILAR CHANGES TO THE SEISMIC MONITORING SYSTEM TO RECEIVE A 10CFR50.59 SAFETY REVIEW/EVALUATION. TECHNICAL SPECIFICATION AMENDMENT REQUEST WILL BE SUBMITTED TO THE NRC TO REDUCE THE OVERLY CONSERVATIVE FREQUENCY RANGE TO ONE THAT WILL NOT ONLY MEET THE REGULATORY GUIDES BUT ONE THAT CAN BE MET BY A STANDARD DESIGN THAT IS AVAILABLE.

[58] DAVIS-BESSE 1 DOCKET 50-346 LER 88-002
BATTERY CHARGER 1N PLACED IN SERVICE AFTER MAINTENANCE WITHOUT THE BALANCE POTENTIOMETERS SEALED AS REQUIRED FOR SEISMIC QUALIFICATIONS.
EVENT DATE: 011188 REPORT DATE: 021088 NSSS: BW TYPE: PWR

(NSIC 208299) ON JANUARY 11, 1988 AT 1900 HOURS, THE STATION PLACED BATTERY CHARGER 1N IN SERVICE FOLLOWING A PLANNED MAINTENANCE OUTAGE. ON JANUARY 12, 1988 AT 1428 HOURS, THE SHIFT SUPERVISOR WAS INFORMED THAT THE BATTERY CHARGER POTENTIOMETERS HAD NOT BEEN SEALED. AT 1600 HOURS IT WAS DETERMINED THAT THE POTENTIOMETERS MUST BE SEALED TO MAINTAIN THE SEISMIC QUALIFICATION OF BATTERY CHARGER 1N. THEREFORE, THE STATION HAD OPERATED WITH AN INOPERABLE BATTERY CHARGER FOR APPROXIMATEL6 19-1/2 HOURS. THE BATTERY CHARGER 1N POTENTIOMETERS WERE SUBSEQUENTLY SEALED. THIS EVENT IS A PERSONNEL ERROR DUE IN PART TO A DEFICIENT PROCEDURE. THE BATTERY CHARGER MAINTENANCE PROCEDURE DID NOT CONTAIN A STEP TO CONFIRM THE POTENTIOMETERS WERE SEALED PRIOR TO COMPLETING MAINTENANCE. A SIGNOFF STEP IS BEING ADDED TO THE PROCEDURE FOR THIS REQUIREMENT. IN ADDITION THE ASSOCIATED MAINTENANCE WORK ORDER (MWO) WAS SIGNED AS COMPLETE BY A MAINTENANCE FOREMAN WHEN ADDITIONAL WORK WAS STILL REQUIRED. THE IMPORTANCE OF THE MWO'S REFLECTING THE CURRENT STATUS OF WORK WILL BE EMPHASIZED WITH THE MAINTENANCE FOREMEN AND SUPERVISORS. OPERATIONS PERSONNEL WILL ALSO BE APPRISED OF THIS EVENT. POTENTIOMETERS ON THE REMAINING BATTERY CHARGERS HAVE BEEN VERIFIED AS SEALED.

[59] DAVIS-BESSE 1 DOCKET 50-346 LER 88-003
FIRE DETECTION SYSTEM SUPERVISION NOT TESTED FOR GROUND FAULT DETECTION DUE TO AN INADEQUATE PROCEDURE.
EVENT DATE: 011388 REPORT DATE: 021888 NSSS: BW TYPE: PWR
VENDOR: PRECIPITATION COMPANY
PYROTRONICS

(NSIC 208335) ON JANUARY 13, 1988 DURING RESEARCH TO RESOLVE REQUEST FOR ASSISTANCE 87-0411-00, FIRE PROTECTION PERSONNEL DETERMINED THAT THE SUPERVISED CIRCUITS SUPERVISION GROUND FAULT DETECTION ON FIRE DETECTION PANEL C- 4720 WAS INOPERABLE AND THE TECHNICAL SPECIFICATION REQUIREMENT TO VERIFY GROUND FAULT DETECTION ON THE REQUIRED INSTRUMENTATION HAD NOT BEEN MET SINCE 1980. A MODIFICATION TO THE SYSTEM WAS IMPLEMENTED IN 1980 WHICH INCLUDED GROUND FAULT DETECTION CAPABILITY. THE SURVEILLANCE TEST REVISION IMPLEMENTED DID NOT TEST FOR GROUND FAULT DETECTION CAPABILITIES DUE TO A LACK OF UNDERSTANDING OF NFPA 72D SUPERVISION REQUIREMENTS. WHEN THIS TEST WAS SPLIT INTO THREE NEW REQUIRED TECHNICAL SPECIFICATION SEPARATE PROCEDURES IN MARCH 1987, GROUND FAULT DETECTION CAPABILITY WAS INCLUDED. WHEN THE NEW PROCEDURE WAS PERFORMED IN MARCH 1987, AND SEPTEMBER 1987 A TEST DEFICIENCY RELATED TO GROUND FAULT DETECTION WAS IDENTIFIED. THIS TEST DEFICIENCY WAS INCORRECTLY EVALUATED AS NOT AFFECTING SYSTEM OPERABILITY DUE TO AN INADEQUATE UNDERSTANDING OF NFPA 72D REQUIREMENTS AND A LACK OF SPECIFIC ACCEPTANCE CRITERIA IN THE TEST PROCEDURE. TRAINING OF FIRE PROTECTION, INSTRUMENT AND CONTROL, AND OPERATIONS PERSONNEL WILL OCCUR. SPECIFIC ACCEPTANCE CRITERIA WILL BE INCORPORATED INTO THE TEST PROCEDURES.

[60] DAVIS-BESSE 1 DOCKET 50-346 LER 88-004
INTENTIONALLY MISSED HOURLY FIRE WATCH DUE TO RADIOLOGICAL CONTROL AIR EVACUATION.
EVENT DATE: 012088 REPORT DATE: 021888 NSSS: BW TYPE: PWR

(NSIC 208336) ON JANUARY 20, 1988 AT 1900 HOURS, THE HOURLY FIRE PATROL WAS NOT PERFORMED AS REQUIRED BY THE ACTION STATEMENTS OF TECHNICAL SPECIFICATIONS 3.3.3.8, AND 3.7.10. THE HOURLY FIRE PATROL WAS MISSED BECAUSE OF AN INADVERTENT GASEOUS RELEASE INSIDE THE AUXILIARY BUILDING. AS A RESULT OF THE RELEASE, THE RADIOLOGICAL CONTROL AREA (RCA) WAS EVACUATED AS A PRECAUTIONARY MEASURE FOR ALARA CONCERNS. THE GASEOUS RELEASE RESULTED A RADIATION DOSE LESS THAN 100

TIMES THE TECHNICAL SPECIFICATION REPORTING LIMITS. AT 2100 HOURS, THE HOURLY FIRE PATROLS RESUMED.

[61] DAVIS-BESSE 1 DOCKET 50-346 LER 88-005
INOPERABLE FIRE BARRIER WITH INOPERABLE FIRE DETECTION ON BOTH SIDES.
EVENT DATE: 012188 REPORT DATE: 022288 NSSS: BW TYPE: PWR

(NSIC 208337) ON JANUARY 21, 1988, AT 2000 HOURS, THE SHIFT SUPERVISOR WAS NOTIFIED THAT AN INOPERABLE FIRE BARRIER, 426-N/427-S, DID NOT HAVE OPERABLE FIRE DETECTION ON EITHER SIDE. THIS CONDITION HAD EXISTED FOR APPROXIMATELY 45 HOURS. THIS IS A VIOLATION OF TECHNICAL SPECIFICATION 3.7.10 BECAUSE A CONTINUOUS FIRE WATCH HAD NOT BEEN ESTABLISHED WITHIN 1 HOUR. THE SHIFT SUPERVISOR RESET THE ALARMS, WHICH RETURNED THE FIRE DETECTION ZONES TO AN OPERABLE STATUS. OPERATIONS PERSONNEL DID NOT RECOGNIZE THAT FIRE DETECTORS WERE INOPERABLE ON BOTH SIDES OF AN INOPERABLE FIRE BARRIER. A LIST OF FIRE DETECTION ZONES AND THEIR CORRESPONDING FIRE BARRIERS WILL BE DEVELOPED FOR OPERATIONS PERSONNEL. UNTIL THE LIST IS DEVELOPED, A CONTINUOUS FIRE WATCH WILL BE ESTABLISHED WHEN A FIRE DETECTION ALARM CAN NOT BE CLEARED WITHIN 1 HOUR. THESE WATCHES WILL BE MAINTAINED UNTIL THE CONDITION IS EVALUATED AND IT IS DETERMINED THAT A CONTINUOUS FIRE WATCH IS NOT REQUIRED. THE TECHNICAL SPECIFICATION FIRE WATCH PROCEDURE DOES NOT REQUIRE RESETTING THE FIRE ALARM IF THE CAUSE OF THE ALARM HAS CLEARED. THIS PROCEDURE WILL BE REVISED TO INCLUDE THIS REQUIREMENT.

[62] DIABLO CANYON 1 DOCKET 50-275 LER 86-001
FAILURE TO PERFORM PLANT VENT AIR SAMPLER FLOW ESTIMATE REQUIRED BY TECHNICAL SPECIFICATION 3.3.3.10 DUE TO PERSONNEL ERROR.
EVENT DATE: 010188 REPORT DATE: 020188 NSSS: WE TYPE: PWR

(NSIC 208220) ON JANUARY 1, 1988, AT 0430 PST, WITH UNIT 1 IN MODE 1 (POWER OPERATION), THE TIME REQUIREMENT OF TECH SPEC 3.3.3.10, TABLE 3.3-13, ITEM 3, ACTION STATEMENT 51, WAS EXCEEDED. THIS TECH SPEC REQUIRES THE PLANT VENT SYSTEM IODINE SAMPLER FLOWRATE MONITOR FLOW TO BE ESTIMATED AT LEAST ONCE PER 4 HOURS IF THE FLOWRATE MONITOR IS INOPERABLE. AT 0745 PST, IT WAS DISCOVERED THAT THE FLOWRATE ESTIMATE REQUIRED, AT 0430 PST, HAD NOT BEEN PERFORMED. AT 0810 PST, THE SAMPLER FLOWRATE WAS DETERMINED TO BE CORRECT AND UNCHANGED FROM THE FLOW ESTABLISHED AT 0030 PST. THE ROOT CAUSE OF THE EVENT WAS PERSONNEL ERROR (COGNITIVE) IN FAILURE TO PERFORM ACTIONS IDENTIFIED IN ADMINISTRATIVE PROCEDURE (AP) A-101S1, "RELIEVING THE WATCH." THIS PROCEDURE REQUIRES THAT FLOW READINGS BE TAKEN EVERY 4 HOURS BY THE PLANT SHIFT CHEMISTRY TECHNICIAN. TO PREVENT RECURRENCE, THE INDIVIDUAL INVOLVED WAS COUNSELED AND THE EVENT REVIEWED WITH ALL CHEMISTRY TECHNICIANS CONCERNING THE NEED TO REVIEW AP A-101S1 SHIFT TURNOVER REQUIREMENTS PROPERLY AND PERFORM ALL TECH SPEC REQUIRED ACTIONS.

[63] DIABLO CANYON 1 DOCKET 50-275 LER 88-003
SPURIOUS ACTUATION OF CONTAINMENT VENTILATION ISOLATION.
EVENT DATE: 010988 REPORT DATE: 020888 NSSS: WE TYPE: PWR

(NSIC 208119) ON JANUARY 9, 1988, AT 1415 PST, WITH THE UNIT IN MODE 3 (HOT STANDBY) AN AUTOMATIC ACTUATION OF THE CONTAINMENT VENTILATION ISOLATION SYSTEM (CVIS) OCCURRED. DURING MAINTENANCE ON STEAM DUMP CONTROL MODULE 1PC-516E, THE CONNECTING PLUG WAS INADVERTENTLY BUMPED AND IT'S METAL JACKET MOMENTARILY SHORTED THE ADJACENT MODULE'S POWER LEAD TO GROUND. THIS GROUNDING CAUSED AN ARC WHICH CREATED ELECTRICAL NOISE ON THE 120 VAC VITAL INSTRUMENT BUS WHICH SUPPLIES POWER TO THE RADIATION MONITORING CHANNELS. THE NOISE WAS SUFFICIENT FOR THE RADIATION MONITORING CHANNELS TO ACTUATE THE CVIS. AS REQUIRED BY 10 CFR 50.72 (B)(2)(II), A 4-HOUR NONEMERGENCY NOTIFICATION WAS COMPLETED BY 1715 PST, JANUARY 9, 1988. THE CVIS ACTUATION WAS ATTRIBUTED TO THE SUSCEPTIBILITY OF THE RADIATION MONITORING CHANNELS TO ACTUATE BECAUSE OF NOISE ON THEIR POWER SOURCE. THE CVIS

WAS RESET AT 1425 PST, JANUARY 9, 1988. TO PRECLUDE RECURRENCE, TIME DELAY CIRCUITRY, SIMILAR TO THAT ALREADY INSTALLED IN UNIT 2, WILL BE INSTALLED IN UNIT 1 IN THOSE RADIATION MONITORS THAT CAN ACTUATE THE CVIS AND ARE KNOWN TO BE SUSCEPTIBLE TO ELECTRICAL NOISE.

[64] DIABLO CANYON 1 DOCKET 50-275 LER 88-005
CONTAINMENT VENTILATION ISOLATION DUE TO ELECTRONIC NOISE AND LATE ISSUANCE OF A 10 CFR 50.72 REQUIRED REPORT DUE TO INADEQUATE PROCEDURAL GUIDANCE.
EVENT DATE: 020388 REPORT DATE: 030488 NSSS: WE TYPE: PWR

(NSIC 208452) ON FEBRUARY 3, 1988, AT 1523 PST, WITH UNIT 1 IN MODE 1 (POWER OPERATION), A CONTAINMENT VENTILATION ISOLATION (CVI) ACTUATION SIGNAL WAS GENERATED BY RADIATION MONITOR RM-14A DUE TO ELECTRONIC NOISE GENERATED WHEN AN I&C TECHNICIAN ENERGIZED RM-13 DURING MAINTENANCE. CVI VALVES NOT ALREADY IN THE CLOSED POSITION PRIOR TO THE EVENT AUTOMATICALLY CLOSED AS REQUIRED. THE CAUSE OF THE CVI ACTUATION WAS IMMEDIATELY KNOWN TO THE CONTROL OPERATOR WHO WAS VERBALLY IN CONTACT WITH THE I&C TECHNICIAN IN THE CONTROL ROOM. THE CONTROL OPERATOR RESET THE CVI ALARM, OPENED THE AFFECTED VALVES, AND VERIFIED THAT THE AFFECTED EQUIPMENT WAS RETURNED TO OPERATION. TO PREVENT RECURRENCES A TIME DELAY CIRCUITRY CHANGES SIMILAR TO THAT ALREADY INSTALLED IN UNIT 2, WILL BE INSTALLED IN THOSE RADIATION MONITORS THAT ACTUATE THE CVI SYSTEM. THE 4-HOUR NON-EMERGENCY REPORT REQUIRED BY 10 CFR 50.72(B)(2)(II) WAS NOT MADE UNTIL FEBRUARY 5, 1988, AT 1503 PST. THEREFORE, THE REPORTING REQUIREMENTS OF TECHNICAL SPECIFICATION 6.9.1 WERE NOT MET. .THIS LATE REPORTING WAS DUE TO INADEQUATE GUIDANCE IN ADMINISTRATIVE PROCEDURE AP C-11S2 REGARDING EXEMPTIONS TO REPORTING REQUIREMENTS. TO PREVENT RECURRENCE, AP C-11S2 HAS BEEN REVISED TO CLARIFY THE REPORTING REQUIREMENTS.

[65] DRESDEN 2 DOCKET 50-237 LER 87-010 REV 02
UPDATE ON CORE SPRAY SYSTEM ANALYTICAL PIPING STRESSES EXCEED FINAL SAFETY ANALYSIS REPORT DESIGN REQUIREMENTS DUE TO DESIGN AND CONSTRUCTION ERRORS.
EVENT DATE: 031787 REPORT DATE: 012788 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 208057) WHILE PERFORMING PIPING SYSTEM INSPECTIONS WITH DRESDEN UNIT 2 IN COLD SHUTDOWN FOR A REFUELING OUTAGE, THE EMBEDMENT PLATE FOR A SUPPORT LOCATED ON THE UNIT 2 SYSTEM "A" CORE SPRAY PIPING WAS FOUND DEGRADED. A REVIEW OF THE CORE SPRAY PIPING WAS INITIATED AND THE FOLLOWING ADDITIONAL DISCREPANCIES WERE FOUND; 1) THE DEGRADED EMBEDMENT PLATE HAD NOT BEEN FABRICATED PROPERLY, 2) A TEST RETURN LINE TEE HAD A WELDED CONNECTION WHERE THE DESIGN ANALYSIS INDICATED A FORGED CONNECTION, 3) A SUPPORT WHICH HAD PREVIOUSLY BEEN SPECIFIED FOR REMOVAL WAS STILL IN PLACE AND A SUPPORT INTENDED TO REPLACE IT HAD BEEN INSTALLED INCORRECTLY, 4) A NON-EXISTENT SNUBBER WAS INDICATED IN THE DESIGN ANALYSIS, 5) CERTAIN PIPING SECTION WERE A DIFFERENT SCHEDULE TYPE THAN THE DESIGN ANALYSIS INDICATED, AND 6) ANALYSIS REVEALED TWO ADDITIONAL EMBEDMENT PLATES NEEDING REINFORCEMENT. ON MARCH 17, 1987 THE CORPORATE ENGINEERING DEPARTMENT INDICATED THAT THE AS-FOUND CONDITION WAS IN EXCESS OF FINAL SAFETY ANALYSIS REPORT (FSAR) LIMITS BUT MET OPERABILITY CRITERIA FOR ALL DESIGN BASIS EVENTS AND THUS WAS OF MINIMAL SAFETY SIGNIFICANCE. REPAIRS WERE COMPLETED PRIOR TO UNIT STARTUP TO ENSURE FSAR COMPLIANCE. SUBSEQUENT INSPECTIONS AND ANALYSES HAVE REVEALED OTHER INCIDENCES OF OVERSTRESSED PIPING SUPPORTS AND CONNECTIONS.

[66] DRESDEN 2 DOCKET 50-237 LER 88-002
FAILURE TO TEST THE APRM INOP FUNCTION IN THE STARTUP MODE DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 020188 REPORT DATE: 022988 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 208439) ON FEBRUARY 1, 1988 DURING NORMAL, STEADY-STATE OPERATIONS, AN AUDIT OF TECH SPEC ADHERENCE IDENTIFED A DEFICIENCY IN THE WEEKLY AVERAGE POWER RANGE MONITOR (APRM) SURVEILLANCE PROCEDURE DRESDEN OPERATING SURVEILLANCE (DOS) 500-3, APRM ROD BLOCK AND SCRAM FUNCTIONAL TEST. THE DEFICIENCY RESULTED IN OMITTING THE APRM INOPERATIVE (INOP) FUNCTION TEST WHENEVER THE PROCEDURE WAS PERFORMED WITH THE REACTOR IN THE STARTUP MODE. THIS IS A VIOLATION OF TECH SPEC TABLE 4.1.1. BECAUSE THE NORMAL WEEKLY TESTING OF THE APRM INOP FUNCTION PROVIDES AMPLE MARGIN TO THE DESIGN VALVE OF .99999 RELIABILITY, AN ISOLATED CASE OF NOT PERFORMING THE WEEKLY TEST DOES NOT COMPROMISE THE INOP FUNCTION. THE MOST RECENT INSTANCE OF PERFORMING THE SURVEILLANCE IN THE STARTUP MODE (AND CONSEQUENTLY OMITTING THE INOP FUNCTION TEST) WAS ON 10/22/87 AND 7/14/87 FOR UNITS 2 AND 3 RESPECTIVELY. THE CORRECTIVE ACTION WILL BE TO REVISE THE PROCEDURE TO ENSURE THAT THE APRM INOP FUNCTION IS TESTED ON A WEEKLY BASIS INDEPENDENT OF WHETHER THE REACTOR IS IN THE STARTUP OR RUN MODE. ADDITIONALLY, STATION PROCEDURES FOR ALL WORK GROUPS ARE BEING REVIEWED TO ASSURE TECH SPEC COMPLIANCE. THIS IS THE FIRST EVENT OF THIS TYPE AT DRESDEN STATION.

[67] DRESDEN 3 DOCKET 50-249 LER 83-002 REV 1
UPDATE ON STEAM LEAK ON RWCU DRAIN VALVE DUE TO STEAM CUTTING OF VALVE SEAL.
EVENT DATE: 010583 REPORT DATE: 121787 NSSS: GE TYPE: BWR
VENDOR: CRANE COMPANY

(NSIC 207719) DURING UNIT 3 OPERATION (99% POWER) AT 1100 HOURS ON JANUARY 5, 1983, AN OPERATOR FOUND STEAM BLOWING FROM REACTOR WATER CLEANUP (RWCU) INLET LOW POINT DRAIN VALVE 3-1299-13 AND PIPE CAP. THE RWCU SYSTEM WAS ISOLATED TO STOP THE LEAKAGE AND WAS RETURNED TO SERVICE FOLLOWING REPAIRS. ALL LEAKAGE WAS CONTAINED IN THE REACTOR BUILDING FLOOR DRAIN SYSTEM. THERE WAS NO EFFECT ON PUBLIC HEALTH AND SAFETY. A SIMILAR EVENT WAS REPORTED BY REPORTABLE OCCURRENCE 82-58/036-0 UNDER DOCKET #50-237. THE CAUSE OF THE STEAMLEAK WAS DUE TO STEAM CUTTING THE VALVE SEAT AND GALLING OF THE PIPE CAP. DURING THE 1983 REFUELING OUTAGE, VALVE 3-1299-13 WAS REPLACED WITH A 3/4" MANUAL ISOLATION GLOBE VALVE. DURING THE SUBSEQUENT 1985 REFUEL OUTAGE, THIS GLOVE VALVE WAS REMOVED AND A PIPE CAP WELDED ON THE 3/4" LINE.

[68] DRESDEN 3 DOCKET 50-249 LER 88-001
LLRT INTERVALS DISCOVERED IN EXCESS OF 24 MONTHS ON AN INDIVIDUAL TEST BASIS DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 010588 REPORT DATE: 012988 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 2 (BWR)

(NSIC 208242) ON JANUARY 5, 1988, WITH UNIT 3 AT 95.8% RATED THERMAL POWER, WHILE REVIEWING PAST LOCAL LEAK RATE TESTING (LLRT) RECORDS, IT WAS DISCOVERED THAT 41 INDIVIDUAL TYPE "B" AND "C" LLRTS PERFORMED DURING THE PREVIOUS REFUELING OUTAGE HAD INDIVIDUAL TESTING INTERVALS IN EXCESS OF 24 MONTHS. AS 10 CFR 50 APPENDIX J REQUIRES THAT THIS TESTING BE PERFORMED ON INTERVALS NO GREATER THAN 24 MONTHS, CONCERN WAS RAISED AS TO WHETHER COMPLIANCE WITH 10 CFR 50 APPENDIX J HAD BEEN DEMONSTRATED. THE ROOT CAUSE WAS ATTRIBUTED TO A PROCEDURAL DEFICIENCY WITHIN THE SCHEDULING OF THE STATION SURVEILLANCE TRACKING PROGRAM; IT HAS BEEN THE STATION'S PRACTICE TO ENTER THE TEST DATE OF THE LAST INDIVIDUAL LLRT AS THE ENTIRE SURVEILLANCE COMPLETION DATE. CORRECTIVE ACTIONS INCLUDED PERFORMING AS MANY OF THE OVERDUE LLRTS AS POSSIBLE DURING REACTOR OPERATION AND UPGRADING THE TRACKING PROGRAM TO VERIFY THE LLRT INTERVALS ON AN INDIVIDUAL TEST BASIS IN THE FUTURE. MANAGEMENT CHANGES WERE ALSO IMPLEMENTED REGARDING ADMINISTRATION OF THE SURVEILLANCE PROGRAM. A SCHEDULAR EXEMPTION WAS REQUESTED FROM THE OFFICE OF NUCLEAR REACTOR REGULATION (NRR) FOR THE REMAINING UNTESTED VALVES. FURTHER REVIEW REVEALED THAT 28 INDIVIDUAL TYPE "B" AND "C" LLRTS PERFORMED DURING A PAST UNIT 2 REFUELLING OUTAGE.

[69] DRESDEN 3 DOCKET 50-249 LER 88-002
DIESEL GENERATOR AUTO START DURING WORK ON AUTO START RELAY DUE TO MANAGEMENT DEFICIENCY.
EVENT DATE: 011388 REPORT DATE: 020488 NSSS: GE TYPE: BWR

(NSIC 208292) ON JANUARY 13, 1988 AT 1940 HOURS, WITH UNIT 3 OPERATING AT 93% RATED CORE THERMAL POWER, THE UNIT 3 EMERGENCY DIESEL GENERATOR (EDG) AUTOMATICALLY (AUTO) STARTED WITH NO INITIATION CONDITIONS INDICATED IN THE CONTROL ROOM. OPERATING PERSONNEL WERE DISPATCHED TO THE UNIT 3 EDG OUTPUT BREAKER AND THE UNIT 3 EDG ROOM. THERE WAS NO WORK ACTIVITY FOUND IN EITHER OF THE TWO AREAS. HOWEVER, THE SHIFT FOREMAN FOUND THE COVER TO THE UNIT 3 EDG AUTO START RELAY LYING ON THE BASE OF THE UPPER SECTION OF THE EDG OUTPUT BREAKER CUBICLE. OPERATING PERSONNEL REVIEWED THEIR COPY OF THE ELECTRICAL MAINTENANCE (EM) DEPARTMENT NIGHT NOTES AND DISCOVERED THAT THE EMS HAD SCHEDULED TO INSTALL A MISSING COVER ON THE AUTO START RELAY PER WORK REQUEST #71752. AT 2035 HOURS THE UNIT 3 EDG WAS LOADED ONTO EMERGENCY BUS IN ACCORDANCE WITH DRESDEN OPERATING SURVEILLANCE (DOS) 6600-1, DIESEL GENERATOR SURVEILLANCE TEST. AFTER VERIFYING SATISFACTORY OPERATION, THE UNIT 3 EDG WAS SECURED AT 2040 HOURS. THE ROOT CAUSE OF THIS EVENT WAS ATTRIBUTED TO A MANAGEMENT DEFICIENCY CONCERNING WORK ACTIVITY ON IN-SERVICE EQUIPMENT. CONTRIBUTING FACTORS TO THE EVENT INCLUDE INSUFFICIENT CLEARANCE BETWEEN THE RELAY WIRES AND ITS COVER AND THAT INSTALLATION OF THE RELAY COVERS INVOLVES SNAPPING THE RELAY COVER TO THE RELAY BASE.

[70] FARLEY 1 DOCKET 50-348 LER 88-001
UNAUTHORIZED INDIVIDUAL ENTERED THE MAIN CONTROL ROOM.
EVENT DATE: 011488 REPORT DATE: 021288 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: FARLEY 2 (PWR)

(NSIC 208238) AT 0643 ON 1-14-88, A CONTRACTOR EMPLOYEE ENTERED THE MAIN CONTROL ROOM WITHOUT AUTHORIZED ACCESS. SINCE THE CONTROL ROOM IS A VITAL AREA, THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR73.71. THE CONTRACTOR EMPLOYEE WAS GOING TO THE WASTE AND DECONTAMINATION (W&D) GROUP'S BREAK ROOM WHERE HIS GROUP'S SHIFT TURNOVERS ARE CONDUCTED. IT WAS HIS SECOND DAY ON THE JOB AND HE DID NOT KNOW HOW TO GET TO THE BREAK ROOM. HE ASKED A PERMANENT FARLEY NUCLEAR PLANT WORKER FOR DIRECTIONS. THE FARLEY NUCLEAR PLANT WORKER, UNAWARE THAT THE CONTRACTOR WAS NOT AUTHORIZED TO ENTER THE CONTROL ROOM, TOLD THE CONTRACTOR TO FOLLOW HIM SINCE HE WAS GOING THAT WAY. SINCE THE CONTRACTOR WAS UNFAMILIAR WITH THIS PART OF THE AUXILIARY BUILDING LAYOUT, HE WAS UNAWARE THAT HE WAS ABOUT TO ENTER THE CONTROL ROOM. WHEN HE INSERTED HIS KEY-CARD, HIS ACCESS TO THE CONTROL ROOM WAS DENIED WHICH CAUSED AN ALARM IN THE SECURITY CENTRAL ALARM STATION. THE MAN DID NOT RECOGNIZE THAT HE WAS DENIED ACCESS AND CONTINUED TO FOLLOW THE FARLEY NUCLEAR PLANT WORKER INTO THE CONTROL ROOM. SECURITY PERSONNEL RESPONDED TO THE ALARM. THEY LOCATED THE INDIVIDUAL AT APPROXIMATELY 0700 IN THE W&D BREAK ROOM AND ESCORTED HIM OUT OF THE AUXILIARY BUILDING.

[71] FARLEY 1 DOCKET 50-348 LER 88-002
TECHNICAL SPECIFICATION ACTION STATEMENT REQUIREMENT NOT MET FOR INOPERABLE FIRE PROTECTION SYSTEMS.
EVENT DATE: 012088 REPORT DATE: 021988 NSSS: WE TYPE: PWR

(NSIC 208338) AT 0830 ON 1-20-88, WITH THE UNIT OPERATING IN STEADY STATE AT 100% REACTOR POWER, IT WAS DISCOVERED THAT THE MASTER OVERRIDE SWITCH FOR THE UNIT 1 PYROTRONICS PANEL WAS IN THE OVERRIDE POSITION. BY BEING IN THIS POSITION, THIS SWITCH RENDERED ALL OF THE UNIT 1 PREACTION WATER PROTECTION SYSTEMS INOPERABLE. TECHNICAL SPECIFICATION 3.7.11.2 REQUIRES THAT CONTINUOUS FIRE WATCHES OR HOURLY FIRE WATCH PATROLS (AS APPROPRIATE) BE POSTED WITHIN ONE HOUR WHENEVER ONE OF THESE SYSTEMS IS INOPERABLE. HOWEVER, THESE FIRE WATCHES WERE NOT ESTABLISHED AS REQUIRED SINCE IT WAS NOT REALIZED THAT THE SWITCH WAS IN OVERRIDE. THE PANEL SWITCH WAS RESET AT 0834 AND AN IMMEDIATE INVESTIGATION WAS BEGUN TO DETERMINE

WHY THE SWITCH WAS IN THE OVERRIDE POSITION. THIS EVENT WAS CAUSED BY PERSONNEL ERROR. THE INDIVIDUAL WHO FAILED TO ENSURE THAT THE SWITCH WAS IN THE CORRECT POSITION HAS BEEN COUNSELED.

[72] FARLEY 1 DOCKET 50-348 LER 88-003
PERSONNEL ERRORS RESULT IN SPECIAL REPORT NOT BEING SUBMITTED FOR AN INOPERABLE FIRE DOOR.
EVENT DATE: 021288 REPORT DATE: 021988 NSSS: WE TYPE: PWR

(NSIC 208410) ON 2-12-88, IT WAS DISCOVERED THAT A SPECIAL REPORT HAD NOT BEEN SUBMITTED AS REQUIRED BY TECHNICAL SPECIFICATION 3.7.12 ACTION STATEMENT A. THIS EVENT WAS CAUSED BY PERSONNEL ERRORS. THE EVENT WAS MISTAKENLY BELIEVED TO NOT BE REPORTABLE AND THUS THE PLANT INCIDENT REPORT WAS NOT AGGRESSIVELY PURSUED. THE INDIVIDUALS WHO WERE RESPONSIBLE FOR THE INVESTIGATION OF THIS EVENT AND DETERMINATION OF REPORTABILITY REQUIREMENTS HAVE BEEN COUNSELED.

[73] FARLEY 2 DOCKET 50-364 LER 88-001
FIRE DAMPER INOPERABLE DUE TO FAILURE TO CLOSE WITH AIR FLOW.
EVENT DATE: 011888 REPORT DATE: 021588 NSSS: WE TYPE: PWR

(NSIC 208196) ON 7-13-87, FNP SUBMITTED A SPECIAL REPORT (LER 87-011-00) CONCERNING INOPERABLE CONTROL ROOM FIRE DAMPERS. AS A RESULT, A FIRE DAMPER MAINTENANCE AND TESTING PROGRAM IS IN PROGRESS. AS A PART OF THIS PROGRAM, ON 1-11-88, FIRE DAMPER 121-116-05 WAS TESTED AND WOULD NOT CLOSE WITH AIR FLOW IN THE SYSTEM. THIS EVENT WAS CAUSED BY DESIGN DEFICIENCY IN THAT THE FIRE DAMPER WILL NOT CLOSE FULLY WITH AIR FLOW. DESIGN CHANGES HAVE BEEN INITIATED TO EVALUATE THE OPTIONS AVAILABLE AND PROVIDE THE APPROPRIATE DESIGN TO ENSURE THE PROPER OPERATION OF THE FIRE DAMPERS. THESE DESIGN CHANGES ARE EXPECTED TO BE IMPLEMENTED WITHIN THE NEXT SIX MONTHS. TECHNICAL SPECIFICATION 3.7.12 REQUIRES THESE FIRE DAMPERS TO BE RETURNED TO OPERABLE STATUS WITHIN SEVEN DAYS OR A SPECIAL REPORT MUST BE SUBMITTED WITHIN THE FOLLOWING 30 DAYS. THEREFORE THIS SPECIAL REPORT IS BEING SUBMITTED. ALL TECHNICAL SPECIFICATION ACTION STATEMENT REQUIREMENTS FOR THE FIRE DAMPER ARE BEING MET.

[74] FERMI 2 DOCKET 50-341 LER 88-001
NEUTRON MONITORING INSTRUMENTS INOPERABLE DUE TO PROCEDURAL INADEQUACY AND TECH SPEC IMPRACTICALITY.
EVENT DATE: 010388 REPORT DATE: 020288 NSSS: GE TYPE: BWR

(NSIC 208233) ON JANUARY 4, 1988, THE AVERAGE POWER RANGE MONITOR (APRM) SETDOWN TRIPS WERE FOUND OUTSIDE THE TECH SPEC ALLOWANCES. THIS WAS DUE TO AN INADEQUACY IN THE CALIBRATION PROCEDURE OF THE APRM FIXED NEUTRON FLUX UPSCALE TRIP. THEY WERE RECALIBRATED UPON DISCOVERY. THE APRM CALIBRATION PROCEDURE IS BEING REVISED. ADDITIONALLY, IT HAS BEEN RECOGNIZED THAT PLACING THE MODE SWITCH IN THE REFUEL POSITION TO PERFORM REQUIRED SOURCE RANGE (SRMS) AND INTERMEDIATE RANGE MONITORS (IRMS) FUNCTIONAL TESTS WAS IN TECHNICAL VIOLATION OF TECH SPECS 3.3.1 AND 3.3.7.6. ALSO NOT PLACING A REACTOR PROTECTION SYSTEM TRIP SYSTEM IN THE TRIP CONDITION PRIOR TO 3 IRM CHANNELS PER TRIP SYSTEM HAVING THEIR TESTING COMPLETED WAS ALSO A TECHNICAL VIOLATION DESPITE ALL CONTROL RODS BEING FULLY INSERTED. PERFORMANCE OF A FULL FUNCTIONAL TEST REQUIRES THE MODE SWITCH BE IN REFUEL SINCE THE ROD BLOCK FUNCTION CANNOT BE VERIFIED WITH THE MODE SWITCH IN SHUTDOWN. THE SRMS AND IRMS ARE NOT REQUIRED TO BE OPERABLE DURING OPERATIONAL CONDITION 1, SO THIS SITUATION BECOMES A PROBLEM FOLLOWING A PLANT SHUTDOWN. A TECH SPEC CHANGE WILL BE SUBMITTED.

[75] FERMI 2 DOCKET 50-341 LER 88-002
MAIN STEAM LINE RADIATION MONITOR SURVEILLANCE PROCEDURE INADEQUACY CAUSES MAIN STEAM ISOLATION VALVE CLOSURE.
EVENT DATE: 010488 REPORT DATE: 020388 NSSS: GE TYPE: BWR

(NSIC 208234) ON JANUARY 3, 1988, A FUSE WAS OPENED FOR THE POWER SUPPLY TO THE MAIN STEAM ISOLATION VALVES (MSIV) DC LOGIC WHILE REPLACING A FUSE FOR THE REACTOR HEAD VENT VALVES. THE CLOSURE LOGIC FOR THE DC SOLENOIDS WAS NOT RESET. ON JANUARY 4, 1988 AT 0950 HOURS, ALL FOUR OUTBOARD MAIN STEAM ISOLATION VALVES STROKED CLOSED TWICE DURING THE PERFORMANCE OF AN I&C SURVEILLANCE TEST THAT REUQIRED DE-ENERGIZING THE AC SOLENOIDS. THE SURVEILLANCE PROCEDURE, WHICH VERIFIES PROPER LOGIC STATE, WAS NOT CLEAR AS TO THE REQUIRED STEPS FOR CORRECTING THE LOW AMPERAGE CONDITION FOR THE MSIV DC SOLENOIDS. THE SURVEILLANCE PROCEDURE WILL BE REVISED. A LABEL WILL BE POSTED ON THE MSIV POWER SUPPLY PANELS. THE LABEL WILL STATE: "THIS CIRCUIT POWERS MSIV LOGIC AND THE NSS MUST BE NOTIFIED IF THIS CIRCUIT IS DE-ENERGIZED OR FOUND DE-ENERGIZED."

[76] FERMI 2 DOCKET 50-341 LER 88-003
SETPOINTS AND HEAD CORRECTION INCORRECT FOR RESIDUAL HEAT REMOVAL INTERFACE VALVES.
EVENT DATE: 010588 REPORT DATE: 020488 NSSS: GE TYPE: BWR

(NSIC 208235) ON JANUARY 5, 1988, AN ENGINEERING VERIFICATION OF THE CALCULATION FOR THE SETPOINTS AND STATIC HEAD CORRECTION FOR THE RESIDUAL HEAT REMOVAL SYSTEM HIGH PRESSURE TO LOW PRESSURE INTERFACE VALVE LEAKAGE PRESSURE MONITORS DETERMINED THAT BOTH CALCULATIONS WERE INCORRECT. THE SETPOINTS FOR THE ALARMS WERE ABOVE THE SETTING OF THE RELIEF VALVES. THEREFORE, THE RELIEF VALVES WOULD HAVE LIFTED PRIOR TO THE ALARMS BEING RECEIVED. THE INCORRECT STATIC HEAD CORRECTIONS RESULTED IN THE INSTRUMENTS BEING OUTSIDE THE VALUES ALLOWED BY THE TECH SPEC. THE TECH SPEC ALARM SETPOINTS WERE ALSO INCORRECT. THIS PROBLEM WAS DISCOVERED DURING THE PERFORMANCE OF A FORMAL VERIFICATION. THE CAUSE WAS ATTRIBUTED TO PERSONNEL ERROR. THE FORMAL VERIFICATION OF THE INSTRUMENTATION AND CONTROL SURVEILLANCES AGAINST THE DESIGN BASIS WAS COMPLETED. NO FURTHER PROBLEMS OF THIS NATURE WERE FOUND.

[77] FERMI 2 DOCKET 50-341 LER 88-004
HIGH PRESSURE COOLANT INJECTION AND REACTOR CORE ISOLATION COOLING ACTUATED FOLLOWING A REACTOR SCRAM.
EVENT DATE: 011088 REPORT DATE: 020988 NSSS: GE TYPE: BWR

(NSIC 208236) ON JANUARY 10, 1988, THE NORTH REACTOR FEED PUMP SPEED CONTROL POWER SUPPLY FAILED. THE PUMP RESPONDED TO THE LOSS OF CONTROL SIGNAL. A LOW REACTOR VESSEL WATER LEVEL RESULTED WHICH CAUSED A REACTOR SCRAM. HIGH PRESSURE COOLANT INJECTION AND REACTOR CORE ISOLATION COOLING SYSTEMS ACTUATED ALONG WITH THE APPROPRIATE ISOLATIONS. THE CAUSE FOR THE FEEDWATER TRANSIENT WAS THAT THE OVERVOLTAGE TRIP FOR THE POWER SUPPLY HAD DRIFTED INTO THE NORMAL OPERATING RANGE. THE POWER SUPPLY WAS REPLACED AND THE OVERVOLTAGE PROTECTION WAS SET TO THE MAXIMUM VALUE. DURING RESTORATION OF THE RWCU SYSTEM, A TECH SPEC REQUIREMENT WAS EXCEEDED. THE TECH SPEC ACTION STATEMENT FOR DEFEATING THE HIGH TEMPERATURE ISOLATION FUNCTION OF THE RWCU ISOLATION VALVES DOES NOT ALLOW SUFFICIENT TIME TO RESTART THE RWCU SYSTEM DURING HIGH OPERATOR ACTIVITY PERIODS. A TECH SPEC CHANGE WILL BE SUBMITTED TO DELETE THE REQUIREMENT FOR RWCU ISOLATIONS ON HIGH NON-REGENERATIVE HEAT EXCHANGER OUTLET TEMPERATURE INDICATION.

[78] FERMI 2 DOCKET 50-341 LER 88-005
REACTOR PRESSURE EXCEEDS 150 PSIG WITHOUT HIGH PRESSURE COOLING INJECTION OR REACTOR CORE ISOLATION COOLING BEING IN STANDBY.
EVENT DATE: 011188 REPORT DATE: 021088 NSSS: GE TYPE: BWR

(NSIC 208237) ON JANUARY 11, 1988 AT 0543 HOURS, REACTOR PRESSURE EXCEEDED 150 PSIG WITHOUT THE HIGH PRESSURE COOLING INJECTION SYSTEM OR THE REACTOR CORE ISOLATION COOLING SYSTEM BEING OPERABLE BY PLACING THEM IN THE STANDBY LINEUP. THIS RESULTED IN ENTRY INTO TECH SPEC 3.0.3. BY 0610 HOURS, BOTH SYSTEMS HAD BEEN PLACED IN STANDBY LINEUP. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR BY THE NUCLEAR SHIFT SUPERVISOR. HE ASSUMED THAT HE HAD TWELVE HOURS TO PLACE BOTH SYSTEMS IN SERVICE ONCE THE PLANT EXCEEDED 150 PSIG. THE INDIVIDUAL INVOLVED HAS BEEN DISCIPLINED IN ACCORDANCE WITH COMPANY POLICY. REQUIRED READING ON THIS EVENT WAS ISSUED TO OPERATIONS PERSONNEL. DISCUSSIONS OF THIS EVENT HAVE BEEN HELD BETWEEN OPERATIONS MANAGEMENT AND THE NUCLEAR SHIFT SUPERVISOR.

[79] FITZPATRICK DOCKET 50-333 LER 87-013 REV 01
UPDATE ON REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM, STEAM SUPPLY ISOLATION DUE TO SPURIOUS ANALOG TRANSMITTER TRIP UNIT GROSS FAILURE ALARM/TRIP.
EVENT DATE: 090587 REPORT DATE: 022488 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 208377) UPDATE REPORT: AT APPROXIMATELY 1655 ON 9/5/87 WHILE AT 98% RATED POWER, A SPURIOUS TRIP UNIT GROSS FAILURE ALARM AND AUTO ISOLATION OF THE REACTOR CORE ISOLATION COOLING (RCIC) (BN) SYSTEM OCCURRED. THE INITIATING SIGNAL CAME FROM RCIC HIGH STEAM LINE FLOW (HSLF) ANALOG TRANSMITTER TRIP SYSTEM (ATTS) MASTER TRIP UNIT (MTU). THE CAUSE OF THE MTU SPURIOUS GROSS FAILURE COULD NOT BE DETERMINED. AFTER COMPLETION OF FUNCTIONAL AND CALIBRATION TESTS, THE SYSTEM WAS RETURNED TO NORMAL. AT APPROXIMATELY 2100 HOURS ON 9/5/87 THE RCIC SYSTEM AGAIN ISOLATED DUE TO A SPURIOUS GROSS FAILURE TRIP FROM THE SAME MTU. TROUBLESHOOTING BY INSTRUMENT AND CONTROL'S TECHNICIAN YIELDED NO CAUSE. AT THIS TIME IT WAS DETERMINED TO DECLARE RCIC ADMINISTRATIVELY INOPERABLE, CONNECT A STRIP CHART RECORDER TO THE TRANSMITTER OUTPUT AND MONITOR FOR 24 HOURS. NO GROSS FAILURE OCCURRED AFTER MONITORING WAS INITIATED. AT 1330 HOURS ON 9/7/87 RCIC WAS DECLARED OPERABLE AND RETURNED TO SERVICE. ON 9/10/87 DURING AN UNSCHEDULED PLANT SHUTDOWN (LER-87-012) THE ELECTRONIC TRANSMITTER (13-DPT-83) AND MTU (13-MTU-283) WERE REPLACED. BOTH UNITS WERE RETURNED TO THE MANUFACTURER FOR EVALUATION. NO PROBLEMS WERE FOUND AND BOTH UNITS PASSED THE MANUFACTURER'S ACCEPTANCE TEST. LER 85-028 IS A SIMILAR EVENT.

[80] FT. CALHOUN 1 DOCKET 50-285 LER 88-001
FAILURE TO CONTROL A VERY HIGH RADIATION AREA.
EVENT DATE: 012588 REPORT DATE: 022288 NSSS: CE TYPE: PWR

(NSIC 208323) ON 1/25/88, A LOCKED BUT IMPROPERLY LATCHED DOOR TO THE WASTE DISPOSAL FILTERING ROOM (A VERY HIGH RADIATION AREA) WAS DISCOVERED BY THE LICENSEE AT 1620 (CST). THE PLANT WAS IN MODE 1, AT APPROXIMATELY 100% POWER AT THE TIME OF THE EVENT. RADIOACTIVE WASTE TECHNICIANS ENTERED THE WASTE DISPOSAL FILTERING ROOM, ROOM 11, AT 1400 (CST) TO CHECK FOR THE PRESENCE OF EXPENDED FILTERS. THE SPENT FILTERS WERE THEN TRANSFERRED TO ANOTHER ROOM TO BE PREPARED FOR OFF-SITE SHIPPING. IT IS BELIEVED THAT NO UNAUTHORIZED PERSONNEL ENTERED ROOM 11 PRIOR TO THE LICENSEE DISCOVERING THE LOCKED BUT UNLATCHED DOOR APPROXIMATELY 2 HOURS AFTER THE ROOM WAS ACCESSED. IT WAS FURTHER CONCLUDED THAT THE DOOR WAS NOT PROPERLY SECURED UPON COMPLETION OF THE AFOREMENTIONED WORK IN ROOM 11 AT 1420 (CST). TO PREVENT FUTURE OCCURRENCES, CHANGES HAVE BEEN MADE TO THE RADIATION PROTECTION MANUAL REQUIRING THE SHIFT HEALTH PHYSICS TECHNICIAN TO PERFORM A DOCUMENTED SECURITY CHECK ENSURING DOORS TO VERY HIGH RADIATION AREAS ARE PROPERLY LOCKED. DUAL VERIFICATION IS REQUIRED TO ENSURE THE DOOR IS PROPERLY SECURED AFTER ENTRIES. PADLOCKS HAVE BEEN ADDED TO DOORS WHICH PRESENTLY ALLOW ENTRANCE INTO A VERY HIGH RADIATION AREAS. NEW LATCH BOLT MONITORS WILL BE ADDED TO THE DOORS TO THESE AREAS OUTSIDE CONTAINMENT BY 9/1/88 WITH ALARM ANNUNCIATION AT THE SECURITY PANEL.

[81]   FT. CALHOUN 1   DOCKET 50-285   LER 88-002
INOPERABILITY OF ISOLATION VALVE ON HIGH PRESSURE SAFETY INJECTION.
EVENT DATE: 012588   REPORT DATE: 022588   NSSS: CE   TYPE: PWR
VENDOR: CHICAGO FLUID POWER

(NSIC 208324) ON 1/28/88, THE HIGH PRESSURE SAFETY INJECTION ALTERNATE DISCHARGE HEADER ISOLATION VALVE, HCV-2987, WAS DECLARED INOPERABLE AT 1500 (CST) DUE TO FAILURE TO MEET THE DESIGN CRITERIA FOR OPERABILITY. THE PLANT WAS IN MODE 1, AT APPROXIMATELY 100% POWER AT THE TIME OF THE EVENT. A TEMPORARY MECHANICAL JUMPER WAS INSTALLED TO BYPASS THE VALVE'S AIR INTENSIFIER. A NITROGEN GAS BOTTLE SUPPLYING 320 PSIG PRESSURE TO THE VALVE'S RECEIVER RESERVOIR WAS UTILIZED TO SUPPLY SUFFICIENT PRESSURE FOR VALVE OPERATIONS. THE VALVE WAS TESTED AND SUCCESSFULLY MET THE DESIGN CRITERIA AT 1745 (CST) THE SAME DAY ENDING THE 24 HOUR LIMITING CONDITION FOR OPERATION ENTERED BY TECHNICAL SPECIFICATION 2.3.2(E). THIS TEMPORARY MECHANICAL JUMPER WILL REMAIN IN PLACE UNTIL THE NEW REPLACEMENT AIR INTENSIFIER CAN BE INSTALLED. THE TEMPORARY MECHANICAL JUMPER AND NITROGEN SUPPLY PRESSURE WILL BE MONITORED BY OPERATIONS PERSONNEL DURING THEIR ROUTINE PLANT TOURS.

[82]   FT. CALHOUN 1   DOCKET 50-285   LER 88-003
INADEQUATE KEY CONTROL TO VERY HIGH RADIATION AREAS.
EVENT DATE: 020588   REPORT DATE: 030788   NSSS: CE   TYPE: PWR

(NSIC 208455) DURING AN NRC EXIT FOR A SPECIAL INSPECTION ON HEALTH PHYSICS HELD FEBRUARY 1-5, 1988, THE LICENSEE WAS NOTIFIED OF A POTENTIAL VIOLATION OF TECHNICAL SPECIFICATION 5.11.2 FOR FAILURE TO HAVE ADEQUATE KEY CONTROL TO VERY HIGH RADIATION AREAS. THE NRC INSPECTION CITED THE LICENSEE FOR BEING IN VIOLATION OF TECHNICAL SPECIFICATION 5.11.2 PRIOR TO THE ROOM 11 INCIDENT ON JANUARY 25, 1988. THE PLANT WAS IN MODE 2 AT 100 PERCENT POWER DURING THIS PERIOD. RED HEALTH PHYSICS PADLOCKS WERE ADDED TO ALL DOORS BARRING ACCESS TO VERY HIGH RADIATION AREAS.

[83]   GRAND GULF 1   DOCKET 50-416   LER 88-001
RPS ACTUATION DUE TO INADVERTENT GROUNDING OF A POWER SUPPLY.
EVENT DATE: 010388   REPORT DATE: 02..88   NSSS: GE   TYPE: BWR

(NSIC 208202) ON JANUARY 3, 1988 WITH THE PLANT IN OPERATIONAL CONDITION 4 AND ALL CONTROL RODS INSERTED, MAINTENANCE ELECTRICIANS WERE REPLACING A METER RELAY WHICH WAS MOUNTED DIRECTLY BELOW A POWER SUPPLY IN AN UPPER CABLE SPREADING ROOM PANEL. THE POWER SUPPLY POWERS THE TRIP UNITS AND THE LOGIC RELAYS OF THE NEWLY INSTALLED ALTERNATE ROD INSERTION/RECIRCULATION PUMP TRIP (ARI/RPT) SYSTEM. THE TIP OF THE SOCKET BEING USED FOR THE RELAY REPLACEMENT MADE CONTACT WITH THE POWER SUPPLY HEAT SINK AND A PANEL SCREW, GROUNDING THE POWER SUPPLY. THE POWER DISTURBANCE CAUSED AN ACTUATION OF THE ARI/RPT LOGIC WHICH TRIPPED BOTH REACTOR RECIRCULATION PUMPS AND VENTED THE SCRAM VALVE PILOT AIR HEADER. THE SCRAM VALVES OPENED ON THE LOSS OF AIR HEADER PRESSURE CAUSING A REACTOR PROTECTION SYSTEM (RPS) ACTUATION ON SCRAM DISCHARGE VOLUME HIGH WATER LEVEL. THE POWER SUPPLY WAS ONE OF TWO RECENTLY INSTALLED AS PART OF THE NEW ARI/RPT SYSTEM. THE ENERGIZED HEAT SINKS WERE NOT PROTECTED AGAINST INCIDENTAL CONTACT. PROTECTIVE GUARDS WILL BE INSTALLED ON THE HEAT SINKS OF THE TWO POWER SUPPLIES. PROCUREMENT OF NONCONDUCTIVE TOOLS FOR USE IN SUCH AREAS IS BEING PURSUED.

[84]   GRAND GULF 1   DOCKET 50-416   LER 88-003
EMERGENCY CORE COOLING SYSTEMS DELTA PRESSURE INSTRUMENTATION NOT CALIBRATED IN ACCORDANCE WITH TECH SPECS.
EVENT DATE: 010988   REPORT DATE: 020888   NSSS: GE   TYPE: BWR

(NSIC 208276) ON JANUARY 6, 1988 GRAND GULF NUCLEAR STATION UNIT 1 RESUMED POWER

OPERATIONS FOLLOWING THE SECOND REFUELING OUTAGE. ON JANUARY 9, 1988 THE LINE BREAK INSTRUMENTATION FOR RESIDUAL HEAT REMOVAL (RHR) "A" HIGH DIFFERENTIAL PRESSURE (DP) ALARMED AND SEALED IN. AN INVESTIGATION OF THE INSTRUMENTATION REVEALED THAT IT WAS WORKING AS DESIGNED AND SENSING THE ACTUAL DP BETWEEN THE RHR "A" AND LPCS INJECTION LINES. IT WAS CONCLUDED THAT NO ACTUAL LINE BREAK HAD OCCURRED IN EITHER RHR "A" OR LOW PRESSURE CORE SPRAY (LPCS) PIPING IN THE REACTOR DOWNCOMER ANNULUS. THE CAUSE OF THE ALARM WAS DETERMINED TO BE A CHANGE IN THE NORMAL INDICATED DP BETWEEN THESE TWO EMERGENCY CORE COOLING SYSTEM INJECTION LINES. THE INSTRUMENT HAD NOT BEEN CALIBRATED TO ACTUAL PRESSURES EXPERIENCED AT 100 PERCENT POWER FOLLOWING THE SECOND REFUELING OUTAGE. AN ANALYSIS IS BEING PERFORMED TO DETERMINE NORMAL INDICATED DP AT NOMINAL FULL POWER. THIS WILL SUPPORT CORRECTIONS, IF NECESSARY, TO THE INSTRUMENT SETPOINTS TO BE MADE. THIS ANALYSIS WILL BE COMPLETE BY FEBRUARY 18, 1988.

[85] GRAND GULF 1 DOCKET 50-416 LER 88-002
REACTOR SCRAM DUE TO MAIN OUTPUT TRANSFORMER FAULT.
EVENT DATE: 011088 REPORT DATE: 020988 NSSS: GE TYPE: BWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 208203) ON JANUARY 10, 1988 THE "B" PHASE GENERATOR OUTPUT TRANSFORMER FAILED CAUSING THE GENERATOR OUTPUT BREAKERS TO OPEN ON A TRANSFORMER DIFFERENTIAL TRIPPING THE MAIN TURBINE/GENERATOR. THE REACTOR AUTOMATICALLY SCRAMMED ON THE TURBINE CONTROL VALVE FAST CLOSURE SIGNAL. AN INVESTIGATION DETERMINED THAT A WINDING FAULT OCCURRED ON THE HIGH VOLTAGE SIDE OF THE TRANSFORMER. THE "B" PHASE TRANSFORMER WAS DISCONNECTED AND A SPARE TRANSFORMER CONNECTED TO THE "B" PHASE. ELECTRICAL TESTS AND OIL SAMPLES OF THE "A" PHASE, "C" PHASE, AND THE SPARE TRANSFORMERS INDICATED ACCEPTABLE INSULATION LEVELS AND OIL CHARACTERISTICS. PREVIOUS TEST RESULTS OF THE "B" PHASE TRANSFORMER WERE REVIEWED TO DETERMINE IF THE FAILURE COULD HAVE BEEN ANTICIPATED. BASED ON THE DATA AVAILABLE, A PRELIMINARY INVESTIGATION CONCLUDED THAT THE FAULT COULD NOT HAVE BEEN ANTICIPATED OR PREVENTED AND THAT THE EXISTING PREVENTATIVE MAINTENANCE PROGRAM OF OIL SAMPLING AND ANALYSIS IS ADEQUATE. ALTHOUGH SYSTEM ENERGY RESOURCES, INC. BELIEVES ITS CURRENT PROGRAM IS ADEQUATE, AN INDEPENDENT REVIEW BY HIGH VOLTAGE SPECIALISTS WILL BE PERFORMED TO SEE IF ANY ENHANCEMENT TO THE CURRENT PROGRAM WOULD BE BENEFICIAL.

[86] GRAND GULF 1 DOCKET 50-416 LER 88-005
REACTOR SCRAM ON LOW WATER LEVEL DUE TO MALFUNCTION OF TURBINE BYPASS VALVE SETPOINT CONTROLLER.
EVENT DATE: 011188 REPORT DATE: 021088 NSSS: GE TYPE: BWR

(NSIC 208204) ON JANUARY 11, 1988 WHILE IN HOT SHUTDOWN WITH ALL CONTROL RODS INSERTED, A REACTOR PROTECTION SYSTEM (RPS) ACTUATION OCCURRED ON A REACTOR LOW WATER LEVEL. REACTOR PRESSURE WAS BEING CONTROLLED BY THE TURBINE BYPASS VALVES. INITIALLY OPERATORS OBSERVED AN INCREASE IN REACTOR WATER LEVEL. THIS WAS DISCOVERED TO BE CAUSED BY A DECREASE IN THE REACTOR PRESSURE CONTROLLER SETPOINT WHICH OPENED THE TURBINE BYPASS VALVES REDUCING REACTOR PRESSURE. OPERATORS RAISED THE SETPOINT WHICH CLOSED THE BYPASS VALVES. THE WATER LEVEL THEN DROPPED BELOW THE LOW LEVEL SCRAM SETPOINT AS VOID FORMATION DECREASED. THE REACTOR WATER LEVEL REACHED A MINIMUM OF 9 INCHES ABOVE INSTRUMENT ZERO (175.7 INCHES ABOVE THE TOP OF ACTIVE FUEL). MAINTENANCE PERSONNEL INVESTIGATING THE CAUSE OF THE TURBINE BYPASS VALVE PRESSURE SETPOINT SHIFT COULD NOT REPEAT THE OCCURRENCE DURING TESTING OF THE SETPOINT CONTROLLER. ALTHOUGH NO SPECIFIC COMPONENT MALFUNCTION COULD BE IDENTIFIED, THE MOST LIKELY COMPONENTS THAT COULD HAVE CAUSED THE MALFUNCTION IN THE SETPOINT CONTROLLER ARE THE MOTOR DRIVEN POTENTIOMETER AND THE SETPOINT RAISE AND LOWER PUSHBUTTONS. THESE COMPONENTS WERE REPLACED AND RETESTED.

[87] GRAND GULF 1 DOCKET 50-416 LER 88-004
REACTOR WATER CLEANUP SYSTEM ISOLATION DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 011288 REPORT DATE: 021188 NSSS: GE TYPE: BWR

(NSIC 208277) AT 2325 ON JANUARY 12, 1988 A REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATION OCCURRED AS OPERATORS PREPARED TO SECURE ONE OF TWO OPERATING RWCU PUMPS. BECAUSE OF THE LOW LEVEL OF DECAY HEAT PRESENT SHORTLY AFTER THE REFUELING OUTAGE, REACTOR PRESSURE WAS DECREASING. OPERATIONS PERSONNEL HAD BEEN INSTRUCTED TO MINIMIZE COOLDOWN DUE TO THE ANTICIPATED TRANSITION TO OPERATIONAL CONDITION 2 ON JANUARY 13, 1988. AFTER OBSERVING SYSTEM FLOW FLUCTUATIONS, OPERATORS REMOVED ONE FILTER/DEMINERALIZER FROM SERVICE. THE OPERATORS NOTED THAT REACTOR PRESSURE HAD DECREASED TO 87 PSIG. THE RWCU SYSTEM FLOW FLUCTUATIONS WERE INDUCED BY HAVING BOTH RWCU PUMPS OPERATING. ONE RWCU PUMP IS NORMALLY REMOVED FROM SERVICE WHEN REACTOR PRESSURE DECREASES TO 100 PSIG. WHEN THE OPERATOR BEGAN TO THROTTLE THE FILTER/DEMINERALIZER BYPASS VALVE, A DIFFERENTIAL FLOW SIGNAL WAS SENSED BY LEAK DETECTION INSTRUMENTATION. THE OPERATOR SECURED BOTH RWCU PUMPS IN AN ATTEMPT TO CLEAR THE HIGH DIFFERENTIAL FLOW ALARM BEFORE THE 45 SECOND TIME DELAY EXPIRED; HOWEVER, THE SIGNAL DID NOT CLEAR AND ALL GROUP 8 CONTAINMENT ISOLATION VALVES CLOSED. OPERATORS PERFORMED A SYSTEM WALKDOWN INSPECTION FOR ABNORMAL LEAKAGE AND VERIFIED SYSTEM INTEGRITY. ONE RWCU PUMP WAS RETURNED TO OPERATION AT 0105 ON JANUARY 13, 1988.

[88] GRAND GULF 1 DOCKET 50-416 LER 88-006
CONDENSER MANWAY LEAKAGE ON HOTWELL LOW LEVEL SWITCHES TRIPS ALL CONDENSATE AND CONDENSATE BOOSTER PUMPS WHICH LED TO REACTOR SCRAM ON LOW WATER LEVEL.
EVENT DATE: 012088 REPORT DATE: 021988 NSSS: GE TYPE: BWR
VENDOR: MERCOID CORP.
SOUTHWESTERN ENGINEERING COMPANY

(NSIC 208356) ON 1/20/88 AT 0439 THE REACTOR AUTOMATICALLY SCRAMMED ON LOW REACTOR WATER LEVEL (LEVEL 3, +11.4 INCHES). THE LOW REACTOR WATER LEVEL OCCURRED WHEN THE CONDENSATE AND CONDENSATE BOOSTER PUMPS TRIPPED RESULTING IN LOSS OF FEEDWATER FLOW. THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM WAS MANUALLY INITIATED AND THE HIGH PRESSURE CORE SPRAY (HPCS) SYSTEM AUTOMATICALLY INITIATED ON LOW LOW REACTOR WATER LEVEL (LEVEL 2, -41.6 INCHES) AND INJECTED TO THE REACTOR VESSEL. A CONDENSER CIRCULATING WATER MANWAY COVER LEAKED AND SPRAYED WATER ON THE HOTWELL LOW LEVEL SWITCHES CAUSING A FALSE LOW HOTWELL LEVEL SIGNAL. THE LEAK WAS FROM A RUBBER GASKET WHICH SLIPPED FROM ITS GROOVE ON A MANWAY COVER. THE GASKET WAS REPLACED AND GLUED IN PLACE PREVENTING SLIPPAGE BETWEEN THE COVER AND THE MATING SURFACE. PROTECTIVE SHIELDS WERE INSTALLED AROUND THE LEVEL SWITCHES TO PROTECT THEM AGAINST WATER SPRAY EVENTS.

[89] GRAND GULF 1 DOCKET 50-416 LER 88-007
FAILURE TO TAKE THE REQUIRED ACTION IN TECHNICAL SPECIFICATIONS FOR INOPERABLE RADIATION MONITOR ON STANDBY SERVICE WATER SYSTEM.
EVENT DATE: 020488 REPORT DATE: 030488 NSSS: GE TYPE: BWR

(NSIC 208464) 1/27/88 THE PROCESS RADIATION MONITORING SAMPLE PUMP FOR STANDBY SERVICE WATER (SSW) "A" RETURN FLOW WAS LOGGED OUT-OF-SERVICE. IT WAS NOT IDENTIFIED AS BEING TECH SPEC RELATED AND THE LIMITING CONDITION REQUIRED ACTION WAS NOT TAKEN. THE CONDITION REMAINED FOR OPERATION (LCO) UNDETECTED UNTIL FEBRUARY 4, AT WHICH TIME THE APPROPRIATE LCO WAS ENTERED AND GRAB SAMPLES TAKEN IN ACCORDANCE WITH TECH SPECS UNTIL THE SAMPLE PUMP WAS RETURNED TO OPERATION. THE AUXILIARY BUILDING OPERATOR PERFORMING NORMAL ROUNDS ASSUMED THAT THE SAMPLE PUMP WAS SECURED BECAUSE OF A DEFICIENCY TAG ON THE PUMP SUCTION VALVE. IT IS BELIEVED THAT THE SAMPLE PUMP HAD BEEN INTENTIONALLY REMOVED FROM SERVICE RATHER THAN HAVING FAILED. SUBSEQUENT INVESTIGATION REVEALED THAT THE PUMP WAS NOT RUNNING DUE TO A BLOWN FUSE. THE SHIFT SUPERVISOR ON SHIFT AT THE TIME OF THE OCCURRENCE AND SUBSEQUENT SHIFT SUPERVISORS FAILED TO IDENTIFY THE CONDITION AS

REQUIRING ENTRY INTO AN LCO ACTION STATEMENT WHEN REVIEWING THE ROUND SHEETS. ADDITIONALLY, A COMMON "LOW SAMPLE FLOW" ANNUNCIATOR WAS IN CONTINUOUS ALARM DUE TO AN ASSOCIATED RADWASTE SAMPLE PUMP BEING NORMALLY SECURED. THEREFORE CONTROL ROOM OPERATORS COULD NOT DETECT LOW SSW "A" SAMPLE FLOW.

[90]   HATCH 2   DOCKET 50-366   LER 87-013 REV 01
UPDATE ON UNINSULATED STEAM PIPING RAISES DIFFERENTIAL AIR TEMPERATURE CAUSING ESF VALVE ISOLATIONS.
EVENT DATE: 101987   REPORT DATE: 022988   NSSS: GE   TYPE: BWR

(NSIC 208308) ON 10/19/87 AND AGAIN ON 11/12/87, PLANT OPERATIONS PERSONNEL RECEIVED AN ALARM IN THE MAIN CONTROL ROOM INDICATING A REACTOR CORE ISOLATION COOLING (RCIC EIIS CODE BN) PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS EIIS CODE JM) LOGIC ACTUATION HAD OCCURRED. THESE WERE UNPLANNED LOGIC ACTUATIONS OF AN ENGINEERED SAFETY FEATURE (ESF). THE ROOT CAUSE OF THESE EVENTS IS DUE TO MISSING INSULATION ON SOME HIGH PRESSURE COOLANT INJECTION (HPCI EIIS CODE BJ) STEAM SUPPLY PIPING. APPROXIMATELY SIX FEET OF PIPE WAS NOT INSULATED AND THIS PIPING WAS NEAR ONE OF THE RCIC TORUS CHAMBER DIFFERENTIAL AIR TEMPERATURE SENSORS. CORRECTIVE ACTIONS FOR THESE EVENTS INCLUDED: 1) CHECKING FOR STEAM LEAKS AND INITIATING REQUIRED TECHNICAL SPECIFICATION ACTIONS, 2) INVESTIGATING THE EFFECT OF TEMPORARY DUCTS, 3) VERIFYING CORRECT VENTILATION SYSTEM OPERATION, 4) REVIEWING WORK IN PROGRESS, 5) DEVELOPING AND ANALYZING TREND DATA, 6) PERFORMING AN ENGINEERING INVESTIGATION, 7) INSTALLING TEMPORARY FANS, 8) INVESTIGATING WHY THE INSULATION WAS MISSING, AND 9) INSTALLING INSULATION AND REMOVING THE TEMPORARY FANS.

[91]   HATCH 2   DOCKET 50-366   LER 88-001
INADEQUATE PROCEDURE CAUSES MIS-ASSEMBLY OF VALVE RESULTING IN ESF SYSTEM INOPERABILITY.
EVENT DATE: 010688   REPORT DATE: 012988   NSSS: GE   TYPE: BWR
VENDOR: AMETEK INC. (SCHUTTE & KOERTTING DIV)

(NSIC 208197) ON 1/6/88 AT APPROXIMATELY 0800 CST, UNIT 2 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 1826 MWT (APPROXIMATELY 75 PERCENT OF RATED THERMAL POWER). THE REACTOR CORE ISOLATION COOLING (RCIC EIIS CODE BN) SYSTEM WAS OUT OF SERVICE FOR VALVE MAINTENANCE AND TESTING. AT THAT TIME, PLANT OPERATIONS PERSONNEL DETERMINED THAT THE HIGH PRESSURE COOLANT INJECTION (HPCI EIIS CODE BJ) SYSTEM WAS INOPERABLE. HAVING BOTH THE HPCI AND RCIC SYSTEMS INOPERABLE IN THE RUN MODE IS PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS. THE ROOT CAUSE OF THIS EVENT IS AN INADEQUATE PROCEDURE. SPECIFICALLY, A MAINTENANCE PROCEDURE DID NOT PROVIDE SUFFICIENT GUIDANCE TO ENSURE THAT PROPER CLEARANCES WERE MAINTAINED ON A TURBINE STOP VALVE. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED: 1) INVESTIGATING THE STATUS OF THE HPCI TURBINE STOP VALVE, 2) RETURNING RCIC TO AN OPERABLE STATUS AND PERFORMING TECHNICAL SPECIFICATIONS REQUIRED ACTIONS, 3) REPAIRING THE HPCI TURBINE STOP VALVE, 4) INITIATING PROCEDURE REVISIONS, AND 5) VERIFYING THAT THE UNIT 1 HPCI TURBINE STOP VALVE DOES NOT HAVE A SIMILAR PROBLEM.

[92]   HATCH 2   DOCKET 50-366   LER 88-003
RWCU CONTAINMENT ISOLATION VALVE CLOSES SPURIOUSLY.
EVENT DATE: 011788   REPORT DATE: 021088   NSSS: GE   TYPE: BWR

(NSIC 208268) ON 1/17/88 AT APPROXIMATELY 1622 CST, UNIT 2 WAS IN THE REFUELING MODE OF OPERATION AT AN APPROXIMATE POWER LEVEL OF 0 MWT (APPROXIMATELY 0 PERCENT OF RATED THERMAL POWER). AT THAT TIME, PLANT OPERATIONS PERSONNEL WERE PERFORMING AN OPERATIONS PROCEDURE AND NOTED THAT ONE OF THE REACTOR WATER CLEAN UP (RWCU EIIS CODE CE) PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS EIIS CODE JM) VALVES HAD CLOSED. THIS WAS AN UNANTICIPATED ACTUATION OF AN ENGINEERED SAFETY FEATURE. THE ROOT CAUSE OF THIS IS A SPURIOUS ISOLATION OF THE VALVE. NO ALARMS

WERE RECEIVED WHEN THE VALVE ISOLATED. A THOROUGH INVESTIGATION OF THE EVENT DID NOT DETERMINE ANY POSSIBLE CAUSE FOR THE ACTUATION. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED: 1) REVIEWING WORK ACTIVITIES IN THE MAIN CONTROL ROOM, 2) REVIEWING THE PROCEDURE AND WORK PERFORMED BY THE PROCEDURE, 3) CHECKING SYSTEM TEMPERATURES, 4) CHECKING CALIBRATION AND OPERABILITY OF ISOLATION INSTRUMENTS, 5) VERIFYING ANNUNCIATOR FUNCTION, 6) REVIEWING OTHER SYSTEM OPERATIONS, AND 7) SCHEDULING LOGIC SYSTEM FUNCTION TESTING AS PART OF THE STANDARD OUTAGE RELATED WORK.

[93] HATCH 2 DOCKET 50-366 LER 88-004
PRIMARY CONTAINMENT PENETRATIONS FAIL LOCAL LEAK RATE TESTS DUE TO NORMAL EQUIPMENT WEAR.
EVENT DATE: 012088 REPORT DATE: 021688 NSSS: GE TYPE: BWR
VENDOR: ROCKWELL MANUFACTURING COMPANY

(NSIC 208411) ON 1/20/88, UNIT 2 WAS IN THE REFUELING MODE OF OPERATION AT AN APPROXIMATE POWER LEVEL OF 0 MWT (APPROXIMATELY 0 PERCENT OF RATED THERMAL POWER), IN PREPARATION FOR A SCHEDULED REFUELING OUTAGE. AT THAT TIME, PLANT PERSONNEL WERE PERFORMING LEAK RATE TESTING ON SOME PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS EIIS JM) PENETRATIONS AND DETERMINED THAT SOME OF THE PENETRATIONS WOULD NOT MEET THE LEAKAGE REQUIREMENTS ALLOWED BY THE PLANT'S TECHNICAL SPECIFICATIONS. THE ROOT CAUSE OF THE LEAKAGE IS BELIEVED TO BE NORMAL EQUIPMENT USE AND WEAR. THE DETAILS OF THE CAUSE AND THE CORRECTIVE ACTIONS PERFORMED FOR EACH FAILED VALVE ARE BEING EVALUATED AND WILL BE INCLUDED IN A REVISION TO THIS LER. THE REVISION WILL BE SUBMITTED WITHIN APPROXIMATELY 30 DAYS FOLLOWING PLANT START UP.

[94] HOPE CREEK 1 DOCKET 50-354 LER 88-001
FAILURE OF THE LIQUID AND GASEOUS RADWASTE DISCHARGE MONITORS TO PASS FUNCTIONAL TESTS DUE TO DESIGN AND PROCEDURE DEFICIENCIES.
EVENT DATE: 010688 REPORT DATE: 020588 NSSS: GE TYPE: BWR

(NSIC 208239) ON JANUARY 6, 1988 AT 1600 HOURS THE PLANT WAS IN OPERATIONAL CONDITION 1 (POWER OPERATION) AT 100% POWER GENERATING 1066 MWE AND FUNCTIONAL TESTING OF THE LIQUID RADWASTE DISCHARGE MONITOR WAS IN PROGRESS. DURING THE TEST, THE VALVES WHICH ISOLATE THE LIQUID RADWASTE DISCHARGE LINE TO THE COOLING TOWER BLOWDOWN LINE DID NOT CLOSE WHEN A RADWASTE MONITOR DOWNSCALE FAILURE WAS SIMULATED AND THE INSTRUMENTATION CHANNEL WAS DECLARED INOPERABLE. THE ROOT CAUSE OF THIS OCCURRENCE WAS THAT THE LOGIC FOR THE VALVES WHICH ISOLATE THE LIQUID RADWASTE DISCHARGE LINE TO THE COOLING TOWER BLOWDOWN LINE WAS NOT DESIGNED TO CLOSE THE VALVES ON RADWASTE MONITOR DOWNSCALE FAILURE. THIS DESIGN CAPABILITY WAS ADDED TO THE VALVE ISOLATION LOGIC. ON JANUARY 28, 1988 AT 0825 HOURS THE PLANT WAS IN OPERATIONAL CONDITION 1 (POWER OPERATION) AT 100% POWER GENERATING 1106 MWE. DURING THE IMPLEMENTATION OF CORRECTIVE ACTIONS FROM THE JANUARY 6, 1987 EVENT, INVESTIGATION UNCOVERED SURVEILLANCE PROCEDURES FOR THE NORTH AND SOURTH PLANT VENTS AND FRVS GASEOUS RADIATION MONITORS WHICH DID NOT VERIFY CONTROL ROOM ANNUNCIATION WHEN A DOWNSCALE MONITOR FAILURE WAS SIMULATED. THE GASEOUS MONITOR DOWNSCALE SURVEILLANCES WERE DEEMED TO HAVE NEVER BEEN PERFORMED AND THE NORTH PLANT VENT, SOUTH PLANT VENT AND FRVS WERE DECLARED INOPERABLE.

[95] HUMBOLDT BAY DOCKET 50-133 LER 88-001
ACTUATION OF THE GAS TREATMENT SYSTEM DUE TO INADVERTENT DE-ENERGIZING OF AN ISOLATION MONITOR.
EVENT DATE: 013088 REPORT DATE: 022488 NSSS: GE TYPE: BWR

(NSIC 208313) ON JANUARY 30, 1988, AT 2010 PST, WHILE THE UNIT WAS IN MODE N (SHUTDOWN) IN THE PROCESS OF BEING DECOMMISSIONED, THE GAS TREATMENT SYSTEM WAS ACTUATED WHEN AN OPERATOR, CLEANING THE CONTROL BOARD, INADVERTENTLY DE-ENERGIZED

ONE OF THE REFUELING BUILDING ISOLATION MONITORS. NO ABNORMAL PLANT RADIATION LEVELS EXISTED AT THE TIME OF THE ACTUATION. OPERATING PERSONNEL VERIFIED THAT THE PLANT CONDITIONS WERE NORMAL AND RESTORED THE GAS TREATMENT SYSTEM TO THE STANDBY MODE. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR IN THAT THE OPERATOR DID NOT EXERCISE PROPER CARE WHILE CLEANING THE CONTROL BOARD. CORRECTIVE ACTION TO PREVENT RECURRENCE INCLUDED COUNSELING THE OPERATOR. ALSO, A PROTECTIVE GUARD WILL BE PLACED AROUND THE TOGGLE SWITCH.

[96] INDIAN POINT 2 DOCKET 50-247 LER 87-020
ENVIRONMENTAL QUALIFICATION OF RESISTANCE TEMPERATURE DETECTORS.
EVENT DATE: 123187 REPORT DATE: 013088 NSSS: WE TYPE: PWR
VENDOR: RDF CORP.

(NSIC 208059) DURING THE 1987 REFUELING OUTAGE, IN REPLACING A FAILED WIDE RANGE RESISTANCE TEMPERATURE DETECTOR (RTD), STATION PERSONEL COULD NOTE COMPLETE THE HOOK UP TO THE TERMINAL BLOCK DUE TO INSUFFICIENT LEAD LENGTH SUPPLIED WITH THE NEW RTD. INVESTIGATION INTO EXTENDING THE RTD LEADS TO MAKE THE HOOKUP REVEALED THAT THE MANUFACTURER'S (RDF) VAPOR TIGHT REQUIREMENT FOR THESE LEADS WAS NOT NET. ON DECEMBER 31, 1987 THE RESULTS OF AN ENGINEERING REVIEW OF THE RDF WIDE RANGE REACTOR COOLANT SYSTEM (RCS) HOT LEG AND COLD LEG RTD'S INDICAED THAT NON VAPOR TIGHT LEADS COULD HAVE COMPROMISED THEIR ENVIRONMENTAL QUALIFICATION DURING A HIGH ENERGY LINE BREAK (HELB) INSIDE CONTAINMENT. IN PARALLEL WITH THE ENGINEERING EVALUATION, THE EXISTING RTD LEAD WIRES WERE SPLICED TO QUALIFIED EXTENSION WIRING AND ENCLOSED IN A CONDULET CONNECTED TO THE TERMINAL BOX. THESE SPLICES WERE THEN TOTALLY ENCAPSULATED IN ENVIRONMENTALLY QUALIFIED RTV-7403 SEALANT WITHIN THE CONDULET, RA ING THE RTD LEADS VAPOR TIGHT AND QUALIFIED FOR SERVICE DURING A HELB. THE PUBLIC HEALTH AND SAFETY WERE NOT AFFECTED.

[97] INDIAN POINT 2 DOCKET 50-247 LER 88-001
REACTOR TRIP DURING VALVE STROKE TEST.
EVENT DATE: 011788 REPORT DATE: 021688 NSSS: WE TYPE: PWR

(NSIC 208317) ON JANUARY 17, 1988, WHILE AT HOT SHUTDOWN, A MAIN STEAM SAFETY VALVE LIFTED DURING THE PERFORMANCE OF AN ATMOSPHERIC RELIEF VALVE STROKE TEST. THE RESULTING COMPARATIVE PRESSURE REDUCTION IN THE AFFECTED STEAM GENERATOR CAUSED A SAFETY INJECTION SIGNAL. THIS SIGNAL INITIATED THE GENERATION OF THE REACTOR TRIP AND CONTAINMENT ISOLATION SIGNALS, AND THE OPENING OF THE TRIP BREAKERS (ALL RODS IN PRIOR TO THE EVENT). SINCE REACTOR COOLANT SYSTEM (RCS) PRESSURE REMAINED ABOVE THE SHUTOFF HEAD FOR SAFETY INJECTION DELIVERY, NO SAFETY INJECTION FLOW WAS PROVIDED TO THE CORE. ALL OTHER SAFETY-RELATED EQUIPMENT OPERATED AS REQUIRED. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED. THE SETPOINTS OF THE SAFETY VALVES WERE SUBSEQUENTLY TESTED AND FOUND TO BE WITHIN SPECIFICATIONS. AS A PRECAUTION, IT WAS DECIDED THAT FULL STROKE TESTING OF ATMOSPHERIC RELIEF VALVES SHOULD NOT BE UNDERTAKEN AT HOT SHUTDOWN CONDITIONS WITHOUT MANUALLY ISOLATING THE ATMOSPHERIC RELIEF VALVES.

[98] INDIAN POINT 2 DOCKET 50-247 LER 88-002
REACTOR TRIP ON INTERMEDIATE RANGE HIGH FLUX.
EVENT DATE: 012588 REPORT DATE: 022488 NSSS: WE TYPE: PWR

(NSIC 208402) ON JANUARY 25, 1988 DURING PLANT START-UP OPERATIONS, A REACTOR TRIP OCCURRED WHEN THE INTERMEDIATE RANGE HIGH FLUX TRIP SETPOINT WAS EXCEEDED SHORTLY AFTER SYNCHRONIZING THE TURBINE-GENERATOR TO THE POWER GRID. A HIGH RATE OF INCREASE IN TURBINE POWER (STEAM DEMAND) AFTER SYNCHRONIZATION INDUCED A REACTOR COOLANT SYSTEM COOLDOWN AND PRESSURE DECREASE. THE OPERATOR RESPONDED BY MANUALLY WITHDRAWING CONTROL RODS TO RAISE REACTOR POWER AND THEREBY INCREASE REACTOR COOLANT TEMPERATURE AND PRESSURE. THE REACTOR TRIPPED AT APPROXIMATELY 15% POWER VIA ONE OF TWO INTERMEDIATE RANGE NUCLEAR POWER CHANNELS (N 36). POST

TRIP TESTING REVEALED THAT CHANNEL N 36 TRIP SETPOINT WAS CONSERVATIVELY SET AT 15% POWER (NOMINALLY 25%). ALL SYSTEMS OPERATED AS PER DESIGN. THE INTERMEDIATE RANGE HIGH FLUX TRIP SETPOINT FOR CHANNEL N 36 WAS SUBSEQUENTLY READJUSTED CLOSER TO BUT LESS THAN 25% REACTOR POWER. THE OPERATING PERSONNEL WERE READVISED CONCERNING THE RATE OF STEAM DEMAND INCREASE DURING PLANT START-UPS. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED.

[99] KEWAUNEE DOCKET 50-305 LER 87-010 REV 01
UPDATE ON VIOLATION OF TECHNICAL SPECIFICATION ON CONTAINMENT INTEGRITY DUE TO UNCLEAR TAGOUTS.
EVENT DATE: 090487 REPORT DATE: 022688 NSSS: WE TYPE: PWR

(NSIC 208306) ON 9/4/87, AT 0226 CDT WITH THE PLANT AT 100% POWER, TECH SPECS PERTAINING TO CONTAINMENT INTEGRITY PROVISIONS WERE VIOLATED. THE REDUNDANT CONTAINMENT ISOLATION (CI) SUMP A DISCHARGE CONTROL VALVES (MD(R)-134 AND MD(R)-135) WERE OPENED FOR 2.9 MINUTES WHILE VALVE MD(R)-134 WAS CONSIDERED INOPERABLE. THE REACTOR OPERATOR OPENED BOTH VALVES IN RESPONSE TO A HIGH CONTAINMENT SUMP LEVEL ALARM PER OPERATING PROCEDURE A-MDS-30. VALVE MD(R)-134 WAS ADMINISTRATIVELY INOPERABLE BECAUSE IT HAD NOT BEEN COMPLETELY RETESTED FOLLOWING REPLACEMENT OF ITS ASSOCIATED SOLENOID VALVE. AFTER SATISFACTORILY COMPLETING THE RETEST REQUIREMENTS WITHOUT ANY FURTHER ADJUSTMENTS, THE VALVE WAS DECLARED OPERABLE AT 1210 THE SAME DAY. THIS EVENT OCCURRED BECAUSE CLEAR INSTRUCTIONS ADDRESSING THE VALVE'S INOPERABLE STATUS AND THE REASONS FOR THE INOPERABLE STATUS WERE NOT INCLUDED ON THE DANGER TAG. IN ADDITION DESIGN CHANGE PROCEDURE 1544-15 FAILED TO ADEQUATELY IDENTIFY VALVES MD(R)-134 AND MD(R)-135 AS REDUNDANT CONTAINMENT ISOLATION VALVES WHICH HAVE OPERABILITY REQUIREMENTS DEFINED IN THE TECH SPECS. CLOSURE OF VALVE MD(R)-135 TO ALLOW RETEST CYCLING OF VALVE MD(R)-134 WAS NOT COVERED BY THE PROCEDURE. IMMEDIATE CORRECTIVE ACTIONS INCLUDED AN INFORMAL REVIEW WITH THE PERSONNEL INVOLVED.

[100] KEWAUNEE DOCKET 50-305 LER 87-012 REV 01
UPDATE ON POTENTIAL FOR CONTROL VALVE FAILURE IN THE NON-SAFE MODE DUE TO OVERPRESSURIZED SOLENOID VALVES.
EVENT DATE: 112887 REPORT DATE: 030488 NSSS: WE TYPE: PWR
VENDOR: ASCO VALVES

(NSIC 208397) ON 11/28/87, WITH THE PLANT AT FULL POWER, TWO INDEPENDENT, CONTAINMENT ISOLATION CONTROL VALVES FAILED TO CLOSE FROM THE CONTROL ROOM DURING PERFORMANCE OF THEIR INSERVICE TESTING (IST). THE TWO VALVES, MU-1010-1 AND RC-507, FAILED TO CLOSE BECAUSE THEIR RESPECTIVE SOLENOID VALVES WERE UNABLE TO BLOCK THE INSTRUMENT AIR FLOW TO THE CONTROL VALVES WHEN DEENERGIZED. THE SOLENOID VALVES, WHICH HAVE A MAXIMUM OPERATING PRESSURE DIFFERENTIAL (MOPD) OF 70 PSI, WERE FOUND TO HAVE THEIR CORRESPONDING AIR LINE REGULATORS SET AT 80 PSI. THIS RESULTED IN THE SOLENOID VALVES BEING OVERPRESSURIZED. THESE SOLENOID VALVES HAD BEEN REPLACED AS A PART OF A RECENT DESIGN CHANGE DURING WHICH TIME THE INCORRECT MODELS, WITH TOO LOW AN MOPD, WERE PURCHASED AND INSTALLED. A FIELD WALKDOWN OF OTHER SAFETY RELATED SOLENOID VALVES IN THE PLANT AND THEIR AIR REGULATOR SETTINGS WAS PERFORMED. NO OTHER CASES WERE FOUND WHERE THE AIR REGULATOR SETTING WOULD HAVE PREVENTED THE SOLENOID FROM OPERATING PROPERLY. HOWEVER, OTHER CASES WERE FOUND WHERE THE SOLENOID VALVES HAD AN MOPD LESS THAN INSTRUMENT AIR PRESSURE. PLANNED CORRECTIVE ACTIONS INCLUDE REPLACING SOLENOID VALVES AND ESTABLISHING A PROGRAM TO FORMALIZE INSTRUMENT AIR REGULATOR SETTINGS. THIS EVENT IS BEING REPORTED IN THE "OTHER" CATEGORY BECAUSE OF ITS POTENTIAL INTEREST TO OTHER LICENSEES.

[101] LA SALLE 1 DOCKET 50-373 LER 87-033 REV 01
UPDATE ON FAILURE OF DIESEL GENERATOR TO CLOSE ONTO 142Y DURING SURVEILLANCE TESTING.
EVENT DATE: 091787 REPORT DATE: 012588 NSSS: GE TYPE: BWR
VENDOR: POTTER & BRUMFIELD

(NSIC 208065) ON SEPTEMBER 17, 1987 AT APPROXIMATELY 1115 HOURS, WITH UNIT 1 SHUTDOWN, LASALLE OPERATING SURVEILLANCE LOS-DG-M2 ("1A" DIESEL GENERATOR OPERABILITY TEST) WAS BEING PERFORMED. THE UNIT 1 OPERATOR (LICENSED REACTOR OPERATOR) HAD STARTED THE "1A" DIESEL GENERATOR (DG) AND ATTEMPTED TO CLOSE THE OUTPUT BREAKER AND SYNCHRONIZE TO BUS 142Y. HOWEVER, THE OUTPUT BREAKER WOULD NOT CLOSE. SEVERAL ATTEMPTS WERE MADE AND ALL WERE UNSUCCESSFUL. THE "1A" DG WAS THEN SHUT DOWN AND DECLARED OPERABLE. TROUBLESHOOTING EFFORTS ON THE OUTPUT BREAKER CLOSING CIRCUITRY REVEALED NO DISCREPANCIES. ALL BREAKER COMPONENTS, INCLUDING ASSOCIATED CLOSURE PERMISSIVE CONTACTS, WERE VERIFIED TO OPERATE AS DESIGNED FOLLOWING THE EVENT. LOS-DG-M2 WAS THEN PERFORMED SUCCESSFULLY AND THE "1A" DG WAS DECLARED OPERABLE AT 1645 HOURS ON SEPTEMBER 17, 1987. THIS EVENT COULD NOT BE REPEATED UNDER TEST CONDITIONS, HOWEVER, A SIMILAR EVENT OCCURRED ON DECEMBER 18, 1987. THE CAUSE OF THE DECEMBER EVENT WAS THE FAILURE OF A CONTACT ON THE 27X RELAY, WHICH PROVIDES A PERMISSIVE IN THE "1A" DG OUTPUT BREAKER CLOSING CIRCUITRY. SINCE THE 27X RELAY CONTACT FAILURE WAS OF AN INTERMITTENT NATURE, IT IS BELIEVED THAT THIS WAS THE CAUSE OF THIS EVENT (SEPTEMBER 17, 1987).

[102] LA SALLE 2 DOCKET 50-374 LER 87-016 REV 01
UPDATE ON DEFECTIVE LOW PRESSURE CORE SPRAY MINIMUM FLOW SWITCH.
EVENT DATE: 071387 REPORT DATE: 012788 NSSS: GE TYPE: BWR
VENDOR: STATIC-O-RING

(NSIC 208035) LOW PRESSURE CORE SPRAY (LPCS) MINIMUM FLOW SWITCH, FS-2E21-N004, WAS FOUND TO BE LEAKING WATER THROUGH THE SWITCH INTERNALS DURING THE PERFORMANCE OF INSTRUMENT SURVEILLANCE LIS-LP-202, "UNIT 2 LPCS MINIMUM FLOW BYPASS CALIBRATION," ON JULY 13, 1987, AT 1045 HOURS. AT THE TIME OF THIS EVENT, UNIT 2 WAS IN OPERATIONAL CONDITION 1 (RUN) AT 95% RATED POWER. THE LEAKAGE WAS VERY SMALL AND DID NOT AFFECT THE OPERATION OF THE SWITCH AS DEMONSTRATED DURING THE CALIBRATION. A WORK REQUEST WAS WRITTEN TO REPLACE THE DEFECTIVE SWITCH WITH ANOTHER ONE, LIKE-FOR-LIKE. THE REPLACEMENT SWITCH WAS INSTALLED, CALIBRATED AND PLACED IN SERVICE BY 2300 HOURS ON JULY 15, 1987. THE LEAKING INSTRUMENT HAS BEEN DISASSEMBLED AND INSPECTED BY SOR, INC. AND COMMONWEALTH EDISON'S SYSTEM MATERIALS ANALYSIS DIVISION. THESE INSPECTIONS REVEALED TWO SMALL SLITS IN THE SWITCH'S DIAPHRAGM. THE CAUSE OF THESE DEFECTS COULD NOT BE DETERMINED. THIS EVENT IS REPORTED TO THE NUCLEAR REGULATORY COMMISSION AS A VOLUNTARY LICENSEE EVENT REPORT IN ACCORDANCE WITH THE REQUIREMENTS OF IE BULLETIN 86-02, "STATIC-O-RING DIFFERENTIAL PRESSURE SWITCHES."

[103] LACROSSE DOCKET 50-409 LER 88-001
CONTAINMENT BUILDING VENTILATION ISOLATION DUE TO CONTAINMENT BUILDING DELAYED PARTICULATE MONITOR.
EVENT DATE: 012988 REPORT DATE: 021688 NSSS: AC TYPE: BWR

(NSIC 208246) CONTAINMENT VENTILATION VALVES CLOSED AUTOMATICALLY DUE TO A HIGH ACTIVITY ALARM ON THE CONTAINMENT BUILDING DELAYED PARTICULATE MONITOR. THE OPERATOR IMMEDIATELY CHECKED OTHER MONITORS TO ENSURE THERE WAS NO OTHER INDICATION OF AN ACTUAL INCREASE IN ACTIVITY. ALL INDICATIONS WERE NORMAL AND THE DELAYED PARTICULATE MONITOR HAD RETURNED TO NORMAL. WELDING OPERATIONS WERE IN PROGRESS AT THE TIME IN THE MACHINE SHOP. CONTAINMENT BUILDING VENTILATION ISOLATION HAS OCCURRED BEFORE DURING WELDING OPERATION.

[104] LACROSSE DOCKET 50-409 LER 88-002
TEMPORARY ABSENCE OF DUTY SHIFT SUPERVISOR FROM SITE.
EVENT DATE: 012988 REPORT DATE: 021888 NSSS: AC TYPE: BWR

(NSIC 208349) THE SHIFT SUPERVISOR ON DUTY THE MORNING OF 01/29/88 WAS TEMPORARILY ASSUMING THE DUTY FOR THE REGULARLY SCHEDULED RELIEF SHIFT SUPERVISOR. THE RELIEF SUPERVISOR HAD CALLED AND ASKED TO BE RELIEVED OF HIS DUTIES FROM 0800 UNTIL 1200 BECAUSE HE ALSO HAD TO RELIEVE FOR THE 1600-2400 SHIFT AND WANTED TO KEEP HIS CONSECUTIVE HOURS AT A MAXIMUM OF 12 HOURS IN KEEPING WITH DAIRYLAND'S POLICY OF MINIMIZING THE TOTAL NUMBER OF OVERTIME HOURS PER DAY FOR NUCLEAR PLANT WORKERS. AT 1205 THE DUTY SHIFT SUPERVISOR, WHO HAD BEEN WORKING ON LICENSING ITEMS WITH OTHER LACBWR STAFF MEMBERS THROUGHOUT THE MORNING, ACCOMPANIED THEM TO LUNCH IN NEARBY GENOA (APPROXIMATELY 1 MILE FROM THE SITE) WITHOUT HAVING BEEN RELIEVED OF HIS DUTIES. AT 1225 THE REGULARLY SCHEDULED SHIFT SUPERVISOR ARRIVED ONSITE AND, UPON REACHING THE CONTROL ROOM, REALIZED THE DUTY SUPERVISOR WAS ABSENT AND ASSUMED THE DUTY HIMSELF. IN THIS REGARD, A DUTY SHIFT SUPERVISOR WAS NOT PRESENT ONSITE FOR ABOUT 20 MINUTES. THE ORIGINAL SHIFT SUPERVISOR ARRIVED BACK ONSITE AT 1250, AND A PROPER SHIFT TURNOVER WAS CONDUCTED.

[105] LIMERICK 1 DOCKET 50-352 LER 87-023 REV 02
UPDATE ON ENGINEERED SAFETY FEATURE ACTUATION DUE TO BATTERY CHARGER FAILURE.
EVENT DATE: 061187 REPORT DATE: 022988 NSSS: GE TYPE: BWR
VENDOR: BROWN BOVERI
C & D BATTERIES, DIV OF ELTRA CORP.

(NSIC 208438) ON JUNE 11, 1987, THE STANDBY GAS TREATMENT AND REACTOR ENCLOSURE RECIRCULATION SYSTEMS (ENGINEERED SAFETY FEATURES) INITIATED AS A CONSEQUENCE OF ACTIONS TAKEN DUE TO FAILURE OF THE 1A1D103 STATION BATTERY CHARGER. THE 125 VDC STATION BATTERIES (1A1) WERE DISCONNECTED FROM THE BUS AT THE TIME OF THE EVENT TO ACCOMMODATE MAINTENANCE WORK. THE BATTERY CHARGER FAILURE IS BELIEVED TO BE A RESULT OF AN INTEGRATED CIRCUIT CONTROLLER CARD FAILURE WHICH RESULTED IN DC VOLTAGE FLUCTUATION. HOWEVER, WHEN THE CARD MANUFACTURER PERFORMED A FAILURE ANALYSIS, NO DEFECT COULD BE FOUND. A TEMPORARY CIRCUIT ALTERATION (TCA) WAS INSTALLED TO PROVIDE AN ALTERNATE POWER SUPPLY TO THE DE-ENERGIZED BUS. DURING REENERGIZATION OF THE BUS, A REACTOR PROTECTION SYSTEM SERIES BREAKER TRIPPED DUE TO A SPURIOUS UNDERVOLTAGE RELAY TRIP SIGNAL AND CAUSED THE INBOARD INSTRUMENT GAS VALVE TO CLOSE. BROWN BOVERI, THE UNDERVOLTAGE RELAY MANUFACTURER, HAS FILED A PART 21 REPORT REGARDING THE RELAY FALSE ACTUATION. THE AFFECTED UNDERVOLTAGE RELAYS CURRENTLY INSTALLED IN UNIT 1 WILL BE REPLACED WITH RELAYS MODIFIED BY BROWN BOVERI DURING THE NEXT OUTAGE OF SUFFICIENT DURATION. THE CONSEQUENCES OF THIS EVENT WERE MINIMAL BECAUSE THE ENGINEERED SAFETY FEATURES INITIATED AS DESIGNED AND THE UNIT WAS SHUTDOWN WITH THE CORE OFFLOADED AT THE TIME OF THE EVENT.

[106] LIMERICK 1 DOCKET 50-352 LER 88-001
TECHNICAL SPECIFICATION REQUIREMENTS MISSED DUE TO PERSONNEL ERROR.
EVENT DATE: 010588 REPORT DATE: 020888 NSSS: GE TYPE: BWR

(NSIC 208267) ON JANUARY 6, 1988, THE REQUIREMENT OF TECHNICAL SPECIFICATION 3.3.2.B WAS NOT MET. DURING A REVIEW OF SURVEILLANCE TEST ST-6-107-590-1 "DAILY SURVEILLANCE LOG" IT WAS NOTICED THAT A REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM PIPE-ROUTING AREA TEMPERATURE READING WAS DEVIATING FROM PREVIOUS SHIFT READINGS. SUBSEQUENT INVESTIGATION REVEALED THAT DURING THE PERFORMANCE OF AN ST THE THERMOCOUPLE LEADS WERE INADVERTENTLY REVERSED. THIS CONDITION WENT UNDETECTED DURING THE PERFORMANCE OF THE INDEPENDENT VERIFICATION OF RESTORATION PORTION OF THE ST. ADDITIONALLY, OPERATIONS PERSONNEL RECOGNIZED THE ABNORMALLY LOW TEMPERATURE INDICATION BUT ATTRIBUTED THE READING TO EXTREME COLD WEATHER. THERE WERE NO ADVERSE CONSEQUENCES AND NO RELEASE OF RADIOACTIVE MATERIAL AS A

RESULT OF THIS EVENT. HAD A STEAM LEAK OCCURRED IN THE RCIC PIPE-ROUTING AREA WITH THE SUBJECT THERMOCOUPLE LEADS REVERSED THE ISOLATION LOGIC WOULD STILL SERVE ITS DESIGN FUNCTION DUE TO THE PRESENCE OF AN ALTERNATE TRIP SYSTEM CONTAINING A REDUNDANT THERMOCOUPLE. THE THERMOCOUPLE LEADS WERE PROPERLY RELANDED AT 1600 HOURS ON JANUARY 6, 1988 AND INDICATION WAS VERIFIED TO REFLECT CURRENT PLANT CONDITIONS. THE TECHNICIANS INVOLVED WERE COUNSELED AS TO THEIR ERROR AND DISCUSSION OF THIS EVENT WILL BE INCLUDED IN ANNUAL TECHNICIAN TRAINING.

[107] MAINE YANKEE DOCKET 50-309 LER 88-001
PLANT TRIP ON HEATER DRAIN TANK LEVEL SWITCH FAILURE.
EVENT DATE: 010588 REPORT DATE: 020588 NSSS: CE TYPE: PWR
VENDOR: JO-BELL PRODUCTS

(NSIC 208225) ON JANUARY 5, 1988, AN UNSCHEDULED REACTOR TRIP OCCURRED ON LOSS OF LOAD DUE TO A MAIN TURBINE TRIP. THE MAIN TURBINE TRIPPED AUTOMATICALLY WHEN THE TURBINE DRIVEN MAIN FEEDWATER PUMP TRIPPED ON LOW SUCTION PRESSURE. THE LOW TURBINE-DRIVEN FEEDWATER PUMP SUCTION PRESSURE WAS THE RESULT OF LOSS OF FLOW FROM THE HEATER DRAIN PUMPS. THE OPERATING HEATER DRAIN PUMP WAS TRIPPED AND ITS STANDBY PUMP WAS PREVENTED FROM STARTING BECAUSE THE HEATER DRAIN TANK LOW LEVEL SWITCH FAILED. THE LOW LEVEL SWITCH FAILURE WAS DUE TO CORROSION OF THE SWITCH FLOAT, WHICH LEAKED AND SANK, LOCKING IN A LOW HEATER DRAIN TANK LEVEL SIGNAL. THE LOW LEVEL SWITCH AND FLOAT WAS REPLACED ON JANUARY 5, 1988.

[108] MAINE YANKEE DOCKET 50-309 LER 88-002
UNLOCKED EMERGENCY CORE COOLING SYSTEM (ECCS) VALVES.
EVENT DATE: 011588 REPORT DATE: 021888 NSSS: CE TYPE: PWR

(NSIC 208293) DURING A REVIEW OF THE EMERGENCY CORE COOLING SYSTEM (ECCS) SURVEILLANCE PROCEDURE, IT WAS DETERMINED THAT SIXTEEN VALVES WERE NOT PROPERLY CONTROLLED AS REQUIRED BY THE PLANT'S TECH SPEC. FOR OPERATION AT POWER, ECCS VALVES THAT AFFECT OPERABILITY MUST BE ALIGNED AND LOCKED IN THE POSITION REQUIRED FOR PROPER SAFEGUARDS OPERATION. ALL THESE VALVES WERE ALIGNED AND CONTROLLED BY PROCEDURE AS REQUIRED BUT WERE NOT LOCKED. THE VALVES WERE LOCKED. AS A RESULT OF THE INCIDENT, SOME LOCKED VALVE POLICY INCONSISTENCIES AND CONCERNS WERE IDENTIFIED. THE LICENSEE WILL REVIEW THE POLICY TO ENSURE THAT IT DOES NOT UNNECESSARILY REDUCE OPERATIONAL FLEXIBILITY AND THAT IT IS CONSISTENTLY APPLIED IN THE PROCEDURES. A TECH SPEC CHANGE IS BEING CONSIDERED TO PROVIDE A LOCKED VALVE POLICY THAT MINIMIZES IMPACT ON ECCS OPERATIONAL FLEXIBILITY.

[109] MCGUIRE 1 DOCKET 50-369 LER 87-020 REV 01
UPDATE ON RESIDUAL HEAT REMOVAL PUMP INOPERABLE DUE TO THE FLOW INSTRUMENT CONTROLLING THE PUMP RECIRC VALVE LEFT ISOLATED DUE TO PERSONNEL ERROR.
EVENT DATE: 090587 REPORT DATE: 102787 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)
VENDOR: INGERSOLL-RAND CO.

(NSIC 208256) ON 08/19/87 A WORK REQUEST (WR) WAS ISSUED TO CALIBRATE FLOW INSTRUMENTS 1MNDPG5050 AND 1MNDPG5051. ON 08/28/87 OPERATIONS (OPS) EVALUATED THE WR AND AUTHORIZED THE WORK TO BEGIN. THAT EVENING INSTRUMENT AND ELECTRICAL (IAE) COMPLETED THE CALIBRATION WORK BUT FAILED TO UNISOLATE THE FLOW INSTRUMENT 1MNDPG5050 (USED TO OPERATE THE RESIDUAL HEAT REMOVAL (ND) PUMP 1B RECIRC VALVE). AT 0005 ON 09/05/87, OPS STARTED ND PUMP 1B AND THE PUMP RECIRC VALVE FAILED TO OPEN. OPS DECLARED ND TRAIN 1B INOPERABLE AND INITIATED A WORK REQUEST. IAE DISCOVERED THE ISOLATED FLOW INSTRUMENT AT 0146 AND UNISOLATED THE VALVE. OPS FUNCTIONALLY VERIFIED ND TRAIN 1B OPERABLE AND DECLARED IT OPERABLE AT 0230. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR WITH CONTRIBUTORY MANAGEMENT DEFICIENCY. THE STANDING WORK REQUESTS FOR THE FLOW INSTRUMENTS WERE REVISED WITH PRECAUTIONS THAT THE ITEM IS RELATED TO TECH SPECS. THE EVENT WILL BE COVERED DURING REQUAL

TRAINING WITH ALL OPS SHIFT PERSONNEL. THIS EVENT WILL BE EVALUATED TO DETERMINE IF THERE ARE GENERIC IMPLICATIONS. STATION DIRECTIVE 4.2.1 WILL BE REVISED REGARDING USE OF PROCEDURES.

[110] MCGUIRE 1 DOCKET 50-369 LER 88-001
REACTOR TRIP/TURBINE TRIP DUE TO A MALFUNCTION IN THE EXCITATION SWITCHGEAR SILICON CONTROLLED RECTIFIER FIRING CIRCUIT.
EVENT DATE: 010788 REPORT DATE: 020888 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 208198) ON 01/07/88 AT APPROXIMATELY 0700, THE UNIT 1 GENERATOR STARTED TO EXPERIENCE AN ABNORMAL VOLTAGE AND REACTIVE LOAD TRANSIENT. OPERATIONS ATTEMPTED TO COMBAT THE TRANSIENT BY MANUALLY ADJUSTING THE EXCITATION SWITCHGEAR OUTPUT. THE TRANSIENT BECAME TOO SEVERE, AND AT 0708, THE GENERATOR AND TURBINE TRIPPED BECAUSE OF A LOSS OF EXCITATION TO THE GENERATOR FIELD. THE TURBINE TRIP CAUSED A REACTOR TRIP. THE ABNORMAL VOLTAGE AND REACTIVE LOAD TRANSIENT WAS CAUSED BY A MALFUNCTIONING FIRING CIRCUIT IN THE EXCITATION SWITCHGEAR. THIS FIRING CIRCUIT CONTROLS THE AMOUNT OF EXCITATION APPLIED TO THE GENERATOR TO CONTROL VOLTAGE AND REACTIVE LOAD. UNIT 1 STABILIZED FROM THE REACTOR TRIP BY APPROXIMATELY 0740. THE TRANSMISSION DEPARTMENT REPAIRED THE FIRING CIRCUIT PORTION OF THE EXCITATION SWITCHGEAR AND UNIT 1 WAS RETURNED TO POWER OPERATION ON 01/07/88 AT 2253. THIS EVENT IS ASSIGNED A CAUSE OF OTHER BECAUSE THE TRANSIENT/TRIP WAS CAUSED BY A MALFUNCTION IN THE EXCITATION SWITCHGEAR SILICON CONTROLLED RECTIFIER (SCR) FIRING CIRCUIT. THE FAILURE MODE OF THE PHASE SHIFTER CIRCUITS WILL BE EVALUATED AND FILTERED COOLING TO THE EXCITATION SWITCHGEAR WILL BE REVIEWED.

[111] MCGUIRE 1 DOCKET 50-369 LER 88-002
A HANGER CLAMP INSTALLED ON CONTAINMENT ISOLATION VALVE IN AN UNACCEPTABLE LOCATION DUE TO APPARENT PERSONNEL ERROR.
EVENT DATE: 011188 REPORT DATE: 021588 NSSS: WE TYPE: PWR
VENDOR: KEROTEST MANUFACTURING CORP.
ROTORK INC.

(NSIC 208342) ON 01/11/88, DUKE DESIGN ENGINEERING (DE) DETERMINED THAT THE LOCATION OF A CLAMP ATTACHING A HANGER TO VALVE 1NM-217B, STEAM GENERATOR D UPPER SHELL SAMPLE CONTAINMENT INSIDE ISOLATION, WAS UNACCEPTABLE BECAUSE THE CLAMP COULD POTENTIALLY DAMAGE THE VALVE OPERATOR IN A SEISMIC EVENT. DE CONTACTED THE MANUFACTURER OF THE VALVE OPERATOR WHO CONFIRMED THE LOCATION OF THE CLAMP TO BE UNSATISFACTORY ON 01/11/88. DE THEN CONTACTED OPERATIONS WHO DECLARED VALVE 1NM-217B INOPERABLE AT 1050. THE VALVE WAS DEENERGIZED IN THE SAFETY (CLOSED) POSITION AT 1155 TO COMPLY WITH TECH SPECS. THIS EVENT WAS DUE TO PERSONNEL ERROR BECAUSE THE LOCATION OF THE CLAMP ATTACHING THE HANGER TO VALVE INM-217B WAS APPARENTLY CHANGED WITHOUT APPROPRIATE APPROVAL OR DOCUMENTATION BETWEEN THE ORIGINAL HANGER INSTALLATION IN 12/80, AND 8/83. THE HANGER REMOVAL AND REPLACEMENT PROCEDURE WILL BE REVISED TO REQUIRE USE OF APPLICABLE DRAWINGS TO VERIFY PROPER HANGER INSTALLATION. A MODIFICATION WILL BE INITIATED TO REPOSITION THE CLAMP AFTER DE DETERMINED THE APPROPRIATE CLAMP LOCATION. DESIGN DRAWINGS OF VALVES WITH ROTORK OPERATORS WILL BE REVIEWED TO IDENTIFY THOSE WITH HANGERS ATTACHED TO VALVE OPERATOR EXTENSIONS. ALSO, DURING SNUBBER INSPECTIONS, MCGUIRE WILL PHYSICALLY INSPECT FOR VALVE OPERATOR EXTENSIONS WITH HANGERS OF SNUBBERS ATTACHED.

[112] MCGUIRE 2 DOCKET 50-370 LER 87-018 REV 01
UPDATE ON INOPERABLE FIRE BARRIER DUE TO A WALL SECTION BEING CONSTRUCTED WITHOUT PROPER END CONNECTION TREATMENT.
EVENT DATE: 092187 REPORT DATE: 021588 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 1 (PWR)

(NSIC 208257) ON 9/21/87, A QUALITY ASSURANCE (QA) INSPECTOR DISCOVERED SMALL GAPS BETWEEN CONCRETE COLUMNS AND THE ADJOINING WALL SECTION BETWEEN TWO ROOMS ON THE 733' ELEVATION OF THE AUXILIARY BUILDING. THE FIRE BARRIER WAS DECLARED INOPERABLE AT 1430 AND A FIRE WATCH WAS INITIATED. ON 9/23/87 THE FIRE BARRIER WAS REPAIRED IN ACCORDANCE WITH FIELD SEALANT SPECIFICATION 7005 AND WAS DECLARED OPERABLE AT 1230. THE CAUSE OF THE EVENT WAS CONSTRUCTION/INSTALLATION DEFICIENCY BECAUSE THE FIRE BARRIER SECTION WAS CONSTRUCTED WITHOUT PROPER END CONNECTION TREATMENT. A CONTRIBUTORY CAUSE OF QA DEFICIENCY HAS BEEN ASSIGNED BECAUSE OF QA FAILURE TO VERIFY THAT THE WALL SECTION WAS CONSTRUCTED ACCORDING TO APPLICABLE DRAWINGS. THE QA INSPECTION PROCEDURE USED DURING THE CONSTRUCTION (PRE-STARTUP) HAS BEEN INACTIVE SINCE APRIL 1987. THE FIRE BARRIER INSPECTION PROCEDURE WILL BE REVISED TO ENSURE THAT THE ENTIRE FIRE BARRIER IS INSPECTED (NOT JUST PENETRATIONS). THE QA CONDITION 3 FIRE WALL REPAIR PROCEDURE WILL BE REVISED TO INCLUDE SPECIFIC PROVISIONS FOR REPAIR OF GAPS BETWEEN GYPSUM DRYWALL AND CONCRETE. A COMPLETE INSPECTION OF GYPSUM DRYWALL FIRE BARRIERS WILL BE CONDUCTED TO VERIFY THEIR INTEGRITY AND/OR ANY REMAINING CONSTRUCTION DEFECTS. MAINTENANCE PROCEDURES WILL ALSO BE INCLUDED IN ANY FUTURE QA INSPECTIONS OF FIRE BARRIER WALLS FOR PROPER ACCEPTANCE CRITERIA.

[113] MCGUIRE 2 DOCKET 50-370 LER 88-001
MANUAL TURBINE TRIP FOLLOWED BY REACTOR TRIP DUE TO DECREASING STEAM GENERATOR LEVEL AND FEEDWATER CONTROL VALVE CLOSED.
EVENT DATE: 011288 REPORT DATE: 021188 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 1 (PWR)
VENDOR: NORGREN

(NSIC 208269) ON 01/12/88 AT 0420, A UNIT 2 STEAM GENERATOR (S/G) C LEVEL DEVIATION ALARM WAS RECEIVED IN THE CONTROL ROOM. OPERATIONS (OPS) PERSONNEL NOTICED MAIN FEEDWATER (CF) FLOW TO S/G C WAS DECREASING WHILE CF FLOW CONTROL TO S/G C, VALVE 2CF-20, HAD AN OPEN DEMAND SIGNAL. OPS REDUCED TURBINE GENERATOR LOAD AND OPERATED BOTH CF PUMPS IN MANUAL. S/G 2C LEVEL RETURNED TO NORMAL. SHORTLY AFTER, THE LEVEL IN S/G 2C BEGAN TO DECREASE AND VALVE 2CF-20 INDICATED IT WAS CLOSED. AT 0425:01, OPS TRIPPED THE TURBINE AND REACTOR. THE DECREASING LEVEL IN S/G 2C WAS DUE TO VALVE 2CF-20 FAILING IN THE CLOSED POSITION WHICH WAS DUE TO A LOSS OF AIR SUPPLY TO THE VALVE. OPS IMPLEMENTED THE REACTOR TRIP RECOVERY PROCEDURE. THE AIR REGULATORS ON ALL S/G CF FLOW CONTROL VALVES WERE REPLACED. UNIT 2 WAS RETURNED TO POWER OPERATION ON 01/13/88 AT 0211. THIS EVENT IS ASSIGNED A CAUSE OF DESIGN AND CONSTRUCTION/INSTALLATION DEFICIENCY BECAUSE THE AIR REGULATOR FOR VALVE 2CF-20 FAILED DUE TO IT BEING IMPROPERLY MOUNTAD. THE MOUNTING BRACKET USED WAS DESIGNED FOR A FIELD MOUNT MODEL AND ADAPTED FOR A PANEL MOUNT REGULATOR. THE UNIT 1 REGULATORS WILL BE REPLACED. CONTROLS WILL BE DEVELOPED TO PREVENT PANEL MOUNTED REGULATORS BEING USED IN FIELD APPLICATIONS.

[114] MILLSTONE 1 DOCKET 50-245 LER 87-044
IMPROPER PERFORMANCE OF STANDBY GAS TREATMENT FOR DISTRIBUTION TEST.
EVENT DATE: 122987 REPORT DATE: 012588 NSSS: GE TYPE: BWR

(NSIC 208058) ON DECEMBER 29, 1987, AT 1145 HOURS, WHILE OPERATING AT 100% POWER (529 DEGREES F, 1032 PSIG) IT WAS DETERMINED THAT TECH SPEC REQUIREMENT 4.7.B.1.C IS NOT FULLY SATISFIED DURING FLOW DISTRIBUTION TESTING OF THE STANDBY GAS TREATMENT (SBGT) SYSTEM. THIS IS A ONCE PER CYCLE TEST THAT REQUIRES FLOW TESTS ACROSS BOTH HIGH EFFICIENCY PARTICULATE AIR (HEPA) AND CHARCOAL ABSORBERS. MILLSTONE'S SURVEILLANCE ADDRESSES THE HEPA FILTER ONLY. A REVIEW OF ANSI N510 AND REG. GUIDE 1.52 INDICATES THIS TEST NEED ONLY BE PERFORMED ON SYSTEM INSTALLATION. THIS DEFICIENCY WAS DISCOVERED AS A RESULT OF A SELF INITIATED STUDY. THE SBGT SYSTEM WAS SUCCESSFULLY TESTED AS SPECIFIED IN PREOPERATIONAL TEST NO. C-17 BY THE MANUFACTURER AND AIR FLOW DISTRIBUTION TESTING PERFORMED DURING THIS TEST DID MEET ANSI N510. A PROPOSAL HAS BEEN SUBMITTED TO DELETE

TECH SPEC REQUIREMENT 4.7.B.1.C SINCE ANSI N510 ONLY REQUIRES FLOW TESTS TO BE PERFORMED FOLLOWING INSTALLATION OR MAJOR MODIFICATIONS.

[115] MILLSTONE 1 DOCKET 50-245 LER 88-001
INADVERTENT DISCHARGE OF UNSAMPLED FLOOR DRAIN SAMPLE TANK 'B'.
EVENT DATE: 012088 REPORT DATE: 021988 NSSS: GE TYPE: BWR

(NSIC 208316) ON JANUARY 20, 1988 AT 1800 HOURS, WHILE OPERATING AT 100% POWER (530 DEGREES F, 1030 PSIG) A RADWASTE PLANT EQUIPMENT OPERATOR (PEO) WAS PREPARING TO DISCHARGE THE 'A' FLOOR DRAIN SAMPLE TANK (FDST) TO LONG ISLAND SOUND USING LIQUID DISCHARGE PERMIT 1016. THE OPERATOR INADVERTENTLY OPENED THE DISCHARGE VALVE TO THE UNSAMPLED 'B' FDST, STARTED THE 'B' FDST PUMP AND INITIATED THE DISCHARGE. AT 1805 HOURS THE DISCHARGE WAS TERMINATED WHEN THE EFFLUENT RADIATION MONITOR TRIPPED AND CLOSED THE COMMON DISCHARGE VALVE. THE 'B' FDST WAS SAMPLED AND ANOTHER LIQUID DISCHARGE PERMIT WAS PROCESSED TO DOCUMENT THE INADVERTENT DISCHARGE. THE TOTAL ACTIVITY AND CONCENTRATIONS OF RADIONUCLIDES RELEASED WERE BELOW ALLOWABLE LIMITS.

[116] MILLSTONE 2 DOCKET 50-336 LER 86-010 REV 01
UPDATE ON INCONSISTENCY BETWEEN SAFETY ANALYSIS AND TECHNICAL SPECIFICATIONS.
EVENT DATE: 100686 REPORT DATE: 021088 NSSS: CE TYPE: PWR

(NSIC 208068) WHILE IN MODE 6 ON OCTOBER 6, 1986 AT 0815 AN INVESTIGATION IDENTIFIED AN INCONSISTENCY BETWEEN THE NUMBER OF REACTOR COOLANT PUMPS REQUIRED TO BE OPERATING IN MODES 3, 4 AND 5, AND THE ASSUMPTIONS USED IN THE SAFETY ANALYSIS FOR THE CEA WITHDRAWAL FROM SUBCRITICAL ACCIDENT. THE SAFETY ANALYSIS FOR THE CEA WITHDRAWAL FROM SUBCRITICAL ACCIDENT DOES NOT EXPLICITLY CONSIDER MODE 3, 4 AND 5 EVENTS BASED ON THE ASSUMPTION THAT THE HOT ZERO POWER RESULTS BOUND THESE OPERATING MODES. THEREFORE, THE SAFETY ANALYSIS FOR THIS ACCIDENT ASSUMES THAT ALL FOUR REACTOR COOLANT PUMPS ARE OPERATING, WHICH IS INCONSISTENT WITH THE TECH SPEC REQUIREMENTS FOR MODES 3, 4 AND 5. IN ADDITION, PLANT PROCEDURES DO NOT ALLOW THE OPERATION OF ALL FOUR REACTOR COOLANT PUMPS BELOW 500 DEGREES F DUE TO CORE UPLIFT CONSIDERATIONS. ADMINISTRATIVE CONTROLS WERE INSTALLED TO ENSURE THAT THE CONTROL ELEMENT DRIVEN MECHANISMS A RE DE-ENERGIZED WHEN LESS THAN FOUR REACTOR COOLANT PUMPS ARE OPERATING. IN ADDITION, WESTINGHOUSE HAS REPERFORMED THE SAFETY ANALYSIS FOR THE CEA WITHDRAWAL FROM SUBCRITICAL ACCIDENT TO SUPPORT A CHANGE TO THE TECH SPECS. THIS CHANGE TO THE MILLSTONE UNIT 2 TECH SPECS WAS APPROVED BY THE NRC ON APRIL 21, 1987 AS AMENDMENT NO. 116.

[117] MILLSTONE 2 DOCKET 50-336 LER 87-014 REV 01
UPDATE ON MAIN STEAM VALVE SETPOINT DRIFT.
EVENT DATE: 123187 REPORT DATE: 022688 NSSS: CE TYPE: PWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV

(NSIC 208379) THIS LICENSEE EVENT REPORT (LER) IS SUBMITTED FOR INFORMATION ONLY. ON DECEMBER 31, 1987 AT 1200, THE PLANT CONDUCTED ROUTINE MAIN STEAM SAFETY VALVE SIMMER TESTING AS REQUIRED BY TECH SPEC SURVEILLANCE 4.7.1.1. PLANT CONDITIONS WERE AS FOLLOWS: HOT STANDBY (MODE 3), 0% POWER, 515 DEGREES FAHRENHEIT, AND 2010 PSI. TWELVE OF THE SIXTEEN VALVES TESTED FAILED THE INITIAL SIMMER TEST; TWO WITH SETPOINTS BELOW THE REQUIRED RANGE AND TEN WITH SETPOINTS ABOVE. THE IMMEDIATE CAUSE OF THE SETPOINT DRIFT IS NOT KNOWN. SIX OF THE VALVES WERE SENT TO AN OUTSIDE CONTRACTOR FOR ROUTINE REFURBISHMENT. THE SETPOINT OF EACH REMAINING VALVE WAS ADJUSTED AS IT WAS DETERMINED TO BE OUT OF SPECIFICATION. THE PLANT COMPLIED WITH THE REQUIREMENTS OF TECH SPEC ACTION S TATEMENT 3.7.1.1.A AT ALL TIMES. SIMILAR EVENTS: 86-008, 83-021

[118] MILLSTONE 2 DOCKET 50-336 LER 88-001
STEAM GENERATOR TUBE PLUGGING ERROR.
EVENT DATE: 010888 REPORT DATE: 020588 NSSS: CE TYPE: PWR

(NSIC 208191) ON JANUARY 8, 1988, WITH THE UNIT IN COLD SHUTDOWN, THREE TUBE ENDS WHICH SHOULD HAVE BEEN PLUGGED DURING THE 1986 REFUELING OUTAGE WERE FOUND UNPLUGGED. THREE ADJACENT TUBES, NOT SCHEDULED FOR PLUGGING WERE PLUGGED ON ONE END. THE UNPLUGGED TUBES CONTAINED FLAWS WITH GREATER THAN 40% THROUGH WALL PENETRATION.

[119] MILLSTONE 2 DOCKET 50-336 LER 88-002
LOSS OF NORMAL POWER.
EVENT DATE: 011988 REPORT DATE: 021888 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: MILLSTONE 1 (BWR)
VENDOR: CENTRALAB
SOLID STATE CONTROLS, INC.

(NSIC 208330) WITH THE PLANT IN MODE 6 AND THE REACTOR COOLANT SYSTEM AT APPROXIMATELY 80 DEGREES FAHRENHEIT, A LOSS OF NORMAL POWER (LNP) OCCURRED ON THE FACILITY 2 VITAL 4160 VOLT ELECTRICAL BUS (24D). THE LNP FOLLOWED A SENSED UNDERVOLTAGE CONDITION THAT CAUSED AN ACTUATION OF THE ENGINEERED SAFETY FEATURES (ESF) SYSTEM. AS EXPECTED, ACTUATION OF THE UNDERVOLTAGE ESF AUTOMATICALLY DEENERGIZED THE IN SERVICE VITAL 4160 VOLT BUS, STRIPPED OFF ALL LOADS, STARTED THE 13U EMERGENCY DIESEL GENERATOR, AND SEQUENCED LOADS BACK ON THE VITAL 4160 VOLT BUS TO BE POWERED FROM THE 13U EMERGENCY DIESEL. THE EVENT OCCURRED WHILE ONE OF FOUR VITAL 120 VOLT AC PANELS, VA-30, SUPPLYING POWER FOR THE ESF INPUTS, WAS OUT OF SERVICE FOR MAINTENANCE. THE EVENT WAS CAUSED WHEN THE LOSS OF A SECOND VITAL 120 VOLT AC PANEL VA-10 SATISFIED THE 2 OUT OF 4 CRITERION FOR ACTUATION OF THE ESF SYSTEM ON UNDERVOLTAGE. VA-10 LOST POWER WHEN ITS SUPPLY INVERTER, INV-1, SHUTDOWN DUE TO A SUDDEN FAILURE OF ONE OF ITS DC INPUT CAPACITORS. THE CAUSE OF THE CAPACITOR FAILURE IS NOT KNOWN, BUT IS BELIEVED TO BE A RANDOM FAILURE. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO RESTORE NORMAL POWER TO VITAL BUS 24D. ADDITIONAL CORRECTIVE ACTION INVOLVED TESTING ALL REPLACEMENT CAPACITORS FOR INVERTERS, REPLACING THE FAILED CAPACITOR IN INV-1 AND RETURNING INV-1 TO NORMAL SERVICE.

[120] MILLSTONE 2 DOCKET 50-336 LER 88-004
INADVERTENT AUXILIARY EXHAUST ACTUATION.
EVENT DATE: 012688 REPORT DATE: 022288 NSSS: CE TYPE: PWR
VENDOR: CONSOLIDATED CONTROLS CORP.
LAMBDA ELECTRONICS

(NSIC 208331) ON JANUARY 26, 1988, AT 1037 HOURS, WITH THE PLANT REFUELING (MODE 6), 0% POWER, 21 PSIA AND 100 DEGREES FAHRENHEIT (F), AN INADVERTENT ENGINEERED SAFETY FEATURES ACTUATION OCCURRED. THE EVENT OCCURRED WHEN AN INSTRUMENT AND CONTROLS TECHNICIAN CONNECTED TEST EQUIPMENT TO AN ENGINEERED SAFEGUARDS ACTUATION SYSTEM (ESAS) POWER SUPPLY. AN ESAS COMMON WAS GROUNDED AND AN AUXILIARY EXHAUST ACTUATION SIGNAL (AEAS) OCCURRED, CAUSING THE ENCLOSURE BUILDING VENTILATION DAMPERS TO GO TO THEIR AEAS POSITIONS. ALL PERSONNEL AND EQUIPMENT RESPONDED AS REQUIRED. NO UNANTICIPATED SYSTEM RESPONSES OCCURRED. THERE WERE NO SAFETY CONSEQUENCES RESULTING FROM THIS EVENT. THE ROOT CAUSE OF THE EVENT WAS PERSONNEL ERROR. THE RETEST PROCEDURE CAUTIONS AGAINST CONNECTING THE ESAS CABINET TO GROUND. THE TEST PROCEDURE WAS REVISED TO PROVIDE MORE EXPLICIT INSTRUCTIONS ON INSTALLING THE TEST EQUIPMENT. NO FURTHER CORRECTIVE ACTION IS REQUIRED.

[121]   MILLSTONE 3   DOCKET 50-423   LER 87-031 REV 01
UPDATE ON REACTOR TRIP DUE TO TURBINE TRIP ON LOW LUBE OIL HEADER PRESSURE.
EVENT DATE: 061487   REPORT DATE: 022688   NSSS: WE   TYPE: PWR
VENDOR: GEN ELEC CO (STEAM TURB/ENGRD PROD)
        MORRIS PUMPS, INC.

(NSIC 208312) ON JUNE 14, 1987 AT 0320 WITH THE PLANT AT 100% POWER (MODE 1) THE REACTOR TRIPPED AS A RESULT OF A TURBINE TRIP. THE TURBINE TRIPPED ON LOW BEARING LUBE OIL HEADER PRESSURE IMMEDIATELY FOLLOWING A TRIP OF THE TURNING GEAR OIL PUMP (TGOP). THE TGOP HAD BEEN AUTOSTARTED APPROXIMATELY 9 SECONDS EARLIER AS PART OF WEEKLY TURBINE/GENERATOR PREVENTIVE MAINTENANCE TESTING, AND WAS NOT THE PRIMARY SOURCE OF LUBE OIL HEADER PRESSURE. PRESSURE WAS PRIMARILY SUPPLIED BY THE (TURBINE) SHAFT DRIVEN OIL PUMP, WHICH WAS OPERATING CORRECTLY. ALL EQUIPMENT OPERATED AS EXPECTED IN RESPONSE TO THE TRIP AND THE PLANT RETURNED TO POWER OPERATION (MODE 1) AT APPROXIMATELY 2012 HOURS. THE CAUSE FOR THIS EVENT WAS A DEFECTIVE BEARING HEADER RELIEF VALVE AND AN OVERSIZED TURNING GEAR OIL PUMP. AS CORRECTIVE ACTION, THE BEARING HEADER RELIEF VALVE WAS REPAIRED AND THE IMPELLER ON THE TGOP WAS SHAVED TO LOWER PRESSURE.

[122]   MILLSTONE 3   DOCKET 50-423   LER 87-046
SAMPLE RIG ACTION STATEMENT SURVEILLANCE MISSED.
EVENT DATE: '12487   REPORT DATE: 121887   NSSS: WE   TYPE: PWR

(NSIC 207482) AT 1900 ON 11/24/87, AND 1200 ON 11/27/87, WHILE REFUELING IN MODE 6, THE AUXILIARY SAMPLE RIG AT THE VENTILATION VENT WAS NOT PROPERLY MONITORED. WHILE EFFORTS TO RETURN THE NORMAL RADIATION MONITOR TO SERVICE PROCEEDED, CONTINUOUS IODINE AND PARTICULATE SAMPLES AND 12 HOUR GAS GRAB SAMPLES WERE BEING COLLECTED IN ACCORDANCE WITH TECHNICAL SPECIFICATION ACTION STATEMENTS, USING THE AUXILIARY SAMPLE RIG. SAMPLE FLOW WAS BEING VERIFIED EVERY 4 HOURS. ON 11/24/87 THE AUXILIARY SAMPLE RIG WAS ISOLATED DUE TO IMPROPER MANUAL VALVE LINEUP. GAS GRAB SAMPLES BEING COLLECTED EVERY 12 HOURS WERE UNAFFECTED BY THE ISOLATION. CONTINUOUS IODINE AND PARTICULATE SAMPLING WAS INTERRUPTED LESS THAN 3 HOURS. ON 11/27/87 SAMPLE FLOW VERIFICATION EVERY 4 HOURS WAS MISSED ON TWO OCCASIONS. READINGS TAKEN BEFORE AND AFTER THE MISSED READINGS INDICATE SATISFACTORY FLOW WAS MAINTAINED.

[123]   MILLSTONE 3   DOCKET 50-423   LER 88-001
INADVERTENT SAFETY INJECTION DUE TO SENSITIVE EQUIPMENT.
EVENT DATE: 010588   REPORT DATE: 020188   NSSS: WE   TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 208240) ON JANUARY 5, 1988 AT 1630 WITH THE PLANT AT 0% POWER, 110 DEGREES AND AMBIENT PRESSURE, AN INADVERTENT SAFETY INJECTION SIGNAL WAS RECEIVED ON TRAIN A. THE SIGNAL WAS NOT REQUIRED FOR SAFETY AND NO INJECTION TO THE CORE TOOK PLACE. THE IMMEDIATE CAUSE OF THE EVENT WAS AN IMPROPER SWITCH POSITION DURING RESTORATION FROM INSTRUMENT AND CONTROLS TESTING. THE MAIN BOARD OPERATOR WAS COGNIZANT OF THE RESTORATION BEING PERFORMED AND THE POTENTIAL FOR AN INADVERTENT SAFETY INJECTION. HIS ACTIONS, AND THOSE OF THE INSTRUMENT TECHNICIAN, WERE IN COMPLIANCE WTIH THE RESTORATION PROCEDURE. IMMEDIATE ACTION WAS TO VERIFY THAT A SAFETY INJECTION WAS NOT REQUIRED AND RESET THE SIGNAL. NO ENGINEERED SAFETY FEATURES ACTUATIONS WERE REQUIRED FOR EXISTING PLANT CONDITIONS, AND NONE OCCURRED. THE ROOT CAUSE OF THE EVENT IS A SENSITIVE SWITCH. AS AN INTERIM SOLUTION, THE OPERATING PROCEDURE HAS BEEN CHANGED TO VERIFY BY INDICATION THAT APPROPRIATE BLOCKS AND RESETS ARE IN PLACE PRIOR TO TAKING AN ACTION THAT COULD RESULT IN A SAFETY INJECTION. TO PREVENT RECURRENCE OF THIS PROBLEM, THIS SWITCH WILL BE REPLACED BY A LESS SENSITIVE DESIGN PRIOR TO THE END OF THE SECOND REFUELING OUTAGE. SIMILAR SWITCHES ON THE MAIN CONTROL BOARDS WILL BE EVALUATED AND ALSO CHANGED IF REQUIRED.

[124] MILLSTONE 3 DOCKET 50-423 LER 88-002
INSUFFICIENT SEISMIC SUPPORT OF REACTOR COOLANT PUMP OIL COLLECTION SYSTEM.
EVENT DATE: 011388 REPORT DATE: 020488 NSSS: WE TYPE: PWR

(NSIC 208205) ON JANUARY 13, 1988 AT 1200 HOURS WITH THE PLANT SHUTDOWN IN MODE 6 AT A TEMPERATURE OF 87F AND AT ATMOSPHERIC PRESSURE, IT WAS DETERMINED THAT DOCUMENTATION WAS INSUFFICIENT TO ASSURE EACH REACTOR COOLANT PUMP (RCP) OIL COLLECTION SYSTEM WAS SEISMICALLY SUPPORTED AS REQUIRED BY DESIGN. AN INVESTIGATION REVEALED THAT THE ROOT CAUSE WAS A LACK OF INTEGRATION NECESSARY TO ACCOMMODATE ONE CONTRACTOR'S DESIGN WITH ANOTHER CONTRACTOR'S CONSTRUCTION WHEN THE CALCULATIONS WERE PERFORMED. CONSEQUENTLY, ONE AREA OF SEISMIC SUPPORT WAS OVERLOOKED. AN EVALUATION OF ALL RCP OIL COLLECTION SYSTEMS WAS COMPLETED, AND THE REQUIRED HANGERS (ONE ON EACH OIL COLLECTION SYSTEM) WERE ADDED TO SATISFY THE SEISMIC CRITERIA.

[125] MILLSTONE 3 DOCKET 50-423 LER 88-003
DIESEL SEQUENCED START DUE TO SPURIOUS RELAY ACTUATION.
EVENT DATE: 011688 REPORT DATE: 020488 NSSS: WE TYPE: PWR

(NSIC 208206) ON JANUARY 16, 1988 AT 1542, WITH THE PLANT AT 0% POWER IN COLD SHUTDOWN (110 DEGREES, 65 PSIA WATER SOLID), THE B TRAIN EMERGENCY DIESEL GENERATOR STARTED WHEN A RELAY, WHICH WAS BUMPED, OPERATED, ALLOWING A TEST SIGNAL TO COMMENCE EMERGENCY LOAD SEQUENCING. INSTRUMENT AND CONTROL TECHNICIANS WERE PERFORMING TIME RESPONSE TESTING OF THE EXTERNAL SAFETY INJECTION SIGNAL TO THE EMERGENCY GENERATOR LOAD SEQUENCER. THE SEQUENCER WAS ALIGNED INTO THE APPROPRIATE TEST CONFIGURATION AND SHOULD NOT HAVE COMMENCED SEQUENCING UPON RECEIPT OF THE TEST SIGNAL. THE LICENSED MAIN CONTROL BOARD OPERATOR RECEIVED A DIESEL START SIGNAL AND NOTED THE B SEQUENCER STRIPPING BUS LOADS AND THE START OF THE B EMERGENCY DIESEL. THE OPERATOR STOPPED AND RESTORED THE DIESEL AND RESTARTED THE VENTILATION LOADS THAT THE LOAD SEQUENCER HAD STRIPPED FROM THE BUSSES. THE TOTAL ELAPSED TIME WAS LESS THAN 15 MINUTES. THE ROOT CAUSE OF THE EVENT WAS A RELAY FAILURE. IMMEDIATE CORRECTIVE ACTION WAS TO REPLACE THE RELAY WITH A SPARE. AS ACTION TO PREVENT RECURRENCE, SURVEILLANCE PROCEDURES HAVE BEEN CHANGED TO REMOVE THE DEPENDENCY ON THE DEFECTIVE RELAY. FUTURE TESTS WILL HAVE CONTROL POWER REMOVED TO THE OUTPUT RELAY RATHER THAN RELY ON THE BYPASS MODE OF THE SEQUENCER.

[126] MILLSTONE 3 DOCKET 50-423 LER 88-004
CONTROL BUILDING ISOLATION SIGNAL DUE TO CHLORINE DETECTOR FAILURE.
EVENT DATE: 011888 REPORT DATE: 021088 NSSS: WE TYPE: PWR
VENDOR: STONE & WEBSTER ENGINEERING CORP.

(NSIC 208207) AT 2319 ON JANUARY 18, 1988, WITH THE PLANT IN COLD SHUTDOWN AT 113 DEGREES FAHRENHEIT AND 360 PSI, A CONTROL BUILDING ISOLATION (CBI) SIGNAL WAS RECEIVED FROM THE TRAIN B CONTROL BUILDING VENTILATION SYSTEM SUPPLY AIR DUCT CHLORINE DETECTOR. ISOLATION OF THE CONTROL BUILDING OCCURRED PROPERLY, AND THE CONTROL BUILDING WAS PLACED IN FILTERED RECIRCULATION PER PLANT TECHNICAL SPECIFICATION REQUIREMENTS. THE CBI SIGNAL WAS THE RESULT OF A FAILURE OF A CHLORINE DETECTOR DUE TO A FAILED SENSING ELEMENT. THE CAUSE OF THE FAILED SENSING ELEMENT IS UNKNOWN. THE SENSING ELEMENT WAS REPLACED AND THE DETECTOR RETURNED TO SERVICE. THE FAILURE IS CONSIDERED AN ISOLATED INCIDENT. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV).

[127] MILLSTONE 3 DOCKET 50-423 LER 88-005
COLD OVERPRESSURE PROTECTION SYSTEM FAILS TO OPERATE DURING PRESSURE TRANSIENT.
EVENT DATE: 011988 REPORT DATE: 021888 NSSS: WE TYPE: PWR

(NSIC 208357) AT 1056 ON 1/19/88 WITH THE PLANT IN COLD SHUTDOWN (MODE 5),

TEMPERATURE 135 DEGREES, PRESSURE 350 PSIA, A PRESSURE TRANSIENT OCCURRED WHICH CHALLENGED THE COLD OVERPRESSURE PROTECTION SYSTEM (COPPS). THE COPPS FAILED TO OPERATE WHEN REQUIRED. THE TRANSIENT WAS MITIGATED BY MANUAL OPERATOR ACTION. THE REACTOR COOLANT SYSTEM WAS INITIALLY IN A COLD CONDITION WITH BOTH CHARGING AND LETDOWN IN OPERATION. THE LETDOWN PATH WAS THROUGH B TRAIN RESIDUAL HEAT REMOVAL (RHR) WITH THE A TRAIN INOPERABLE DUE TO A GENERAL OUTAGE OF THE A TRAIN. THE PRESSURE TRANSIENT WAS INITIATED WHEN A CONTRACT TECHNICIAN REMOVED FUSES FOR WORK IN WHAT WAS BELIEVED TO BE AN UNRELATED SYSTEM RESULTING IN ISOLATION OF THE RHR LETDOWN PATH. THE FAILURE OF COPPS TO OPERATE WAS DUE TO A LACK OF INPUT FROM THE SOLID STATE PROTECTION SYSTEM, WHICH WAS NOT AVAILABLE. THE ROOT CAUSE OF THE EVENT WAS A LACK OF A PROCEDURE FOR THE ARMING OF COLD OVERPRESSURE PROTECTION. A PROCEDURE WOULD HAVE PROVIDED THE OPERATORS WITH A SET OF PREREQUISITES NEEDED TO ENSURE THAT THE SYSTEM WAS OPERABLE. [illegible] CTION TO PREVENT RECURRENCE OF THIS INCIDENT IS THE PREPARATION OF A PROCEDURE FOR THE USE OF THE COLD OVERPRESSURE PROTECTION AND INCREASED TRAINING TO PREVENT RECURRENCE OF THE EVENTS LEADING TO THE TRANSIENT.

[128] MILLSTONE 3 DOCKET 50-423 LER 88-007
MANUAL REACTOR TRIP DUE TO INOPERABLE DIGITAL ROD POSITION INDICATOR.
EVENT DATE: 020388 REPORT DATE: 030488 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 208477) ON 2/3/88 AT 0548 HOURS, WHILE IN HOT STANDBY, WITH THE REACTOR SUBCRITICAL AT 2250 PSIA AND 557F, A MANUAL REACTOR TRIP WAS INITIATED DUE TO AN INOPERABLE DIGITAL ROD POSITION INDICATION (DRPI) INDICATOR. AT THE TIME OF THE TRIP, SHUTDOWN BANK D HAD JUST BEEN FULLY WITHDRAWN. THE REACTOR OPERATOR NOTICED THAT ROD N-11 HAD DUAL POSITION INDICATION AT FULL OUT AND PARTIALLY INSERTED POSITIONS. THE SHIFT SUPERVISOR ORDERED THE REACTOR TRIP BREAKERS OPENED. THE OPERATORS VERIFIED OPENING OF THE TRIP BREAKERS AND INSERTION OF ALL RODS. TRIP RESPONSE PROCEDURES WERE IMPLEMENTED AND THE PLANT WAS STABILIZED. DRPI NOW INDICATED ROD N-11 AT BOTTOM, WITH NO DUAL POSITION INDICATION. ROD E-13, HOWEVER, INDICATED BOTH AT BOTTOM AND PARTIALLY INSERTED. THE OPERATORS VERIFIED THAT ADEQUATE SHUTDOWN MARGIN (ASSUMING E-13 FULLY STUCK OUT) WAS PRESENT. THE ROOT CAUSE OF THE EVENT WAS EQUIPMENT FAILURE. INVESTIGATION REVEALED THAT THE DISPLAY CARD FOR ROD N-11 WAS WARPED. THIS CAUSED PINS ON THE N-11 CARD TO CONTACT THE ADJACENT CARD (E-13), RESULTING IN THE FAULTY POSITION INDICATION. THE CARD WAS REPLACED AND ALL RODS INDICATED AT BOTTOM.

[129] MONTICELLO DOCKET 50-263 LER 87-023 REV 01
UPDATE ON REACTOR RECIRCULATION SYSTEM DECONTAMINATION CONNECTION DUE TO HIGH CYCLE FATIGUE.
EVENT DATE: 112587 REPORT DATE: 022988 NSSS: GE TYPE: BWR
VENDOR: SANDVIK STEEL INC.

(NSIC 208446) WHILE PERFORMING THE REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE TEST, A CRACK WAS DISCOVERED IN A TWO INCH DECONTAMINATION CONNECTION ON THE REACTOR RECIRCULATION SYSTEM A LOOP. THE PLANT WAS SHUTDOWN FOR A REFUELING OUTAGE. METALLOGRAPHIC EXAMINATION REVEALED THAT THE CRACK WAS CAUSED BY HIGH CYCLE FATIGUE. IT IS BELIEVED THAT THE FATIGUE FAILURE RESULTED FROM CYCLIC STRESSES CAUSED BY FORCED RESONANT VIBRATION IN THE DECONTAMINATION CONNECTION. THE DECONTAMINATION CONNECTION WAS MODIFIED SO THAT ITS NATURAL FREQUENCY DOES NOT COINCIDE WITH THE FREQUENCY OF ANY EXPECTED RECIRCULATION SYSTEM FORCING FUNCTIONS. OTHER RECIRCULATION SYSTEM BRANCH CONNECTIONS WERE EVALUATED. TWO OTHER DECONTAMINATION CONNECTIONS WERE IN THE RANGE OF CONCERN AND WERE MODIFIED. IT IS BELIEVED THAT NO OTHER CONNECTIONS ARE SUSCEPTIBLE TO THIS FAILURE MODE.

[130] MONTICELLO DOCKET 50-263 LER 88-001
FIRE WATCH INTERVAL EXCEEDED DUE TO INADEQUATE CONTROLS.
EVENT DATE: 010188 REPORT DATE: 020188 NSSS: GE TYPE: BWR

(NSIC 208116) DURING A PERIOD OF 89 HOURS FROM DECEMBER 31, 1987, UNTIL JANUARY 4, 1988, THE FIRE DETECTION FOR THE TORUS AREA FIRE ZONE WAS INOPERABLE. A FIRE WATCH PATROL WAS ESTABLISHED. ALTHOUGH THE AREA WAS INSPECTED 93 TIMES DURING THE 89 HOUR PERIOD, SOME INTERVALS BETWEEN INSPECTIONS EXCEEDED THE ONE HOUR INTERVAL REQUIRED BY THE TECHNICAL SPECIFICATIONS. THE LONGEST INTERVAL WAS 1 HOUR AND 55 MINUTES. THE CAUSE OF THE PROBLEM WAS INADEQUATE CONTROLS OVER THE FIRE WATCH PATROL FREQUENCIES. IT WAS APPARENT FROM THE 93 INSPECTIONS OVER THE 89 HOUR PERIOD THAT PERSONNEL WERE AWARE OF THE ONE HOUR REQUIREMENT, HOWEVER THERE WERE NO ADMINISTRATIVE CONTROLS CONCERNING FIRE WATCH PATROLS AND ACCEPTABLE PATROL FREQUENCIES. AN OPERATIONS POLICY WAS ISSUED REQUIRING A PERSON BE ASSIGNED WHOSE PRIMARY DUTIES ARE THE FIRE WATCH PATROL (IN THIS CASE THE DUTY REACTOR BUILDING OPERATOR, WHO HAD OTHER CONCURRENT DUTIES, WAS ASSIGNED). THE POLICY ALSO STATES THAT AN ACCEPTABLE INSPECTION FREQUENCY IS 1 HOUR PLUS OR MINUS 25% (45 MINUTES TO 75 MINUTES). THE 25% IS CONSISTENT WITH OTHER TECHNICAL SPECIFICATIONS SURVEILLANCE REQUIREMENTS.

[131] NINE MILE POINT 1 DOCKET 50-220 LER 88-001
TECHNICAL SPECIFICATION VIOLATION DUE TO ISI PROGRAM DEFICIENCIES.
EVENT DATE: 011588 REPORT DATE: 021688 NSSS: GE TYPE: BWR

(NSIC 208391) ON JANUARY 15, 1988, WITH NINE MILE POINT UNIT 1 (NMP1) AT 0% POWER AND THE MODE SWITCH IN REFUEL, A REVIEW OF THE FIRST TEN YEAR INSERVICE INSPECTION (ISI) INTERVAL WAS COMPLETED. THIS REVIEW IDENTIFIED SEVERAL INSPECTION DEFICIENCIES REQUIRING RESOLUTION, I.E. FAILURE TO COMPLETE THE FIRST TEN YEAR INTERVAL INSPECTION REQUIREMENTS AND FAILURE TO PROPERLY DISPOSITION DEFICIENCY/CORRECTIVE ACTION (DCA) NOTICES AND OTHER EXAMINATION RESULTS. THE FIRST TEN YEAR INTERVAL WAS SCHEDULED FOR COMPLETION DURING THE 1986 REFUELING OUTAGE. BY NOT COMPLETING AND RESOLVING ALL THE INSPECTION REQUIREMENTS REQUIRED BY SECION XI OF THE ASME BOILER AND PRESSURE VESSEL CODE, A VIOLATION OF TECH SPEC 3.2.6 HAS OCCURRED. THE ROOT CAUSE OF THE EVENT IS MANAGEMENT INEFFECTIVENESS IN IMPLEMENTING THE ISI PROGRAM PLAN. INITIAL CORRECTIVE ACTION WAS TO DETERMINE WHAT OMITTED ITEMS ARE NECESSARY TO COMPLETE THE FIRST TEN YEAR ISI INTERVAL. ALL THESE ITEMS WILL BE COMPLETED DURING THE PRESENT REFUELING OUTAGE. ALSO, ALL THE OUTSTANDING DCA'S AND INSPECTION REPORTS WILL BE PROPERLY DISPOSITIONED. THE QUALITY ASSURANCE DEPARTMENT HAS ISSUED A CORRECTIVE ACTION REQUEST FOR THIS EVENT. BECAUSE OF THE SIMILARITY OF THIS EVENT TO THE EVENT DESCRIBED IN LICENSEE EVENT REPORT 87-27.

[132] NINE MILE POINT 2 DOCKET 50-410 LER 87-075
CONDENSATE STORAGE TANK RUPTURE DUE TO HIGH STRESS WHEN FILLED CAUSED BY CONSTRUCTION DEFICIENCIES.
EVENT DATE: 112887 REPORT DATE: 021288 NSSS: GE TYPE: BWR
VENDOR: ENGDAHL ENTERPRISES
METAL CLADDING, INC.

(NSIC 208381) ON NOVEMBER 28, 1987 AT 1824 HOURS, CONTROL ROOM OPERATORS AT NINE MILE POINT UNIT 2 (NMP2) WERE INFORMED THAT ONE OF TWO CONDENSATE STORAGE TANKS (2CNS-TK1A) WAS LEAKING WATER FROM A LARGE CRACK NEAR ITS BOTTOM. AT THE TIME, NMP2 WAS IN THE COLD SHUTDOWN CONDITION WITH REACTOR COOLANT AT APPROXIMATELY 114 DEGREES FAHRENHEIT AND AMBIENT PRESSURE. THE LEAKING WATER CAUSED A SUMP PUMP TO OVERFLOW AND A SECONDARY CONTAINMENT PENETRATION SEAL TO FAIL, WHICH ESTABLISHED A FLOW PATH TO THE REACTOR BUILDING (RB) 175 FOOT ELEVATION. FLOODING ON RB 175 FOOT ELEVATION CAUSED THE FAILURE OF A SEISMIC MONITOR. THE CAUSE OF THE RUPTURE OF 2CNS-TK1A WAS A CONSTRUCTION DEFICIENCY. THE CAUSE OF THE PENETRATION SEAL FAILURE WAS ASSUMED TO BE IMPROPER WORK PRACTICES/PERSONNEL ERROR. THE

CONSTRUCTION DEFICIENCY HAS BEEN CORRECTED AND 2CNS-TK1A HAS BEEN REFILLED AND RETESTED. THE PENETRATION SEAL AND FLOODED SEISMIC MONITOR HAVE BEEN REPAIRED. PROCEDURES WILL ALSO BE DEVELOPED TO ENSURE THAT WATERTIGHT PENETRATION SEALS ARE VERIFIED OPERABLE ONCE PER 18 MONTHS. THIS EVENT HAS BEEN DETERMINED TO NOT BE REPORTABLE PER 10CFR21, "REPORTING OF DEFECTS AND NONCOMPLIANCE", NOR 10CFR50.73, "LICENSEE EVENT REPORT SYSTEM", BUT IS BEING SUBMITTED AS A VOLUNTARY LER.

[133] NINE MILE POINT 2 DOCKET 50-410 LER 88-001
REACTOR SCRAM DUE TO A LOSS OF FEEDWATER FLOW CAUSED BY PERSONNEL ERROR.
EVENT DATE: 012088 REPORT DATE: 021788 NSSS: GE TYPE: BWR

(NSIC 208350) ON JANUARY 20, 1988 AT 0944 HOURS WITH THE REACTOR OPERATING AT APPROXIMATELY 41% POWER, NINE MILE POINT UNIT 2 EXPERIENCED A SCRAM DUE TO AN ACTUAL LOW (LEVEL 3) WATER LEVEL CONDITION. THE LOW WATER LEVEL WAS CAUSED BY A LOSS OF FEEDWATER FLOW TO THE REACTOR. AN OPERATOR WHILE PLACING A MARKUP (TAG OUT), VALVED AN INSTRUMENT AIR SYSTEM PREFILTER OUT OF SERVICE WITHOUT ENSURING THAT A REDUNDANT PREFILTER WAS IN SERVICE. THIS ISOLATED THE AIR COMPRESSORS FROM THE REMAINDER OF THE SYSTEM. INSTRUMENT AIR PRESSURE THROUGHOUT THE PLANT DECAYED TO THE POINT OF CAUSING THE MINIMUM FLOW VALVES FOR THE CONDENSATE, CONDENSATE BOOSTER, AND FEEDWATER PUMPS TO FAIL OPEN. AS A RESULT, REDUCED FEEDWATER FLOW TO THE REACTOR CAUSED REACTOR WATER LEVEL TO RAPIDLY DECREASE TO THE LEVEL 3 SCRAM SETPOINT (159.3 INCHES). BOTH THE HIGH PRESSURE CORE SPRAY (HPCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEMS ACTUATED ON LOW LEVEL INSTRUMENTATION WHICH RESTORED NORMAL WATER LEVEL IN THE VESSEL. THE MAIN TURBINE-GENERATOR ALSO TRIPPED DUE TO RCIC INJECTION. DESIGN AND PERSONNEL ERRORS WERE ENCOUNTERED DURING THE EVENT. CORRECTIVE ACTIONS HAVE BEEN IMPLEMENTED AND/OR INCORPORATED TO MINIMIZE THE POTENTIAL OF A RECURRENCE.

[134] NINE MILE POINT 2 DOCKET 50-410 LER 88-002
REACTOR WATER CLEANUP SYSTEM DEFICIENCY RESULTS IN AN ISOLATION ON A HIGH DIFFERENTIAL FLOW SIGNAL.
EVENT DATE: 012088 REPORT DATE: 021888 NSSS: GE TYPE: BWR

(NSIC 208273) ON JANUARY 20, 1988 AT 2019 HOURS, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED THE ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF), SPECIFICALLY, ISOLATION OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM ON AN ERRONEOUS HIGH DIFFERENTIAL FLOW SIGNAL. AT THE TIME OF THE EVENT, THE PLANT WAS IN THE COLD SHUTDOWN CONDITION WITH THE REACTOR MODE SWITCH IN THE "SHUTDOWN" POSITION. REACTOR PRESSURE AND TEMPERATURE WERE APPROXIMATELY 0 POUNDS PER SQUARE INCH GAUGE (PSIG) AND 185F, RESPECTIVELY. THE IMMEDIATE CAUSE FOR THE EVENT WAS A SHIFT IN THE DIVISION I RWCU SUCTION FLOW TRANSMITTER SIGNAL. THIS SHIFT RESULTED IN AN INCREASE IN THE OUTPUT VOLTAGE OF THE TRANSMITTER. THE ROOT CAUSE IS A DESIGN DEFICIENCY OF THE RWCU SYSTEM. INITIAL CORRECTIVE ACTIONS WERE FOR THE OPERATORS TO VERIFY THE PLANT STATUS AS NORMAL, BYPASS THE ISOLATION SIGNAL AND RESTORE THE RWCU SYSTEM TO SERVICE. ADDITIONAL CORRECTIVE ACTION WAS TO ISSUE A WORK REQUEST (WR 131121) FOR THE DIVISION I RWCU DIFFERENTIAL FLOW INSTRUMENTATION (COMPRISED OF SEVERAL INSTRUMENTS). UNDER THIS WORK REQUEST, THE DIVISION I RWCU SUCTION FLOW TRANSMITTER WAS VENTED AND RECALIBRATED. THE DETAILS OF THIS EVENT WILL BE FORWARDED TO THE NMP2 SPECIAL TASK FORCE ASSIGNED TO EVALUATE AND TROUBLESHOOT THE RWCU SYSTEM. CORRECTIVE ACTIONS WILL BE IMPLEMENTED AS WARRANTED BY THE TASK FORCE RECOMMENDATIONS.

[135] NINE MILE POINT 2 DOCKET 50-410 LER 88-003
TECHNICAL SPECIFICATION VIOLATION DUE TO THE FAILURE TO SUBMIT A SPECIAL REPORT AND PROGRAMMATIC DEFICIENCY.
EVENT DATE: 012188 REPORT DATE: 021788 NSSS: GE TYPE: BWR

(NSIC 208351) ON JANUARY 21, 1988 WITH THE REACTOR IN COLD SHUTDOWN (OPERATIONAL

CONDITION 4), IT WAS DISCOVERED THAT NINE MILE POINT UNIT 2 (NMP2) WAS NOT IN COMPLIANCE WITH TECHNICAL SPECIFICATION (TS) SECTION 3.5.1(F). SPECIFICALLY A SPECIAL REPORT REQUIRED BY TS SECTION 3.5.1(F) WAS NOT SUBMITTED IN THE DESIGNATED TIME PERIOD FOR AN EMERGENCY CORE COOLING SYSTEM LOW PRESSURE COOLANT INJECTION EVENT WHICH OCCURRED ON DECEMBER 18, 1986. THE DUE DATE FOR THIS SPECIAL REPORT WAS MARCH 19, 1987. THE ROOT CAUSE FOR THIS EVENT IS A PROGRAMMATIC DEFICIENCY. THE CORRECTIVE ACTIONS FOR THIS EVENT ARE: 1. THE INFORMATION NEEDED TO SATISFY THE TS (SPECIAL REPORT) REPORTABILITY REQUIREMENTS FOR THE DECEMBER 18, 1986 EVENT HAS BEEN INCLUDED IN THIS REPORT. 2. A TS REPORTABILITY REQUIREMENTS DOCUMENT HAS BEEN PREPARED. 3. A SUMMARY OF THE EVENT WILL BE INCLUDED IN THE OPERATIONS AND TECHNICAL SUPPORT DEPARTMENTS' LESSONS LEARNED BOOKS. 4. A REVIEW OF ALL NMP2 OCCURRENCE REPORTS HAS BEEN CONDUCTED.

[136] NINE MILE POINT 2 DOCKET 50-410 LER 88-004
PRIMARY CONTAINMENT ISOLATION ACTUATION INSTRUMENTATION SET LESS CONSERVATIVELY THAN TECHNICAL SPECIFICATION REQUIRED TRIP SETPOINT CAUSED BY PROCEDURAL DEFICIENCY.
EVENT DATE: 012188 REPORT DATE: 021888 NSSS: GE TYPE: BWR

(NSIC 208274) ON JANUARY 21, 1988 AT 1800 HOURS A TECHNICAL SPECIFICATION (TS) VIOLATION WAS DISCOVERED AT NINE MILE POINT UNIT 2 (NMP2). THE TS VIOLATION WAS A RESULT OF THE TRIP SETPOINTS OF CERTAIN PRIMARY CONTAINMENT ISOLATION ACTUATION INSTRUMENTATION BEING SET LESS CONSERVATIVELY THAN THE TS TRIP SETPOINT. AT THE TIME THE TS VIOLATION WAS DISCOVERED, NMP2 WAS IN THE COLD SHUTDOWN CONDITION WITH THE MODE SWITCH IN THE "SHUTDOWN" POSITION AND REACTOR COOLANT TEMPERATURE LESS THAN 200F. THE CAUSE OF THE EVENT WAS AN ERROR IN AN APPROVED PROCEDURE. THE MONTHLY FUNCTIONAL TEST AND OPERATING CYCLE CALIBRATION PROCEDURES FOR THE INSTRUMENTATION CONTAINED MISCALCULATED GUIDE VALUE SETPOINTS WHICH RESULTED IN THE TRIP SETPOINTS BEING SET LESS CONSERVATIVELY THAN TS TRIP SETPOINTS. THE FOLLOWING CORRECTIVE ACTIONS HAVE BEEN COMPLETED: 1. CORRECTION OF THE DEFICIENT PROCEDURES TO INCLUDE GUIDE VALUE SETPOINTS WHICH ARE ACCEPTABLE PER NMP2 TS. 2. RECALIBRATION OF THE INSTRUMENTATION SO THAT THE TRIP SETPOINTS COMPLY WITH CORRECTED PROCEDURE VALUES. 3. REVIEW OF GUIDE VALUE CALCULATIONS FOR ALL OTHER LEVEL, PRESSURE, AND TEMPERATURE INSTRUMENT CHANNELS WHICH HAVE TS REQUIRED SETPOINTS. PROCEDURE CHANGES WERE MADE AS REQUIRED. 4. THE I&C PROCEDURE WRITING GUIDE IS IN THE PROCESS OF BEING REVISED.

[137] NINE MILE POINT 2 DOCKET 50-410 LER 88-005
NON-COMPLIANCE WITH TECHNICAL SPECIFICATIONS AND WITH 10CFR50.73 DUE TO MOTOR FAILURE AND FAILURE TO SUBMIT AN LER AND IMPROPER MAINTENANCE/INCORRECT TECHNICAL SPECIFICATION INTERPRETATION.
EVENT DATE: 012688 REPORT DATE: 022588 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 208352) ON JANUARY, 26, 1988 WITH,THE REACTOR IN COLD SHUTDOWN (OPERATIONAL CONDITION 4), THE NINE MILE POINT UNIT 2 CHEMISTRY DEPARTMENT REPORTED TWO INCIDENTS (WHICH OCCURRED ON NOVEMBER 17, 1987 AND ON JANUARY 12, 1988) WHERE THE AUXILIARY SAMPLING EQUIPMENT FOR THE GASEOUS EFFLUENT MONITORING SYSTEMS (GEMS) WERE INOPERABLE. THESE INCIDENTS CONSTITUTED A NON-COMPLIANCE WITH TECHNICAL SPECIFICATION (TS) SECTION 3.3.7.10. ADDITIONALLY, THESE INCIDENTS WERE NOT REPORTED VIA 10CFR50.73(A)(2)(I)(B) IN A TIMELY MANNER. THE ROOT CAUSES FOR THESE EVENTS ARE, (1) IMPROPER EQUIPMENT MAINTENANCE, AND (2) AN INCORRECT INTERPRETATION OF TS SECTION 3.3.7.10 ACTION STATEMENT 138. THE CORRECTIVE ACTIONS FOR THESE EVENTS ARE: 1. THE DISCHARGE FILTER FOR THE AUXILIARY SAMPLING PUMPS IS BEING CLEANED OR REPLACED ON A WEEKLY BASIS. 2. A PROBLEM REPORT HAS BEEN SUBMITTED REQUESTING AN EVALUATION OF THE GEMS AUXILIARY SAMPLING EQUIPMENT. 3. A LESSONS LEARNED DOCUMENT HAS BEEN PREPARED AND WILL BE INCLUDED IN THE CHEMISTRY DEPARTMENT'S LESSONS LEARNED BOOK.

[138] NINE MILE POINT 2 DOCKET 50-410 LER 88-009
ESF ACTUATION INSTRUMENTATION SET LESS CONSERVATIVELY THAN TECHNICAL SPECIFICATION REQUIRED TRIP SETPOINT CAUSED BY PERSONNEL ERROR.
EVENT DATE: 012988 REPORT DATE: 022688 NSSS: GE TYPE: BWR

(NSIC 208353) ON 1/29/88 AT 1520 HOURS, A TECH SPEC VIOLATION WAS DISCOVERED AT NINE MILE POINT UNIT 2 (NMP2). THE TS VIOLATION 3.3.3 WAS A RESULT OF THE TRIP INSTRUMENTATION FOR THE HIGH PRESSURE CORE SPRAY (CSH) SUCTION TRANSFER CHANNELS BEING SET LESS CONSERVATIVELY THAN THE TS TRIP SETPOINT. AT THE TIME THE TS VIOLATION WAS DISCOVERED, NMP2 WAS IN THE COLD SHUTDOWN CONDITION WITH THE MODE SWITCH IN THE "SHUTDOWN" POSITION. REACTOR COOLANT TEMPERATURE AND PRESSURE WERE LESS THAN 200F, AND AMBIENT, RESPECTIVELY. THE ROOT CAUSE OF THE EVENT WAS A PERSONNEL ERROR. THE CALCULATION USED TO DETERMINE THE BASIS FOR THE TS TRIP SETPOINT FOR THE INSTRUMENTATION PROVIDED TWO DIFFERENT CALCULATION METHODS. THE FOLLOWING CORRECTIVE ACTIONS WERE TAKEN: 1. THE SETPOINTS OF THE INSTRUMENTATION HAS BEEN RECALIBRATED SO THAT IT IS CONSISTENT WITH TS REQUIREMENTS. THE CALIBRATION PROCEDURES AND SETPOINT CALCULATION HAVE BEEN REVISED ACCORDINGLY. 2. THE MINIMUM REQUIRED CONDENSATE STORAGE TANK LEVEL HAS BEEN REVISED UPWARD TO MAINTAIN THE REQUIRED CONDENSATE RESERVE OF 135,000 GALLONS. 3. THE LEVEL SETPOINT FOR A SIMILARLY DESIGNED SYSTEM CONFIGURATION HAS BEEN REVIEWED. 4. A CLARIFICATION OF THE ENGINEERING BACKGROUND CONCERNING THESE SETPOINTS SHALL BE INSERTED INTO THE NMP2 TS.

[139] NINE MILE POINT 2 DOCKET 50-410 LER 88-007
SHUTDOWN COOLING ISOLATION OCCURS WHILE VENTING AN INSTRUMENT LINE.
EVENT DATE: 020188 REPORT DATE: 030188 NSSS: GE TYPE: BWR

(NSIC 208463) WHILE IN COLD SHUTDOWN ON FEBRUARY 1, 1988, THE SHUTDOWN COOLING (SDC) SYSTEM ISOLATED ON A FALSE HIGH REACTOR PRESSURE SIGNAL AT 0403 HOURS. DURING THE EVENT REACTOR PRESSURE WAS 0 PSIG AND COOLANT TEMPERATURE WAS 115 F. AN INSTRUMENT AND CONTROLS (I&C) TECHNICIAN WAS BACKFILLING A COMMON SENSING LINE TO THE HIGH REACTOR PRESSURE SENSING TRANSMITTER TO REMOVE ENTRAPPED AIR. THE FILL PUMP DISCHARGE PRESSURE WAS SUFFICIENT TO INCREASE THE SENSING LINE PRESSURE AT THE TRANSMITTER AND EXCEED THE PRESSURE SETPOINT FOR SDC OPERATION. AS A RESULT, PRIMARY CONTAINMENT ISOLATION GROUP 5 VALVES AND SDC ISOLATED ON A HIGH REACTOR PRESSURE SIGNAL (128 PSIG). THE CAUSE OF THE EVENT IS UNKNOWN AT THIS TIME, AND THE INVESTIGATION WILL CONTINUE. IMMEDIATE CORRECTIVE ACTION WAS TO VERIFY THAT AN ALTERNATE METHOD FOR COOLANT CIRCULATION AND DECAY HEAT REMOVAL WAS AVAILABLE. NORMAL SDC WAS RESTORED AT 0600 HOURS. A SUPPLEMENTAL REPORT WILL BE SUBMITTED BY JUNE 2, 1988 THAT DISCUSSES THE RESULT OF THE INVESTIGATION AND INCLUDES THE CAUSE AND ALL PLANNED CORRECTIVE ACTIONS.

[140] NORTH ANNA 1 DOCKET 50-338 LER 87-022 REV 01
UPDATE ON INADVERTENT OPENING OF A PRESSURIZER POWER OPERATED RELIEF VALVE.
EVENT DATE: 101087 REPORT DATE: 022488 NSSS: WE TYPE: PWR

(NSIC 208380) ON OCTOBER 10, 1987, AT 2117 HOURS, WITH UNIT 1 IN MODE 5 (195 DEGREES F AND PRIMARY SYSTEM PRESSURE OF 375 PSIG), A PRESSURIZER POWER OPERATED RELIEF VALVE OPENED DUE TO ACTUATION OF THE OVERPRESSURE PROTECTION SYSTEM. THE EVENT WAS CAUSED BY A BRIEF REDUCTION IN PRIMARY SYSTEM TEMPERATURE DURING THE PERFORMANCE OF AN INSERVICE INSPECTION TEST OF THE RESIDUAL HEAT REMOVAL SYSTEM VALVES. THIS TEMPERATURE REDUCTION AUTOMATICALLY REDUCED THE SETPOINT OF THE OVERPRESSURE PROTECTION SYSTEM. THE REDUCED SETPOINT WAS BELOW THE REACTOR COOLANT SYSTEM PRESSURE AND THE OVERPRESSURE PROTECTION WAS INITIATED WHICH AUTOMATICALLY OPENED AT LEAST ONE POWER OPERATED RELIEF VALVE. THIS EVENT HAD NO IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC. THIS EVENT IS REPORTABLE PURSUANT TO TECHNICAL SPECIFICATION 3.4.9.3, ACTION C, AND SUBMITTED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 6.9.2.

[141] NORTH ANNA 1 DOCKET 50-338 LER 88-002
MANUAL REACTOR TRIP IN ANTICIPATION OF LOSS OF THE MAIN CONDENSER.
EVENT DATE: 010888 REPORT DATE: 020388 NSSS: WE TYPE: PWR

(NSIC 208192) AT 0438 HOURS ON JANUARY 8, 1988, UNIT 1 WAS MANUALLY TRIPPED FROM APPROXIMATELY 100 PERCENT POWER (MODE 1). THE MANUAL TRIP WAS INITIATED IN ANTICIPATION OF LOSS OF THE MAIN CONDENSER AFTER THE THREE RUNNING CIRCULATING WATER (CW) PUMPS TRIPPED SIMULTANEOUSLY AND CONDENSER VACUUM WAS OBSERVED TO BE DECREASING RAPIDLY. FOLLOWING THE REACTOR TRIP, AN ELECTRICAL TROUBLESHOOTING PROCEDURE WAS DEVELOPED AND PERFORMED TO DUPLICATE THE EVENTS IMMEDIATELY PRIOR TO THE SIMULTANEOUS TRIPPING OF THE THREE RUNNING CW PUMPS. NO EQUIPMENT PROBLEMS (RELAYS, TIMERS, ETC.) WERE IDENTIFIED WHICH COULD ACCOUNT FOR THE SIMULTANEOUS TRIPPING OF THE CW PUMPS. ADDITIONAL TESTING WAS PERFORMED ON THE CW PUMP/CONDENSER WATERBOX TRIP INTERLOCK PROTECTION CIRCUITRY IN AN ATTEMPT TO DETERMINE THE POTENTIAL CAUSE FOR THE LOSS OF ALL THE CW PUMPS. AS A RESULT, THE EXACT CAUSE OF THE SIMULTANEOUS TRIPPING OF THE CW PUMPS COULD NOT BE DETERMINED. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE ALL SAFETY RELATED EQUIPMENT FUNCTIONED AS DESIGNED, WITH THE EXCEPTION OF THE STEAM DRIVEN AUXILIARY FEEDWATER PUMP WHICH SUCCESSFULLY STARTED AND OPERATED FOR ABOUT 40 MINUTES BEFORE TRIPPING UNEXPECTEDLY. ALSO, KEY REACTOR PARAMETERS STABILIZED, AS EXPECTED, FOLLOWING THE REACTOR TRIP.

[142] NORTH ANNA 1 DOCKET 50-338 LER 88-005
AUTOMATIC REACTOR TRIP DUE TO HI-HI STEAM GENERATOR LEVEL.
EVENT DATE: 011388 REPORT DATE: 021088 NSSS: WE TYPE: PWR

(NSIC 208266) AT 0313 HOURS ON JANUARY 13, 1988, UNIT 1 AUTOMATICALLY TRIPPED FROM APPROXIMATELY 15 PERCENT POWER (MODE 1). THE INITIATING SIGNAL FOR THIS REACTOR TRIP WAS A TURBINE SOLENOID TRIP WHICH RESULTED WHEN A HI-HI LEVEL (GREATER THAN 75 PERCENT) WAS DETECTED ON 2 OUT OF 3 LEVEL CHANNELS IN THE "B" STEAM GENERATOR (S/G). THE HI-HI LEVEL IN THE "B" S/G RESULTED FROM S/G LEVEL OSCILLATIONS DUE TO THE LOW TEMPERATURE OF THE FW (APPROXIMATELY 81 DEGREES F) ENTERING THE S/G'S. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR 50.73(A)(2)(IV). WHEN A HI-HI LEVEL WAS REACHED IN THE "B" S/G, A TURBINE SOLENOID TRIP OCCURRED WHICH RESULTED IN A TURBINE TRIP, REACTOR TRIP, AND A MAIN FEEDWATER (FW) ISOLATION. AS A RESULT OF THE MAIN FW ISOLATION, THE AUXILIARY FW SYSTEM AUTOMATICALLY STARTED, AS DESIGNED. AS CORRECTIVE ACTIONS, MAIN FW WAS RESTORED AND AUXILIARY FW WAS SECURED AND THEN RESTORED TO THE AUTOMATIC POSITION. TO PREVENT RECURRENCE OF THIS TYPE EVENT, OPERATING PROCEDURE 2.1, UNIT POWER OPERATION MODE 2 TO MODE 1, WILL BE REVISED TO INCLUDE A CAUTION ABOUT FW TEMPERATURE AND INCREASING POWER ABOVE TEN PERCENT WITH S/G LEVEL OSCILLATIONS. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE ALL SAFETY RELATED EQUIPMENT RESPONDED AS DESIGNED AND KEY REACTOR PARAMETERS STABILIZED FOLLOWING THE REACTOR TRIP.

[143] NORTH ANNA 1 DOCKET 50-338 LER 88-007
UNIDENTIFIED FIRE BARRIER PENETRATIONS.
EVENT DATE: 012688 REPORT DATE: 022488 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)

(NSIC 208332) AT 1845 HOURS ON JANUARY 26, 1988, WITH UNIT 1 AT 0 PERCENT POWER (MODE 5) AND UNIT 2 AT 100 PERCENT POWER (MODE 1), IT WAS DISCOVERED THAT A FIRE BARRIER BREACH EXISTED BETWEEN THE QUENCH SPRAY PUMP HOUSE AND THE PIPING TUNNEL TO THE UNIT 1 TURBINE BUILDING. AT 1600 HOURS ON JANUARY 27, 1988 SEVEN ADDITIONAL FIRE BARRIER BREACHES, CONNECTING WITH THE UNIT 1 AND UNIT 2 QUENCH SPRAY PUMP HOUSES, WERE DISCOVERED THROUGH SUPPLEMENTARY WALKDOWNS. TECHNICAL SPECIFICATION 3.7.15 REQUIRES ALL FIRE BARRIER PENETRATIONS PROTECTING SAFETY RELATED AREAS TO BE FUNCTIONAL. SINCE THESE BARRIERS WERE NOT FUNCTIONAL, FIRE WATCHES WERE ESTABLISHED IN COMPLIANCE WITH TECHNICAL SPECIFICATIONS. THIS EVENT

IS REPORTABLE PURSUANT TO 10CFR50.73 (A)(2)(I). AS A CORRECTIVE ACTION, THE BREACHED FIRE BARRIER PENETRATIONS HAVE BEEN SEALED IN ACCORDANCE WITH STATION DESIGN REQUIREMENTS. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED AT ANY TIME DURING THIS EVENT.

[144] NORTH ANNA 1 DOCKET 50-338 LER 88-008
INADVERTENT PORV ACTUATION DURING SOLID WATER OPERATIONS.
EVENT DATE: 020288 REPORT DATE: 022488 NSSS: WE TYPE: PWR

(NSIC 208333) ON FEBRUARY 2, 1988, UNIT 1 WAS IN MODE 5 PREPARING TO RETURN TO POWER OPERATIONS FOLLOWING A MAINTENANCE OUTAGE. REACTOR COOLANT SYSTEM (RCS) PRIMARY TEMPERATURE AND PRESSURE WERE APPROXIMATELY 110 DEGREES F AND 320 PSIG, RESPECTIVELY. THE PRIMARY SYSTEM WAS IN SOLID WATER OPERATION AND THE "B" REACTOR COOLANT PUMP (RCP) WAS RUNNING. AT 0252 HOURS THE "A" RCP WAS STARTED AND AS A RESULT, ONE PRESSURIZER POWER OPERATED RELIEF VALVE, PCV-1445C, (PORV) MOMENTARILY LIFTED. THIS EVENT IS REPORTABLE AS A SPECIAL REPORT PURSUANT TO TECHNICAL SPECIFICATION 3.4.9.3, ACTION C. THIS EVENT WAS A RESULT OF A PRESSURE TRANSIENT WHICH OCCURRED WHEN A SECOND RCP WAS STARTED WHILE OPERATING CLOSE TO THE PORV SETPOINT. OPERATION NEAR THE PORV SETPOINT WAS NECESSARY TO MAINTAIN THE REQUIRED DIFFERENTIAL PRESSURE OF THE #1 "A" RCP SEAL ABOVE 200 PSID. NO SIGNIFICANT SAFETY CONSEQUENCES EXISTED DURING THIS EVENT BECAUSE PCV-1455C RESPONDED AS EXPECTED TO PROVIDE OVERPRESSURE PROTECTION AT LOW RCS TEMPERATURES. THE OPENING OF PCV-1455C AND IMMEDIATE OPERATOR ACTION RAPIDLY REDUCED THE RCS OVERPRESSURE CONDITION. THE LOW TEMPERATURE, LOW PRESSURE SETPOINT FOR PCV-1455C IS 345 PSIG WITH PRIMARY SYSTEM TEMPERATURE LESS THAN 185 DEGREES F. THE REDUNDANT PORV, PCV-1456, WILL OPEN AT 350 PSIG; HOWEVER, PRESSURE WAS NEVER HIGH ENOUGH FOR IT TO OPEN. THE HEALTH AND SAFETY OF THE GENERAL PUBLIC WERE NOT AFFECTED.

[145] NORTH ANNA 1 DOCKET 50-338 LER 88-010
INADVERTENT PORV ACTUATION DURING SOLID WATER OPERATIONS.
EVENT DATE: 021088 REPORT DATE: 022388 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)
VENDOR: KAMAN SCIENCES CORP.

(NSIC 208334) AT 0659 HOURS ON 2/10/88, WITH UNITS 1 AND 2 IN MODE 1 AT 30 AND 100 PERCENT POWER, RESPECTIVELY, THE KAMAN VENT STACK "B" RADIATION MONITOR, RI-VG-180, WAS DECLARED INOPERABLE DUE TO HIGH INDICATED RADIATION LEVELS. TWO ADDITIONAL RADIATION MONITORS IN THE SAME EFFLUENT RELEASE PATH WERE INDICATING NORMAL RADIATION LEVELS DURING THIS PERIOD. ACTION STATEMENT 35 OF TECH SPEC 3.3.3.1 REQUIRES THAT THE RADIATION MONITOR BE RETURNED TO OPERABLE STATUS WITHIN 72 HOURS, OR INITIATE THE PREPLANNED ALTERNATE METHOD OF MONITORING AND PREPARE A SPECIAL REPORT. SINCE THIS ACTION STATEMENT EXPIRED AT 0659 HOURS ON 2/10/88, WITH THE RADIATION MONITOR STILL DECLARED INOPERABLE, THIS EVENT IS REPORTABLE PURSUANT TO TECH SPEC 6.9.2. INVESTIGATION INTO THE CAUSE FOR THE ERRONEOUS INDICATION OF RI-VG-180 REVEALED THAT THE CENTRAL PROCESSING UNIT (CPU) BOARD HAD MALFUNCTIONED. AN ANALYSIS WILL BE PERFORMED TO DETERMINE THE ROOT CAUSE OF THE BOARD FAILURE. THE CPU BOARD WAS REPLACED AND A PENDING RELIABILITY UPGRADE MODIFICATION WAS INITIATED. THE RADIATION MONITOR WAS RETURNED TO OPERABLE STATUS ON 2/19/88, AFTER THE UPGRADE MODIFICATION, FUNCTIONAL TESTING AND THE MONTHLY PERIODIC TEST WERE COMPLETED WITH SATISFACTORY RESULTS. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE THERE WERE BACKUP RADIATION MONITORS.

[146] OCONEE 1 DOCKET 50-269 LER 87-012
TECHNICAL SPECIFICATION VIOLATION DUE TO AN EXCEEDED INSERVICE INSPECTION INTERVAL RESULTING FROM A QUALITY ASSURANCE DEFICIENCY.
EVENT DATE: 120387 REPORT DATE: 020588 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)

OCONEE 3 (PWR)

(NSIC 208249) ON DECEMBER 3, 1987 DURING A PROCEDURE IMPLEMENTATION REVIEW IT WAS DISCOVERED THAT THE EXAMINATION FREQUENCY OF THE REACTOR COOLANT PUMP (RCP) FLYWHEELS HAD EXCEEDED THE REQUIREMENTS OF TECHNICAL SPECIFICATION 4.2.3. THIS SPECIFICATION STATES THE FREQUENCY OF INSPECTION FOR THE RCP FLYWHEELS TO BE APPROXIMATELY THREE YEARS. UPON REVIEW OF THIS INCIDENT, IT WAS DISCOVERED THAT THE REQUIRED FREQUENCY WAS VIOLATED FOUR TIMES INVOLVING ALL THREE UNITS. DURING EACH INSPECTION, THE UNIT BEING INSPECTED WAS SHUTDOWN FOR REFUELING. THE ROOT CAUSE OF THIS INCIDENT WAS DETERMINED TO BE A QUALITY ASSURANCE DEFICIENCY BECAUSE QA-OPERATIONS INTERPRETED THAT THE REQUIRED THREE YEAR INSPECTION PERIOD ALLOWED A TIME PERIOD OF 3 TO 5 YEARS BETWEEN INSPECTIONS. THIS EXCEEDED THE INTERVAL INTENDED BY SPECIFICATION 4.2.3 AND ASME SECTION XI. THE IMMEDIATE CORRECTIVE ACTION WAS TO REPORT THE VIOLATION TO QA-OPERATIONS. SUBSEQUENT CORRECTIVE ACTIONS INVOLVED AN EVALUATION OF PAST INSPECTION RESULTS TO ENSURE THE RCP FLYWHEEL HAD BEEN INSPECTED WITHIN THE REQUIREMENTS ESTABLISHED BY SPECIFICATION 4.2.3. PLANNED CORRECTIVE ACTIONS INCLUDE PREPARATION OF A TECHNICAL SPECIFICATION INTERPRETATION AND REVISION TO INSERVICE INSPECTION PLANS.

[147] OYSTER CREEK DOCKET 50-219 LER 83-024 REV 01
UPDATE ON TORQUE SWITCH SETPOINTS ON LIMITORQUE MOTOR OPERATED VALVES SET BELOW MANUFACTURER'S RECOMMENDATIONS.
EVENT DATE: 122083 REPORT DATE: 022688 NSSS: GE TYPE: BWR
VENDOR: LIMITORQUE CORP.

(NSIC 208471) AFTER A REVIEW OF HISTORICAL DATA ON TORQUE SWITCH SETPOINTS OF THE LIMITORQUE MOTOR OPERATED VALVES, IT WAS DISCOVERED THAT THE TORQUE SWITCHES WERE SET BELOW THE MANUFACTURERS RECOMMENDED SETPOINT. ALTHOUGH THIS IS STILL UNDER EVALUATION, THIS OCCURRENCE MAY HAVE PREVENTED THE VALVES FROM OPERATING DURING ACCIDENT CONDITIONS. THIS EVENT IS CONSIDERED TO BE A REPORTABLE OCCURRENCE AS DEFINED IN THE TECH SPEC, PARAGRAPH 6.9.2.A.9. FOLLOW UP REPORT ON ORIGINAL DESIGN OF LIMITORQUE TORQUE SWITCH SETPOINTS, RECALCULATION FOR DESIRED VALVES, RESETTING OF THE SWITCHES, AND ADMINISTRATIVE CONTROLS TO PRECLUDE RECURRENCE.

[148] OYSTER CREEK DOCKET 50-219 LER 86-009 REV 01
UPDATE ON SCRAM SIGNAL RECEIVED DUE TO NEUTRON INSTRUMENTATION NOISE.
EVENT DATE: 050186 REPORT DATE: 030488 NSSS: GE TYPE: BWR

(NSIC 208419) ON MAY 1, 1986, AT APPROXIMATELY 1108 HOURS A FULL SCRAM SIGNAL WAS RECEIVED. THE CAUSE OF THE SCRAM SIGNAL WAS ELECTRONIC NOISE INDUCED SPIKES ON THE INTERMEDIATE RANCE MONITORS (IRMS) OF THE NUCLEAR-INSTRUMENTATION. PRIOR TO THIS EVENT, THE REACTOR WAS IN THE REFUEL MODE WITH ALL CONTROL RODS FULLY INSERTED, OR AT THE FULL OUT POSITION WITH THE DRIVE UNIT VALVED OUT AND CONTROL CELL UNLOADED. A POST TRIP REVIEW WAS PERFORMED ON THIS EVENT WITH THE FOLLOWING OUTCOME; ALL PLANT SYSTEMS RESPONDED AS EXPECTED FOR SIGNALS RECEIVED AND CONTROL ROOM PERSONNEL RESPONDED PROPERLY AND IN ACCORDANCE WITH APPROPRIATE PROCEDURES. THE CAUSE OF THE NOISE SPIKES ON THE IRMS HAS BEEN DETERMINED. THE NOISE SPIKES ARE GENERATED DURING SCHEDULED UNDER-VESSEL MAINTENANCE DUE TO THE MANIPULATION OF IRM CABLES AND CONNECTORS. NOISE SPIKES ON THE IRMS HAVE BEEN A PROBLEM IN THE PAST AND ARE MORE FREQUENT AND SEVERE WHEN SHUTDOWN WITH INCREASED MAINTENANCE ACTIVITIES. OCCASIONAL SCRAMS DUE TO UNDER-VESSEL MAINTENANCE CAN BE EXPECTED AND THEIR POSSIBLE OCCURRENCE SHOULD BE RECOGNIZED PRIOR TO AUTHORIZING THIS WORK. PROCEDURES WILL BE REVIEWED TO ENSURE APPROPRIATE PRECAUTIONS EXIST TO ALERT WORKERS TO THE POTENTIAL EFFECTS OF MOVEMENT OF THE IRM CABLES.

[149] OYSTER CREEK DOCKET 50-219 LER 86-035 REV 01
UPDATE ON CONTAINMENT PENETRATION FOUND DEGRADED DUE TO ISOLATION VALVES ACTUATOR/VALVE LINKAGES OUT OF ADJUSTMENT.
EVENT DATE: 123186 REPORT DATE: 022588 NSSS: GE TYPE: BWR

(NSIC 208302) AFTER EXPERIENCING PROBLEMS VENTING THE TORUS FOLLOWING A SHUTDOWN, A DIAGNOSTIC LEAK TEST WAS PERFORMED ON TWO CONTAINMENT ISOLATION VALVES. THE RESULTS OF THE LEAK TEST PERFORMED ON 12/31/86, REVEALED THAT BOTH VALVES WERE LEAKING EXCESSIVELY. THE MAIN CONTRIBUTOR TO THE LEAKAGE EXPERIENCED WAS THE LINKAGE BETWEEN THE ACTUATOR AND VALVE BEING OUT OF ADJUSTMENT. MAINTENANCE PERSONNEL REBUILT AND ADJUSTED THE VALVE/ACTUATOR LINKAGE ON ONE VALVE AND ADJUSTED THE STEM/ACTUATOR COUPLING ON THE OTHER VALVE. FOLLOWING MAINTENANCE A LOCAL LEAK RATE TEST WAS PERFORMED AND TEST RESULTS WERE ACCEPTABLE. THE CAUSES OF THIS EVENT ARE AS FOLLOWS: A POST MAINTENANCE LOCAL LEAK RATE TEST (LLRT) PERFORMED 9/9/86 FOR BOTH VALVES WAS NOT VALID AS IT DID NOT IDENTIFY EXISTING LEAKAGE FOUND IN SUBSEQUENT LEAK RATE TESTS; AND A VALVE HUB SEAL WAS TIGHTENED BEYOND VENDOR RECOMMENDED LIMITS AND WITHOUT A POST MAINTENANCE LLRT DURING THE 10/27/86 PRIMARY CONTAINMENT INTEGRATED LEAK RATE TEST BECAUSE SPECIFIC INSTRUCTIONS AND A LLRT WERE NOT SPECIFIED ON THE STANDING WORK ORDER USED TO PERFORM THIS MAINTENANCE. THESE TWO DEFICIENCIES RESULTED IN A DEGRADED PENETRATION. TO PREVENT RECURRENCE OF THIS EVENT ADDITIONAL CHECKS WILL BE PERFORMED TO ASSURE THE TEST VOLUME IS VERIFIED TO BE PRESSURIZED FOR LLRTS.

[150] OYSTER CREEK DOCKET 50-219 LER 87-041
FUEL POOL COOLING SYSTEM PIPING RADIATION SHIELDING INADVERTENTLY REMOVED RESULTING IN AN UNLOCKED HIGH RADIATION AREA DUE TO PERSONNEL ERROR.
EVENT DATE: 111987 REPORT DATE: 020188 NSSS: GE TYPE: BWR

(NSIC 208253) ON NOVEMBER 18, 1987 AT APPROXIMATELY 0000 HOURS A RADIOLOGICAL CONTROL TECHNICIAN (RCT) INADVERTENTLY REMOVED FUEL POOL COOLING SYSTEM PIPING RADIATION SHIELDING WHICH WAS REQUIRED TO MAINTAIN AREA DOSE RATE LEVELS BELOW THE TECHNICAL SPECIFICATION LIMIT FOR LOCKED, CONTROLLED ACCESS AREAS (1000 MREM/HR). AN AREA RADIATION SURVEY PERFORMED AFTER THE SHIELDING WAS REMOVED WAS INADEQUATE, RESULTING IN AN AREA WITH A DOSE RATE EXCEEDING 1000 MREM/HR BEING LEFT UNLOCKED AND UNGUARDED. AT THE TIME OF THIS OCCURRENCE THE PLANT WAS SHUT DOWN, DEPRESSURIZED, AND IN THE REFUEL MODE. THE ROOT CAUSE OF THIS EVENT IS PERSONNEL ERROR IN THAT THE RCT REMOVED SHIELDING WITHOUT PERFORMING AN ADEQUATE SURVEY TO DETERMINE RADIOLOGICAL CONDITIONS. FAILURE TO REVIEW THE SHIELDING AUTHORIZATION AND INADEQUATE LABELING OF THE SHIELDING CONTRIBUTED TO THE CAUSE OF THIS EVENT. AFTER THIS CONDITION WAS DISCOVERED, SHIELDING WAS INSTALLED TO LOWER THE AREA DOSE RATE BELOW 1000 MREM/HR. THIS LER AND A CRITIQUE OF THIS EVENT IS REQUIRED READING FOR ALL PERSONNEL INVOLVED WITH RADIATION SHIELDING INSTALLATION OR REMOVAL.

[151] PALISADES DOCKET 50-255 LER 87-039 REV 01
UPDATE ON POTENTIAL FOR OPERATION OUTSIDE OF DESIGN BASIS WITH RESPECT TO MSLB ANALYSIS.
EVENT DATE: 103087 REPORT DATE: 021688 NSSS: CE TYPE: PWR

(NSIC 208287) DURING EFFORTS TO CLOSE OUT AN NRC OPEN ITEM IDENTIFIED THROUGH THE PALISADES SYSTEM FUNCTIONAL EVALUATION (SFE) PROGRAM, IT WAS DETERMINED THAT CHARGING PUMP P-55B (CB;P) WOULD NOT AUTOMATICALLY ACTUATE UPON A PRESSURIZER (AB;PZR) LOW LEVEL SIGNAL WITH COINCIDENT SAFETY INJECTION SIGNAL (SIS) AS PREVIOUSLY THROUGHT. THIS DISCOVERY RESULTED IN THE POTENTIAL FOR PAST PLANT OPERATION OUTSIDE OF ITS DESIGN BASIS AS DESCRIBED IN SECTION 14.14, "STEAM LINE RUPTURE INCIDENT" OF THE PALISADES FINAL SAFETY ANALYSIS REPORT WHILE OPERATING WITHIN CURRENT PLANT TECH SPEC. THE PLANT WAS IN COLD SHUTDOWN CONDITION WHEN THIS ITEM WAS IDENTIFIED. THE MAXIMUM CHARGING FLOW RATE CALLED FOR IN EXISTING MSLB - FSAR ANALYSES IS 68 GALLONS PER MINUTE. THIS IS EQUIVALENT TO TWO

CHARGING PUMP FLOW. ANALYSES COMPLETED BY SAFETY ANALYSIS AND VENDOR PERSONNEL HAVE DEMONSTRATED THAT SCENARIOS CALLING FOR 68 GALLONS PER MINUTE WILL REMAIN BOUNDED BY CURRENT ANALYSES IF CHARGING FLOW WAS REDUCED TO ONE PUMP EQUIVALENT FLOW OF 34 GALLONS PER MINUTE. ADMINISTRATIVE CONTROLS HAVE BEEN IMPLEMENTED VIA AN OPERATIONS DEPARTMENT STANDING ORDER WHICH REQUIRE ONE CHARGING PUMP BE MAINTAINED IN AN OPERABLE STATUS ON EACH SAFETY RELATED BUS (EB;BU) UNLESS A 24 HOUR LIMITING CONDITION OF OPERATION IS ENTERED.

[152] PALO VERDE 1 DOCKET 50-528 LER 87-018 REV 01
UPDATE ON REACTOR TRIP OCCURS DURING SHUTDOWN DUE TO PRESSURE BOUNDARY LEAKAGE.
EVENT DATE: 082787 REPORT DATE: 021988 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 2 (PWR)
PALO VERDE 3 (PWR)

(NSIC 208445) THIS IS A SUPPLEMENT TO LER 87-018-00. AT APPROXIMATELY 2037 MST ON AUGUST 27, 1987 PALO VERDE UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 8 PERCENT REACTOR POWER WHEN A REACTOR TRIP OCCURRED DUE TO THE AXIAL SHAPE INDEX OF THE REACTOR CORE. THE TRIP OCCURRED AS THE UNIT WAS SHUTTING DOWN TO EVALUATE A POSSIBLE REACTOR COOLANT SYSTEM (RCS) PRESSURE BOUNDARY LEAK. THE PRESENCE OF A PRESSURE BOUNDARY LEAK WAS CONFIRMED AFTER THE UNIT WAS STABILIZED IN MODE 3 (HOT STANDBY). THE ROOT CAUSE OF THE REACTOR TRIP WAS A DEFICIENT PROCEDURE. TO PREVENT RECURRENCE PROCEDURAL ENHANCEMENTS HAVE BEEN IMPLEMENTED. THE CAUSE OF THE LEAK WAS A CRACKED SOCKET WELD ON THE UPSTREAM SIDE OF THE ISOLATION VALVE FOR THE FLANGED REFUELING WATER LEVEL INDICATION. THE ROOT CAUSE FOR THE CRACKED WELD WAS FATIGUE FAILURE, INDUCED IN PART BY CYCLIC LOADING. AS CORRECTIVE ACTION THE VALVE WAS CUT OUT AND REPLACED, AND PIPE SUPPORTS WERE ADDED TO THIS PIPE AND TO FIVE ADDITIONAL PIPES OF SIMILAR CONFIGURATION TO REDUCE CYCLIC LOADING.

[153] PALO VERDE 1 DOCKET 50-528 LER 88-001
PERSONNEL ERROR RESULTS IN LATE SPECIAL REPORT SUBMITTAL.
EVENT DATE: 010488 REPORT DATE: 020388 NSSS: CE TYPE: PWR
VENDOR: TERRA TECHNOLOGY CORP.

(NSIC 208216) THIS LER INCLUDES INFORMATION FOR SPECIAL REPORT 1-SR-87-026 PREPARED PURSUANT TO TECHNICAL SPECIFICATION 3.3.3.3 ACTION "A" AND 6.9.2. AT APPROXIMATELY 0830 EST ON JANUARY 4, 1988 PALO VERDE UNIT 1 WAS IN MODE 5 (COLD SHUTDOWN) WHEN IT WAS DETERMINED THAT A SPECIAL REPORT REQUIRED PURSUANT TO TECHNICAL SPECIFICATION 3.3.3.3 HAD NOT BEEN SUBMITTED WITHIN THE TIME FRAME PRESCRIBED. THE SPECIAL REPORT WAS REQUIRED TO REPORT THE SEISMIC MONITORING SYSTEM (IN) BEING INOPERABLE FOR GREATER THAN 30 DAYS. THE MONITOR'S MALFUNCTIONING SEISMIC SWITCH (VIS) WAS IDENTIFIED AS BEING INOPERABLE DURING SURVEILLANCE TESTING. THERE WERE NO OTHER COMPONENT OR SYSTEM MALFUNCTIONS THAT CONTRIBUTED TO THE EVENT. THERE WERE NO SAFETY RESPONSES AND NONE WERE NECESSARY. THE ROOT CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE A COGNITIVE PERSONNEL ERROR IN THAT A REPORTABILITY EVALUATION DOCUMENT WAS NOT PREPARED WHEN THE SEISMIC MONITORING SYSTEM WAS DECLARED INOPERABLE FOR REQUIRED SURVEILLANCE TESTING. AS CORRECTIVE ACTION TO PREVENT RECURRENCE, THIS EVENT WILL BE REVIEWED BY APPROPRIATE PERSONNEL. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED.

[154] PALO VERDE 1 DOCKET 50-528 LER 88-003
ESFAS ACTUATIONS FROM A LOSS OF POWER.
EVENT DATE: 011688 REPORT DATE: 021688 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 2 (PWR)
VENDOR: CALVERT COMPANY INC., THE
GENERAL ELECTRIC CO.

(NSIC 208285) AT APPROXIMATELY 0100 MST ON JANUARY 16, 1988, PALO VERDE UNIT 1

WAS IN MODE 5 (COLD SHUTDOWN) AT APPROXIMATELY 107F, 50 PSIA AND ON SHUTDOWN COOLING TRAIN A. PALO VERDE UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 100 PERCENT POWER WHEN A LOSS OF POWER OCCURRED IN UNITS 1 AND 2. STARTUP TRANSFORMER AE-NAN-X03 SHED ITS UNIT 1 AND 2 LOADS DUE TO A FAULT IN A CURRENT TRANSFORMER AND IN UNIT 1 13.8 KV BUS TE-NAN-S0L. DIESEL GENERATOR A STARTED IN UNIT 1 AND DIESEL GENERATOR B STARTED IN UNIT 2. BOTH DIESEL GENERATORS SUPPLIED POWER TO THEIR RESPECTIVE 4.16 KV CLASS 1E BUSSES. DUE TO THE LOSS OF POWER TO RADIATION MONITORS IN UNIT 1 THE FOLLOWING ENGINEERED SAFETY FEATURE ACTUATION SYSTEM ACTUATIONS OCCURRED: CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNAL, CONTAINMENT PURGE ISOLATION ACTUATION SIGNAL AND FUEL BUILDING ESSENTIAL VENTILATION ACTUATION SIGNAL. ADDITIONALLY HALF LEG TRIPS WERE RECEIVED ON CONTAINMENT ISOLATION ACTUATION SIGNAL, SAFETY INJECTION ACTUATION SIGNAL, CONTAINMENT SPRAY ACTUATION SIGNAL AND RECIRCULATION ACTUATION SIGNAL. THE ROOT CAUSE WAS DETERMINED TO BE A FAULT IN A 13.8 KV BUS CONCURRENT WITH A FAULT IN A CURRENT TRANSFORMER WHICH ISOLATED STARTUP TRANSFORMER AE-NAN-X03. AS CORRECTIVE ACTION THE CURRENT TRANSFORMER WAS REPLACED AND THE FAULT CORRECTED IN THE 13.8 KV BUS.

[155] PALO VERDE 1 DOCKET 50-528 LER 88-005
BROKEN POST ISOLATION VALVE (PIV) RENDERS FIRE WATER SUPPRESSION LOOP INOPERABLE DUE TO CONTINOUS FIRE WATCHES WERE LATE.
EVENT DATE: 012388 REPORT DATE: 021988 NSSS: CE TYPE: PWR

(NSIC 208370) AT APPROXIMATELY 1530 MST ON JANUARY 23, 1988, WITH PALO VERDE UNIT 1 IN MODE 5 (COLD SHUTDOWN), A BACKHOE OPERATOR (UTILITY, NON-LICENSED) INADVERTENTLY DAMAGED A POST INDICATION VALVE (PIV)(KP)(ISV) AND CAUSED FIRE SUPPRESSION WATER TO DISCHARGE INTO THE EAST SIDE OF THE UNIT 1 PROTECTED AREA. FIRE PROTECTION PERSONNEL (UTILITY, NON-LICENSED) WERE UNABLE TO ISOLATE THE LEAK DUE TO PERSONNEL ERROR, CONTRIBUTED TO BY AN INADEQUATE PROCEDURE, IN THAT THE PIVS NEEDED TO ISOLATE THE LEAK WERE NOT CORRECTLY IDENTIFIED AND CLOSED. AS A RESULT THE VALVE LINEUP USED TO ISOLATE THE UNIT 1 FIRE WATER SUPPRESSION LOOP TO EFFECT REPAIRS FOR THE DAMAGED VALVE, THE UNIT 1 SPRINKLER/DELUGE SYSTEMS, HOSE STATIONS AND FIRE HYDRANTS WERE RENDERED INOPERABLE. A COGNITIVE PERSONNEL ERROR CONTRARY TO APPROVED PROCEDURE ALSO OCCURRED, IN WHICH HOURLY FIRE WATCHES WERE UTILIZED IN LIEU OF REQUIRED CONTINUOUS FIRE WATCHES, FOR APPROXIMATELY 4 HOURS. AS CORRECTIVE ACTION, PROCEDURAL CONTROLS WILL BE IMPLEMENTED TO PROVIDE ADDITIONAL GUIDANCE FOR ISOLATING SELECTED PORTIONS OF THE FIRE WATER SUPPRESSION SYSTEM, AND THE REQUIREMENTS FOR ESTABLISHING CONTINUOUS FIRE WATCHES WILL BE REVIEWED BY APPROPRIATE FIRE PROTECTION PERSONNEL. NO SIMILAR EVENTS HAVE BEEN REPORTED.

[156] PALO VERDE 2 DOCKET 50-529 LER 88-001
PERSONNEL ERROR RESULTS IN INCOMPLETE TECHNICAL SPECIFICATION SAMPLE ANALYSIS.
EVENT DATE: 010688 REPORT DATE: 012588 NSSS: CE TYPE: PWR

(NSIC 208217) AT APPROXIMATELY 0924 MST ON JANUARY 6, 1988, PALO VERDE UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT 100% POWER WHEN IT WAS DETERMINED THAT A WEEKLY CONDENSER VACUUM PUMP/GLAND SEAL EXHAUST RADIATION MONITOR (SH)(IL)(RI) SAMPLE HAD NOT BEEN RETAINED FOR INCLUSION WITH THE QUARTERLY COMPOSITE STRONTIUM - 89/90 SAMPLE. THIS RESULTED IN AN INABILITY TO FULLY MEET THE SAMPLING REQUIREMENTS OF TECHNICAL SPECIFICATION 4.11.2.1.2. THE ROOT CAUSE OF THIS EVENT WAS A PERSONNEL ERROR IN THAT INADEQUATE MEASURES WERE ESTABLISHED TO PROVIDE POSITIVE CONTROLS OVER THE SAMPLES WHILE THEY WERE BEING STORED AWAITING SHIPPING FOR OFF-SITE ANALYSIS. PROCEDURAL CONTROLS WERE EVALUATED AND DETERMINED TO BE ADEQUATE. AS CORRECTIVE ACTION TO PREVENT RECURRENCE, ADDITIONAL CONTROLS FOR TRACKING AND STORING SAMPLES WILL BE IMPLEMENTED. UNIT CHEMISTRY DEPARTMENT PERSONNEL WILL RECEIVE APPROPRIATE TRAINING. SIMILAR EVENTS WERE REPORTED IN UNIT 1 LER'S 86-007-00 AND 87-028-00.

[157] PALO VERDE 2 DOCKET 50-529 LER 88-002
ASME SECTION XI SURVEILLANCE TEST INTERVAL EXCEEDED.
EVENT DATE: 011688 REPORT DATE: 021688 NSSS: CE TYPE: PWR

(NSIC 208218) ON JANUARY 18, 1988 AT APPROXIMATELY 1354 MST, PALO VERDE UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER WHEN IT WAS DETERMINED THAT ASME SECTION XI SURVEILLANCE TESTING HAD NOT BEEN CONDUCTED WITHIN ALLOWABLE TIME CONSTRAINTS ON TWO CONTAINMENT POWER ACCESS ISOLATION VALVES (VA)(ISV). PURSUANT TO TECHNICAL SPECIFICATION 4.0.2 THE VALVES WERE REQUIRED TO HAVE BEEN TESTED BY JANUARY 16, 1988. THE VALVES WERE SATISFACTORILY SURVEILLANCE TESTED BY 0137 MST ON JANUARY 19, 1988. THE ROOT CAUSE WAS INADEQUATE ADMINISTRATIVE CONTROLS ESTABLISHED FOR THE TRACKING AND COMPLETION OF PARTIALLY COMPLETED SURVEILLANCE TESTS. AS CORRECTIVE ACTION TO PREVENT RECURRENCE, APPROPRIATE GUIDANCE WILL BE IMPLEMENTED TO REQUIRE APPROPRIATE TRACKING OF PARTIALLY COMPLETED SURVEILLANCE TESTS. THERE HAVE BEEN NO PREVIOUS LER'S SUBMITTED INVOLVING THE IDENTIFIED ROOT CAUSE.

[158] PALO VERDE 2 DOCKET 50-529 LER 88-003
TECHNICAL SPECIFICATION ACTION REQUIREMENT PERFORMED LATE DUE TO PERSONNEL ERROR.
EVENT DATE: 012088 REPORT DATE: 021688 NSSS: CE TYPE: PWR

(NSIC 208286) AT 0958 MST ON JANUARY 20, 1988, PALO VERDE UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER WHEN EMERGENCY DIESEL GENERATOR "A" (DG) WAS DECLARED INOPERABLE FOR PREPLANNED PREVENTIVE MAINTENANCE. THE DIESEL GENERATOR WAS RETURNED TO OPERABLE STATUS AT 1107 MST ON JANUARY 20, 1988. AT APPROXIMATELY 1200 MST ON JANUARY 20, 1988 IT WAS IDENTIFIED THAT TECHNICAL SPECIFICATION 3.8.1.1 ACTION B WAS NOT COMPLIED WITH. THE OPERABILITY OF THE A.C. OFFSITE SOURCES WAS NOT VERIFIED WITHIN 1 HOUR AFTER DECLARING THE DIESEL GENERATOR INOPERABLE. THE SURVEILLANCE WAS SATISFACTORILY COMPLETED AT 1223 MST ON JANUARY 20, 1988. THE ROOT CAUSE OF THE EVENT WAS COGNITIVE PERSONNEL ERROR ON THE PART OF THE CONTROL ROOM PERSONNEL (UTILITY, LICENSED) IN THAT THEY DID NOT PERFORM THE SPECIFIED ACTION REQUIREMENT WITHIN 1 HOUR. AS CORRECTIVE ACTION APPROPRIATE DISCIPLINARY ACTION WAS ADMINISTERED. PREVIOUS SIMILAR EVENTS WERE REPORTED IN PALO VERDE UNIT 1 LER'S: 85-054-00, 85-072-00 AND 86-001-00.

[159] PALO VERDE 2 DOCKET 50-529 LER 88-005
INADVERTENT SAFETY INJECTION RESULTING FROM PERSONNEL ERROR.
EVENT DATE: 022188 REPORT DATE: 022688 NSSS: CE TYPE: PWR
VENDOR: BORG-WARNER CORP.

(NSIC 208371) ON FEBRUARY 21, 1988, PALO VERDE UNIT 2 WAS IN MODE 5 (COLD SHUTDOWN) AT APPROXIMATELY 170F AND 125 PSIA BEING COOLED-DOWN AND DEPRESSURIZED TO BEGIN A REFUELING OUTAGE. AT APPROXIMATELY 0719 MST AN INADVERTENT SAFETY INJECTION (JE) FROM THE SAFETY INJECTION TANKS (BP)(ACC) OCCURRED AS A RESULT OF LOW PRESSURIZER PRESSURE SIGNALS NOT BEING PROPERLY BYPASSED. THE SAFETY INJECTION WAS ACCOMPANIED BY A CONTAINMENT ISOLATION (BP)(JE) ENGINEERED SAFETY FEATURES (ESF) ACTUATION. THERE WERE NO OTHER ESF ACTUATIONS AND NONE WERE NECESSARY. DURING THE EVENT A HIGH PRESSURE SAFETY INJECTION (HPSI) VALVE (INV) DID NOT FULLY OPEN. ALL OTHER EQUIPMENT OPERATED PER DESIGN. THE ROOT CAUSE OF THE EVENT WAS A COGNITIVE PERSONNEL ERROR ON THE PART OF UTILITY, LICENSED PERSONNEL. ADDITIONALLY DURING THE EVENT, THE HPSI LOOP INJECTION VALVE DID NOT OPEN DUE TO A BLOWN FUSE (FU). AS CORRECTIVE ACTION, APPROPRIATE DISCIPLINARY MEASURES WILL BE TAKEN. THE HPSI LOOP INJECTION VALVE WAS VERIFIED TO OPERATE PROPERLY AFTER REPLACING THE MALFUNCTIONING FUSE. A ROOT CAUSE OF FAILURE HAS BEEN INITIATED FOR THE BLOWN FUSE. THERE HAVE BEEN NO PREVIOUS SIMILAR OCCURRENCES.

[160] PALO VERDE 3 DOCKET 50-530 LER 88-001
POST ACCIDENT MONITORING CONTAINMENT PURGE INDICATOR FOUND ISOLATED.
EVENT DATE: 020488 REPORT DATE: 030488 NSSS: CE TYPE: PWR

(NSIC 208469) AT APPROXIMATELY 1847 MST ON FEBRUARY 4, 1988, PALO VERDE UNIT 3 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 100 PERCENT POWER WHEN IT WAS DISCOVERED THAT AN ISOLATION VALVE (ISV) IN A CONTAINMENT (CTMT) PRESSURE TRANSMITTER (PT) SENSING LINE WAS SHUT. THE PRESSURE TRANSMITTER PROVIDES AN INPUT INTO INSTRUMENTATION UTILIZED FOR POST ACCIDENT MONITORING OF CONTAINMENT PRESSURE PURSUANT TO REG. GUIDE 1.97. THE TRANSMITTER IS REQUIRED TO BE OPERABLE TO SATISFY THE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.3.3.6. UPON DISCOVERY, THE APPROPRIATE ACTION REQUIREMENT WAS ENTERED, THE PRESSURE TRANSMITTER PROPERLY ALIGNED AND RETURNED TO SERVICE. AN INVESTIGATION WAS CONDUCTED TO DETERMINE WHEN THE PRESSURE TRANSMITTER MAY HAVE BEEN ISOLATED. BASED UPON THE INVESTIGATION, IT COULD NOT BE DETERMINED WHEN THE TRANSMITTER WAS ISOLATED. IT WAS VERIFIED THAT THE TRANSMITTER WAS PROPERLY ALIGNED AND CALIBRATED AS OF DECEMBER 1, 1987. AS CORRECTIVE ACTION TO PREVENT RECURRENCE, THIS EVENT WILL BE REVIEWED BY RESPONSIBLE MAINTENANCE DEPARTMENT PERSONNEL. A SIMILAR EVENT OCCURRED AS REPORTED IN UNIT 1 LER 85-050-00.

[161] PEACH BOTTOM 2 DOCKET 50-277 LER 87-027
PRIMARY CONTAINMENT ISOLATION DUE TO PERSONNEL ERROR WHILE APPLYING BLOCKING PERMIT.
EVENT DATE: 043087 REPORT DATE: 011288 NSSS: GE TYPE: BWR

(NSIC 207882) ON APRIL 30, 1987 AT 1405 HOURS WITH THE CORE OFFLOADED, A GROUP III OUTBOARD ISOLATION OCCURRED. A GROUP III ISOLATION INVOLVES THE VENTILATION SYSTEM OF THE PRIMARY AND SECONDARY CONTAINMENT. THE EVENT OCCURRED DURING APPLICATION OF A PERMIT WHICH REQUIRED THE LEADS ON A TERMINAL BLOCK TO BE LIFTED. THE CAUSE OF THE EVENT WAS A PERSONNEL ERROR. WHILE APPLYING THE PERMIT, AN INADVERTENT CONTACT INDUCED A GROUND WHICH CAUSED A FUSE TO BLOW, THEREBY INITIATING THE ISOLATION. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THE EVENT. ALL EXPECTED ACTIONS EITHER OCCURRED OR WERE ELECTRICALLY BLOCKED. THE FUSE WAS REPLACED AND THE ISOLATION RESET WITHIN APPROXIMATELY 70 MINUTES. THE INDIVIDUAL WHO APPLIED THE PERMIT WAS UNAWARE THAT HE CAUSED A GROUND, AND WAS REMINDED OF THE IMPORTANCE OF ATTENTION TO DETAIL DURING PERFORMANCE OF DUTIES. OPERATIONS MANAGEMENT STRESSES THAT LIFTING LEADS IS THE LEAST DESIRABLE METHOD FOR ELECTRICALLY BLOCKING EQUIPMENT. FOR THIS EVENT, LIFTING LEADS WAS WARRANTED. CONTINUED ADHERENCE TO THIS POLICY WILL MINIMIZE THE NUMBER OF BLOCKING PERMITS, THEREBY MINIMIZING THE OPPORTUNITY FOR PERSONNEL ERROR. THIS EVENT IS REPORTABLE DUE TO ACTUATION OF AN ENGINEERED SAFETY FEATURE.

[162] PEACH BOTTOM 2 DOCKET 50-277 LER 87-017 REV 01
UPDATE ON PARTIAL VENTILATION ISOLATION DURING THE TEMPORARY CLEARANCE OF A SAFETY BLOCK.
EVENT DATE: 091687 REPORT DATE: 020588 NSSS: GE TYPE: BWR

(NSIC 208061) ON SEPTEMBER 16, 1987 AT 0600 HOURS, THE TEMPORARY CLEARING OF BLOCKED SYSTEM LOGIC INITIATED AN AUTO-START OF THE STANDBY GAS TREAMENT SYSTEM (SBGTS) AND AN OUTBOARD ISOLATION OF THE UNIT 2 REACTOR BUILDING VENTILATION SYSTEM (RBVS). THE ISOLATION WAS RESET, AND THE SBGTS FAN WAS TRIPPED TO RE-ESTABLISH THE NORMAL VENTILATION CONFIGURATION. THE UNEXPECTED ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF) MAKES THIS EVENT REPORTABLE. THE CAUSE OF THIS EVENT IS THE INADEQUATE REVIEW OF BLOCKING PERMITS AND HOW THEIR APPLICATION OR REMOVAL AFFECTS ESF LOGIC SYSTEMS. THIS PERMIT WAS SUBSEQUENTLY REVISED TO INCLUDE INSTRUCTIONS TO RESET THE ISOLATION LOGIC OR NOTIFY THE OPERATOR THAT SBGTS WILL AUTO-START AND AN RBVS ISOLATION WILL RESULT WHEN THE BLOCK IS REMOVED. TO PREVENT SIMILAR EVENTS IN THE FUTURE, THE PLANT STAFF IS DEVELOPING A PROCESS WHICH WILL ALLOW THE APPLICATION OR CLEARANCE OF COMPLICATED ELECTRICAL

BLOCKS WITHOUT RESULTING IN UNEXPECTED ACTUATIONS. AS A RESULT OF THIS EVENT, NO PERSONNEL OR EQUIPMENT WERE ENDANGERED, AND NO EQUIPMENT OR SYSTEMS WERE RENDERED INOPERABLE.

[163] PEACH BOTTOM 2 DOCKET 50-277 LER 87-026
GROUP II AND GROUP III ISOLATION OF THE RWCU AND SHUTDOWN COOLING MODE DUE TO THE LEAKING OF AN INSTRUMENT VALVE.
EVENT DATE: 120687 REPORT DATE: 010588 NSSS: GE TYPE: BWR
VENDOR: DRAGON VALVE, INC.

(NSIC 207674) ON DECEMBER 6, 1987 AT 2100 HOURS WITH THE REACTOR IN THE COLD SHUTDOWN CONDITION, A SCRAM SIGNAL AND SUBSEQUENT GROUP II AND III ISOLATIONS WERE GENERATED FROM THE ACTUATION OF REACTOR LEVEL TRANSMITTERS LT-101A AND B. THE GROUP II AND III ISOLATIONS TERMINATED THE OPERATION OF THE SHUTDOWN COOLING MODE OF THE RESIDUAL HEAT REMOVAL SYSTEM AND THE REACTOR WATER CLEANUP SYSTEM. THE ACTUATION OF THE LEVEL TRANSMITTERS WAS CAUSED BY A LEAKING SHUTOFF VALVE WHICH, DURING THE CALIBRATION OF THE DIAPHRAGM TYPE PRESSURE SWITCH PS-2-2-3-102B, PRESSURIZED THE REFERENCE LEG AND CAUSED LT-101A AND 101B TO SENSE FALSE REACTOR LEVEL OSCILLATIONS. THE CALIBRATION OF THE PRESSURE SWITCH WAS TEMPORARILY HALTED. THE GROUP II AND III ISOLATIONS WERE RESET, THE SHUTDOWN COOLING MODE WAS RESTARTED, THE SCRAM SIGNAL WAS RESET, AND THE REACTOR WATER CLEANUP SYSTEM WAS RESTARTED. THE SHUTDOWN COOLING MODE WAS OUT OF SERVICE FOR APPROXIMATELY ELEVEN MINUTES AND THE REACTOR WATER CLEANUP SYSTEM WAS OUT OF SERVICE FOR APPROXIMATELY 26 MINUTES. THE SHUTOFF VALVE FOR PRESSURE SWITCH PS-2-2-3-102B WAS REPLACED AND THE REFERENCE LEG WAS BACKFILLED. THE PRESSURE TESTING OF PS-2-2-3-102B WAS COMPLETED AND THE PRESSURE SWITCH WAS RETURNED TO SERVICE.

[164] PEACH BOTTOM 2 DOCKET 50-277 LER 87-028
DESIGN DEFICIENCY THAT COULD PERMIT DIESEL GENERATOR TRIPS DURING A SEISMIC EVENT.
EVENT DATE: 121787 REPORT DATE: 022288 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 208304) A DESIGN DEFICIENCY WAS DISCOVERED WHICH COULD RESULT IN DIESEL GENERATOR TRIPS DURING A LOSS OF OFFSITE POWER (LOOP) EVENT. FOUR RELAYS IN THE DIESEL GENERATOR ROOM CARBON DIOXIDE FIRE SUPPRESSION (CARDOX) SYSTEM CONTROL CIRCUITS, WHICH ARE NOT CLASSIFIED AS SAFETY- RELATED OR SEISMIC, COULD INITIATE DIESEL GENERATOR TRIP SIGNALS DURING A LOOP EVENT IF ACTUATED BY SEISMIC CONDITIONS. THERE ARE FOUR DIESEL GENERATORS COMMON TO UNIT 2 AND UNIT 3, AND EACH DIESEL GENERATOR COULD BE TRIPPED BY ITS RESPECTIVE CARDOX SYSTEM RELAY. THE ORIGINAL DESIGN DOES NOT PREVENT A SEISMIC-INDUCED DIESEL GENERATOR TRIP SIGNAL FROM THESE RELAYS. ON DECEMBER 17, 1987 IT WAS DETERMINED THAT THIS CONDITION WAS REPORTABLE PURSUANT TO 10 CFR 50.73 (A)(2)(VI). THE CONDITION WAS DISCOVERED APPROXIMATELY ONE MONTH EARLIER. THIS CONDITION COMPROMISED THE ABILITY TO SAFELY SHUT DOWN THE PLANT DURING A LOOP EVENT CONCURRENT WITH A SEISMIC EVENT. BOTH PEACH BOTTOM UNITS ARE SHUT DOWN. THE CARDOX SYSTEM/DIESEL GENERATOR INTERFACE IS BEING REVIEWED. UPON COMPLETION OF THIS REVIEW, CORRECTIVE ACTIONS WILL BE ESTABLISHED AND A REVISED LER WILL BE SUBMITTED BY APRIL 22, 1988.

[165] PEACH BOTTOM 2 DOCKET 50-277 LER 87-030
ACTUATION OF PRIMARY CONTAINMENT ISOLATION SYSTEM RESULTING DURING AN INTERRUPTION IN OFFSITE POWER.
EVENT DATE: 123087 REPORT DATE: 012988 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 208062) AT 0910 HOURS ON DECEMBER 30, 1987, A PARTIAL LOSS OF OFFSITE POWER INITIATED THE ACTUATION OF THE PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) OF

BOTH UNITS 2 AND 3. THE UNEXPECTED ACTUATION OF AN ENGINEERED SAFETY FEATURE, THE PCIS, MAKES THIS EVENT REPORTABLE. OFFSITE POWER WAS INTERRUPTED WHEN A CRANE MADE CONTACT WITH AN ENERGIZED TRANSMISSION LINE MAINTAINED BY ANOTHER UTILITY. THE LOSS OF POWER FROM THIS LINE RESULTED IN A FAST TRANSFER OF FOUP OF THE EIGHT 4KV BUSSES TO THE ALTERATE SOURCE OF OFFSITE POWER. THE PCIS AND FAST TRANSFER FUNCTIONED AS DESIGNED AND THE DIESEL GENERATORS WERE AVAILABLE, BUT UNCHALLENGED. THE "2A" REACTOR PROTECTION SYSTEM MOTOR GENERATOR (RPS M/G SET) SET TRIPPED, RESULTING IN PCIS GROUP III AND RBVS INBOARD ISOLATIONS AND A HALF-SCRAM SIGNAL TO UNIT 2. NO CONTROL ROD MOTION OCCURRED, AND THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THIS EVENT. THE CAUSE OF THE ISOLATIONS NOT ASSOCIATEΓ WITH THE M/G SET WAS THE TRANSMISSION LINE INCIDENT WHICH IS BEYOND PECO'S CONTROL. CONSEQUENTLY, NO ACTION WILL BE TAKEN TO PREVENT RECURRENCE; HOWEVER, THE OTHER UTILITY WAS NOTIFIED OF THE EVENT. THE OTHER ISOLATIONS WERE CAUSED BY A FAULURE OF THE M/G SET TO FUNCTION AS DESIGNED.

[166] PEACH BOTTOM 2 DOCKET 50-277 LER 87-031
CONDITION OUTSIDE THE DESIGN BASIS OF THE PLANT RESULTED FROM A MODIFICATION TO THE FEEDWATER HEATERS.
EVENT DATE: 123187 REPORT DATE: 020188 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 208063) DURING THE UNIT 2 REFUELING OUTAGE IN 1982 AND THE UNIT 3 REFUELING OUTAGE IN 1983, EXTRACTION STEAM BLOCK VALVES WERE INSTALLED ON THE 3RD, 4TH AND 5TH FEEDWATER HEATERS IN EACH OF THE THREE FEEDWATER STRINGS. IT WAS RECENTLY RECOGNIZED THAT THESE VALVES HAVE A COMMON ELECTRICAL FEED AND WILL FAIL CLOSED ON A LOSS OF POWER. ON DECEMBER 31, 1987 WITH UNIT 2 IN COLD SHUTDOWN AND UNIT 3 IN THE REFUELING MODE WITH THE CORE OFFLOADED, IT WAS DETERMINED THAT A LOSS OF POWER TO THESE BLOCK VALVES COULD RESULT IN A LOSS OF FEEDWATER HEATING (LOFWH) EVENT OUTSIDE THE DESIGN BASIS OF THE PLANT. CALCULATIONS PERFORMED ASSUMING THE FAILURE INDICATE THAT THERE WOULD HAVE BEEN NO CHANGE IN PLANT OPERATING LIMITS AND THAT ALL DESIGN AND LICENSING CRITERIA WOULD HAVE BEEN SATISFIED WITH NO REDUCTION IN SAFETY. IN ADDITION, NEITHER UNIT EXPERIENCED A LOSS OF POWER TO THESE BLOCK VALVES. THE CAUSE OF THE EVENT APPEARS TO BE AN ERROR DURING THE DESIGN PROCESS. A PLANT MODIFICATION WHICH WILL PRECLUDE A SINGLE FAILURE FROM RESULTING IN A LOFWH EVENT OUTSIDE THE DESIGN BASIS OF THE PLANT WILL BE IMPLEMENTED ON EACH UNIT PRIOR TO STARTUP OF THAT UNIT. IN ADDITION, A PROGRAM IS BEING DEVELOPED TO REVIEW A SAMPLE OF NONSAFETY-RELATED MODIFICATIONS TO ASCERTAIN WHETHER THIS IS AN ISOLATED INCIDENT OR A PROGRAMMATIC PROBLEM.

[167] PERRY 1 DOCKET 50-440 LER 88-001
REACTOR SCRAM RESULTS FROM INTERMEDIATE RANGE NEUTRON MONITORS UPSCALE TRIP DUE TO EXCESSIVE FEEDWATER FLOW WITH MANUAL CONTROL OF A TURBINE DRIVEN FEEDWATER PUMP.
EVENT DATE: 010388 REPORT DATE: 020288 NSSS: GE TYPE: BWR

(NSIC 208208) ON JANUARY 3, 1988 AT 0630, A REACTOR SCRAM OCCURRED DUE TO AN UPSCALE TRIP ON THE INTERMEDIATE RANGE NEUTRON MONITORS (IRM). THE SCRAM WAS A RESULT OF EXCESSIVE FEEDWATER INJECTION INTO THE REACTOR VESSEL WHILE PLANT OPERATORS WERE ADJUSTING THE MANUAL SPEED CONTROL POTENTIOMETER FOR THE B TURBINE DRIVEN FEEDWATER PUMP (TDFP) TURBINE. THE CAUSES OF THE EVENT WERE INSTRUCTION INADEQUACY AND DESIGN DEFICIENCY. INTEGRATED OPERATING INSTRUCTION (IOI)-4 "SHUTDOWN" DID NOT PROVIDE ADEQUATE GUIDANCE AND A PROPER SEQUENCE FOR FEEDWATER CONTROL DURING A PLANT SHUTDOWN WITH A TDFP. THE TDFP MINIMUM FLOW VALVES ARE UNDERSIZED MAKING MANUAL CONTROL OF THE TDFP NECESSARY DURING LOW FLOW CONDITIONS. THE LOW FLOW CONTROLLER PROVIDES VERY LITTLE OVERLAP WITH THE STARTUP LEVEL CONTROLLER AND THEREFORE, WAS NOT BEING USED DURING THIS EVENT. IN ORDER TO PREVENT RECURRENCE IOI-4 WILL BE REVISED TO INCLUDE SPECIFIC GUIDANCE FOR FEEDWATER CONTROL DURING SHUTDOWN WHEN THE MOTOR DRIVEN FEEDWATER PUMP IS UNAVAILABLE. THE TDFP MINIMUM FLOW VALVES WILL BE REPLACED DURING THE FIRST

REFUEL OUTAGE. AN ENGINEERING EVALUATION IS BEING CONDUCTED TO DETERMINE IF THE TOTAL FLOW THROUGH THE LOW FLOW CONTROL VALVE CAN BE INCREASED IN ORDER TO PROVIDE GREATER OVERLAP WITH THE STARTUP LEVEL CONTROLLER.

[168] PERRY 1 DOCKET 50-440 LER 88-002
OVERSENSITIVE FLOW CONTROL VALVES RESULT IN INDICATED HIGH DIFFERENTIAL FLOW AND REACTOR WATER CLEANUP SYSTEM CONTAINMENT ISOLATION.
EVENT DATE: 010388 REPORT DATE: 020288 NSSS: GE TYPE: BWR

(NSIC 208209) ON JANUARY 3, 1988 AT 1401, AN UNEXPECTED REACTOR WATER CLEANUP (RWCU) INBOARD CONTAINMENT ISOLATION OCCURRED DUE TO INDICATED HIGH DIFFERENTIAL FLOW. THE CAUSE OF THIS EVENT IS OVERSENSITIVE FLOW CONTROL VALVES. DURING RWCU SYSTEM MANIPULATIONS, DIFFICULTIES IN ADJUSTING AND MAINTAINING THE REQUIRED FLOW CONDITIONS RESULTS IN FLOW OSCILLATIONS. THE PLANT OPERATORS ARE NOT ALWAYS ABLE TO RECOVER FROM THE FLOW OSCILLATIONS BEFORE THE SYSTEM ISOLATES ON HIGH DIFFERENTIAL FLOW. AS A RESULT OF THE ISOLATION, PLANT OPERATORS VERIFIED THAT NO ACTUAL SYSTEM LEAKAGE EXISTED AND RETURNED THE RWCU SYSTEM TO SERVICE AT 1404. OVERSENSITIVE RWCU FLOW CONTROL VALVES HAVE BEEN IDENTIFIED IN PREVIOUS RWCU CONTAINMENT ISOLATIONS. AN ENGINEERING DESIGN CHANGE TO REPLACE THE RWCU FLOW CONTROL VALVES HAD BEEN INITIATED. HOWEVER, DUE TO OPERATIONAL CONSTRAINTS, THIS ENGINEERING DESIGN CHANGE IS NOT EXPECTED TO BE IMPLEMENTED UNTIL THE FIRST REFUELING OUTAGE. ADDITIONALLY, AS A RESULT OF PREVIOUS EVENTS, AN INCREASE OF THE DIFFERENTIAL FLOW TRIP SETPOINT AND/OR TIME DELAY HAS BEEN UNDER EVALUATION TO ALLOW ADDITIONAL OPERATING MARGIN FOR THE RWCU INDICATED DIFFERENTIAL FLOW.

[169] PERRY 1 DOCKET 50-440 LER 88-003
FAILURE TO DECLARE RADIATION MONITOR INOPERABLE RESULTS IN VIOLATION OF TECHNICAL SPECIFICATION ACTION STATEMENT REQUIREMENTS.
EVENT DATE: 010388 REPORT DATE: 020288 NSSS: GE TYPE: BWR

(NSIC 208210) BETWEEN 1130, JANUARY 3, 1988 AND 1140, JANUARY 5, 1988, EMERGENCY SERVICE WATER LOOP A WAS DISCHARGING TO THE ENVIRONMENT WITHOUT THE REQUIRED RADIATION MONITORING SYSTEM OPERABLE. SAMPLE ANALYSES REQUIRED BY TECHNICAL SPECIFICATION ACTION STATEMENTS WERE NOT PERFORMED. THE CAUSES OF THIS EVENT WERE DESIGN DEFICIENCY AND PERSONNEL ERROR. HIGH BACKGROUND RADIATION LEVELS FROM CRUD DEPOSITS IN NEARBY PIPING CAUSED A HIGH-HIGH ALARM ON THE EMERGENCY SERVICE WATER LOOP A RADIATION MONITOR WHEN THE RESIDUAL HEAT REMOVAL SYSTEM WAS ALIGNED FOR SHUTDOWN COOLING. CONTROL ROOM PERSONNEL FAILED TO RECOGNIZE THAT THE RADIATION MONITOR WAS UNABLE TO DETECT AND ALARM HIGH ACTIVITY LEVELS IN THE EMERGENCY SERVICE WATER SYSTEM WITH THE CONTINUOUS ALARM DUE TO HIGH BACKGROUND AREA RADIATION LEVELS. THE RADIATION MONITOR WAS NOT DECLARED INOPERABLE AS REQUIRED BY TECHNICAL SPECIFICATIONS, AND THE REQUIRED GRAB SAMPLE ANALYSES WERE NOT PERFORMED. CORRECTIVE ACTIONS TO PREVENT RECURRENCE INCLUDE DESIGN MODIFICATIONS TO REDUCE THE HIGH BACKGROUND RADIATION LEVELS DURING SHUTDOWN COOLING OPERATIONS, CHANGES TO APPROPRIATE PLANT OPERATING INSTRUCTIONS, AND TRAINING OF ALL CONTROL ROOM OPERATORS.

[170] PERRY 1 DOCKET 50-440 LER 88-004
FAILED LOCAL LEAK RATE TEST RESULTS IN EXCEEDING ALLOWABLE SECONDARY CONTAINMENT BYPASS LEAKAGE DUE TO VALVE SEATING SURFACE DEGRADATION.
EVENT DATE: 010688 REPORT DATE: 020588 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 208211) ON JANUARY 6, 1988 AT APPROXIMATELY 0400 IT WAS IDENTIFIED THAT THE TOTAL COMBINED SECONDARY CONTAINMENT BYPASS LEAKAGE RATE DEFINED BY TECHNICAL SPECIFICATION 3.6.1.2 HAD BEEN EXCEEDED. THIS WAS A RESULT OF THE POST ACCIDENT SAMPLING SYSTEM (PASS) INSTRUMENT SAMPLE LINE CONTAINMENT ISOLATION VALVES FAILING TO MEET THEIR EXPECTED LEAK RATE. THE VALVES WERE SUBSEQUENTLY RCPLACED

AUD WERE SATISFACTORILY LEAK TESTED ON JANUARY 13. INSPECTION OF THE FAILED VALVES REVEALED DEGRADATION AND DISCOLORATION OF THE DISKS DUE TO POSSIBLE ELECTRICAL ARCING AND FOREIGN MATERIAL IN THE VALVE BODIES. THE CAUSE OF THE ELECTRICAL ARCING ON THE VALVE DISKS HAS NOT BEEN DETERMINED. THE FOREIGN MATERIAL IS BELIEVED TO BE FROM REACTOR WATER CLEANUP SYSTEM SAMPLING. ALSO CONTRIBUTING TO THE EXCESSIVE LEAKAGE IS VALVE SEATING WEAR DUE TO THE FREQUENCY OF OPERATION OF THE VALVES. ENGINEERING EVALUATIONS WERE UNDERWAY PREVIOUS TO THIS EVENT TO CHANGE THE VALVES TO NORMALLY OPEN. THIS WOULD REDUCE THE CYCLING REQUIRED FOR THESE VALVES. ADDITIONALLY, THE NEW VALVE BODIES WERE MAINTAINED LESS THAN 350 DEGREES WHILE BEING WELDED IN PLACE. IN ORDER TO FURTHER EVALUATE ANY POTENTIAL LEAKAGE PROBLEMS WITH THESE VALVES, A LEAK RATE TEST WILL BE PERFORMED DURING THE NEXT AVAILABLE OUTAGE OF SUFFICIENT DURATION.

[171] PERRY 1 DOCKET 50-440 LER 88-005
PERSONNEL ERROR DURING REACTOR PROTECTION SYSTEM BUS POWER SUPPLY TRANSFER RESULTS IN AN RESIDUAL HEAT REMOVAL SHUTDOWN COOLING ISOLATION.
EVENT DATE: 010788 REPORT DATE: 020588 NSSS: GE TYPE: BWR

(NSIC 208278) ON JANUARY 7, 1988 AT 2006, RESTORATION OF PLANT COMPONENTS FOLLOWING A TRANSFER OF REACTOR PROTECTION SYSTEM (RPS) BUS A FROM ITS NORMAL POWER SUPPLY TO ITS ALTERNATE SUPPLY RESULTED IN A RESIDUAL HEAT REMOVAL (RHR) SYSTEM INBOARD CONTAINMENT ISOLATION. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. ALTHOUGH THE ISOLATION WAS A PREPLANNED PART OF THE BUS TRANSFER PROCEDURE, THE OPERATORS TOOK STEPS TO PREVENT THESE UNDESIRABLE EXPECTED ISOLATIONS. HOWEVER, FAILURE TO ADEQUATELY REVIEW THE EVOLUTION AND SEQUENCE THE RESTORATION STEPS RESULTED IN THE INBOARD ISOLATION SIGNAL NOT BEING RESET. CONSEQUENTLY THIS VOLUNTARY REPORT IS BEING SUBMITTED. A CONTRIBUTING FACTOR WAS THE ABSENCE OF PROCEDUREAL GUIDANCE FOR PREVENTING UNDESIRABLE ISOLATIONS WHEN TRANSFERRING RPS BUS POWER SUPPLIES. THE OPERATING PERSONNEL INVOLVED IN THIS EVENT HAVE BEEN COUNSELED. ALL SHIFT OPERATING PERSONNEL WILL BE TRAINED TO THIS EVENT. ADDITIONALLY, PLANT INSTRUCTIONS WILL BE ENHANCED TO PROVIDE GUIDANCE FOR PREVENTING UNDESIRABLE ISOLATIONS WHEN PERFORMING RPS BUS POWER SUPPLY TRANSFERS.

[172] PERRY 1 DOCKET 50-440 LER 88-006
LOSS OF REACTOR PROTECTION SYSTEM BUS DUE TO AN OVER VOLTAGE TRIP OF THE MOTOR-GENERATOR OUTPUT CIRCUIT BREAKER RESULTS IN A DIVISION II BALANCE OF PLANT ISOLATION.
EVENT DATE: 011988 REPORT DATE: 021888 NSSS: GE TYPE: BWR

(NSIC 208360) ON JANUARY 19, 1988 AT 1510, THE B REACTOR PROTECTION SYSTEM (RPS) MOTOR-GENERATOR (MG) SET OUTPUT CIRCUIT BREAKER TRIPPED OPEN ON OVERVOLTAGE, RESULTING IN THE DEENERGIZATION OF RPS BUS B AND A NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) DIVISION II BALANCE OF PLANT (BOP) ISOLATION. AFFECTED SYSTEMS AND COMPONENTS WERE RESTORED BY 1610. THE ROOT CAUSE OF THIS EVENT HAS NOT BEEN DETERMINED. INVESTIGATION HAS IDENTIFIED SMALL RANDOM DISTURBANCES IN THE RPS BUS VOLTAGE COMBINED WITH A RESTRICTIVE OVERVOLTAGE RELAY TRIP SETPOINT AS THE MOST LIKELY CAUSE. THE VOLTAGE REGULATOR IS NOT RESPONDING PROMPTLY TO PREVENT THESE FLUCTUATIONS. IT IS BELIEVED THAT THE EXISTING VOLTAGE REGULATOR IS SUFFICIENT TO ADEQUATELY CONTROL THE RPS BUS VOLTAGE, BUT HAS NOT BEEN OPTIMALLY CALIBRATED. CORRECTIVE ACTIONS TO BE TAKEN IN RESPONSE TO THIS EVENT INCLUDE THE DEVELOPMENT OF A DETAILED PROCEDURE FOR CALIBRATION OF THE VOLTAGE REGULATOR. THE OVERVOLTAGE RELAY TRIP SETPOINT HAS BEEN INCREASED. ADDITIONALLY, A CALIBRATION CHECK WILL BE PERFORMED PERIODICALLY TO IDENTIFY ANY DEGRADATION IN VOLTAGE REGULATOR PERFORMANCE. A SUPPLEMENTAL REPORT WILL BE SUBMITTED UPON POSITIVE DETERMINATION OF THE PROPER ACTIONS TO PREVENT RECURRENCE OF THIS TYPE OF EVENT.

[173] PERRY 1 DOCKET 50-440 LER 88-007
REACTOR PROTECTION SYSTEM ACTUATION RESULTS FROM INADVERTENT ISOLATION OF INSTRUMENT AIR TO CONTAINMENT DURING SYSTEM RESTORATION.
EVENT DATE: 012288 REPORT DATE: 021888 NSSS: GE TYPE: BWR

(NSIC 208361) ON JANUARY 22, 1988 AT 0626 A REACTOR PROTECTION SYSTEM (RPS) ACTUATION OCCURRED DUE TO HIGH SCRAM DISCHARGE VOLUME (SDV) LEVEL. DURING RESTORATION OF A TAGOUT ON THE INSTRUMENT AIR SYSTEM, THE SUPPLY TO CONTAINMENT WAS INADVERTENTLY ISOLATED, DEPRESSURIZING THE SCRAM VALVE AIR HEADER. DEPRESSURIZATION OF THE AIR HEADER CAUSED THE SCRAM VALVES AND SDV DRAIN VALVE TO REPOSITION. CONSEQUENTLY, THE SDV FILLED RESULTING IN A RPS ACTUATION ON HIGH LEVEL. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR. THE UNIT SUPERVISOR, AUTHORIZING CLEARANCE OF THE TAGOUT, FAILED TO SPECIFY THE CORRECT SEQUENCE TO PREVENT IMPACT UPON THE PLANT DURING THE REVIEW. CONSEQUENTLY, TAG REMOVAL AND THE RESTORATION OF THE INSTRUMENT AIR SYSTEM WAS PERFORMED IN AN INCORRECT ORDER CAUSING THE LOSS OF INSTRUMENT AIR TO CONTAINMENT. TO PREVENT RECURRENCE, THE UNIT SUPERVISOR INVOLVED WITH THIS EVENT HAS BEEN COUNSELED ON THE NEED TO PERFORM A MORE THOROUGH REVIEW AND ENSURE PROPER CONTROLS ARE UTILIZED AND SUFFICIENT GUIDANCE GIVEN WHEN PERFORMING TAG CLEARANCE AND/OR SYSTEM RESTORATION.

[174] PERRY 1 DOCKET 50-440 LER 88-008
EQUIPMENT TESTABILITY DEFICIENCY RESULTS IN PARTIAL NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM BALANCE OF PLANT ISOLATION.
EVENT DATE: 012388 REPORT DATE: 021888 NSSS: GE TYPE: BWR

(NSIC 208362) ON JANUARY 23, 1988 AT 1915, AN UNEXPECTED PARTIAL DIVISION 2 NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) BALANCE OF PLANT (BOP) ISOLATION OCCURRED DURING THE PERFORMANCE OF A REACTOR PROTECTION SYSTEM (RPS) SURVEILLANCE INSTRUCTION (SVI). THIS EVENT OCCURRED WHEN ONE OF THE ELECTRICAL JUMPERS PLACED TO PREVENT THE NSSSS BOP ISOLATION FAILED TO PROVIDE CONTINUITY OF THE CIRCUIT. THE ROOT CAUSE OF THIS EVENT WAS A DEFICIENCY INVOLVING THE TESTABILITY OF THE EQUIPMENT. THE AREA OF EXPOSED METAL AVAILABLE TO ATTACH A JUMPER IS LIMITED AND NOT SUITED FOR JUMPER CONNECTION SUCH THAT ADDITIONAL EFFORT IS REQUIRED TO ASSURE CONNECTION. ADDITIONALLY, EXTREME CAUTION IS REQUIRED BY THE TECHNICIANS DURING THE INSTALLATION AND REMOVAL OF THE TEMPORARY JUMPERS TO PREVENT CONTACT OF THE JUMPER WITH OTHER CIRCUITS OR GROUNDS. DESPITE THESE EFFORTS, PLACEMENT OF THE ELECTRICAL JUMPER DID NOT PROVIDE CONTINUITY OF THE CIRCUIT TO PREVENT THE ISOLATIONS. AS A RESULT OF PREVIOUS SIMILAR EVENTS, A REVIEW OF SVIS HAD BEEN PERFORMED WHICH IDENTIFIED TEST POINTS WHICH SHOULD BE UPGRADED TO ADD TEST CONNECTION LUGS TO AID TEST PERFORMANCE. APPROXIMATELY 110 OF THE 190 IDENTIFIED TEST POINTS HAD BEEN UPGRADED PRIOR TO THIS EVENT. HOWEVER, THE TERMINALS INVOLVED IN THIS EVENT HAD NOT YET BEEN UPGRADED.

[175] PILGRIM 1 DOCKET 50-293 LER 87-022
SECONDARY CONTAINMENT ISOLATION SIGNAL DUE TO FAILED RELAY.
EVENT DATE: 060787 REPORT DATE: 022288 NSSS: GE TYPE: BWR
VENDOR: AMF POTTER & BRUMFIELD DIV

(NSIC 208396) ON 6/7/87, AT 1100 HOURS, AN AUTOMATIC ACTUATION OF THE REACTOR BUILDING ISOLATION CONTROL SYSTEM (RBIS) OCCURRED. THE ACTUATION OCCURRED DURING PERFORMANCE OF OPER-9, TEST NO. 18, "REFUELING FLOOR VENT EXHAUST MONITORS HIGH-LOW FUNCTIONAL TESTS." AT THE TIME OF THE EVENT, CHANNEL "A" WAS BEING TESTED WHEN A SPURIOUS HIGH LEVEL UPSCALE TRIP WAS RECEIVED ON CHANNEL "B". THE HIGH LEVEL TRIP SIGNAL PRESENT ON CHANNEL "A" DUE TO TESTING, CONCURRENT WITH THE SPURIOUS HIGH LEVEL TRIP OF CHANNEL "B" SATISFIED THE "ONE-OUT-OF TWO TAKEN TWICE" LOGIC THAT RESULTED IN THE SECONDARY CONTAINMENT ISOLATION SIGNAL. ALTHOUGH THE RBIS FUNCTIONED AS DESIGNED TO ISOLATE THE SECONDARY CONTAINMENT, THE FANS OF THE STANDBY GAS TREATMENT SYSTEM (SGTS) PORTION OF THE SECONDARY CONTAINMENT SYSTEM (SCS) DID NOT START AS THEY WERE TAGGED OUT-OF-SERVICE FOR

MAINTENANCE ACTIVITIES. IMMEDIATE ACTION WAS TAKEN TO CLEAR THE TRIP SIGNAL AND RESET THE ISOLATION SIGNAL. THE CAUSE OF THE SPURIOUS CHANNEL "B" HIGH LEVEL TRIPS WAS DETERMINED TO BE A FAILED RELAY (K2) IN THE CHANNEL "B" TRIP CIRCUIT. THE FAILED RELAY WAS REPLACED UNDER MAINTENANCE REQUEST 87-46-256. AT THE TIME OF THIS EVENT, THE PLANT WAS IN AN EXTENDED OUTAGE WITH THE MODE SWITCH IN THE REFUEL POSITION. THE REACTOR VESSEL WAS DEFUELED AND ALL CONTROL RODS WERE FULLY INSERTED INTO THE REACTOR VESSEL.

[176] PILGRIM 1 DOCKET 50-293 LER 87-008 REV 01
UPDATE ON UNPLANNED ISOLATION OF SHUTDOWN COOLING DUE TO PERSONNEL ERROR.
EVENT DATE: 101587 REPORT DATE: 022988 NSSS: GE TYPE: BWR

(NSIC 208433) ON OCTOBER 15, 1987 AT 2100 HOURS AN UNPLANNED ISOLATION OF THE RESIDUAL HEAT REMOVAL SYSTEM (SHUTDOWN COOLING MODE (SDC)) OCCURRED. THE ISOLATION WAS CAUSED WHEN AN ENERGIZED POWER SUPPLY LEAD TO RELAY 16A-K28 BECAME DISCONNECTED DURING A WORK TASK. DE-ENERGIZING THE RELAY CAUSED THE AUTOMATIC CLOSURE OF TWO ISOLATION VALVES AND THE UNPLANNED ISOLATION OF THE SDC SYSTEM. IMMEDIATE ACTION WAS TAKEN TO RECONNECT THE POWER SUPPLY LEAD TO RELAY 16A-K28 AND THE SDC SYSTEM WAS RESTORED APPROXIMATELY FIVE MINUTES AFTER THE ISOLATION. THERE WERE NO COMPONENT OR SYSTEM FAILURES THAT CAUSED THIS EVENT OR RESULTED FROM THIS EVENT. THE CAUSE OF THIS EVENT WAS ELECTRICAL CRAFT PERSONNEL ERROR DUE TO A DEFICIENT WORK PLAN AND INSPECTION RECORD (WP&IR). THE WP&IR SPECIFIED AN INCOMPLETE ELECTRICAL ISOLATION. A SECONDARY CAUSE OF THIS EVENT WAS THE FAILURE OF THE RESPONSIBLE ELECTRICAL CRAFT PERSONNEL TO PERFORM AN ADEQUATE CHECK TO VERIFY AN ISOLATED CIRCUIT. IMMEDIATE ACTION WAS TAKEN TO REVISE THE ELECTRICAL ISOLATION TAGOUT AND WP&IR. THIS EVENT OCCURRED DURING AN EXTENDED OUTAGE WHILE IN THE COLD SHUTDOWN CONDITION, THE REACTOR VESSEL COMPLETELY REFUELED AND WITH THE MODE SWITCH IN THE REFUEL POSITION. ALL CONTROL RODS WERE FULLY INSERTED PRIOR TO THIS EVENT.

[177] PILGRIM 1 DOCKET 50-293 LER 88-001
ENGINEERED SAFETY FEATURES ACTUATION DUE TO THE INADVERTENT OPENING OF A REACTOR TRIP BREAKER AS A RESULT OF PERSONNEL ERROR.
EVENT DATE: 011388 REPORT DATE: 020588 NSSS: GE TYPE: BWR
VENDOR: AGASTAT RELAY CO.
GENERAL ELECTRIC CO.

(NSIC 208120) ON JANUARY 6, 1988 AT 1450 HOURS, AUTOMATIC ACTUATIONS OF PORTIONS OF THE PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM (PCIS AND REACTOR BUILDING ISOLATION CONTROL SYSTEM (RBIS) OCCURRED. THE ACTUATIONS RESULTED IN THE FOLLOWING AUTOMATIC RESPONSES. THE TRAIN "B" PRIMARY CONTAINMENT SYSTEM (PCS) GROUP 2 ISOLATION VALVES RECEIVED AN ISOLATION SIGNAL. THE OUTBOARD (TRAIN "B") VENTILATION SYSTEM DAMPERS OF THE SECONDARY CONTAINMENT SYSTEM (SCS) CLOSED. THE "B" TRAIN OF THE SCS/STANDBY GAS TREATMENT SYSTEM (SGTS) STARTED. FOLLOWING IMMEDIATE INVESTIGATION THE SGTS WAS SECURED, THE VENTILATION DAMPERS WERE OPENED, AND THE ISOLATION RESET ON JANUARY 7, 1988 AT 0200 HOURS. THE CAUSE FOR THE ACTUATIONS WAS THE FAILURE OF THE COIL IN A LOGIC RELAY THAT IS PART OF THE OUTBOARD PCIS/RBIS LOGIC CIRCUITRY. THE RELAY COIL WAS REPLACED. BASED ON ANALYSIS, ADDITIONAL RELAYS OR RELAY COILS HAVE BEEN SELECTED FOR REPLACEMENT PRIOR TO STARTUP. THE ACTUATIONS OCCURRED DURING AN EXTENDED OUTAGE WHILE IN THE COLD SHUTDOWN CONDITION WITH NEGLIGIBLE CORE DECAY HEAT AND WITH THE MODE SWITCH IN THE SHUTDOWN POSITION. NO CONTROL RODS WERE IN THE WITHDRAWN POSITION AT THE TIME OF THE ACTUATIONS. THE ACTUATIONS POSED NO THREAT TO THE PUBLIC HEALTH AND SAFETY.

[178] PILGRIM 1 DOCKET 50-293 LER 88-002
FULL SCRAM TRIP SIGNAL DURING SURVEILLANCE AND RESULTING INCOMPLETE AUTOMATIC ACTUATIONS.
EVENT DATE: 011788 REPORT DATE: 021688 NSSS: GE TYPE: BWR
VENDOR: BORG-WARNER CORP.
GENERAL ELECTRIC CO.

(NSIC 208325) ON JANUARY 17, 1988 AT 0113 HOURS, A FULL REACTOR PROTECTION SYSTEM (RPS) SCRAM TRIP SIGNAL OCCURRED DURING A SURVEILLANCE TEST. THE AUTOMATIC ACTUATIONS RESULTING FROM THE TRIP SIGNAL WERE INCOMPLETE IN THAT PORTIONS OF THE "B" TRAINS OF THE PRIMARY CONTAINMENT SYSTEM (PCS) AND SECONDARY CONTAINMENT SYSTEM (SCS) DID NOT ACTUATE AS DESIGNED. A CONTROL ROD DRIVE (CRD) SYSTEM VENT VALVE DID NOT CLOSE AUTOMATICALLY. FOLLOWING IMMEDIATE INVESTIGATION, THE RPS TRIP SIGNAL AND ISOLATIONS WERE RESET AT 0135 HOURS. THE CAUSE FOR THE TRIP SIGNAL WAS INADEQUACY OF THE PROCEDURE BEING USED FOR THE SURVEILLANCE. THE CAUSE FOR THE INCOMPLETE AUTOMATIC ACTUATIONS WAS HIGH CONTACT RESISTANCE OF TWO CONTACTS OF A LOGIC RELAY. THE CAUSE FOR THE CRD SYSTEM VENT VALVE NOT CLOSING AUTOMATICALLY WAS MECHANICAL BINDING OF THE VALVE. CORRECTIVE ACTIONS CONSISTED OF REVISING THE SURVEILLANCE PROCEDURE AND REPLACING THE LOGIC RELAY. CORRECTIVE ACTIONS FOR THE CRD SYSTEM VENT VALVE IS PENDING THE DISASSEMBLY OF THE VALVE ON SITE WITH A REPRESENTATIVE OF THE VALVE MANUFACTURER. THIS EVENT OCCURRED DURING AN EXTENDED OUTAGE WITH NEGLIGIBLE CORE DECAY HEAT AND WITH THE MODE SELECTOR SWITCH IN THE SHUTDOWN POSITION. THERE WERE NO CONTROL RODS IN THE WITHDRAWN POSITION AT THE TIME OF THE TRIP SIGNAL. THIS EVENT POSED NO THREAT TO THE HEALTH AND SAFETY OF THE PUBLIC.

[179] PILGRIM 1 DOCKET 50-293 LER 88-003
LOW SETPOINTS OF DEGRADED GRID VOLTAGE RELAYS DUE TO ERROR IN MODEL USED FOR ANALYSIS.
EVENT DATE: 013088 REPORT DATE: 022988 NSSS: GE TYPE: BWR

(NSIC 208407) ON JANUARY 30, 1988 AT 1630 HOURS, THE SETPOINTS OF DEGRADED GRID VOLTAGE RELAYS WERE IDENTIFIED AS BEING TOO LOW. THE LOW SETPOINTS WERE IDENTIFIED BY THE BOSTON EDISON COMPANY NUCLEAR ENGINEERING DEPARTMENT DURING A DESIGN REVIEW AND ANALYSIS OF THE ELECTRICAL POWER DISTRIBUTION SYSTEM. THE CAUSE FOR THE LOW SETPOINTS WAS AN INCORRECT ASSUMPTION MADE IN THE MODEL USED FOR DEGRADED GRID VOLTAGE ANALYSIS ORIGINALLY PERFORMED IN 1976. THE NUCLEAR ENGINEERING DEPARTMENT IS PERFORMING ADDITIONAL ANALYSIS AND REVIEW. BASED ON THE ANALYSIS AND REVIEW, NEW SETPOINTS ARE EXPECTED TO BE IDENTIFIED. WHEN THE NEW SETPOINTS ARE IDENTIFIED, A CHANGE TO APPROPRIATE TECHNICAL SPECIFICATIONS WILL BE SUBMITTED AND A MODIFICATION WILL BE MADE TO RESET THE SETPOINTS OF THE DEGRADED GRID VOLTAGE RELAYS. THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED BY THE LOW SETPOINTS OF THE DEGRADED GRID VOLTAGE RELAYS. A SELECTED REVIEW OF AVAILABLE RECORDS INDICATED THAT THE GRID VOLTAGE HAD NOT DEGRADED TO LESS THAN 342 KV WHILE PILGRIM STATION WAS IN POWER OPERATION. THERE WERE NO COMPONENT OR SYSTEM FAILURES THAT WERE CAUSED BY OR RESULTED FROM THE LOW SETPOINTS OF THE DEGRADED GRID VOLTAGE RELAYS. THE LOW SETPOINTS WERE IDENTIFIED DURING AN EXTENDED OUTAGE WHILE IN COLD SHUTDOWN CONDITIONS WITH THE REACTOR MODE SELECTOR SWITCH IN THE SHUTDOWN POSITION.

[180] PILGRIM 1 DOCKET 50-293 LER 88-004
UNPLANNED AUTOMATIC ACTUATIONS OF PORTIONS OF PRIMARY CONTAINMENT, SECONDARY CONTAINMENT AND STANDBY GAS TREATMENT SYSTEMS.
EVENT DATE: 020288 REPORT DATE: 030288 NSSS: GE TYPE: BWR

(NSIC 208456) ON FEBRUARY 2, 1988 AT 1115 HOURS, UNPLANNED AUTOMATIC ACTUATIONS OF PORTIONS OF THE PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM PCIS AND REACTOR BUILDING ISOLATION CONTROL SYSTEM (RBIS) OCCURRED. THE ACTUATIONS RESULTED IN ISOLATION SIGNALS AND AUTOMATIC CLOSURE OF APPROPRIATE PRIMARY

CONTAINMENT SYSTEM (PCS) ISOLATION VALVES, THE AUTOMATIC CLOSURE OF THE "A" TRAIN DAMPERS OF THE SECONDARY CONTAINMENT SYSTEM (SCS), AND THE AUTOMATIC START OF THE "A" TRAIN OF THE SCS/STANDBY GAS TREATMENT SYSTEM. THE OPERATION OF THE REACTOR WATER CLEANUP (RHCU) SYSTEM WAS TEMPORARILY INTERRUPTED. FOLLOWING IMMEDIATE INVESTIGATION, THE ISOLATIONS WERE RESET AT 1315 HOURS AND THE AFFECTED SYSTEMS WERE RETURNED TO NORMAL. THE CAUSE FOR THE ACTUATIONS WAS INADEQUATE INSTRUCTIONS FOR WORK BEING PERFORMED IN A PCIS/RBIS LOGIC PANEL. CORRECTIVE ACTIONS RESULTING FROM THE ACTUATIONS CONSISTED OF REVIEWING AND REVISING INSTRUCTIONS RELATED TO THE WORK THAT CAUSED THE ACTUATIONS, INITIATING A CHANGE THAT SUPPLEMENTS THE WORK PLAN PROCESS, AND INSTALLING APPROPRIATE CAUTIONARY SIGNS IN THE PCIS/RBIS PANELS. THE ACTUATIONS OCCURRED DURING AN EXTENDED OUTAGE WHILE IN COLD SHUTDOWN. SWITCH WAS IN THE SHUTDOWN POSITION.

[181] PILGRIM 1 DOCKET 50-293 LER 88-005
AUTOMATIC ACTUATION OF PORTIONS OF PRIMARY CONTAINMENT, SECONDARY CONTAINMENT AND STANDBY GAS TREATMENT SYSTEMS.
EVENT DATE: 020288 REPORT DATE: 030288 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 208457) ON FEBRUARY 2, 1988 AT 1908 HOURS, AN AUTOMATIC ACTUATION OF PORTIONS OF THE PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM (PCIS) AND REACTOR BUILDING ISOLATION CONTROL SYSTEM (RBIS) OCCURRED. THE ACTUATIONS RESULTED IN THE FOLLOWING AUTOMATIC RESPONSES. THE "A" TRAIN PRIMARY CONTAINMENT SYSTEM (PCS) GROUP 2 ISOLATION VALVES RECEIVED AN ISOLATION SIGNAL. THE TRAIN "A" VENTILATION DAMPERS OF THE SECONDARY CONTAINMENT SYSTEM (SCS) CLOSED. THE "A" TRAIN OF THE SCS/STANDBY GAS TREATMENT SYSTEM (SGTS) STARTED. THE ISOLATIONS WERE RESET AND THE SYSTEMS RETURNED TO NORMAL ON FEBRUARY 3, 1988 AT 0500 HOURS. THE CAUSE FOR THE ACTUATIONS WAS THE FAILURE OF THE COIL IN A LOGIC RELAY THAT IS PART OF THE INBOARD PCIS/RBIS LOGIC CIRCUITRY. THE RELAY COIL WAS REPLACED. BASED ON ANALYSIS, ADDITIONAL RELAYS OR RELAY COILS HAVE BEEN SELECTED FOR REPLACEMENT PRIOR TO STARTUP. THE ACTUATIONS OCCURRED DURING AN EXTENDED OUTAGE WITH PLANT CONDITIONS THAT WERE AS FOLLOWS. THE REACTOR MODE SELECTOR SWITCH WAS IN THE SHUTDOWN POSITION. THE REACTOR WATER TEMPERATURE WAS APPROXIMATELY 101 DEGREES FAHRENHEIT WITH NEGLIGIBLE CORE DECAY HEAT. THERE WERE NO CONTROL RODS IN THE WITHDRAWN POSITION. THE ACTUATIONS POSED NO THREAT TO THE HEALTH AND SAFETY OF THE PUBLIC.

[182] PILGRIM 1 DOCKET 50-293 LER 88-006
ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) DIVISION 2 TRIP SIGNAL AND SUBSEQUENT SCRAM SIGNAL.
EVENT DATE: 020388 REPORT DATE: 030288 NSSS: GE TYPE: BWR

(NSIC 208458) ON 2/3/88 AT 1936 HOURS, A TRIP SIGNAL WAS GENERATED UNEXPECTEDLY FROM THE ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) DIVISION 2 (TWO) CIRCUITRY. THE TRIP SIGNAL RESULTED IN THE EXPECTED AUTOMATIC TRIP OF THE RECIRCULATION SYSTEM PUMPS, AND A SUBSEQUENT REACTOR PROTECTION SYSTEM (RPS) SCRAM SIGNAL. FOLLOWING IMMEDIATE INVESTIGATION, THE SCRAM SIGNAL WAS RESET AT 1938 HOURS. THE ROOT CAUSE FOR THE ATWS TRIP SIGNAL HAS NOT BEEN IDENTIFIED AT THE TIME OF SUBMITTAL OF THIS REPORT BUT IS BEING INVESTIGATED. THE RPS SCRAM SIGNAL WAS THE SUBSEQUENT AND EXPECTED DESIGNED RESPONSE TO THE ATWS TRIP SIGNAL. A STOP WORK ORDER WAS ISSUED HALTING FURTHER IMPLEMENTATION OF MODIFICATIONS BEING MADE TO THE ATWS CIRCUITRY. A ROOT CAUSE INVESTIGATION PLAN WAS DEVELOPED FOR THE TESTING OF ATWS PANELS AND RELATED CIRCUITRY. THE TRIP FUNCTIONS OF THE ATWS CIRCUITRY WERE MADE INOPERABLE FOR INVESTIGATION PURPOSES. A SUPPLEMENT TO THIS REPORT WILL BE SUBMITTED FOLLOWING THE INVESTIGATION. THIS EVENT OCCURRED DURING AN EXTENDED OUTAGE WHILE IN COLD SHUTDOWN WITH PLANT CONDITIONS THAT WERE AS FOLLOWS. THE REACTOR MODE SELECTOR SWITCH WAS IN THE SHUTDOWN POSITION. THE CONTROL RODS WERE IN THE INSERTED POSITION PRIOR TO THE TRIP SIGNAL. THE REACTOR

WATER TEMPERATURE WAS APPROXIMATELY 95 DEGREES FAHRENHEIT WITH NEGLIGIBLE CORE DECAY HEAT.

[183] POINT BEACH 1 DOCKET 50-266 LER 88-001
SINGLE FAILURE POTENTIAL IN 4160 VOLT SAFEGUARDS SWITCHGEAR.
EVENT DATE: 011088 REPORT DATE: 020888 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: POINT BEACH 2 (PWR)

(NSIC 208263) IN RESPONSE TO IE INFORMATION NOTICE 87-61, "FAILURE OF WESTINGHOUSE W-2 TYPE CIRCUIT BREAKER CELL SWITCHES," POINT BEACH NUCLEAR PLANT PERSONNEL DISCOVERED A POTENTIAL "SINGLE FAILURE" DESIGN ERROR IN THE USE OF A CELL SWITCH IN THE 4160 VOLT SAFEGUARDS BUS TIE BREAKERS. FAILURE OF THIS CELL SWITCH WOULD PREVENT BOTH EMERGENCY DIESEL GENERATORS FROM AUTOMATICALLY SUPPLYING POWER TO BOTH TRAINS OF SAFEGUARDS EQUIPMENT IN THE EVENT OF A LOSS OF OFFSITE AC. THE CELL SWITCHES WERE VERIFIED TO BE IN THEIR REQUIRED POSITION AND ADMINISTRATIVELY CONTROLLED THERE UNTIL A TEMPORARY MODIFICATION COULD BE INSTALLED TO BYPASS THE CELL SWITCH SUCH THAT FAILURE OF THE SWITCH WILL NOT PREVENT THE DIESEL GENERATORS FROM AUTOMATICALLY SUPPLYING POWER TO SAFEGUARDS EQUIPMENT. ADMINISTRATIVE CONTROL OF THE TIE BREAKERS POSITION WAS ALSO REQUIRED AS A RESULT OF THE TEMPORARY MODIFICATION. A PERMANENT MODIFICATION WHICH WILL ELIMINATE THE SINGLE FAILURE PROBLEM ASSOCIATED WITH THE TIE BREAKER CELL SWITCH IS BEING DESIGNED.

[184] PRAIRIE ISLAND 1 DOCKET 50-282 LER 87-020
AUTO START OF 12 COMPONENT COOLING WATER PUMP.
EVENT DATE: 101987 REPORT DATE: 020488 NSSS: WE TYPE: PWR

(NSIC 208250) ON OCTOBER 19, 1987, WITH BOTH UNITS OPERATING AT FULL POWER, MOTOR OPERATED VALVE MV-32121 (NO. 12 COMPONENT COOLING WATER HEAT EXCHANGER OUTLET VALVE) WAS CLOSED AS PART OF PREWORK TESTING FOR PERIODIC MAINTENANCE. DUE TO A PROCEDURAL INADEQUACY, WHEN MV-32121 WAS CLOSED, NO. 12 COMPONENT COOLING (CC) WATER PUMP AUTOMATICALLY STARTED DUE TO LOW DISCHARGE HEADER PRESSURE. PRIOR TO CLOSING MV-32121, THE NO. 12 CC PUMP DISCHARGE HEADER WAS BEING SUPPLIED AND PRESSURIZED BY NO. 11 CC PUMP. WHEN MV-32121 WAS CLOSED THE NO. 12 CC PUMP DISCHARGE LINE WAS ISOLATED FROM THE NO. 11 CC PUMP AND THE PRESSURE IN THE LINE DROPPED TO THE NO. 12 CC PUMP AUTO-START SETPOINT. MV-32121 WAS IMMEDIATELY REOPENED AND NO. 12 CC PUMP WAS STOPPED. WORK WAS STOPPED AND THE SUBJECT PREWORK TEST PROCEDURE WAS MODIFIED TO REQUIRE THE MANUAL STARTING OF NO. 12 CC PUMP PRIOR TO THE CYCLING OF MV-32121. SUBSEQUENT TESTING OF COMPONENT COOLING HEAT EXCHANGER OUTLET VALVES WAS PERFORMED WITH BOTH CC PUMPS FOR THE ASSOCIATED UNIT IN OPERATION. THIS EVENT HAD NO SAFETY SIGNIFICANCE.

[185] QUAD CITIES 1 DOCKET 50-254 LER 88-001
TWO CONTRACTOR PERSONNEL OVEREXPOSURES IN THE FOURTH QUARTER OF 1980 DUE TO DOSIMETER INACCURACY.
EVENT DATE: 112380 REPORT DATE: 012888 NSSS: GE TYPE: BWR
VENDOR: LANDAUER, R. S. JR. & COMPANY
VICTOREEN INSTRUMENT DIVISION
XTEX

(NSIC 208319) ON JANUARY 14, 1988 IT WAS DETERMINED FROM A RECORDS REVIEW THAT TWO APPARENT OVEREXPOSURES OF CONTRACTOR PERSONNEL OCCURRED DURING THE FOURTH QUARTER OF 1980. SINCE ADEQUATE INFORMATION DOES NOT EXIST TODAY RELATIVE TO THE 1980 CASES, THESE COULD NOT BE DISPROVED. THIS IDENTIFICATION OF APPARENT OVEREXPOSURES WAS THE RESULT OF A SPECIAL COMPUTER PROGRAM THAT PRODUCED A LIST OF ALL INDIVIDUALS THAT HAD BEEN DOCUMENTED AS BEING OVEREXPOSED AT COMMONWEALTH EDISON (CECO) SITES. ONE INDIVIDUAL RECEIVED 3190 MREM/QUARTER AND THE SECOND INDIVIDUAL RECEIVED 3180 MREM/QUARTER BASED ON DOCUMENTED FILM BADGE READINGS.

THIS IS IN EXCESS OF 10CFR20 EXPOSURE LIMITS AND IS REPORTABLE PER 10CFR20.405(A)(1)(I). THE CAUSE FOR THE APPARENT OVEREXPOSURE WAS THE INACCURACY OF THE SECONDARY AND/OR PRIMARY DOSIMETRY IN USE. SINCE THIS EVENT, SIGNIFICANT IMPROVEMENTS HAVE BEEN MADE IN BOTH SECONDARY AND PRIMARY DOSIMETRY SYSTEMS AND DISCREPANCIES OF THIS MAGNITUDE DO NOT OCCUR. IN ADDITION, THERE ARE IMPROVED ADMINISTRATIVE CONTROLS TO TRACK INDIVIDUAL DOSES AND TO PROVIDE FOR IMMEDIATE RECOGNITION OF ANYONE APPROACHING THE LIMITS.

[186] QUAD CITIES 1 DOCKET 50-254 LER 88-002
MISSED REACTOR CORE ISOLATION COOLING LOW PRESSURE FUNCTIONAL TEST DUE TO INADEQUATE PROCEDURE.
EVENT DATE: 011288 REPORT DATE: 020288 NSSS: GE TYPE: BWR

(NSIC 208219) ON JANUARY 12, 1988, QUAD CITIES UNIT ONE. AT 0900 HOURS, IT WAS DETERMINED THAT A QUARTERLY FUNCTIONAL TEST OF THE REACTOR CORE ISOLATION (RCIC) SYSTEM LOW PRESSURE ISOLATION HAD NOT BEEN PERFORMED PRIOR TO UNIT ONE STARTUP FROM ITS REFUEL OUTAGE AS REQUIRED BY TECHNICAL SPECIFICATION TABLE 4.2-1. THE FUNCTIONAL TEST WAS IMMEDIATELY PERFORMED ON JANUARY 12. 1988 AFTER THIS WAS IDENTIFIED. THE CAUSE FOR THE MISSED FUNCTIONAL TEST WAS AN INADEQUATE PROCEDURE. THE CALIBRATION AND FUNCTIONAL TESTS ARE NORMALLY COMPLETED CONCURRENTLY. IN THIS CASE, THE CALIBRATION WAS COMPLETED AS REQUIRED ON NOVEMBER 24. 1987 BUT A FUNCTIONAL TEST COULD NOT BE PERFORMED DUE TO THE RCIC LOW PRESSURE LOGIC SYSTEM BEING OUT-OF-SERVICE. FUNCTIONAL TESTING WAS OVERLOOKED DURING THE SUBSEQUENT STARTUP FROM THE REFUEL OUTAGE BECAUSE THE PROCEDURE CHECKLIST IN USE DID NOT DIFFERENTIATE BETWEEN CALIBRATION AND FUNCTIONAL TESTING. THE PROCEDURE CHECKLIST USED AND OTHER SIMILAR CHECKLISTS WILL BE REVISED TO DIFFERENTIATE BETWEEN CALIBRATION AND FUNCTIONAL TESTING. A MEMORANDUM HAS BEEN ISSUED TO INSTRUMENT MAINTENANCE PERSONNEL DETAILING THIS EVENT AND STATING THAT AN APPROPRIATE NOTATION MUST BE MADE ON THE SCHEDULING CHECKLIST TO ENSURE THAT THIS TYPE OF CONDITION DOES NOT RECUR. THIS REPORT IS PROVIDED TO SATISFY 10 CFR 50.73 (A)(2)(I)(CB).

[187] QUAD CITIES 1 DOCKET 50-254 LER 88-003
MOTOR OPERATOR VALVE FAILED TO OPEN DURING TESTING DUE TO TORQUE SWITCH BYPASS LIMIT SWITCH SETTING TOO CLOSE TO THE CLOSED POSITION.
EVENT DATE: 012588 REPORT DATE: 021988 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)
VENDOR: CRANE VALVE CO.
LIMITORQUE CORP.

(NSIC 208390) ON JANUARY 25, 1988, QUAD CITIES UNIT ONE WAS IN THE RUN MODE AT APPROXIMATELY 100% POWER. AT 1315 HOURS, WHILE PERFORMING THE REACTOR CORE ISOLATION COOLING (RCIC) MOTOR-OPERATED (MO) VALVE OPERABILITY TEST, MO 1-1301-48 WOULD NOT INITIALLY OPEN FROM ITS CLOSED POSITION. THE VALVE DID OPEN ON THE THIRD ATTEMPT BUT RCIC WAS DECLARED INOPERABLE. TECH SPEC 4.5.E.2. REQUIREMENTS WERE THEN SATISFACTORILY PERFORMED. NRC NOTIFICATION OF THIS EVENT WAS COMPLETED AT 1415 HOURS. THE ELECTRICAL MAINTENANCE DEPARTMENT (EMD) INVESTIGATED THIS EVENT AND DETERMINED THAT THE TORQUE SWITCH BYPASS LIMIT SWITCH WAS SET TOO CLOSE TO THE FULL CLOSED POSITION. IN ADDITION, THE REDUNDANT ISOLATION VALVE WAS LEAKING BY WHICH CAUSED A DIFFERENTIAL PRESSURE CONDITION ON THE MO 1-1301-48. EMD ADJUSTED THE TORQUE SWITCH BYPASS LIMIT SWITCH AND RCIC WAS DECLARED OPERABLE AT 1010 HOURS ON JANUARY 26, 1988 FOLLOWING TESTING DESIGNED TO SIMULATE ORIGINAL CONDITIONS. WORK REQUESTS WILL BE COMPLETED TO ADJUST UNIT ONE AND TWO RCIC AND HIGH PRESSURE COOLANT INJECTION (HPCI) MO VALVES LIMIT SWITCHES BASED ON THE ANALYSIS OF STRIP CHART RECORDS. THESE VALVES ARE NOT ADDRESSED BY I.E. BULLETIN 85-03. THIS REPORT IS PROVIDED TO COMPLY WITH 10 CFR 50.73(A)(2)(V)(A).

[188] QUAD CITIES 1 DOCKET 50-254 LER 88-004
REACTOR HEAD VENT LINE OUTSIDE SAFETY ANALYSIS CRITERIA FOR ALLOWABLE STRESS DUE TO DESIGN ERROR.
EVENT DATE: 012688 REPORT DATE: 021788 NSSS: GE TYPE: BWR

(NSIC 208389) ON JANUARY 26, 1988, UNIT ONE WAS IN THE RUN MODE AT 99% THERMAL POWER. AT 1430 HOURS, THE STATION WAS NOTIFIED BY IMPELL CORPORATION THAT LINE 1-0216-1/2" WAS NOT QUALIFIED TO FINAL SAFETY ANALYSIS REPORT (FSAR) CRITERIA FOR ALLOWABLE STRESS. HOWEVER, ANALYSIS INDICATED THE LINE WAS STILL OPERABLE. NRC NOTIFICATION OF THIS CONDITION WAS COMPLETED AT 1510 HOURS TO SATISFY 10CFR50.72. THE CAUSE FOR THIS CONDITION WAS DESIGN ERROR INVOLVING ARCHITECT/ENGINEERING (A/E) PERSONNEL. THE PIPING WAS ORIGINALLY INSTALLED USING SPAN CRITERIA WITHOUT CONSIDERATION TO COMPONENT WEIGHTS. TWO AIR-OPERATED VALVES (VESSEL HEAD VENTS) ON THIS LINE WEIGH 150 POUNDS, WHICH COULD CAUSE AN OVERSTRESS CONDITION DURING A SEISMIC EVENT. IMPELL IS DEVELOPING A MODIFICATION THAT WILL CORRECT THIS CONDITION. IT WILL LIKELY INVOLVE MODIFYING THE EXISTING SUPPORTS WITHOUT REQUIRING ANY NEW SUPPORTS. THE MODIFICATION WILL BE IMPLEMENTED DURING THE NEXT UNIT ONE REFUEL OUTAGE. THIS REPORT IS PROVIDED TO COMPLY WITH THE REQUIREMENTS OF 10CFR50.73(A)(2)(II)(B).

[189] QUAD CITIES 2 DOCKET 50-265 LER 87-021
STANDBY COOLANT SUPPLY SYSTEM OUTSIDE SAFETY ANALYSIS REPORT DUE TO POSITION INDICATION SHORT CIRCUIT.
EVENT DATE: 123087 REPORT DATE: 011988 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 208060) ON DECEMBER 30, 1987, QUAD CITIES UNIT TWO WAS IN THE RUN MODE AT 89 PERCENT THERMAL POWER. AT 0025 HOURS THE CLOSED POSITION INDICATOR LIGHT SOCKET FOR THE 2B SERVICE SUPPLY VALVE TO THE CONDENSER HOTWELL (STANDBY COOLANT SUPPLY) DEVELOPED A SHORT IT. THIS CREATED AN OVERCURRENT CONDITION TO THE CONTROL TRANSFORMER WHICH DISTROYED VARIOUS COMPONENTS AT THE MOTOR CONTROL CENTER CUBICLE. THE CUBICLE WAS DEENERGIZED. THIS CAUSED UNIT TWO TO BE IN A CONDITION OUTSIDE THE DESIGN BASIS OF THE AND NRC NOTIFICATION OF THIS CONDITION WAS COMPLETED AT 0115 HOURS PER 10 CFR THE STANDBY COOLANT SUPPLY SYSTEM WAS STILL CONSIDERED TO BE OPERABLE BECAUSE VALVE IS LOCATED IN AN ACCESSIBLE AREA AND COULD HAVE BEEN MANUALLY OPENED, IF NECESSARY. REQUEST Q63045 WAS WRITTEN FOR REPAIRS. THE BREAKER CUBICLE AND LIGHT SOCKET WERE REPLACED AND TESTED THE WORK REQUEST WAS COMPLETED ON JANUARY 14, 1988. NO FURTHER ACTIVE ACTION IS DEEMED NECESSARY. THIS REPORT IS PROVIDED PER 10 CFR 50.73)(II)(B).

[190] QUAD CITIES 2 DOCKET 50-265 LER 88-001
REACTOR SCRAM DUE TO TURBINE/GENERATOR LOAD REJECT DUE TO UNDERTERMINED CAUSES.
EVENT DATE: 011188 REPORT DATE: 012788 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 208117) ON JANUARY 11, 1988, QUAD CITIES UNIT TWO WAS IN THE RUN MODE AT 100 PERCENT THERMAL POWER. AT 1055 HOURS, A REACTOR SCRAM OCCURRED DUE TO A TURBINE/GENERATOR LOAD REJECT SIGNAL. THIS GENERATOR TRIP RESULTED FROM A GROUND DETECTED BY THE GENERATOR 18 KV GROUND DETECTION SYSTEM. ALL SAFETY FEATURE ACTUATIONS OCCURRED AS DESIGNED BASED ON LOW REACTOR WATER LEVEL WHICH RESULTED FROM THE COLLAPSE OF VOIDS IN THE REACTOR CORE. A NORMAL SCRAM RECOVERY FOLLOWED. TWO OVERVOLTAGE RELAYS WERE FOUND TRIPPED. NRC NOTIFICATION OF THIS EVENT WAS COMPLETED AT 1120 HOURS TO SATISFY THE REQUIREMENTS OF 10 CFR 50.72. THE CAUSE OF THIS EVENT WAS NOT DETERMINED. STATION ELECTRICAL MAINTENANCE AND THE OPERATIONAL ANALYSIS DEPARTMENT (OAD) PERFORMED EXTENSIVE TESTS BUT NO GROUNDING PROBLEM WAS IDENTIFIED. THE ONLY PROBLEM IDENTIFIED WAS A BLOWN FUSE WHICH COULD EXPLAIN WHY ONE OVERVOLTAGE RELAY WAS TRIPPED, BUT COULD NOT HAVE CAUSED THE OTHER LAY TO TRIP. UNIT TWO GENERATOR WAS PUT ON-LINE AT 0300 HOURS ON

JANUARY 16, 1988. NO PROBLEMS HAVE BEEN OBSERVED SINCE THAT TIME. THIS REPORT IS PROVIDED TO COMPLY WITH 10 CFR 50.73(A)(2)(IV).

[191] RANCHO SECO DOCKET 50-312 LER 86-026 REV 01
UPDATE ON SFAS CALIBRATION PROCEDURE DOES NOT ACCOUNT FOR THE ERROR OF THE TRIPPING BI-STABLE.
EVENT DATE: 110786 REPORT DATE: 022388 NSSS: BW TYPE: PWR
VENDOR: BAILEY INSTRUMENT CO., INC.

(NSIC 208393) THERE IS A TECHNICAL ERROR IN THE I.200 SERIES PROCEDURES WHICH DOES NOT ALLOW FOR THE ERROR INTRODUCED BY THE BI-STABLE IN THE ASSOCIATED PRESSURE-TRIP INSTRUMENT STRING. THE PROCEDURE'S ACCEPTANCE CRITERIA DOES NOT ACCOUNT FOR ERROR INTRODUCED BY THE BI-STABLE WITH RESPECT TO THE FINAL TRIP SETPOINT. THIS PROCEDURAL DEFICIENCY WAS IDENTIFIED DURING ANALYSIS FOR AN UNRELATED OCCURRENCE DESCRIPTION REPORT, BY PLANT INSTRUMENT AND CONTROLS ENGINEERS TO DETERMINE INSTRUMENT ACCURACIES AND THEIR EFFECT ON SFAS SETPOINTS. THE AFFECTED INSTRUMENT STRINGS WERE ADJUSTED THROUGH THE WORK REQUEST SYSTEM PRIOR TO REACTOR COOLANT SYSTEM PRESSURE BEING RAISED ABOVE 1600 PSIG TO ADD ENOUGH BIAS INTO THE CIRCUIT SO THAT THE RESULTING SIGNAL SENSED BY THE BI-STABLES IS CONSERVATIVE TO THE ACTUAL PLANT CONDITION.

[192] RANCHO SECO DOCKET 50-312 LER 87-048
INCORRECT SURVEILLANCE PROCEDURE METHODOLOGY DUE TO MISINTERPRETATION OF TECHNICAL SPECIFICATION REQUIREMENTS.
EVENT DATE: 122887 REPORT DATE: 021088 NSSS: BW TYPE: PWR

(NSIC 208251) DURING THE TECHNICAL SPECIFICATION VERIFICATION PROGRAM (TSVP) TECHNICAL SPECIFICATIONS 3.14 AND 4.18 WERE REVIEWED TO VERIFY COMPLIANCE WITH TECHNICAL SPECIFICATION AMENDMENT NO. 35, DATED AUGUST 20, 1980. ON DECEMBER 28, 1987, THE TSVP REVIEW DETERMINED THAT SURVEILLANCE PROCEDURE SP.711 DID NOT FULLY IMPLEMENT TECHNICAL SPECIFICATION 4.18.3.1.D BECAUSE THE PROCEDURE METHODOLOGY FAILED TO ADEQUATELY MEASURE THE ACTUAL WATER FLOW FROM THE SPRAY AND/OR SPRINKLER HEADER. SPECIFICALLY, PROCEDURE SP.711 DID NOT MEASURE THE DIFFERENCE BETWEEN THE "STATIC" PRESSURE AND "RESIDUAL" PRESSURE AS REQUIRED BY NFPA 13A "FIRE PROTECTION SYSTEMS INSPECTION, TEST, AND MAINTENANCE." ON DECEMBER 30, 1987, PROCEDURE SP.100 WAS ISSUED TO ASSURE THAT THE SPRAY AND/OR SPRINKLER HEADERS ARE TESTED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 4.18.3.1.D. THE SUCCESSFUL PERFORMANCE OF SP.100 ON JANUARY 20 AND 22, 1988, DEMONSTRATED THE OPERABILITY OF THE SYSTEM. FROM AUGUST 20, 1980 THRU JANUARY 22, 1988, THE SYSTEM WAS CONSIDERED TO BE INOPERABLE DUE TO THE PROCEDURAL INADEQUACY OF SP.711. DURING THIS PERIOD THE SYSTEM WAS FUNCTIONAL AS DEMONSTRATED BY THE SUCCESSFUL PERFORMANCE OF SP.100 ON JANUARY 20 AND 22, 1988.

[193] RANCHO SECO DOCKET 50-312 LER 88-001
LOST AIR FILTER SAMPLE.
EVENT DATE: 011488 REPORT DATE: 021088 NSSS: BW TYPE: PWR

(NSIC 208264) ON DECEMBER 26, 1987. A PARTICULATE FILTER AND CHARCOAL CARTRIDGE SAMPLE WAS TAKEN OF THE REACTOR BUILDING (RB) EFFLUENT DURING RB PURGE 87-47. THE SAMPLE LOG IN THE CHEMISTRY HOT LAB SHOWS THAT THE FILTER WAS RECEIVED IN THE LAB ON DECEMBER 26, 1987. THE SAMPLE WAS ANALYZED FOR GAMMA RADIATION. THE RESULTS OF THAT ANALYSIS WERE LESS THAN THE LOWER LIMIT OF DETECTION (LLD) OF BOTH THE TECHNICAL SPECIFICATIONS AND THE EQUIPMENT. FOLLOWING THE GAMMA ANALYSIS, THE FILTER WAS PLACED IN A LABELED ENVELOPE. THE ENVELOPE WAS PLACED IN AN INTERIM STORAGE LOCATION (SAMPLE RACK AWAITING ANALYSIS FOR GROSS ALPHA ACTIVITY. DURING AN ADMINISTRATIVE REVIEW ON JANUARY 14, 1988, RADIATION PROTECTION DISCOVERED THAT THE ALPHA ANALYSIS RESULTS FOR RB PURGE 87-47 WERE MISSING. CHEMISTRY PERSONNEL CONDUCTED A THOROUGH SEARCH OF THE HOT LAB;

HOWEVER, THE FILTER WAS NOT FOUND. THE FAILURE TO PERFORM A GROSS ALPHA ANALYSIS OF THE AIR SAMPLE TAKEN DURING RB PURGE 87-47 IS A VIOLATION OF TECHNICAL SPECIFICATION TABLE 4.22-1. THIS TECHNICAL SPECIFICATION VIOLATION IS REPORTABLE IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(I)(B). A PROGRAM DEFINING ACCOUNTABILITIES AND RESPONSIBILITIES FOR TECHNICAL SPECIFICATION REQUIRED SAMPLES WILL BE ESTABLISHED BY FEBRUARY 15, 1988.

[194] RANCHO SECO DOCKET 50-312 LER 88-002
INADVERTENT ACTUATION OF "A" EMERGENCY DIESEL GENERATOR DUE TO PERSONNEL ERROR.
EVENT DATE: 020388 REPORT DATE: 022588 NSSS: BW TYPE: PWR

(NSIC 208326) ON FEBRUARY 3, 1988, WITH THE PLANT IN COLD SHUTDOWN, A PERSONNEL ERROR DURING THE PERFORMANCE OF SPECIAL TEST PROCEDURE STP.961 "LOSS OF OFFSITE POWER" RESULTED IN THE AUTOSTART OF THE "A" EMERGENCY DIESEL GENERATOR. STP.961, SECTION 6.3 "SUBTRAIN A2 SFAS/LOOP" REQUIRED THE 4160V NUCLEAR SERVICE BUS 4A TO BE DE-ENERGIZED PRIOR TO INITIATING THE SFAS/LOOP SIGNAL. DUE TO OPERATOR ERROR, THE BUS DE-ENERGIZATION PROCEDURE WAS PERFORMED OUT OF SEQUENCE AND THE 4A BUS WAS DE-ENERGIZED BEFORE THE "A EMERGENCY DIESEL GENERATOR WAS DISABLED. THE "A" EMERGENCY DIESEL GENERATOR RECEIVED AN AUTOSTART SIGNAL ON LOSS OF BUS VOLTAGE AT 2310 ON FEBRUARY 3, 1988. THE DIESEL GENERATOR WAS SHUT DOWN LOCALLY AT 2315. TH AUTOSTART OF THE DIESEL GENERATOR CONSTITUTES AN INADVERTENT ACTUATION OF AN ENGINEERED SAFETY FEATURE AND IS REPORTABLE IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(IV).

[195] RIVERBEND 1 DOCKET 50-458 LER 87-012 REV 01
UPDATE ON REACTOR SCRAM ON HIGH LEVEL SETPOINT DUE TO FEEDWATER REGULATING VALVE LOCKUP.
EVENT DATE: 061887 REPORT DATE: 021288 NSSS: GE TYPE: BWR
VENDOR: ELGAR, CORP.

(NSIC 208258) ON 6/18/87 AT 0322, WITH THE UNIT AT APPROXIMATELY 70 PERCENT POWER, A REACTOR TRIP OCCURRED. INITIATION OF THE REACTOR PROTECTION SIGNAL WAS CAUSED BY A REACTOR VESSEL WATER LEVEL - HIGH LEVEL 8 (51 INCHES) CONDITION. A LOSS OF CONTROL POWER TO PANEL 1VBN-PNL01B1 OCCURRED INADVERTENTLY DURING THE TROUBLE SHOOTING OF BATTERY INVERTER 1BYS-INV01BB *INVT*. LOSS OF CONTROL POWER TO THE FEEDWATER REGULATING VALVES CAUSED THEM TO LOCKUP IN A POSITION CONSISTENT WITH 70 PERCENT REACTOR POWER. ALSO, AS A RESULT OF THE LOSS OF THE INVERTER, THE RECIRCULATION SYSTEM FLOW CONTROL VALVES RAN BACK. SIMULTANEOUSLY, THE RECIRCULATION PUMPS RECEIVED A SIGNAL TO TRANSFER TO THE LOW FREQUENCY MOTOR GENERATORS. THIS CAUSED SUFFICIENT FEEDWATER FLOW/STEAM FLOW MISMATCH TO INCREASE THE VESSEL LEVEL TO HIGH LEVEL 8. OPERATIONS PERSONNEL RESPONDED BY SATISFACTORILY IMPLEMENTING THE IMMEDIATE AND SUBSEQUENT ACTIONS REQUIRED BY "REACTOR SCRAM" PROCEDURES. PROCEDURAL REVISIONS HAVE BEEN COMPLETED THAT WILL PRECLUDE RECURRENCE BY REQUIRING THE PLACEMENT OF THE BATTERY INVERTER IN MANUAL BYPASS MODE PRIOR TO TROUBLE SHOOTING. THERE WAS NO ADVERSE IMPACT ON THE SAFE OPERATION OF THE PLANT OR TO THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT. THE PLANT'S RESPONSE WAS IN A CONSERVATIVE DIRECTION WITH NO IMPACT ON SAFETY SYSTEMS. PLANT STAFF'S RESPONSE WAS IN ACCORDANCE WITH APPROVED PROCEDURES.

[196] RIVERBEND 1 DOCKET 50-458 LER 87-028 REV 01
UPDATE ON RESIDUAL HEAT REMOVAL SYSTEM ISOLATION DUE TO HIGH DIFFERENTIAL TEMPERATURE.
EVENT DATE: 110687 REPORT DATE: 022988 NSSS: GE TYPE: BWR

(NSIC 208400) AT APPROXIMATELY 1900 ON 11/6/87, THE UNIT IN OPERATIONAL CONDITION 5 (REFUELING) AND THE RESIDUAL HEAT REMOVAL SYSTEM (RHR) IN THE SHUTDOWN COOLING MODE OF OPERATION AND PUMPING THE UPPER CONTAINMENT FUEL POOLS TO THE RADWASTE

SYSTEM, THE SHUTDOWN COOLING SUCTION VALVE (1E12*MOVF009) AND THE DISCHARGE VALVE TO RADWASTE (1E12*MOVF049) AUTOMATICALLY CLOSED. AS A RESULT, RHR PUMP A (1E12*PC002A) TRIPPED. THE CAUSE OF THESE ISOLATIONS WAS AN ACTUAL HIGH DIFFERENTIAL TEMPERATURE SIGNAL FROM THE RHR A PUMP ROOM LEAK DETECTION SYSTEM AT THE TIME OF THE ISOLATIONS, THE RHR A PUMP ROOM UNIT COOLER, 1HVR*UC6, WAS OUT OF SERVICE. THE UNIT COOLER WAS RETURNED TO SERVICE, THE ISOLATION SIGNAL WAS RESET, AND THE RHR A LOOP WAS SUBSEQUENTLY RETURNED TO SERVICE AT 1905. THE CAUSE OF THE HIGH DIFFERENTIAL TEMPERATURE CONDITION WAS NATURAL CONVECTIVE AIR FLOW THROUGH THE UNIT COOLER. HOWEVER, THE HIGH DIFFERENTIAL TEMPERATURE CONDITION COULD HAVE BEEN PREVENTED IF THE UNIT COOLER HAD BEEN PLACED IN OPERATION. SINCE NO ACTUAL REACTOR COOLANT SYSTEM LEAKAGE EXISTED AND SHUTDOWN COOLING WAS RESTORED IN APPROXIMATELY FIVE MINUTES, THERE WAS NO IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT.

[197] RIVERBEND 1 DOCKET 50-458 LER 87-032
MANUAL REACTOR SCRAM DUE TO CONTROL ROD DRIVE TRIP CAUSED BY DEFICIENT PROCEDURE.
EVENT DATE: 121987 REPORT DATE: 011888 NSSS: GE TYPE: BWR

(NSIC 208067) ON 12/19/87 AT 0918 WITH THE UNIT IN STARTUP (OPERATIONAL CONDITION 2) A MANUAL SCRAM WAS INITIATED BY PLACING THE REACTOR MODE SWITCH IN THE SHUTDOWN POSITION IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.1.3.3, "CONTROL ROD SCRAM ACCUMULATORS". WITH THE UNIT SUBCRITICAL, THE CONTROL ROD DRIVE HYDRAULIC PUMP TRIPPED ON LOW SUCTION PRESSURE. APPROXIMATELY ONE MINUTE LATER, TWO CONTROL ROD ACCUMULATOR FAULT: ALARMS WERE RECEIVED, AND THE OPERATOR PLACED THE REACTOR MODE SWITCH IN THE SHUTDOWN POSITION, INITIATING A REACTOR SCRAM. A LEAK FROM THE SHORT CYCLE CONDENSATE LINE NECESSITATED A SHUTDOWN OF THE CONDENSATE SYSTEM SHUTTING OFF THE NORMAL SUPPLY OF WATER TO THE CONTROL ROD DRIVE (CRD) HYDRAULIC PUMPS. THE BACKUP SUPPLY OF WATER FROM THE CONDENSATE STORAGE TANK WAS UNABLE TO SUPPLY WATER DUE TO A CLOSED ISOLATION VALVE ATTRIBUTED TO A DEFICIENCY IN SYSTEM OPERATING PROCEDURE (SOP)-0002, "CONTROL ROD DRIVE HYDRAULICS". IN ADDITION, THE OPERATOR FAILED TO OBTAIN PROPER AUTHORIZATION FROM THE SHIFT SUPERVISOR PRIOR TO REPOSITIONING THE VALVE, IN VIOLATION OF ADMINISTRATIVE PROCEDURE (ADM)-0020, "PLANT KEY CONTROL", AND OPERATIONS SECTION PROCEDURE (OSP)-0014, AND "CONTROL OF LOCKED VALVES AND DEVICES". THE CRD VALVE LINE-UP WAS IMMEDIATELY RESTORED AND THE PUMP RESTARTED. THE DEFICIENT SOP WAS CORRECTED AND REVISED.

[198] RIVERBEND 1 DOCKET 50-458 LER 88-001
MISSED TECHNICAL SPECIFICATION ACTION ON SCRAM DISCHARGE VOLUME LEVEL TRANSMITTER DUE TO PERSONNEL ERROR DURING THE PERFORMANCE OF A SURVEILLANCE TEST PROCEDURE.
EVENT DATE: 010888 REPORT DATE: 020888 NSSS: GE TYPE: BWR
VENDOR: GOULD SWITCHGEAR DIVISION

(NSIC 208279) ON 1/8/88 AT APPROXIMATELY 1305, WITH THE UNIT IN POWER OPERATION (MODE 1) AT 100 PERCENT POWER, SCRAM DISCHARGE VOLUME (SDV) REACTOR PROTECTION SYSTEM (RPS) LEVEL TRANSMITTER 1C11*LTN012D WAS DISCOVERED TO HAVE BEEN OUT OF TOLERANCE FOR APPROXIMATELY ELEVEN HOURS. TECHNICAL SPECIFICATION 3.3.1 REQUIRES PLACING THE INOPERABLE CHANNEL IN THE TRIPPED CONDITION WITHIN ONE HOUR. INVESTIGATION REVEALED THAT THE OVERSIGHT OF THE OPERATOR RECORDING DATA FOR SURVEILLANCE TEST PROCEDURE (STP)-000-0001, "DAILY OPERATING LOGS," LEAD TO THE OUT-OF-TOLERANCE INDICATION GOING UNNOTICED FOR A GREATER TIME THAN THAT ALLOWED BY TECHNICAL SPECIFICATIONS. UPON DISCOVERY, THE CHANNEL WAS PLACED IN THE TRIPPED CONDITION TO COMPLY WITH REQUIRED TECHNICAL SPECIFICATION ACTION. THE CHANNEL "D" TRANSMITTER, 1C11*LTN012D, WAS DISCOVERED TO HAVE DRIFTED OUT-OF-TOLERANCE AND WAS RECALIBRATED AND RESTORED TO OPERABLE STATUS. THE INVOLVED CONTROL ROOM SHIFT CREW WAS COUNSELED ON THE IMPORTANCE OF STP-000-0001 AND ITS REVIEW PRIOR TO SIGN-OFF. A REVIEW OF THIS EVENT WILL BE INCLUDED IN OPERATOR REQUALIFICATION TRAINING. THE OUT-OF-TOLERANCE CONDITION OF THE TRANSMITTER

PLACED THE SDV HIGH LEVEL SETPOINT IN A MORE CONSERVATIVE CONDITION AND WOULD HAVE FULFILLED ITS INTENDED SAFETY FUNCTION.

[199] RIVERBEND 1 DOCKET 50-458 LER 88-002
ALTERNATE ROD INSERTION/ANTICIPATED TRANSIENT WITHOUT SCRAM (ARI/ATWS) SYSTEM INITIATION CAUSING REACTOR SCRAM DUE TO MISLEADING SURVEILLANCE TEST PROCEDURE WORDING.
EVENT DATE: 011088 REPORT DATE: 020988 NSSS: GE TYPE: BWR

(NSIC 208301) ON 1/10/88 AT APPROXIMATELY 0933 HOURS, WITH THE UNIT IN POWER OPERATION AT APPROXIMATELY 95 PERCENT POWER, INSTRUMENT AND CONTROLS (I&C) TECHNICIANS INADVERTENTLY INITIATED A FULL SCRAM VIA THE ALTERNATE ROD INSERTION (ARI) TRIP SYSTEM, WHILE PERFORMING SURVEILLANCE TEST PROCEDURE (STP)-051-4269 "ATWS RECIRC PUMP TRIP REACTOR VESSEL PRESSURE HIGH MONTHLY CHANNEL FUNCTIONAL, 18 MONTH CHANNEL CAL, 18 MONTH LSFT." INVESTIGATION REVEALED THAT THE CHANNEL CAL, 18 MONTH LSFT." INVESTIGATION REVEALED THAT THE TECHNICIANS LIFTED LEADS FROM THE WRONG TERMINAL BLOCK, WHICH CAUSED TWO ARI TRIP UNITS TO LOSE THEIR SIGNAL AND CAUSED AN ARI/ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) SYSTEM INITIATION. THE STP WAS MISLEADING IN THE LOCATION OF THE TERMINAL BLOCK AND IMPROPERLY LED THE TECHNICIANS TO TERMINAL BLOCK 2B6 INSTEAD OF THE CORRECT TERMINAL BLOCK, TB0006. THE STP WAS REVISED, AND THE PROCEDURE WAS RERUN WITH NO PROBLEM. I&C TECHNICIANS WERE TRAINED ON THIS INCIDENT THROUGH SHOP TRAINING. AFTW STPS WERE REVIEWED, EVALUATED, AND REWORDED TO CLEARLY DEFINE THE CORRECTED TERMINAL BOARD LOCATION. THE INITIATION OF ARI CAUSED ALL CONTROL RODS TO BE INSERTED SHUTTING DOWN THE REACTOR. ALL PLANT RESPONSES OCCURRED AS DESIGNED. THERE WAS NO IMPACT ON THE SAFE OPERATION OF THE PLANT OR TO THE HEALTH AND SAFETY OF THE PUBLIC.

[200] RIVERBEND 1 DOCKET 50-458 LER 88-003
MAIN TURBINE RUNBACK/REACTOR SCRAM CAUSED BY LOSS OF MAIN GENERATOR STATOR COOLING DUE TO FAILED TEMPERATURE CONTROLLER.
EVENT DATE: 012888 REPORT DATE: 022688 NSSS: GE TYPE: BWR

(NSIC 208465) AT 1612 ON 1/28/88 WITH THE UNIT IN POWER OPERATION (MODE 1) AT 100% POWER, VIBRATION INDUCED FAILURE OF A TEMPERATURE CONTROLLER IN THE MAIN GENERATOR STATOR COOLING SYSTEM CAUSED A MAIN TURBINE RUNBACK. THE TURBINE SHED ITS LOAD FASTER THAN REACTOR POWER COULD BE DECREASED CAUSING AN INCREASE IN REACTOR PRESSURE. AT 1615 THE REACTOR SCRAMMED ON HIGH PRESSURE. ALL SYSTEMS RESPONDED PROPERLY DURING THE TRANSIENT WITH THE EXCEPTION OF THE 'B' REACTOR RECIRC PUMP WHICH TRIPPED OFF FROM HIGH SPEED INSTEAD OF SHIFTING TO LOW SPEED. THE STATOR TEMPERATURE CONTROLLER WAS REPAIRED AND RELOCATED TO A BUILDING STRUCTURAL MEMBER TO PROTECT IT FROM VIBRATION. THERE WAS NO ADVERSE IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT.

[201] RIVERBEND 1 DOCKET 50-458 LER 88-004
REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION DUE TO INADVERTENT SHORTING OF LEADS AND PROCEDURAL ERROR.
EVENT DATE: 020288 REPORT DATE: 030388 NSSS: GE TYPE: BWR

(NSIC 208466) ON THE FOLLOWING DATES AND APPROXIMATE TIMES, 2/2/88 AT 1100, 2/4/88 AND 2/23/88 AT 2340, WITH THE UNIT IN POWER OPERATION (MODE 1) AT 100% POWER, ISOLATIONS OF THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM OCCURRED. EACH OF THESE ISOLATIONS OCCURRED DURING THE PERFORMANCE OF A SURVEILLANCE TEST PROCEDURE (STP). THE ISOLATIONS OCCURRING ON 2/2/88 AND 2/4/88 WERE CAUSED, RESPECTIVELY, BY THE INADVERTENT SHORTING OF A LEAD IN A CONFINED AREA AND A WRONG RELAY LISTED IN AN STP CAUSING TECHNICIANS TO DETERMINATE THE WRONG LEAD. CORRECTIVE ACTION FOR THE FIRST TWO ISOLATIONS CONSISTED OF REVISING THE PROCEDURES VIA TEMPORARY CHANGE NOTICES (TCNS). IN IMPLEMENTING THE TCNS TO

ADDRESS THE LEAD LOCATION/ACCESSIBILITY PROBLEM, A PROCEDURAL ERROR WAS INTRODUCED IN WHICH A LEAD WAS REQUIRED BY THE STP TO BE RETERMINATED PRIOR TO RESETTING ISOLATION LOGIC. THIS RESULTED IN THE RCIC ISOLATION ON 2/23/88. IN EACH CASE, THE ISOLATIONS WERE RESET, THE SUBJECT STPS WERE REVISED VIA TCN, AND EACH STP WAS SUCCESSFULLY COMPLETED. THE EVENTS WERE ASSESSED AS NOT SAFETY SIGNIFICANT SINCE THE ISOLATIONS OCCURRED AS DESIGNED, AND HIGH PRESSURE EMERGENCY CORE COOLING SYSTEMS (ECCS) WERE AVAILABLE. THEREFORE, THERE WAS NO ADVERSE IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THESE EVENTS.

[202] ROBINSON 2 DOCKET 50-261 LER 88-001
AUTOMATIC REACTOR TRIP DUE TO TURBINE TRIP ON LOSS OF AUTOSTOP OIL PRESSURE.
EVENT DATE: 011988 REPORT DATE: 021888 NSSS: WE TYPE: PWR
VENDOR: LUNKENHEIMER CO., THE

(NSIC 208262) AN AUTOMATIC REACTOR TRIP OCCURRED AT 0216 HOURS, TUESDAY, JANUARY 19, 1988. THE TRIP WAS INITIATED BY AN AUTOMATIC TURBINE TRIP ON A TWO-OUT-OF-THREE AUTOSTOP OIL PRESSURE LOGIC. THE TURBINE WAS UNDER SURVEILLANCE TESTING AT THE TIME, WITH THE REACTOR AT 66 PERCENT POWER. NORMAL OPERATION OVER A PERIOD OF YEARS HAD APPARENTLY CAUSED THE SEAT OF ONE OF THE TWO AUTOSTOP OIL PRESSURE REGULATING VALVES TO WEAR SO THAT THE VALVE WAS UNABLE TO HOLD PRESSURE. WHEN THE TRIP TEST LEVER WAS RETURNED TO THE RESET POSITION FOLLOWING THE TURBINE OVERSPEED TRIP TEST, THE BACKPRESSURE FURTHER AGGRAVATED THE RELIEF AND SUFFICIENT PRESSURE WAS LOST TO CAUSE THE TURBINE TRIP AND, SUBSEQUENTLY, THE AUTOMATIC REACTOR TRIP. BOTH REGULATING VALVES WERE REPLACED WITH VALVES OF A KNOWN HIGHER RELIABILITY IN ACCORDANCE WITH RECOMMENDATIONS FROM THE TURBINE VENDOR. THE REACTOR AND TURBINE WERE RETURNED TO POWER OPERATION USING STANDARD STARTUP AND OPERATING PROCEDURES. THE LICENSEE NOTIFIED THE NRC EMERGENCY OPERATIONS CENTER VIA THE EMERGENCY NOTIFICATION SYSTEM PURSUANT TO 10CFR50.72(B)(2)(II) FOR A FOUR-HOUR NONEMERGENCY EVENT. THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(IV).

[203] ROBINSON 2 DOCKET 50-261 LER 88-002
POTENTIAL NONCONSERVATIVE REACTOR PROTECTION SETPOINT DUE TO INCORRECT ANALYSIS.
EVENT DATE: 012288 REPORT DATE: 021988 NSSS: WE TYPE: PWR
VENDOR: EXXON NUCLEAR CO., INC.

(NSIC 208322) THIS LER IS SUBMITTED TO PROVIDE INFORMATION OF POTENTIAL INTEREST TO THE INDUSTRY. ON 1/22/88, IT WAS IDENTIFIED THAT A POTENTIALLY NONCONSERVATIVE TRIP SETTING EXISTED IN THE OVERTEMPERATURE DELTA TEMPERATURE SETPOINT AS NOTED IN TECH SPEC 2.3.1.2.D AS A RESULT OF AN ERROR IN THE INSTRUMENT RESPONSE TIME ASSUMED CERTAIN ACCIDENT ANALYSES. THE VALUE USED WAS 2.3 SECONDS VERSUS 6.0 SECONDS WHICH SHOULD HAVE BEEN USED. THE TRIP POINT RESPONSE DELAY TIME IS USED IN THE FUEL VENDORS ACCIDENT ANALYSES FOR THE FOLLOWING EVENTS: 1) LOSS OF EXTERNAL LOAD; 2) UNCONTROLLED ROD WITHDRAWAL; AND 3) ROD DROP WITH ACTIVE TURBINE RUNBACK. THE CAUSE OF THIS CONDITION WAS ATTRIBUTED TO THE CURRENT FUEL VENDORS USE OF THE DELAY TIME DATA USED IN THE ORIGINAL ANALYSIS TO DEVELOP THE REQUIRED ACCIDENT ANALYSIS CODES FOR THE CORE. AN INTERIM SETPOINT WAS DEVELOPED AND IMPLEMENTED USING VERY CONSERVATIVE VALUES FOR USE IN THE PROTECTION SYSTEM UNTIL COMPLETION OF A FINAL ANALYSES, WHICH WAS IN PROGRESS AT THAT TIME. THE FINAL ANALYSIS WAS RECEIVED BY CP&L ON 2/18/88. THE RESULTS OF THE ANALYSIS CONCLUDED THAT APPLICABLE ACCEPTANCE CRITERIA ARE MET WITH THE CURRENT TECH SPEC LIMIT ON F DELTA H OF 1.65 AND THE CURRENT TECH SPEC OVERTEMPERATURE DELTA-T TRIP FUNCTION.

[204] ROBINSON 2 DOCKET 50-261 LER 88-003
LOSS OF SAFETY INJECTION PUMP AUTOSTART DUE TO EIGHT SINGLE-FAILURE SCENARIOS.
EVENT DATE: 012888 REPORT DATE: 022788 NSSS: WE TYPE: PWR
VENDOR: GOULD-NATIONAL BAIT

WORTHINGTON PUMP CORP.

(NSIC 208404) DURING REVIEW FOR AN NRC REQUEST FOR ADDITIONAL INFORMATION, THE
LICENSEE FOUND AN ORIGINAL DESIGN SINGLE-FAILURE DISCREPANCY: FAILURE OF "B"
BATTERY (DC CONTROL 1 POWER) DURING SAFETY INJECTION (S COULD LEAVE ONLY ONE SI
PUMP AVAILABLE FOR AUTOSTART. THE PLANT NUCLEAR SAFETY COMMITTEE WAS CONVENED
AND DETERMINED THIS WAS AN UNANALYZED CONDITIONS SINCE SAFETY ANALYSES ASSUME TWO
SI PUMPS AVAILABLE. THE NRC WAS NOTIFIED AT 1749 HOURS, JANUARY 28, 1988, PER
10CFR50.72(B)(1)(II)(A). HOT SHUTDOWN WAS REQUIRED WITHIN EIGHT HOURS. AT 2356
HOURS, THE NRC WAS NOTIFIED THAT THE DISCREPANCY HAD BEEN RESOLVED AND THE PLANT
WAS RETURNED TO FULL POWER AT 0535 HOURS, JANUARY 29. THEN, AT 1410, THE NRC
WAS NOTIFIED OF ANOTHER ASPECT OF THE UNANALYZED CONDITION: LOSS OF "A" BATTERY
COULD RESULT IN LOSS OF "A" DIESEL GENERATOR AND EMERGENCY BUS E-1, THE POWER
SUPPLY FOR TWO OF THE THREE INJECTION PUMPS. HOT SHUTDOWN WAS ACHIEVED AT 2026
AND COLD SHUTDOWN AT 1942 JANUARY 30. FURTHER REVIEW FOUND SEVEN ADDITIONAL
SCENARIOS FOR A TOTAL OF EIGHT POSTULATED SINGLE-FAILURE EVENTS. SEVEN SCENARIOS
WERE RESOLVED BY FEBRUARY 12, WITH ONE REMAINING, PENDING NRC APPROVAL OF A
LICENSE AMENDMENT RESTRICTING MAXIMUM POWER TO 60% WITH AUTOMATIC OPERATION OF
ONLY ONE INJECTION PUMP ASSUMING A SINGLE FAILURE.

[205] SALEM 1 DOCKET 50-272 LER 84-021 REV 01
UPDATE ON CONTAINMENT ISOLATION VALVE INOPERABLE DUE TO EQUIPMENT FAILURE.
EVENT DATE: 101484 REPORT DATE: 021088 NSSS: WE TYPE: PWR
VENDOR: LIMITORQUE CORP.

(NSIC 208247) ON OCTOBER 14, 1984, DURING A REACTOR STARTUP, 1CC131 (THE
COMPONENT COOLING WATER RETURN ISOLATION VALVE FROM THE REACTOR COOLANT PUMP
THERMAL BARRIERS) CLOSED. 1CC131, IS DESIGNED TO CLOSE UPON A HIGH FLOW
CONDITION IN THE EVENT OF A THERMAL BARRIER FAILURE; HOWEVER, THIS WAS AN
INADVERTENT CLOSURE RESULTING FROM A PRESSURE TRANSIENT INDUCED BY THE FLEXING OF
THE COMPONENT COOLING WATER HEAT EXCHANGER PLATES WHEN A SERVICE WATER PUMP WAS
STARTED. UPON AN OPENING ATTEMPT, THE "OPEN" LIGHT WAS NOT RECEIVED. THE MOTOR
APPARENTLY DID NOT DE-ENERGIZE FOLLOWING THE OPENING ATTEMPT, RESULTING IN
EXCESSIVE CURRENT DRAW AND MOTOR FAILURE. 1CC131 WAS DECLARED INOPERABLE AND
TECHNICAL SPECIFICATION ACTION STATEMENT 3.6.1.D WAS ENTERED. THE REDUNDANT
ISOLATION VALVE (1CC190) WAS MAINTAINED IN AN OPERABLE STATUS AND THE UNIT WAS
PLACED IN HOT STANDBY. 1CC131 WAS REPAIRED, TESTED AND RETURNED TO AN OPERABLE
STATUS. AN ENGINEERING REVIEW OF THIS EVENT HAS BEEN COMPLETED. THE ROOT CAUSE
OF THIS EVENT HAS BEEN ATTRIBUTED TO AN EQUIPMENT FAILURE. CORRECTIVE ACTION
INCLUDED REPAIR OF THE 1CC131 LIMITORQUE MOTOR OPERATOR, MINOR PROCEDURAL
MODIFICATIONS, AND A DESIGN CHANGE TO MODIFY THE CIRCUITRY THAT GOVERNS THE
1CC131 HIGH FLOW CLOSURE.

[206] SALEM 2 DOCKET 50-311 LER 88-001
CHEMISTRY SAMPLE TAKEN LATE DUE TO INADEQUATE ADMINISTRATIVE CONTROLS
EVENT DATE: 011288 REPORT DATE: 020288 NSSS: WE TYPE: PWR

(NSIC 208226) ON JANUARY 12, 1988 AT 0940 HOURS CHEMISTRY WAS REQUIRED TO DRAW A
SAMPLE IN ACCORDANCE WITH TECH SPEC TABLE 3.3-12 ACTION 28.C. HOWEVER, THE
SAMPLE WAS NOT TAKEN UNTIL 1345 HOURS THAT DAY CONTRARY TO THE REQUIREMENTS OF
THE ACTION STATEMENT. TECH SPEC TABLE 3.3-12 ACTION 28.C WAS ENTERED ON 1/9/88
AT 1705 HOURS WHEN THE NOS. 23 AND 25 CFCUS SERVICE WATER EFFLUENT RADIATION
MONITOR (2R13C) WAS DECLARED INOPERABLE DUE TO CHANNEL SPIKING. AS PER TECH
SPECS, SAMPLES WERE TAKEN ON 1/8/88 AT 2010 HOURS, 1/10/88 AT 0945 HOURS AND
1/11/88 AT 0940 HOURS. HOWEVER, ON 1/12/88 THE SAMPLE WAS NOT TAKEN UNTIL 1345
HOURS. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO INADEQUATE ADMINISTRATIVE
CONTROLS. CORRECTIVE ACTIONS INCLUDE ISSUANCE OF A MEMO TO ALL CHEMISTRY
DEPARTMENT PERSONNEL ADDRESSING A NEW FORM WHICH WILL HELP TRACK TECH SPEC ACTION
STATEMENT SAMPLING REQUIREMENTS, A RADIATION PROTECTION/CHEMISTRY DEPARTMENT

MANAGEMENT REVIEW OF THIS EVENT WITH APPLICABLE DEPARTMENT PERSONNEL, AND A REVIEW OF THE RADIATION PROTECTION/CHEMISTRY DEPARTMENT IMPLEMENTING PROCEDURES TO ENSURE THE CRITERIA FOR ACTION STATEMENT COMPLIANCE IS CLEAR.

[207] SALEM 2 DOCKET 50-311 LER 88-002
FOUR SW PUMPS INOPERABLE DUE TO AN EQUIPMENT FAILURE.
EVENT DATE: 011388 REPORT DATE: 020288 NSSS: WE TYPE: PWR

(NSIC 208227) ON JANUARY 13, 1988 AT 1216 HOURS, AN OPERATOR OBSERVED EXCESSIVE LEAKAGE FROM NO. 22 SW PUMP. NO. 22 SW PUMP WAS DECLARED INOPERABLE. WITH THE UNIT ALREADY ENTERED IN TECH SPEC ACTION STATEMENT 3.7.4 DUE TO THE INOPERABILITY OF THREE OTHER SW PUMPS (21, 25 AND 26), TECH SPEC ACTION STATEMENT 3.0.3 WAS ENTERED AT 1314 HOURS. WITH FOUR INOPERABLE SW PUMPS, OPERATIONS PROCEDURE OD-12, "TECH SPEC INTERPRETATIONS" STATES THAT CREDIT FOR AN OPERABLE SW LOOP CANNOT BE TAKEN. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO EQUIPMENT MALFUNCTION. INVESTIGATION OF THE NO. 22 SW PUMP LEAKAGE REVEALED A DISLOCATED TUBE TENSIONER. NO. 22 SW PUMP WAS REPLACED WITH A SPARE PUMP AND NO. 26 SW PUMP PREVENTIVE MAINTENANCE SILT INSPECTIONS WERE COMPLETED. BOTH PUMPS WERE SUBS' 'NTLY RETURNED TO SERVICE.

[2(SAN ONOFRE 1 DOCKET 50-206 LER 88-003
SAFETY INJECTION SYSTEM SUCTION PIPING BORON CONCENTRATION REQUIREMENTS RELAXED.
EVENT DATE: 020888 REPORT DATE: 021988 NSSS: WE TYPE: PWR

(NSIC 208314) THIS REPORT IS BEING SUBMITTED AS AN INFORMATIONAL LER IN ORDER TO PROVIDE NOTIFICATION OF A CONDITION INVOLVING SAFETY INJECTION SYSTEM (SI) PIPING BORON CONCENTRATIONS. DUE TO THE CONFIGURATION OF THE SI PIPING (THE MAIN FEEDWATER PUMPS (MFP) ALSO SERVE AS SI PUMPS), MINOR INLEAKAGE FROM THE MAIN FEEDWATER SYSTEM MAY DILUTE THE FLUID IN THE SI PIPING FROM THE MFP TO A BORON CONCENTRATION BELOW 3750 PPM. BECAUSE IT WAS SCE'S UNDERSTANDING THAT THE MAIN STEAM LINE BREAK (MSLB) ACCIDENT ANALYSIS ASSUMED A MINIMUM CONCENTRATION IN THE SI PIPING OF 3750 PPM, MONTHLY SAMPLING AND, IF NECESSARY, RECIRCULATION OF THE SI SUCTION PIPING HAD BEEN PERFORMED. IN 1985, DUE TO THE DIFFICULTY IN MAINTAINING THIS CONCENTRATION, SCE ASKED WESTINGHOUSE TO REVIEW THE ANALYSIS TO DETERMINE THE MINIMUM ACCEPTABLE CONCENTRATION. ON 5/29/85, SCE WAS INFORMED BY WESTINGHOUSE THAT THE MSLB ANALYSIS ASSUMED 0 PPM BORON CONCENTRATION FOR THE SI PIPING. BASED UPON WESTINGHOUSE'S WRITTEN RESPONSE, SCE RELAXED ITS VIGILANCE OF THE SI PIPING BORON CONCENTRATION AND THERE HAVE BEEN OCCASIONS WHEN BORON CONCENTRATION MAY HAVE BEEN AS LOW AS 1500 PPM FOR SHORT PERIODS OF TIME. IN SUPPORT OF AN ONGOING EVALUATION OF SI SYSTEM CAPABILITIES, WESTINGHOUSE RECENTLY PROVIDED SCE WITH MSLB ANALYSIS INFORMATION.

[209] SAN ONOFRE 2 DOCKET 50-361 LER 88-002
INADVERTENT TOXIC GAS ISOLATION SYSTEM (TGIS) ACTUATION DUE TO HUMAN ERROR IN THE APPLICATION OF ADMINISTRATIVE CONTROLS.
EVENT DATE: 011988 REPORT DATE: 021788 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 208339) ON JANUARY 19, 1988 AT 1345 WI' UNIT 2 IN MODE 1 AT 100% REACTOR POWER, AND UNIT 3 IN MODE 3, THE TRAIN "B" TOXIC GAS ISOLATION SYSTEM (TGIS) STARTED BOTH TRAINS OF THE CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM (CREACUS) ON HIGH AMMONIA GAS LEVEL. CREACUS OPERATED IN THE ISOLATION MODE AS DESIGNED, WHILE IT WAS DETERMINED THAT NO AMMONIA GAS WAS PRESENT. THE ACTUATION OCCURRED BECAUSE THE TGIS TRAIN "B" ACTUATION HAD NOT BEEN BYPASSED DURING RE-CALIBRATION OF THE AMMONIA CHANNEL AS REQUIRED BY PROCEDURE. THE TGIS RE-CALIBRATION WAS REQUIRED BECAUSE OF SET POINT DRIFT FOLLOWING CALIBRATION. THE TGIS AMMONIA CHANNEL WAS NOT BYPASSED FOR THE RE-CALIBRATION DUE TO HUMAN ERROR IN THE APPLICATION OF REQUIRED ADMINISTRATIVE CONTROLS. THIS EVENT HAS BEEN DISCUSSED

WITH THE INDIVIDUALS INVOLVED. FURTHER, A DESCRIPTION OF THIS EVENT AND ITS CAUSES WILL BE REQUIRED READING FOR ALL INSTRUMENTATION AND CONTROL MAINTENANCE PERSONNEL. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE ALL TGIS AND CREACUS COMPONENTS OPERATED AS DESIGNED.

[210] SAN ONOFRE 2 DOCKET 50-361 LER 88-003
INADVERTENT TRAIN "A" TOXIC GAS ISOLATION SYSTEM ACTUATION DUE TO THE USE OF CLEANING SOLVENTS NEAR THE MONITOR.
EVENT DATE: 012088 REPORT DATE: 021888 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 208340) ON JANUARY 20, 1988 AT 1320 WITH UNIT 2 IN MODE 1 AT 100% REACTOR POWER, AND UNIT 3 IN MODE 4, THE TRAIN "A" TOXIC GAS ISOLATION SYSTEM (TGIS) STARTED BOTH TRAINS OF THE CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM (CREACUS) ON HIGH AMMONIA AND BUTANE GAS LEVELS. CREACUS OPERATED IN THE ISOLATION MODE AS DESIGNED, WHILE IT WAS DETERMINED THAT NEITHER AMMONIA NOR BUTANE GAS WAS PRESENT AND THAT THE ACTUATION RESULTED FROM UNAUTHORIZED USE OF A CLEANING SOLVENT NEAR THE TGIS TRAIN "A" MONITOR CABINET AND SUPPORT EQUIPMENT. THE HOUSEKEEPING PERSON USING THE SOLVENT WAS AWARE OF THE POSTED REQUIREMENT TO OBTAIN A CONTROL ROOM OPERATOR'S APPROVAL PRIOR TO USING CLEANING SOLVENTS IN THE AREA NEAR THE TGIS MONITORS. HOWEVER, THE PERSON DID NOT KNOW THAT THE CLEANER HE WAS USING WAS SUCH A CLEANING SOLVENT. CORRECTIVE ACTION HAS BEEN INITIATED TO INFORM APPROPRIATE HOUSEKEEPING PERSONNEL OF THOSE CLEANING MATERIALS WHICH CANNOT BE USED IN THIS AREA WITHOUT NOTIFYING THE CONTROL ROOM. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE TRAIN "A" TGIS CONSERVATIVELY ACTUATED AND BOTH TRAINS OF CREACUS ACTUATED AS DESIGNED.

[211] SAN ONOFRE 3 DOCKET 50-362 LER 87-001 REV 01
UPDATE ON FOUR SNUBBERS NOT INCLUDED IN SNUBBER SURVEILLANCE PROGRAM DUE TO OVERSIGHT DURING INITIAL PROGRAM DEVELOPMENT.
EVENT DATE: 010987 REPORT DATE: 022388 NSSS: CE TYPE: PWR
VENDOR: PACIFIC SCIENTIFIC COMPANY

(NSIC 208398) AS REPORTED IN LER 87-001, REV. 0, ON 1/9/87, WITH UNIT 3 IN MODE 5, A REVIEW OF SNUBBER DOCUMENTATION IDENTIFIED TWO SNUBBERS ON A 3/4 INCH COMPONENT COOLING WATER SYSTEM (EIIS SYSTEM CODE CC) INSTRUMENT LINE WHICH WERE NOT INCLUDED IN THE SNUBBER SURVEILLANCE PROGRAM AND WHICH WERE THEREFORE INOPERABLE BECAUSE THEY HAD NOT BEEN TESTED AS REQUIRED BY TECHNICAL SPECIFICATION 4.7.6. THE SNUBBERS WERE SATISFACTORILY TESTED AND RETURNED TO OPERABLE STATUS ON 1/10/87. NO OTHER UNTESTED UNIT 3 SNUBBERS WERE IDENTIFIED BY THIS REVIEW. SUBSEQUENTLY, A REVIEW OF UNIT 2 SNUBBER DOCUMENTATION WAS INITIATED. THIS REVIEW IDENTIFIED EIGHT UNTESTED UNIT 2 SNUBBERS WHICH WERE SUBSEQUENTLY TESTED SATISFACTORILY, AS REPORTED IN LER 87-025 (DOCKET 50- 361). THIS REVIEW ALSO IDENTIFIED THE POSSIBILITY THAT THE UNIT 3 DOCUMENTATION REVIEW MAY NOT HAVE BEEN COMPLETE. A RE-REVIEW OF UNIT 3 DOCUMENTATION HAS IDENTIFIED TWO UNTESTED SNUBBERS ON A 3/4 INCH VENT LINE FOR A REACTOR COOLANT PUMP SEAL (EIIS SYSTEM CODE AB). THE FIRST SNUBBER WAS IDENTIFIED AND SATISFACTORILY TESTED ON 1/24/88 WITH UNIT 3 IN MODE 3, AND THE SECOND WAS IDENTIFIED AND SATISFACTORILY TESTED ON 1/25/88 WITH UNIT 3 IN MODE 2. ALL OF THE SNUBBERS NOTED IN THIS REPORT ARE PACIFIC SCIENTIFIC PSA-1/4-4 MODELS, RATED AT 350 LBS AND WERE APPARENTLY OVERLOOKED DURING INITIAL REVIEW OF PIPE STRESS CALCULATION.

[212] SAN ONOFRE 3 DOCKET 50-362 LER 88-001
DELINQUENT COLLECTION OF CONTAINMENT PURGE SAMPLES DUE TO INADEQUATE ADMINISTRATIVE CONTROLS.
EVENT DATE: 011888 REPORT DATE: 021788 NSSS: CE TYPE: PWR

(NSIC 208341) ON 1/17/88, AT 0216, FOLLOWING A SHUTDOWN FROM 80% POWER,

CONTAINMENT PURGING WAS INITIATED. TECHNICAL SPECIFICATION (TS) SURVEILLANCE REQUIREMENT 4.11.2 1.2, TABLE 4.11-2 REQUIRES THAT CONTINUOUS IODINE AND PARTICULATE SAMPLES FROM THE CONTAINMENT PURGE STACK BE TAKEN, COLLECTED AT LEAST ONCE PER 24 HOURS FOR AT LEAST 7 DAYS FOLLOWING EACH SHUTDOWN, AND ANALYSES BE COMPLETED WITHIN 48 HOURS OF COLLECTION. CONTRARY TO TS, THE REQUIRED PURGE SAMPLES WERE NOT COLLECTED UNTIL 1/18/88 AT 1100 (THE SURVEILLANCE BECAME DELINQUENT AT 0216 ON 1/18/88, 24 HOURS AFTER PURGE WAS INITIATED). THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE SUBSEQUENT ANALYSES OF THE SAMPLES INDICATED THAT EFFLUENT RELEASES REMAINED WELL WITHIN TS LIMITS. THE IMMEDIATE CAUSE OF THIS EVENT WAS AN ERROR BY A CHEMISTRY TECHNICIAN. RATHER THAN COLLECTING THE PURGE SAMPLES AS INDICATED ON THE "SHIFT REQUIREMENTS" CHECKLIST, THE TECHNICIAN IMPROPERLY ASSUMED THAT THE SAMPLES HAD BEEN COLLECTED DURING THE PREVIOUS SHIFT AND, BELIEVING THAT THE STEPS WERE INVALID, WROTE "N/A" IN BOTH THE "PERFORMED" AND "VERIFIED" COLUMNS OF THE CHECKLIST FOR BOTH THE PARTICULATE AND IODINE SAMPLES. SINCE BOTH COLUMNS WERE MARKED "N/A", AN INDEPENDENT CHECK WHICH COULD HAVE IDENTIFIED THE ERRONEOUS ASSUMPTION WAS NOT PERFORMED.

[213] SEABROOK 1 DOCKET 50-443 LER 88-001
ESF COMPONENT ACTUATION DURING MAINTENANCE.
EVENT DATE: 012988 REPORT DATE: 022988 NSSS: WE TYPE: PWR
VENDOR: GOULD SWITCHGEAR DIVISION

(NSIC 208416) ON JANUARY 29, 1988 AT 1530 EST, DURING THE REPLACEMENT OF A RELAY COIL ASSEMBLY. THE CONTACTS ON THE RELAY CONTACT BLOCK WERE INADVERTENTLY DEPRESSED CAUSING THE ACTUATION OF CBA-FN-16A, THE CONTROL ROOM EMERGENCY AIR CLEANUP SUBSYSTEM FAN. THIS ACTUATION DID NOT RESULT IN ANY OTHER SAFETY SYSTEM RESPONSE OR REQUIRE ANY OPERATOR CORRECTIVE ACTIONS OTHER THAN TO SECURE THE FAN. ALL ASSOCIATED WORK WAS STOPPED FOLLOWED BY NOTIFICATION OF THE OCCURRENCE TO THE CONTROL ROOM. THE SUBJECT RELAY WAS RETURNED TO THE AS FOUND CONDITION PRIOR TO THE START OF WORK.

[214] SEQUOYAH 1 DOCKET 50-327 LER 87-039 REV 02
UPDATE ON CONTROL ROOM EMERGENCY VENTILATION SYSTEM SINGLE FAILURE CRITERIA VIOLATED DUE TO A DESIGN ERROR.
EVENT DATE: 071087 REPORT DATE: 020488 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 208064) ON JULY 10, 1987, WITH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN), IT WAS DETERMINED THAT THE POTENTIAL EXISTED FOR A SINGLE FAILURE OF THE MAIN CONTROL ROOM (MCR) NORMAL PRESSURIZATION SYSTEM, WHEN OPERATING DURING A CONTROL ROOM ISOLATION (CRI) ACTUATION, TO VIOLATE GENERAL DESIGN CRITERIA (GDC)-19 OF 10 CFR 50 APPENDIX A, "CONTROL ROOM." A MALFUNCTION IN THE CONTROLLER OF THE OPERATING NORMAL PRESSURIZATION FAN COULD RESULT IN THE CONTROL BUILDING (CB) PRESSURE OF THE LOWER FLOORS EXCEEDING THE PRESSURE IN THE MCR THEREBY ALLOWING UNFILTERED, POTENTIALLY RADIOACTIVE AIR TO LEAK INTO THE MCR. IN ADDITION, A SINGLE FAILURE OF THE OPERATING NORMAL PRESSURIZATION FAN SUCTION DAMPER IN THE CLOSED POSITION COULD RESULT IN A LOWER THAN DESIGNED PRESSURE IN THE CB LOWER FLOORS, THEREBY CAUSING EXCESSIVE OUTLEAKAGE FROM THE MCR AND THE INABILITY OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) TO MAINTAIN THE MCR HABITABILITY ZONE AT GREATER THAN OR EQUAL TO +0.125 INCH WATER-GAUGE PRESSURE. TVA HAS SUBSEQUENTLY PERFORMED EXTENSIVE TESTING OF CREVS AND IDENTIFIED OTHER DEFICIENCIES THAT COULD AFFECT CREVS OPERABILITY. THE DOSE TO MCR PERSONNEL RESULTING FROM THESE DEFICIENCES IS CURRENTLY BEING ASSESSED TO DETERMINE IF IT COULD EXCEED THE DOSE CALCULATED IN THE FSAR.

[215] SEQUOYAH 1 DOCKET 50-327 LER 87-040 REV 01
UPDATE ON REACTOR SHIELD BUILDING MECHANICAL PENETRATION SEALS NOT QUALIFIED TO ENSURE COMPLIANCE WITH THE DESIGN BASIS DUE TO DESIGN DEFICIENCIES.
EVENT DATE: 072387 REPORT DATE: 022588 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
VENDOR: A & A MANUFACTURING CO.
DOW CORNING CORP.
GENERAL ELECTRIC CO.

(NSIC 208373) THIS REPORT IS REVISED IN ITS ENTIRETY TO ADDRESS ISSUES RELATED TO THE QUALIFICATION OF THE SHIELD BUILDING MECHANICAL PENETRATION SEALS. ON JANUARY 13, 1987, WITH BOTH UNITS IN MODE 5 (0 PERCENT POWER), A SIGNIFICANT CONDITION REPORT (SCR) WAS INITIATED THAT IDENTIFIED A PROBLEM RELATING TO THE UNIT 1 AND UNIT 2 REACTOR SHIELD BUILDING MECHANICAL PENETRATION SEALS UP TO ELEVATION 724 WHICH WERE POTENTIALLY NOT HYDRAULICALLY LEAK TIGHT AS REQUIRED BY SEQUOYAH NUCLEAR PLANT'S (SQN) FINAL SAFETY ANALYSIS REPORT (FSAR), SECTION 2.4A. THE SCR CONCLUDED THAT THE PRESENTLY INSTALLED SEALS ON THE SHIELD BUILDING ARE NOT WATERTIGHT. THE SHIELD BUILDING MECHANICAL SEALS OF UNIT 1 WILL BE REPLACED WITH HYDRAULICALLY QUALIFIED SEALS BEFORE UNIT 1 ENTERS MODE 4 AND THAT OF UNIT 2 WILL BE REPLACED DURING UNIT 2, CYCLE 3, REFUELING OUTAGE. ON FEBRUARY 9, 1987, ANOTHER SCR WAS INITIATED THAT IDENTIFIED DESIGN DEFICIENCIES RELATED TO THE SHIELD BUILDING PENETRATION SEALS DESIGN, DOCUMENTATION, AND EFFECTS OF PIPE MOVEMENT ON SEALS. THE SCR DETERMINED THAT WHEN THE SILICON FOAM SEAL IS LATERALLY LOADED, A GAP MAY BE CREATED BETWEEN THE PIPE OR SLEEVE SURFACES AND FOAM THAT RUNS THE LENGTH OF THE SLEEVE. THIS COULD POTENTIALLY RESULT IN A LOSS OF FIRE AND AIR PRESSURE SEAL INTEGRITY.

[216] SEQUOYAH 1 DOCKET 50-327 LER 87-061 REV 01
UPDATE ON ASSOCIATED CIRCUITS THAT SHARE A COMMON POWER SUPPLY WITH APPENDIX R CIRCUITS LACKED SELECTIVE COORDINATION DUE TO INADEQUATE DESIGN CALCULATIONS.
EVENT DATE: 083187 REPORT DATE: 022688 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 208435) THIS LER IS BEING REVISED IN ITS ENTIRETY TO PROVIDE ADDITIONAL INFORMATION RELATING TO PREVIOUSLY REPORTED 10 CFR 50 APPENDIX R DEFICIENCIES, TO UPDATE THE CORRECTIVE ACTIONS TAKEN BY TVA, AND TO CLARIFY THE CONTROLS THAT ARE CURRENTLY IN PLACE TO PREVENT THE RECURRENCE OF THIS EVENT. ON 8/31/87, WITH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN), IT WAS DETERMINED THAT A FAULT IN APPENDIX R ASSOCIATED CIRCUITS COULD HAVE CAUSED A REQUIRED CIRCUIT ON THE 125-VOLT DC VITAL BATTERY BOARDS AND/OR THE 480-VOLT SHUTDOWN BOARDS TO BE INTERRUPTED. THE CONDITIONS ON THE 125-VOLT BATTERY BOARDS WERE CAUSED BY INADEQUATE DESIGN PROCEDURES WHILE THE CONDITIONS ON THE SHUTDOWN BOARDS WERE CAUSED BY AN INADEQUATE DESIGN CHANGE CONTROL PROCESS. THE DESIGN PROCEDURES THAT WERE USED TO CALCULATE SELECTIVE COORDINATION BETWEEN FUSE AND BREAKER COMBINATIONS IN CIRCUITS POWERED BY THE 125-VOLT BATTERY BOARDS DID NOT REQUIRE ACTUAL CONSTRUCTED CABLE LENGTHS TO BE USED. THE DESIGN CHANGE CONTROL PROCESS THAT WAS FOLLOWED WHEN MODIFICATIONS WERE MADE TO THE CIRCUITS POWERED BY THE 480-VOLT SHUTDOWN BOARDS DID NOT REQUIRE SELECTIVE COORDINATION CALCULATIONS TO BE PERFORMED. THE IDENTIFIED CIRCUITS HAVE BEEN EVALUATED, AND CORRECTIVE ACTIONS TO OBTAIN PROPER SELECTIVE COORDINATION (OR DELETE THE REQUIREMENT FOR SAME) HAS BEEN COMPLETED.

[217] SEQUOYAH 1 DOCKET 50-327 LER 87-078 REV 01
UPDATE ON AN INADEQUATE PROCEDURE FOR REACTOR COOLANT SYSTEM CHEMICAL ADDITION RESULTED IN NON-COMPLIANCE WITH A TECHNICAL SPECIFICATION ACTION STATEMENT.
EVENT DATE: 122187 REPORT DATE: 022588 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 208374) THIS REPORT IS REVISED TO PROVIDE ADDITIONAL INFORMATION ON

COMPLETED CORRECTIVE ACTION. ON DECEMBER 21, 1987, WITH UNIT 1 IN MODE 5 (0 PERCENT POWER, 4 PSIG, 124 DEGREES F), IT WAS DETERMINED THAT A CHEMICAL ADDITION MADE TO THE UNIT 1 REACTOR COOLANT SYSTEM (RCS) ON NOVEMBER 25, 1987, CAUSED A DILUTION OF THE RCS BORON CONCENTRATION AND SUBSEQUENT POSITIVE REACTIVITY CHANGE. SINCE BOTH TRAINS OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) WERE INOPERABLE AT THE TIME OF THE CHEMICAL ADDITION, ACTION STATEMENT (B) TO TECH SPEC 3.7.7, WHICH REQUIRES ALL OPERATIONS INVOLVING A POSITIVE REACTIVITY CHANGE TO BE SUSPENDED, WAS NOT MET. FURTHER INVESTIGATION INTO THIS EVENT IDENTIFIED 4 OTHER OCCASIONS FOR UNIT 1 AND 11 OCCASIONS FOR UNIT 2 WHERE A CHEMICAL ADDITION MADE TO RCS HAVE RESULTED IN SMALL POSITIVE REACTIVITY CHANGES AND DID NOT MEET THE REQUIREMENT OF ACTION STATEMENT (B) TO TECH SPEC 3.7.7. THE EVENT WAS CAUSED BY THE LACK OF AN ADEQUATE PROCEDURE FOR ENSURING THAT RCS CHEMICAL ADDITIONS, WHICH WERE MADE DURING THE PERIOD OF TIME WHEN BOTH TRAINS OF CREVS WERE INOPERABLE, DID NOT RESULT IN POSITIVE REACTIVITY CHANGES. TVA HAS REVISED THE CHEMICAL ADDITION PROCEDURES SUCH THAT RCS CHEMICAL ADDITIONS WILL NOT RESULT IN DILUTION OF THE RCS BORON CONCENTRATION.

[218] SEQUOYAH 1 DOCKET 50-327 LER 87-074 REV 01
UPDATE ON OPERATING PROCEDURES DO NOT ADEQUATELY ADDRESS ECCS REQUIREMENTS IN MODE 4 CONTRARY TO THE REQUIREMENTS OF TECHNICAL SPECIFICATIONS.
EVENT DATE: 122287 REPORT DATE: 022588 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 208437) THE LER WAS REVISED IN ITS ENTIRETY TO ADD A SECTION CONCERNING RESIDUAL HEAT REMOVAL (RHR) MANUAL REALIGNMENT FOR EMERGENCY CORE COOLING SYSTEM (ECCS) OPERATION IN MODE 4 AND TO CHANGE THE REPORTING CRITERIA OF THE ORIGINAL REPORT. ON DECEMBER 22, 1987, WITH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN), IT WAS IDENTIFIED THAT THE BORON INJECTION TANK INLET AND OUTLET VALVES WERE DEENERGIZED IN MODE 4 CONTRARY TO THE REQUIREMENTS OF TECH SPEC 3.5.3. THIS TECH SPEC REQUIRES ONE CENTRIFUGAL CHARGING PUMP (CCP) CAPABLE OF INJECTING EMERGENCY CORE COOLING WATER INTO THE COLD LEGS OF THE REACTOR COOLANT SYSTEM WHILE THE PLANT IS IN MODE 4. ALSO, THE PLANT EMERGENCY INSTRUCTIONS DO NOT ADDRESS MANUAL REALIGNMENT OF THE RHR SYSTEM IN MODE 4 IN ORDER TO COMPLY WITH TECH SPEC 3.5.3. TECH SPEC 3.5.3 ALLOWS MANUAL REALIGNMENT OF THE RHR SYSTEM FOR ECCS OPERATION IN MODE 4. THE CAUSE OF THIS EVENT WAS AN INADEQUATE REVIEW OF A PROCEDURE CHANGE AND AN INADEQUATE PROCEDURE FOR PREPARING AND REVIEWING PROCEDURE CHANGES. THE CAUSE OF THE INADEQUATE EMERGENCY INSTRUCTION WAS A FAILURE TO WRITE SPECIFIC INSTRUCTIONS FOR THE TRANSITORY MODE 4 CONDITION THAT REQUIRED RHR TO BE BOTH A DECAY HEAT REMOVAL SYSTEM AND AN ECCS.

[219] SEQUOYAH 1 DOCKET 50-327 LER 88-002
AN ESSENTIAL RAW COOLING WATER RADIATION MONITOR WAS DECLARED INOPERABLE WITHOUT COMPLYING WITH THE LIMITING CONDITION FOR OPERATION AS A RESULT OF MISINTERPRETATION OF THE LCO.
EVENT DATE: 010588 REPORT DATE: 012988 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 208231) ON JANUARY 5, 1988, WITH UNITS 1 AND 2 IN MODE 5 (0 PERCENT POWER, 4 PSIG, 120 DEGREES F AND 0 PERCENT POWER, 100 PSIG, 128 DEGREES F, RESPECTIVELY), AT APPROXIMATELY 1730 EST IT WAS DISCOVERED THAT A LIMITING CONDITION FOR OPERATION (LCO) WAS NOT COMPLIED WITH AS REQUIRED BY SEQUOYAH NUCLEAR PLANT (SQN) TECH SPECS. ON JANUARY 4, 1988, AT APPROXIMATELY 1630 EST OPERATIONS PERSONNEL DECLARED RADIATION MONITORS (RMS) O-RM-90-134 AND O-RM-90-141 INOPERABLE, HOWEVER, OPERATIONS PERSONNEL DID NOT COMPLY WITH ACTION STATEMENT 32 OF LCO 3.3.3.9 FOR AN INTERVAL OF APPROXIMATELY 13 HOURS. AN INVESTIGATION INTO THIS EVENT CONCLUDED THAT A SUFFICIENT DESCRIPTION TO DETERMINE THE MINIMUM CHANNELS OF MONITORING REQUIRED TO ENSURE COMPLIANCE FOR THE LCO DID NOT EXIST. THIS RESULTED IN OPERATIONS PERSONNEL INCORRECTLY INTERPRETING THE REQUIREMENTS FOR COMPLIANCE OF THE LCO. AS A RESULT, THE ACTION

OF THE LCO WAS NOT ENTERED. AS IMMEDIATE CORRECTIVE ACTION UPON DISCOVERY OF THE NONCOMPLIANCE, OPERATIONS NOTIFIED CHEMISTRY TO INITIATE THE LCO ACTION STATEMENT. TO PREVENT RECURRENCE A TECH SPEC INTERPRETATION IS BEING ISSUED TO STATE THAT ONE CHANNEL FOR EACH ERCW EFFLUENT LINE DISCHARGE HEADER IS REQUIRED TO BE OPERABLE DURING RELEASE THROUGH THESE PATH WAYS.

[220] SEQUOYAH 1 DOCKET 50-327 LER 88-001 REV 01
UPDATE ON AN INACCURATE COMPUTER DATABASE CAUSES TECHNICAL SPECIFICATION SURVEILLANCE INTERVALS TO BE EXCEEDED RESULTING IN INOPERABLE DIESEL GENERATORS AND BORON INJECTION FLOW PATHS.
EVENT DATE: 011188 REPORT DATE: 021888 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 208294) THIS REPORT IS BEING REVISED TO INCLUDE AN ADDITIONAL DEFICIENCY FOUND AND TO PROVIDE ADDITIONAL CORRECTIVE ACTIONS. ON JANUARY 11, 1988, WITH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN) ALL FOUR EMERGENCY DIESEL GENERATORS (D/GS) WERE DECLARED INOPERABLE BECAUSE THE INTERVAL FOR TECH SPEC SURVEILLANCE REQUIREMENT (SR) 4.8.1.2 (CHEMICAL ANALYSIS OF D/G FUEL OIL) WAS EXCEEDED. IN ADDITION, BECAUSE THE HIGH PRESSURE FIRE PUMPS USE THE D/GS AS AN EMERGENCY POWER SOURCE, THE PLANT FIRE SUPPRESSION SYSTEM WAS ALSO DECLARED INOPERABLE, AND A BACKUP FIRE SUPPRESSION SYSTEM WAS ESTABLISHED. BECAUSE OF AN OVERSIGHT DURING THE RECENT CONVERSION TO A NEW COMPUTER PROGRAM USED TO SCHEDULE SURVEILLANCE INSTRUCTIONS (SIS), SI-116, "QUARTERLY CHEMISTRY REQUIREMENTS ON DIESEL GENERATOR FUEL OIL," WAS NOT PERFORMED WITHIN THE TIME INTERVAL REQUIRED BY THE TECH SPEC. IMMEDIATELY UPON DISCOVERY OF THE EVENT, A SPECIAL SI-116 PACKAGE WAS ISSUED AND PERFORMED, AND ALL D/GS AND FIRE SUPPRESSION PUMPS WERE RETURNED TO OPERABLE STATUS ON JANUARY 12, 1988. IN ADDITION, AN IMMEDIATE REVIEW OF ALL SIS WAS INITIATED TO ENSURE THAT THE MODE REQUIREMENTS IN THE SCHEDULING PROGRAM WERE CONSISTENT WITH THE APPLICABILITY SECTION IN THE SIS.

[221] SEQUOYAH 1 DOCKET 50-327 LER 88-004
50-AMP CIRCUIT BREAKERS MAY NOT PRECLUDE AUTO-IGNITION OF ASSOCIATED CABLES CONTRARY TO 10 CFR APPENDIX R DUE TO MISAPPLICATION OF BREAKER CURVES.
EVENT DATE: 011188 REPORT DATE: 021088 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 208295) ON JANUARY 11, 1988, AT APPROXIMATELY 1500 EST, UNITS 1 AND 2 WERE IN MODE 5 (0 PERCENT POWER, 4 PSIG, 125 DEGREES F AND 0 PERCENT POWER, 130 PSIG, 123 DEGREES F, RESPECTIVELY), AN ASSESSMENT WAS INITIATED TO DETERMINE THE CONSEQUENCES OF CLASS 1E CABLES IDENTIFIED AS NOT ADEQUATELY BEING PROTECTED FROM CABLE INSULATION AUTO-IGNITION TEMPERATURES. IT HAD PREVIOUSLY BEEN NOTED DURING SURVEILLANCE TESTING THAT GENERAL ELECTRIC (GE) 50-AMP TYPE TED BREAKER HAD A HIGH RATE OF FAILURE. THE TEST CRITERIA WAS REVIEWED AND RESULTED IN THE DISCOVERY OF TWO DIFFERENT TIME-CURRENT TRIP CHARACTERISTIC CURVES FOR GE 50-AMP TYPE TED CIRCUIT BREAKERS. THIS DISCOVERY RESULTED IN A REVIEW OF THE 10 CFR 50, APPENDIX R CABLE AND PENETRATION CALCULATIONS. THE CALCULATION REVIEW DISCOVERED THAT ALL THE GE 50-AMP TYPE TED BREAKERS WERE ASSUMED TO HAVE THE SAME CHARACTERISTIC CURVE. THE CALCULATION WAS THEN REVISED USING BOTH CURVES, AS APPLICABLE, RESULTING IN THE DISCOVERY OF 24 BREAKERS THAT DID NOT ADEQUATELY PROTECT THE NO. 8 AWG AND NO. 10 AWG CABLES FROM AUTO-IGNITION TEMPERATURES. THE BREAKERS ALLOWING THE CABLES TO BE SUSCEPTIBLE TO AUTO-IGNITION CONDITIONS WERE INSTALLED IN UNIT 1 AND UNIT 2.

[222] SEQUOYAH 1 DOCKET 50-327 LER 88-003
INADVERTENT AUXILIARY BUILDING ISOLATION (ABI) CAUSED BY AN UNKNOWN SOURCE.
EVENT DATE: 011488 REPORT DATE: 020488 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 208232) ON JANUARY 14, 1988, AT APPROXIMATELY 1425 EST WITH BOTH UNITS 1 AND 2 IN MODE 5, AN INADVERTENT AUXILIARY BUILDING ISOLATION (ABI) OCCURRED WHICH ACTUATED EQUIPMENT COMMON TO BOTH UNITS. THE ABI OCCURRED DURING CALIBRATION OF RADIATION MONITOR (RM)-90-101A (PARTICULATE CHANNEL) COMPUTER LOG POINT WHICH REQUIRED RAMPING THE OUTPUT VOLTAGE OF THIS CHANNEL UP THROUGH THE HIGH RADIATION SETPOINT. SINCE A HIGH RADIATION SIGNAL WILL INITIATE AN ABI, THE RM BLOCK SWITCH HS-30-136A3 WAS PULLED TO BLOCK THE 101A CHANNEL SIGNAL. AFTER THE OPERATOR NOTED THAT ALL ASSOCIATED EQUIPMENT ACTUATED AS DESIGNED, AN IMMEDIATE INVESTIGATION ENSUED. NO HIGH RADIATION SIGNALS WERE FOUND ON THE RECORDER CHART FOR THE OTHER TWO CHANNELS OF RM-90-101 NOR ON THE FUEL POOL AREA RMS RM-90-102 AND 103 CONFIRMING THAT NO HIGH RADIATION CONDITION EXISTED. THE OTHER INITIATING SOURCES OF AN ABI (MANUAL TEMPERATURE TO THE AUXILIARY BUILDING (AB) GENERAL SUPPLY FANS ON EITHER UNIT EXCEEDING 115 DEGREES F) WERE INVESTIGATED BUT EVIDENCE INDICATED THAT NONE OF THESE SIGNALS WERE THE CAUSE. ADDITIONAL TROUBLESHOOTING WAS CONDUCTED IN AN EFFORT TO SIMULATE THE PREVIOUS OCCURRENCE; HOWEVER, AN ABI WAS NOT PRODUCED UNTIL THE BLOCK SWITCH WAS ALMOST COMPLETELY PUSHED IN TO THE UNBLOCKED POSITION.

[223] SEQUOYAH 1 DOCKET 50-327 LER 88-005
PERSONNEL NOT PROPERLY IMPLEMENTING APPROVED ADMINISTRATIVE PROCEDURE RESULTING INAPPROPRIATELY EXITING A TECH SPEC ACTION STATEMENT ON RADIATION MONITOR.
EVENT DATE: 011688 REPORT DATE: 021288 NSSS: WE TYPE: PWR

(NSIC 208296) ON JANUARY 16, 1988, AT APPROXIMATELY 1700 EST, IT WAS DISCOVERED THAT THE TURBINE BUILDING SUMP RELEASE LINE RADIATION MONITOR (RM) WAS INAPPROPRIATELY DECLARED OPERABLE ON JANUARY 15, 1988, AT 1400 EST, AND THE TECH SPEC ACTION STATEMENT WAS EXITED. AT THE TIME OF THE DISCOVERY, UNITS 1 AND 2 WERE IN MODE 5 (0 PERCENT POWER, 4 PSIG, 124 DEGREES F AND 0 PERCENT POWER, 90 PSIG, 123 DEGREES F, RESPECTIVELY). THE TURBINE BUILDING SUMP RELEASE LINE RM WAS DECLARED INOPERABLE ON JANUARY 5, 1988, TO CLEAN THE SAMPLE LINE. THE APPLICABLE TECH SPEC ACTION STATEMENT WAS ENTERED, AT THAT TIME, WHICH REQUIRED GRAB SAMPLES TO BE TAKEN FOR TOTAL GAMMA ANALYSES AT LEAST ONCE EVERY 12 HOURS. ON JANUARY 11, A SECOND JOB WAS STARTED TO REPAIR THE RM AFTER FAILING ITS SURVEILLANCE TEST. THE RM JOB WAS COMPLETED AND FUNCTIONALLY TESTED SATISFACTORY ON JANUARY 15. THE CONTROL ROOM OPERATORS (CRO) SUBSEQUENTLY DECLARED THE RM OPERABLE AND EXITED THE TECH SPEC ACTION STATEMENT. OPERABLE AND EXITED THE TECH SPEC ACTION STATEMNET. ON JANUARY 16, AN ASSISTANT UNIT OPERATOR (AUO) WAS DISPATCHED TO INVESTIGATE THE CAUSE OF AN INSTRUMENT MALFUNCTION ALARM ON THE RM. THE AUO REPORTED THE RM VALVES WERE TAGGED SHUT.

[224] SEQUOYAH 1 DOCKET 50-327 LER 88-007
OPENING OF UNIT 1 CONTAINMENT RESULTS IN SECONDARY CONTAINMENT ENVLECPE OUTSIDE THE BOUNDARY SET FOR SURVEILLANCE TESTING OF AUXILIARY BUILDING GAS TREATMENT SYSTEM.
EVENT DATE: 012488 REPORT DATE: 022388 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 208384) ON JANUARY 24, 1988, AT 1800 EST WITH UNITS 1 AND 2 IN MODE 5, IT WAS DISCOVERED, VIA A TOUR OF THE REFUELING FLOOR AND DISCUSSIONS WITH TEST PERSONNEL, THAT THE AUXILIARY BUILDING SECONDARY CONTAINMENT ENCLOSURE (ABSCE) WAS NOT BEING MAINTAINED WITHIN THE CONFIGURATION SET DURING THE IMPLEMENTATION OF TECH SPEC SURVEILLANCE REQUIREMENT (SR) TESTING, USED TO DETERMINE AUXILIARY BUILDING GAS TREATMENT SYSTEM (ABGTS) OPERABILITY. WHEN TECH SPEC SR TESTING WAS DONE TO ENSURE THE ABGTS CAN MAINTAIN THE REQUIRED NEGATIVE PRESSURE IN THE ABSCE, THE BLAST DOORS (BDS) TO UNIT 1 AND 2 REACTOR BUILDINGS (RBS) WERE CLOSED AND THE CONTAINMENT PURGE SYSTEM (CPS) WAS SECURED. OPENING THE BDS WILL ENCOMPASS THE RB IN THE ABSCE BOUNDARY. WHEN A UNIT IS IN MODE 5 OR 6, IT IS NORMAL TO HAVE THE BDS OPEN FOR THAT UNIT, AND IT WAS POSSIBLE THAT THE CPS COULD BE IN OPERATION. INCREASING THE ABSCE BOUNDARY POTENTIALLY CAUSES MORE LEAKAGE

INTO THE ABSCE. CONSIDERING THE ADDITIONAL POTENTIAL LEAKAGE, THERE WAS NO ASSURANCE THAT THE ABGTS WOULD BE ABLE TO SATISFY TECH SPEC SR WITH THE INCREASED ABSCE BOUNDARY OR WHEN THE CPS IS OPERATING. THE CAUSE OF THIS CONDITION, IS THE LACK OF ADEQUATE CONTROLS TO ENSURE THE ABSCE BOUNDARY IS MAINTAINED WITHIN THE CONDITION SET BY SR TESTING.

[225] SEQUOYAH 1 DOCKET 50-327 LER 88-006
INSTRUMENT MAINTENANCE TECHNICIAN INADVERTENTLY REMOVED INCORRECT RADIATION MONITOR MODULE WHICH CAUSED A CONTAINMENT VENTILATION ISOLATION.
EVENT DATE: 012788 REPORT DATE: 022588 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 208441) ON JANUARY 27, 1988, AT 1005 EST WITH UNIT 1 IN MODE 5 (0% POWER, 198 PSIG, 128 DEGREES F) AND UNIT 2 IN MODE 5 (0% POWER, 365 PSIG, 190 DEGREES F), A CONTAINMENT VENTILATION ISOLATION (CVI) (EIIS CODE JM) OCCURRED. AN INSTRUMENT MAINTENANCE (IM) TECHNICIAN WHO WAS PART OF A DEDICATED RADIATION MONITOR (EIIS CODE IL) MAINTENANCE CREW WAS PERFORMING A WORK REQUEST (WR), WHICH HAD BEEN WRITTEN TO RESOLVE A PROBLEM IDENTIFIED BY SURVEILLANCE TESTING, ON THE SHIELD BUILDING EXHAUST SYSTEM RADIATION MONITOR (RM) 2-RM-90-100B. DURING THIS EVENT THE IM INADVERTENTLY REMOVED THE CONTAINMENT BUILDING LOWER CONTAINMENT VENTILATION ROOM (RM-90-106) RATHER THAN 2-RM-90-100B WHICH RESULTED IN A CVI. THE PRIMARY CAUSE OF THE CVI CAN BE ATTRIBUTED TO INATTENTION TO DETAIL BY THE IM AS HE REMOVED THE WRONG MONITOR FROM SERVICE. A CONTRIBUTING CAUSE TO THIS EVENT IS THE AMBIGUOUS METHOD OF LABELING ON THE RM PANEL. THE IM HAS BEEN COUNSELED ON THIS EVENT. THE IMPORTANCE OF CORRECTLY PERFORMING WORK ACTIVITIES AND INDEPENDENTLY VERIFYING THAT THE EQUIPMENT TO BE SERVICED IS CORRECT BEFORE IT WAS REMOVED FROM SERVICE HAS BEEN STRESSED TO THE IMS. ALSO, A HUMAN FACTORS EVALUATION HAS BEEN PERFORMED ON THE RADIATION MONITOR PANEL AS PART OF THE CONTROL ROOM DESIGN REVIEW.

[226] SEQUOYAH 1 DOCKET 50-327 LER 88-008
PERSONNEL NOT PROPERLY IMPLEMENTING APPROVED ADMINISTRATIVE PROCEDURE RESULTING IN INAPPROPRIATELY EXITING A TECH SPEC ACTION STATEMENT ON RADIATION MONITORING.
EVENT DATE: 013088 REPORT DATE: 022588 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 208383) ON JANUARY 30, 1988, AT APPROXIMATELY 1600 EST WITH UNITS 1 AND 2 IN COLD SHUTDOWN CONDITION, IT WAS DISCOVERED BY THE ASSISTANT SHIFT SUPERVISOR THAT THE TRAIN "B" ESSENTIAL RAW COOLING WATER (ERCW) EFFLUENT LINE RADIATION MONITOR (RM) WAS INAPPROPRIATELY DECLARED OPERABLE, AND THE TECH SPEC ACTION STATEMENT WAS PREMATURELY EXITED ON THE PREVIOUS SHIFT. THE RM WAS DECLARED OPERABLE AT APPROXIMATELY 0300 EST ON JANUARY 30, 1988, WITHOUT COMPLETING THE REQUIRED POSTMAINTENANCE TEST (PMT). WHEN BOTH CHANNELS OF THE RM ARE INOPERABLE, THE APPLICABLE TECH SPEC ACTION STATEMENT REQUIRES GRAB SAMPLES TO BE TAKEN AND ANALYZED FOR RADIOACTIVITY AT LEAST ONCE EVERY 12 HOURS. COMPLIANCE WITH THIS REQUIREMENT HAD BEEN MAINTAINED UNTIL THE RM HAD NOT BEEN COMPLETELY TESTED FOLLOWING MAINTENANCE AT 1600 EST, AND SAMPLES WERE RESUMED UNTIL COMPLETION OF THE PMT. APPROXIMATELY 15.5 HOURS HAD TRANSPIRED BETWEEN GRAB SAMPLES. THIS EVENT WAS CAUSED PARTLY BY OPERATIONS PERSONNEL DECLARING THE RM OPERABLE WITHOUT REVIEWING THE ASSOCIATED WORK REQUEST TO ENSURE THE PMT WAS COMPLETE AND PARTLY BY NOT REVIEWING THE PMT REQUIREMENTS BEFORE SIGNING VERIFICATION THAT THE PMT IS COMPLETE. AS CORRECTIVE ACTIONS, THE TECH SPEC ACTION STATEMENT WAS REENTERED, AND THE PERSONNEL INVOLVED WERE COUNSELED.

[227] SEQUOYAH 1 DOCKET 50-327 LER 88-009
AN INADEQUATE PROCEDURE CAUSED INACCURATE PRIMARY TO SECONDARY LEAK RATES TO BE MEASURED RESULTING IN A POTENTIAL NONCOMPLIANCE WITH A LIMITING CONDITION FOR OPERATION.

EVENT DATE: 020488 REPORT DATE: 022688 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 208442) ON FEBRUARY 4, 1988, WITH UNITS 1 AND 2 IN MODE 5 (0 PERCENT POWER, 4 PSIG, 130 DEGREES F AND 0 PERCENT POWER, 320 PSIG, 189 DEGREES F, RESPECTIVELY), IT WAS DETERMINED THAT SURVEILLANCE INSTRUCTION (SI)-137.2, "REACTOR COOLANT SYSTEM WATER INVENTORY," DID NOT ADEQUATELY ADDRESS THE TECH SPEC REQUIREMENT FOR CALCULATING THE PRIMARY TO SECONDARY LEAK RATE IN MODE 1 WHEN THE PLANT WAS OPERATING AT LESS THAN 100 PERCENT POWER OR WHEN THE PLANT WAS IN MODES 2, 3, AND 4. LIMITING CONDITION FOR OPERATION (LCO) 3.4.6.2.C REQUIRES THAT THE PRIMARY TO SECONDARY LEAK RATE BE LIMITED TO 1 GALLON PER MINUTE (GPM) FOR ALL STEAM GENERATORS (EIIS CODE SB) AND 500 GALLONS PER DAY FOR ANY ONE STEAM GENERATOR. ADDITIONAL INVESTIGATION REVEALED THAT THE FULL POWER STEAM GENERATOR WATER VOLUME INCORPORATED INTO THE SI-137.2 MATHEMATICAL MODEL WAS ALSO INCORRECT. A MODIFICATION WHICH OCCURRED BEFORE ORIGINAL STARTUP CHANGED THE STEAM GENERATOR VOLUME; HOWEVER, THE MATHEMATICAL MODEL WAS NOT REVISED. BECAUSE OF THE INCORRECT WATER VOLUME, SI-137.2 UNDERESTIMATED THE FULL POWER PRIMARY TO SECONDARY LEAK RATE BY APPROXIMATELY 28 PERCENT. THE EVENT WAS CAUSED BY AN INADEQUATE INSTRUCTION FOR MEASURING THE PRIMARY TO SECONDARY LEAK RATE.

[228] SEQUOYAH 2 DOCKET 50-328 LER 87-008 REV 02
UPDATE ON ELECTROMAGNETIC INTERFERENCE CAUSED A SPURIOUS HIGH RADIATION SPIKE RESULTING IN A CONTAINMENT VENTILATION ISOLATION.
EVENT DATE: 112787 REPORT DATE: 021888 NSSS: WE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.
LUNDELL CONTROLS - TECHNOLOGY INC.
MALLORY CAPACITOR CO

(NSIC 208375) THIS LER IS BEING REVISED TO PROVIDE ADDITIONAL INFORMATION RELATING TO THE CAUSE OF THIS EVENT AND THE SPECIFIC CORRECTIVE ACTIONS TO BE TAKEN BY TVA. ON NOVEMBER 27, 1987, WITH UNIT 2 IN MODE 5 (COLD SHUTDOWN), A TRAIN "A" CONTAINMENT VENTILATION ISOLATION (CVI) OCCURRED ON UNIT 2. AT APPROXIMATELY 0320 EST, WITH AN INSTRUMENT MALFUNCTION ALARM PRESENT ON THE CONTAINMENT LOWER COMPARTMENT AIR RADIATION MONITOR (RM), OPERATIONS PERSONNEL DETERMINED THAT A SPURIOUS HIGH RADIATION SPIKE HAD OCCURRED AT APPROXIMATELY 0230 EST. SINCE THE SPURIOUS RADIATION SPIKE EXCEEDED THE RM TRIP CRITERIA, A CVI AND INSTRUMENT MALFUNCTION ALARM WERE GENERATED. INSTRUMENT MAINTENANCE (IM) PERSONNEL DETERMINED THAT THE MOST PROBABLE CAUSE OF THE SPURIOUS HIGH RADIATION SPIKE WAS ELECTROMAGNETIC INTERFERENCE (EMI) GENERATED BY THE ACTUATION OF THE LOW SAMPLE FLOW SWITCH. AN IMMEDIATE INVESTIGATION INTO THIS AND OTHER CVIS REVEALED THAT THE DETECTOR CABLE DID NOT HAVE A GOOD GROUND. FURTHER INVESTIGATION SHOWED THAT THE EMI WAS CAUSED BY A CHATTERING REED SWITCH IN THE RM LOW FLOW ALARM CIRCUIT. TVA WILL ISSUE A DESIGN CHANGE AND WILL MODIFY THE RM ALARM CIRCUITS TO SEAL IN THE LOW FLOW ALARM THE FIRST TIME IT IS ACTUATED THEREBY ELIMINATING THE REED SWITCH CHATTER AND SIGNIFICANTLY REDUCING THE EMI.

[229] SEQUOYAH 2 DOCKET 50-328 LER 87-009 REV 01
UPDATE ON ELECTROMAGNETIC INTERFERENCE CAUSED SPURIOUS HIGH RADIATION SPIKES RESULTING IN TWO CONTAINMENT VENTILATION ISOLATIONS.
EVENT DATE: 120587 REPORT DATE: 021888 NSSS: WE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 208289) THIS LER IS BEING REVISED TO PROVIDE ADDITIONAL INFORMATION RELATING TO THE CAUSE OF THESE EVENTS AND THE SPECIFIC CORRECTIONS TO BE TAKEN BY TVA. ON DECEMBER 5, 1987, WITH UNIT 2 IN MODE 5 (COLD SHUTDOWN), TWO CONTAINMENT VENTILATION ISOLATIONS (CVIS) OCCURRED; THE FIRST AT APPROXIMATELY 0911 EST AND THE SECOND AT APPROXIMATELY 1553 EST. AN INVESTIGATION INTO THE EVENTS REVEALED THAT WELDING AND GRINDING WORK WAS BEING PERFORMED IN UNIT 2 CONTAINMENT. AS A RESULT OF THIS WORK, THE CONTAINMENT ATMOSPHERE IN THE VICINITY OF THE RADIATION

MONITORS CONTAINED A HIGH CONCENTRATION OF PARTICULATES. THE PARTICULATES CLOGGED THE RADIATION MONITOR (RM) SAMPLE GAS PREFILTERS AND CREATED A LOW FLOW CONDITION WHICH ACTIVATED THE LOW FLOW SWITCH OF RM 2-RM-90-106. TVA BELIEVED THAT ACTUATION OF THIS SWITCH INDUCED A ELECTROMAGNETIC INTERFERENCE (EMI) SPIKE TO THE RM DETECTOR CABLE AND CAUSED A SPURIOUS RADIATION SPIKE. SINCE THIS RADIATION SPIKE EXCEEDED THE RM TRIP CRITERIA, THE SUBJECT CVIS WERE GENERATED. FURTHER INVESTIGATION INTO THESE EVENTS BY A SPECIAL TASK GROUP HAS DETERMINED THAT THE EMI WAS CAUSED BY A CHATTERING REED SWITCH IN THE RM LOW FLOW ALARM CIRCUIT. TVA WILL ISSUE A DESIGN CHANGE AND WILL MODIFY THE RM ALARM CIRCUITS TO SEAL IN THE LOW FLOW ALARM THE FIRST TIME IT IS ACTUATED.

[230] SEQUOYAH 2 DOCKET 50-328 LER 87-010 REV 01
UPDATE ON ELECTROMAGNETIC INTERFERENCE CAUSED SPURIOUS HIGH RADIATION SPIKES RESULTING IN TWO CONTAINMENT VENTILATION ISOLATIONS.
EVENT DATE: 122187 REPORT DATE: 021888 NSSS: WE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.
LUNDELL CONTROLS - TECHNOLOGY INC.
MALLORY CAPACITOR CO

(NSIC 208376) THIS LER IS BEING REVISED TO PROVIDE ADDITIONAL INFORMATION RELATING TO THE CAUSE OF THESE EVENTS AND THE SPECIFIC CORRECTIVE ACTIONS TO BE TAKEN BY TVA. ON DECEMBER 21, 1987, WITH UNIT 2 IN MODE 5 (COLD SHUTDOWN), TWO CONTAINMENT VENTILATION ISOLATIONS (CVI) OCCURRED. THE FIRST CVI OCCURRED AT APPROXIMATELY 0750 EST AND WAS CAUSED BY A SPURIOUS HIGH RADIATION SPIKE ON CONTAINMENT RADIATION MONITORS (RMS) 2-RM-90-106 AND/OR 2-RM-90-112. SINCE BOTH RM CHART RECORDERS SHOWED A HIGH RADIATION SPIKE AT APPROXIMATELY THE SAME TIME AND NEITHER OF THE HIGH RADIATION ALARMS ASSOCIATED WITH THE RMS ACTUATED, IT WAS NOT POSSIBLE TO DETERMINE WHICH RM GENERATED THE CVI. AT APPROXIMATELY 0810 EST, THE CVI WAS RESET, AND NORMAL MODE 5 OPERATION WAS RESUMED. THE SECOND CVI OCCURRED AT APPROXIMATELY 0900 EST AND WAS CAUSED BY A SPURIOUS HIGH RADIATION SPIKE ON RM 2-RM-90-112. AT APPROXIMATELY 0906 EST, THE CVI WAS RESET, AND NORMAL MODE 5 OPERATION WAS RESUMED. AN INITIAL INVESTIGATION BY INSTRUMENT MAINTENANCE (IM) PERSONNEL DETERMINED THAT THE MOST PROBABLE CAUSE OF THE SPURIOUS HIGH RADIATION SPIKE WAS ELECTROMAGNETIC INTERFERENCE (EMI) GENERATED BY THE ACTUATION OF THE LOW SAMPLE FLOW SWITCH.

[231] SEQUOYAH 2 DOCKET 50-328 LER 88-001
CONTAINMENT SPRAY HEAT EXCHANGERS LOWER SUPPORT FRAME COULD BE OVERSTRESSED DURING A DESIGN BASIS SEISMIC EVENT DUE TO A DESIGN DEFICIENCY.
EVENT DATE: 011188 REPORT DATE: 021188 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 1 (PWR)

(NSIC 208297) ON 1/12/88, WITH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN) (0%, 4 PSIG, 122 DEGREES F AND 0%, 140 PSIG, 126.5 DEGREES F, RESPECTIVELY), TVA HAD RECEIVED A LETTER FROM STONE AND WEBSTER ENGINEERING COMPANY WHICH REVEALED THAT THE LOWER SUPPORT FRAME OF CONTAINMENT SPRAY (CS) HEAT EXCHANGERS 2A AND 2B COULD BE OVERSTRESSED DURING A DESIGN BASIS SEISMIC EVENT. THE LETTER ALSO REVEALED THAT DUE TO CLOSE PROXIMITY OF THE RESIDUAL HEAT REMOVAL (RHR) HEAT EXCHANGERS TO THE CONTAINMENT SPRAY HEAT EXCHANGERS, THERE EXISTED A POTENTIAL FOR PHYSICAL INTERACTION BETWEEN THE TWO COMPONENTS DURING A DESIGN BASIS SEISMIC EVENT. TECH SPECS REQUIRE BOTH RHR LOOPS OPERABLE DURING MODE 5. HOWEVER, THE CS HEAT EXCHANGERS ARE NOT REQUIRED OPERABLE DURING MODE 5. THE ROOT CAUSE OF THE ABOVE SPECIFIED CONDITION WAS DETERMINED TO BE DESIGN INPUT CONTROL PROCEDURES DID NOT EXIST DURING INITIAL DESIGN TO ENSURE VENDOR INTERFACE REVIEW IN THE GENERATION OF MAXIMUM DESIGN LOADS OF THE SUPPORT FRAME. TVA HAS PERFORMED FURTHER EVALUATION OF THE CS HEAT EXCHANGERS POTENTIAL INTERACTION WITH THE RHR HEAT EXCHANGERS AND CONCLUDED THAT WHEN THE EFFECTS OF THE PLATFORMS AND GRATING ON ELEVATION 703, SUPPORT STRUCTURE AT ELEVATION 720 AND THE 4 ATTACHED PIPES ARE CONSIDERED, THERE WOULD BE NO PHYSICAL INTERACTION WITH THE RHR HEAT EXCHANGERS.

[232] SEQUOYAH 2 DOCKET 50-328 LER 88-002
AN INADEQUATE REVIEW OF A PLANT MODIFICATION CAUSED AN INACCURATE RESPONSE TIME MEASUREMENT RESULTING IN NONCOMPLIANCE WITH A TECH SPEC SURVEILLANCE REQUIREMENT.
EVENT DATE: 011888 REPORT DATE: 021788 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 1 (PWR)

(NSIC 208382) ON 1/18/88, WITH UNIT 2 IN MODE 5 (COLD SHUTDOWN), A POSTPERFORMANCE REVIEW OF AN INSTRUMENT MAINTENANCE INSTRUCTION (IMI) REVEALED THAT THE SUBJECT IMI INCORRECTLY MEASURED THE RESPONSE TIME OF THE TWO VALVES THAT OPEN THE CENTRIFUGAL CHARGING PUMPS (CCPS) SUCTION LINE TO THE REFUELING WATER STORAGE TANK (RWST). DURING A SUBSEQUENT SEARCH OF THE SEQUOYAH NUCLEAR PLANT (SQN) RECORDS, IT WAS DISCOVERED THAT AN OCT. 1984 RESPONSE TIME MEASUREMENT WAS INADEQUATE TO VERIFY COMPLIANCE WITH THE SQN TECH SPEC FOR ONE OF THE TWO VALVES. THE EVENT WAS CAUSED BY AN INADEQUATE REVIEW OF A DESIGN CHANGE THAT WAS IMPLEMENTED TO MODIFY APPROXIMATELY 65 OF THE SQN WESTINGHOUSE TYPE W-2 HANDSWITCHES. THIS MODIFICATION CONNECTED THE RED AND GREEN VALVE STATUS LIGHT IN SERIES WITH THE NEUTRAL (AUTO) SWITCH CONTACTS TO PROVIDE PLANT OPERATORS WITH AN IMMEDIATE VISUAL INDICATION OF ELECTRICAL CONTINUITY WHEN THE HANDSWITCH WAS IN THE NEUTRAL POSITION. SINCE THE SUBJECT IMI OBTAINED THE RESPONSE TIME OF THE CCP SUCTION VALVES BY MEASURING THE VOLTAGE DROP ACROSS THE GREEN STATUS LIGHTS, ANY CHANGE TO THE ELECTRICAL WIRING CONFIGURATION OF THE VALVE STATUS LIGHTS COULD AFFECT THE RESPONSE TIME MEASUREMENT. HOWEVER, BECAUSE THIS MODIFICATION WAS NOT ADEQUATELY REVIEWED FOR ITS POTENTIAL EFFECT ON SQN PROCEDURES, NO PERMANENT CHANGES WERE MADE.

[233] SEQUOYAH 2 DOCKET 50-328 LER 88-003
ICE BUILDUP IN THE FLOW PASSAGES OF THE ICE CONDENSER DUE TO SUBLIMATION WHICH COULD RESULT IN INCREASED CONTAINMENT PRESSURES.
EVENT DATE: 011988 REPORT DATE: 022588 NSSS: WE TYPE: PWR

(NSIC 208443) THIS LER IS BEING SUBMITTED AS A VOLUNTARY REPORT TO INFORM NRC OF THE DEGRADATION FOUND WHILE INSPECTING THE UNIT 2 ICE CONDENSER. ON JANUARY 19, 1988, WITH UNITS 1 AND 2 IN MODE 5 (0 PERCENT POWER, 4 PSIG, 124 DEGREES F AND 0 PERCENT POWER, 110 PSIG, 117 DEGREES F, RESPECTIVELY), AN INSPECTION OF THE UNIT 2 ICE CONDENSER REVEALED THAT ACCUMULATION OF ICE OR FROST IN THE FLOW PASSAGES REPRESENTED A DEGRADED CONDITION. TECH SPEC 3.6.5.1 INDICATES MORE THAN ONE RESTRICTED FLOW PASSAGE PER ICE CONDENSER BAY (THERE ARE 24 BAYS IN THE ICE CONDENSER) IS EVIDENCE OF ABNORMAL DEGRADATION OF THE ICE CONDENSER. A SPECIAL MAINTENANCE INSTRUCTION (SMI) WAS WRITTEN BY TVA AND WESTINGHOUSE TO DETERMINE THE EXTENT OF THE DEGRADATION IN ORDER TO JUSTIFY THAT SEQUOYAH NUCLEAR PLANT (SQN) MEET THE INTENT OF TECH SPEC 3.6.5.1. WESTINGHOUSE HAD PREVIOUSLY COMPLETED A COMPUTER ANALYSIS WHICH CONCLUDED WITH 15 PERCENT OF THE ICE CONDENSER FLOW PASSAGES BLOCKED, THE CONTAINMENT PRESSURE WOULD BE WITHIN DESIGN LIMITS. THE SMI WAS WRITTEN WITH AN ACCEPTANCE CRITERIA OF LESS THAN 15 PERCENT TOTAL FLOW PASSAGE BLOCKAGE. THE SMI WAS PERFORMED, ACCEPTANCE CRITERIA WAS MET, AND ICE CONDENSER WAS DETERMINED TO MEET THE REQUIREMENTS OF TECH SPEC 3.6.5.1.

[234] SEQUOYAH 2 DOCKET 50-328 LER 88-004
INADVERTENT START OF ALL EMERGENCY DIESEL GENERATORS DURING SURVEILLANCE TESTING DUE TO DAMAGED LOCKOUT RELAY.
EVENT DATE: 012788 REPORT DATE: 021888 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 1 (PWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 208298) ON JANUARY 27, 1988, AT APPROXIMATELY 1743 EST WITH UNITS 1 AND 2 IN MODE 5 (0 PERCENT POWER, 4 PSIG, 128 DEGREES AND 0 PERCENT POWER, 380 PSIG, 190 DEGREES F, RESPECTIVELY), AN EMERGENCY START OF ALL FOUR EMERGENCY DIESEL GENERATORS (D/GS) OCCURRED DURING SURVEILLANCE TESTING. DURING THE PERFORMANCE OF THE SURVEILLANCE TEST PROCEDURE, IT WAS REQUIRED TO TRANSFER THE 2A-A 6900V

SHUTDOWN BOARD FROM THE ALTERNATE FEEDER BREAKER TO THE NORMAL FEEDER BREAKER FROM THE CONTROL ROOM. THE UNIT OPERATOR (UO) ATTEMPTED TO PERFORM THIS EVOLUTION WHEN THE NORMAL FEEDER BREAKER FAILED TO CLOSE WHILE THE ALTERNATE FEEDER BREAKER WAS OPEN. WITH THE NORMAL AND ALTERNATE FEEDER BREAKER OPEN, THE 2A-A 6900V SHUTDOWN BOARDS DEENERGIZED, RESULTING IN A BLACKOUT CONDITION. THE BLACKOUT CONDITION CAUSED A COMMON START OF ALL FOUR D/GS. THE UO RECOGNIZED THE BLACKOUT CONDITION AND IMMEDIATELY CLOSED THE ALTERNATE FEEDER BREAKER BEFORE THE 2A-A D/G OUTPUT BREAKER CLOSED AND BEFORE LOAD SHEDDING HAD OCCURRED. ALL D/G WERE SUBSEQUENTLY SHUT DOWN. THROUGH INTERVIEWS WITH OPERATIONS AND TEST PERSONNEL, VISUAL INSPECTIONS, AND TROUBLESHOOTING OF THE 2A-A 6900V SHUTDOWN BOARD, THE IMMEDIATE CAUSE OF THIS EVENT WAS THE 2A-A 6900V SHUTDOWN DEENERGIZING DUE TO THE NORMAL FEEDER BREAKER FAILING TO CLOSE.

[235] SHEARON HARRIS 1 DOCKET 50-400 LER 87-05, REV 01
UPDATE ON LOSS OF OFF-SITE POWER DUE TO INCOMING LINE BREAKER OPENING CAUSED BY PERSONNEL WORKING IN SWITCHYARD RELAY CABINET.
EVENT DATE: 101187 REPORT DATE: 022388 NSSS: WE TYPE: PWR
VENDOR: ENVIREX INC (DIV OF REXNORD)

(NSIC 208311) THE PLANT WAS IN MODE 5, COLD SHUTDOWN AT 0% REACTOR POWER ON 10/11/87. ONE OF TWO INCOMING POWER LINES TO 1A START-UP TRANSFORMER FOR OFF-SITE POWER WAS UNDER CLEARANCE FOR RELAY CABINET MODIFICATIONS. AT 1550 HOURS, THE REMAINING INCOMING LINE BREAKER TRIPPED INITIATING A 1A-SA SAFETY BUS BLACKOUT. THE 1A-SA TRAIN SEQUENCER ACTUATED AND 1A-SA EMERGENCY DIESEL GENERATOR (EDG) STARTED AND PROPERLY ASSUMED 1A-SA SAFETY BUS TRAIN LOADS. ALL PLANT SYSTEMS FUNCTIONED AS REQUIRED WITH THE EXCEPTION OF 1A-SA EMERGENCY SERVICE WATER TRAVELING WATER SCREEN WHICH DID NOT START. THE INCOMING BREAKER TRIPPED BECAUSE OF ACCIDENTAL JARRING OF THE PROTECTION RELAYS. THIS WAS DUE TO RELAY PERSONNEL HAMMERING AND DRILLING IN THE RELAY CABINET FOR A MODIFICATION INSTALLATION. THE CORRECTIVE ACTION WAS TO STOP ALL HAMMERING AND DRILLING IN THE RELAY CABINET. OFF-SITE POWER WAS RESTORED AT 1655 HOURS AND 1A-SA DGB SECURED AT 1700 HOURS. SITE MANAGEMENT HAS ISSUED A LETTER STRESSING THE IMPORTANCE OF CAUTIOUS WORK HABITS TO THE RELAY CREW MANAGEMENT WHEN WORKING IN SHNPP SWITCHYARD. SUBSTATION CONSTRUCTION PERSONNEL HAVE BEEN INSTRUCTED TO NOTIFY SITE MANAGEMENT PRIOR TO STARTING WORK ON ANY PROJECT AT THE PLANT AND THE CREW DOING WORK AT THE PLANT WILL NOTIFY THE SHIFT FOREMAN UPON ARRIVAL AT THE PLANT AND BEFORE LEAVING THE PLANT.

[236] SHEARON HARRIS 1 DOCKET 50-400 LER 88-001
EMERGENCY OPERATING PROCEDURE DEFICIENCY FOR SWITCHOVER TO RECIRCULATION AFTER A LOSS OF COOLANT ACCIDENT.
EVENT DATE: 011588 REPORT DATE: 021588 NSSS: WE TYPE: PWR

(NSIC 208345) ON JANUARY 15, 1988, IT WAS DISCOVERED THAT A PROCEDURE DEFICIENCY HAD PREVIOUSLY EXISTED IN THE EMERGENCY OPERATING PROCEDURES (EOPS) WHICH COULD HAVE RESULTED IN THE FAILURE OF THE EMERGENCY CORE COOLING SYSTEM (ECCS) DURING THE RECIRCULATION PHASE OF AN ACCIDENT UNDER A SCENARIO OF A SINGLE FAILURE OF ONE RESIDUAL HEAT REMOVAL PUMP (RHRP). THE DEFICIENCY INVOLVED AN IMPROPER VALVE LINEUP WHICH COULD CAUSE A RHRP TO EXCEED ITS DESIGN FLOW LIMIT DURING ECCS RECIRCULATION, IF ONE OF THE TWO RHRPS HAD FAILED DURING A LOSS OF COOLANT ACCIDENT (LOCA) LEAVING ONLY ONE OPERATING RHRP, AND IF THE LOCA WAS SUFFICIENTLY LARGE TO COMPLETELY DEPRESSURIZE THE REACTOR COOLANT SYSTEM. THIS PROCEDURE DEFICIENCY RESULTED FROM THE FAILURE TO COMPLETELY INCORPORATE INTO PLANT PROCEDURES A CHANGE MADE TO THE FINAL SAFETY ANALYSIS REPORT. THE DEFICIENCY EXISTED BEGINNING IN DECEMBER OF 1986, WHEN THE EOP CONTAINING THE ERROR WAS APPROVED, AND INITIAL OPERATION OF THE PLANT BEGAN. WHEN THE SAFETY SIGNIFICANCE OF THE DISCREPANCY WAS DISCOVERED, THE EOPS WERE REVISED TO CORRECT THE ERROR. THIS OCCURRED ON DECEMBER 16, 1987. OTHER FSAR CHANGES MADE IN THIS TIME PERIOD

INVOLVING A CHANGE TO PROCEDURES WERE REVIEWED TO ENSURE THEY WERE PROPERLY IMPLEMENTED.

[237] SHEARON HARRIS 1 DOCKET 50-400 LER 88-002
MAINTENANCE SURVEILLANCE TEST NOT PERFORMED AS REQUIRED ON RADIATION MONITOR RM #1WV-3547-1.
EVENT DATE: 011988 REPORT DATE: 021888 NSSS: WE TYPE: PWR

(NSIC 208200) ON JANUARY 19, 1988, AT 1600, WHILE THE PLANT WAS OPERATING AT 100% POWER IN MODE 1, THE VENT STACK 5A WIDE RANGE GAS MONITOR, (WRGM), RM #1WV-3547-1, (EIIS:IL) WAS DECLARED INOPERABLE UNDER TECHNICAL SPECIFICATION 3.3.3.11, RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION. THIS WAS DUE TO THE DISCOVERY THAT A MAINTENANCE SURVEILLANCE TEST (MST-I0380) ON THE MONITOR WAS NOT COMPLETED AS SCHEDULED AND WAS OVERDUE ON DECEMBER 27, 1987. BECAUSE OF THE MISSED SURVEILLANCE TEST, THE MONITOR SHOULD HAVE BEEN DECLARED INOPERABLE ON DECEMBER 27, 1987; THEREFORE, THE PLANT MUST ASSUME THAT THE MONITOR HAS BEEN INOPERABLE SINCE THAT TIME. SINCE THE MONITOR WAS NOT RESTORED TO SERVICE WITHIN 7 DAYS OF THAT DATE, A SPECIAL REPORT IS REQUIRED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.3.3.6, ACCIDENT MONITORING INSTRUMENTATION. THE INFORMATION REQUIRED IN A SPECIAL REPORT IS INCLUDED IN THIS LICENSE EVENT REPORT. MST-I0380, WPB STACK 5A ACCIDENT MONITOR RM1WV35471 OPERATIONAL TEST, WAS SUCCESSFULLY COMPLETED ON JANUARY 21, 1988, AND THE MONITOR WAS DECLARED OPERABLE ON JANUARY 26, 1988, AT 1655. THE CAUSE OF THE OVERDUE SURVEILLANCE WAS A PROCEDURAL DEFICIENCY. THE CORRECTIVE ACTION INVOLVED A REVISION TO PLANT PROCEDURE PLP-103, SURVEILLANCE AND PERIODIC TEST PROGRAM.

[238] SHEARON HARRIS 1 DOCKET 50-400 LER 88-003
REACTOR OVERPOWER DUE TO CONDENSER STEAM DUMP VALVES OPENING CAUSED BY FAULTY WIRING IN SSPS CABINET.
EVENT DATE: 012188 REPORT DATE: 022288 NSSS: WE TYPE: PWR
VENDOR: FISHER CONTROLS CO.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 208346) THE PLANT WAS OPERATING IN MODE 1, POWER OPERATION, AT 100% REACTOR POWER ON 1/21/88. PLANT PERSONNEL WERE IN THE PROCESS OF CONDUCTING A LOOP CALIBRATION ON TURBINE FIRST STAGE PRESSURE CHANNEL P-446. AT 0846 HOURS, 5 OF THE 6 CONDENSER STEAM DUMP VALVES OPENED AND CAUSED STEAM FLOW TO INCREASE AND REACTOR POWER TO RAMP UP. A PEAK REACTOR POWER OF 103% WAS REACHED. OPERATOR ACTION WAS TAKEN TO CLOSE THE STEAM DUMPS AND TAKE MANUAL CONTROL OF FEEDWATER REGULATING VALVES TO CONTROL STEAM GENERATOR WATER LEVELS. REACTOR POWER WAS REDUCED TO 98%, WHICH AVOIDED A REACTOR TRIP AND THEN STABILIZED AT 100% POWER. THE LOOP CALIBRATION WAS STOPPED AND THE STEAM DUMPS LEFT IN OFF UNTIL AN EVALUATION OF THE EVENT WAS COMPLETE. THE IMMEDIATE CAUSE OF THE EVENT WAS FOUND TO BE A JUMPER WIRE INSTALLED IN TRAIN A OF THE SOLID STATE PROTECTION SYSTEM (SSPS) CABINET. THIS JUMPER WAS AROUND THE OUTPUT CONTACTS PROVIDING TURBINE TRIP INDICATION TO THE STEAM DUMP LOGIC. THE WIRE WAS INSTALLED AT THE TIME OF MANUFACTURE AND EFFECTIVELY ALLOWED THE STEAM DUMP LOGIC TO OPERATE IN THE TAVG AND STEAM PRESSURE MODE SIMULTANEOUSLY WHEN THE STEAM PRESSURE MODE WAS SELECTED. AN EXTENSIVE WIRE CHECK OF SSPS WAS CONDUCTED FOR ADDITIONAL WIRING ERRORS AND AN EVALUATION MADE FOR SAFETY SIGNIFICANCE.

[239] SHEARON HARRIS 1 DOCKET 50-400 LER 88-004
LOW LEVEL IN CONTAINMENT SPRAY ADDITIVE TANK DUE TO INCORRECT LEVEL INDICATION.
EVENT DATE: 012588 REPORT DATE: 022488 NSSS: WE TYPE: PWR

(NSIC 208347) ON JANUARY 25, 1988, AT 1315, THE PLANT WAS IN MODE 1 AT 100% POWER. AFTER COMPLETING THE SCHEDULED CALIBRATION OF THE CONTAINMENT SPRAY (EIIS:BE) ADDITIVE TANK LEVEL INDICATORS (LT-1CT-7150SA AND LT-1CT- 7166SB), THE

RESULTANT LEVEL INDICATION REVEALED THAT THE ACTUAL TANK LEVEL WAS LESS THAN THAT REQUIRED BY TECHNICAL SPECIFICATION 3.6.2.2. WITH THE LEVEL OUT-OF-SPECIFICATION, THE CONTAINMENT SPRAY ADDITIVE SYSTEM WAS DECLARED INOPERABLE, AND ACTIONS WERE TAKEN TO RESTORE THE LEVEL TO NORMAL. THE LOW LEVEL IN THE TANK WAS DUE TO INCORRECT LEVEL INDICATION WHICH WAS ATTRIBUTED TO AIR IN THE SENSING LINES TO THE LEVEL TRANSMITTERS. APPARENTLY, THIS CONDITION HAS EXISTED SINCE INITIAL PLANT STARTUP, RESULTING IN A VIOLATION OF TECHNICAL SPECIFICATION 3.6.2.2. THE LEVEL TRANSMITTERS WERE CALIBRATED AND THE SENSING LINES PROPERLY VENTED. APPROXIMATELY 165 GALLONS OF SODIUM HYDROXIDE (NAOH) SOLUTION WAS PUMPED INTO THE TANK TO RESTORE THE PROPER LEVEL, AND THE SYSTEM WAS DECLARED OPERABLE ON JANUARY 26, 1988 AT 1542.

[240] SHEARON HARRIS 1 DOCKET 50-400 LER 88-005
TURBINE BUILDING VENT STACK MONITOR SAMPLING ROOM AIR INSTEAD OF VENT STACK AIR DUE TO MISPOSITIONED VALVES CAUSED BY INADEQUATE PROCEDURE.
EVENT DATE: 013088 REPORT DATE: 022988 NSSS: WE TYPE: PWR

(NSIC 208348) THE PLANT WAS OPERATING IN MODE 1, POWER OPERATION, AT 100% REACTOR POWER FEBRUARY 1, 1988. AT 0930 HOURS, IT WAS DISCOVERED THAT TWO VALVES FOR THE TURBINE BUILDING VENT STACK 3A WIDE RANGE GAS MONITOR RM-01TV-3536-1 WERE MISPOSITIONED. AS RESULT OF THESE MISPOSITIONED VALVES, THE MONITOR WAS SAMPLING ROOM AIR INSTEAD OF VENT STACK AIR AS REQUIRED. IT HAS BEEN DETERMINED THAT THIS CONDITION HAD EXISTED SINCE JANUARY 30, 1988 AT 1831 HOURS WHEN THE MONITOR WAS RETURNED TO SERVICE FOLLOWING A MAINTENANCE ACTIVITY. THE CAUSE OF THE MISPOSITIONED VALVES WAS A PROCEDURAL DEFICIENCY IN THAT RESTORATION OF THE MISALIGNED VALVES WAS NOT ADEQUATELY ADDRESSED IN APPLICABLE PROCEDURES. NO SAFETY CONSEQUENCES RESULTED FROM THIS EVENT SINCE NO DETECTABLE RADIOACTIVITY HAS BEEN RELEASED VIA THIS STACK PRIOR TO OR AFTER THIS EVENT. ALSO, NO PRIMARY TO SECONDARY LEAKAGE CURRENTLY EXISTS IN THE PLANT. CORRECTIVE ACTIONS ARE: THE VALVE LINEUP FOR THE MONITOR SKID WAS IMMEDIATELY CORRECTED, HEALTH PHYSICS PROCEDURES ARE BEING REVISED TO INCLUDE VALVE LINEUPS AND INDEPENDENT VERIFICATION, AND APPLICABLE PERSONNEL WILL BE INSTRUCTED ON PROPER VALVE LINEUP PRIOR TO DECLARING MONITOR OPERABLE. THIS EVENT IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(I)(B) AS A VIOLATION OF TECHNICAL SPECIFICATIONS.

[241] SHEARON HARRIS 1 DOCKET 50-400 LER 88-006
BOTH EMERGENCY SERVICE WATER SYSTEMS INOPERABLE DUE TO ISOLATION VALVE FAILURES AND DESIGN DEFICIENCY.
EVENT DATE: 020888 REPORT DATE: 030888 NSSS: WE TYPE: PWR
VENDOR: ROCKWELL MANUFACTURING COMPANY
TARGET ROCK CORP.

(NSIC 208462) ON 2/8/88, DURING SURVEILLANCE TESTING OF THE EMERGENCY SERVICE WATER SYSTEM (ESWS), THE NONSAFETY PORTION OF THE ESW PUMP SEAL WATER SUPPLY SYSTEM FAILED TO ISOLATE AS REQUIRED WHEN TWO SOLENOID VALVES STUCK OPEN AND A CHECK VALVE FAILED TO SEAT. BOTH ESW TRAINS WERE DECLARED INOPERABLE DUE TO THE PIPING CONFIGURATION OF THE SEAL WATER SUPPLY AND THE LOCATIONS OF THE FAILED VALVES. MANUAL VALVES WERE CLOSED TO ISOLATE THE NONSAFETY PIPING AND PERMIT CONTINUED OPERATION WHILE REPAIRS TOOK PLACE. DEBRIS WAS DISCOVERED DURING DISASSEMBLY OF THE FAILED CHECK VALVE, AND IS ALSO SUSPECTED OF CAUSING THE SOLENOID VALVE FAILURES. REPAIRS WERE COMPLETED THE NEXT DAY, THE SURVEILLANCE TESTING WAS COMPLETED AND THE ESW SYSTEM WAS RETURNED TO SERVICE. ON FEBRUARY 12, 1988, A CONCERN REGARDING THE SEAL WATER PIPING CONFIGURATION VULNERABILITY TO SINGLE PASSIVE FAILURES WAS RAISED, AND A MANUAL VALVE WAS IMMEDIATELY CLOSED TO SEPARATE THE TRAINS WHILE AN EVALUATION WAS CONDUCTED. THIS EVALUATION CONCLUDED ON FEBRUARY 25, 1988, THAT THE PIPING CONFIGURATION WAS VULNERABLE TO SINGLE PASSIVE FAILURES WHICH COULD DISABLE BOTH ESW TRAINS, IN CONFLICT WITH THE SYSTEM DESIGN REQUIREMENTS SPECIFIED IN THE FINAL SAFETY ANALYSIS REPORT. THE

TWO TRAINS REMAIN ISOLATED BY LOCK CLOSED MANUAL VALVE PENDING A PERMANENT DESIGN CHANGE TO THE SEAL WATER SYSTEM.

[242] SHOREHAM DOCKET 50-322 LER 88-001
LOSS OF CONTINUOUS MONITORING/SAMPLING OF STATION VENTILATION EXHAUST DUE TO A LOSS OF POWER TO THE ALTERNATE MONITOR SAMPLE PUMP RESULTING FROM PERSONNEL ERROR.
EVENT DATE: 012888 REPORT DATE: 022688 NSSS: GE TYPE: BWR

(NSIC 208440) ON JANUARY 28, 1988, AT APPROXIMATELY 1700, IT WAS DISCOVERED BY A RADIOCHEMISTRY TECHNICIAN THAT THE SAMPLE PUMP FOR THE STATION VENTILATION EXHAUST RADIATION MONITOR WAS NOT RUNNING. THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) WITH THE MODE SWITCH IN SHUTDOWN WITH ALL RODS INSERTED IN THE CORE. THE ROOT CAUSE OF THE EVENT WAS PERSONNEL ERROR. THE MONITOR, 1D11*PNL-021 WAS BEING UTILIZED AS THE AUXILIARY SAMPLING EQUIPMENT REQUIRED TO CONTINUOUSLY COLLECT IODINE AND PARTICULATE SAMPLES TO SATISFY ACTION STATEMENT 122 OF TECH SPEC 3.3.7.11 DUE TO THE NORMAL STATION VENT MONITOR (1D11*PNL-041) BEING OUT OF SERVICE FOR TESTING. IT WAS DETERMINED, UPON REVIEW OF THE RMS COMPUTER PRINTOUT, THAT THE PUMP WAS TURNED OFF AT 1422, DUE TO A LOSS OF POWER TO THE MONITOR. MAINTENANCE MECHANICS WORKING IN THE EMERGENCY DIESEL GENERATOR (EDG) ROOM 102 HAD OPENED THE BREAKER WHILE INVESTIGATING AN UNRELATED PROBLEM. WHEN THE BREAKER WAS RECLOSED (APPROXIMATELY THREE SECONDS LATER), POWER WAS RESTORED TO THE MONITOR. SINCE THE PUMP AUTOMATIC START LOGIC REQUIRES AN RBSVS SIGNAL, THE PUMP DID NOT AUTOMATICALLY RESTART. THE OPERATOR IN THE CONTROL ROOM ASSUMED THAT PANEL 21 WAS BACK IN OPERATION, WHEN IN FACT, IT WASN'T. THE PUMP WAS TURNED BACK ON BY THE RADCHEM TECH. AT 1703.

[243] SOUTH TEXAS 1 DOCKET 50-498 LER 88-010
INOPERABILITY OF REACTOR COOLANT PUMP SEAL INJECTION CONTAINMENT ISOLATION VALVES.
EVENT DATE: 092387 REPORT DATE: 022588 NSSS: WE TYPE: PWR

(NSIC 208418) ON 9/23/87, A MAINTENANCE WORK REQUEST WAS PREPARED STATING THAT THE CHARGING HEADER PRESSURE INDICATOR (PI-204) WAS READING 1000 PSI MORE THAN THE LOCAL PROCESS PRESSURE GAUGE. THE SHIFT SUPERVISOR DID NOT REALIZE THAT THE OUTBOARD CONTAINMENT ISOLATION VALVES ON THE RCP SEAL INJECTION PIPING SYSTEM COULD BE AFFECTED. THE PRESSURE TRANSMITTER COULD NOT BE RECALIBRATED AND A REPLACEMENT WAS ORDERED. ON 1/26/88, WHILE THE PLANT WAS IN MODE 4, PRIOR TO INITIAL CRITICALITY ENGINEERING DISCOVERED DURING A TECH SPEC SURVEILLANCE REVIEW MEETING, THAT THE CHARGING HEADER PRESSURE INSTRUMENTATION LOOP DID NOT HAVE SURVEILLANCE REQUIREMENTS WITH RESPECT TO CONTAINMENT ISOLATION. THE PRESSURE TRANSMITTER (PT-204) MONITORS THE CHARGING HEADER PRESSURE AND PROVIDES AN ISOLATION SIGNAL TO THE REACTOR COOLANT PUMP (RCP) SEAL INJECTION CONTAINMENT ISOLATION VALVES SHOULD A LOW CHARGING PUMP DISCHARGE HEADER PRESSURE OCCUR. THE CHARGING HEADER PRESSURE INSTRUMENTATION LOOP WAS RETESTED WITH A REPLACEMENT TRANSMITTER IN SERVICE AND THE LOOP WAS DECLARED OPERABLE ON 1/28/88. THERE WERE NO ADVERSE SAFETY OR RADIOLOGICAL CONSEQUENCES AS A RESULT OF THIS EVENT SINCE THE PLANT HAD NOT ACHIEVED INITIAL CRITICALITY AND NO RADIOACTIVITY HAD BEEN PRODUCED. THE EVENT DID NOT PRODUCE ANY ADDITIONAL RISK TO THE PUBLIC. THE EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B).

[244] SOUTH TEXAS 1 DOCKET 50-498 LER 88-001
REACTOR COOLANT PUMP START WITH SECONDARY WATER TEMPERATURE GREATER THAN 50 DEGREES F ABOVE THE REACTOR COOLANT SYSTEM, AND PRESSURIZER PORV ACTUATION.
EVENT DATE: 010288 REPORT DATE: 020188 NSSS: WE TYPE: PWR

(NSIC 208244) AT APPROXIMATELY 1451 HOURS ON JANUARY 2, 1988, WITH UNIT 1 IN MODE 5 WHILE FILLING AND VENTING THE REACTOR COOLANT SYSTEM (RCS), WITH THE PRESSURIZER WATER SOLID REACTOR COOLANT PUMP (RCP) 1A WAS STARTED WITH RCS COLD LEG TEMPERATURE LESS THAN 350 DEGREES F AND SECONDARY WATER TEMPERATURES GREATER

THAN 50 DEGREES ABOVE THE RCS COLD LEG TEMPERATURES. THESE CONDITIONS EXCEEDED THE LIMITS OF TECH SPECS 3.4.1.4.1. DUE TO INSUFFICIENT PROCEDURAL GUIDANCE AND TRAINING, PLANT OPERATORS ERRONEOUSLY CONCLUDED THAT THE TEMPERATURE DIFFERENCE WAS LESS THAN 50 DEGREES F. ACTUATION OF RCP 1A UNDER THESE CONDITIONS RESULTED IN RCS PRESSURE EXCEEDING THE COLD OVERPRESSURE MITIGATION SYSTEM (COMS) PRESSURE SETPOINT, CAUSING PRESSURIZER PORV PCV-0656A TO OPEN MOMENTARILY AND THEN CLOSE. THE COMS ALARM SOUNDED IN THE CONTROL ROOM AND RCP 1A WAS SHUT DOWN. THE RCS PRESSURE WAS REDUCED TO LESS THAN 50 PSIG BY MANUALLY CONTROLLING LETDOWN PRESSURE CONTROL VALVE PCV-0135. TO PREVENT RECURRENCE OF THE EVENT, PROCEDURES HAVE BEEN REVISED TO PROVIDE DIRECTION ON TEMPERATURE MEASUREMENT, AND OPERATORS WILL RECEIVE ADDITIONAL TRAINING.

[245] SOUTH TEXAS 1 DOCKET 50-498 LER 88-002
FAILURE TO PERFORM LOCAL LEAKAGE RATE TESTING ON CONTAINMENT ISOLATION VALVES.
EVENT DATE: 010588 REPORT DATE: 020488 NSSS: WE TYPE: PWR
VENDOR: HILLS-MCCANNA COMPANY

(NSIC 208214) ON JANUARY 5, 1988, WITH UNIT 1 IN MODE 5 PRIOR TO INITIAL CRITICALITY, IT WAS DETERMINED THAT A REQUIRED POST MAINTENANCE TEST (PMT) HAD NOT BEEN PERFORMED ON A PAIR OF CONTAINMENT ISOLATION VALVES (CIVS) AS REQUIRED BY TECHNICAL SPECIFICATION 4.0.5 PRIOR TO ENTERING MODE 4. THE UNIT HAD BEEN OPERATED IN MODE 4 AFTER THE MAINTENANCE WORK AND PRIOR TO DISCOVERY OF THE INADEQUATE PMT. SUBSEQUENT TESTING REVEALED THAT ONE OF THE TWO CIVS EXCEEDED ITS LOCAL LEAKAGE RATS REQUIREMENTS. THIS EVENT OCCURRED AS A RESULT OF A LACK OF TRAINING WITH REGARD TO THE POST MAINTENANCE TESTING REQUIREMENTS FOR CONTAINMENT ISOLATION VALVES AND FAILURE OF SUPERVISORY PERSONNEL TO IDENTIFY THAT CONTAINMENT ISOLATION VALVES HAD NOT RECEIVED ADEQUATE POST MAINTENANCE TESTING. THE VALVES WERE RETESTED AND REWORKED AS APPROPRIATE TO A SATISFACTORY CONDITION WITHIN APPROXIMATELY TWENTY- SEVEN HOURS OF DISCOVERY OF THE EVENT. MAINTENANCE SUPERVISORY PERSONNEL AND MAINTENANCE WORK PLANNERS HAVE BEEN COUNSELLED WITH REGARD TO THE IMPORTANCE OF PMT REQUIREMENTS OF CIVS. PLANT PROCEDURES WILL BE CLARIFIED AND ENHANCED. TRAINING WILL BE GIVEN ON THE PROCEDURES AND THE PMT REQUIREMENTS FOR CIVS. NO ADVERSE AFFECT UPON THE SAFETY OF THE PUBLIC OCCURRED AS A RESULT OF THIS EVENT.

[246] SOUTH TEXAS 1 DOCKET 50-498 LER 88-003
CONTROL ROOM VENTILATION ACTUATION TO RECIRCULATION MODE DUE TO IMPROPER OPERATOR ACTION.
EVENT DATE: 010688 REPORT DATE: 020588 NSSS: WE TYPE: PWR

(NSIC 208215) AT APPROXIMATELY 2257 HOURS ON JANUARY 5, 1988 WITH THE UNIT 1 IN MODE 5, THE PRIMARY POWER SUPPLY UNDERVOLTAGE RELAY ASSOCIATED WITH TRANSFER SWITCH ESW002 FAILED. THIS RELAY FAILURE TRANSFERRED THE NON-CLASS 1E DISTRIBUTION PANEL DP002 TO AN EMERGENCY POWER SUPPLY. AT APPROXIMATELY 1021 HOURS ON JANUARY 6, 1988 AN OPERATOR (NON-LICENSED) WAS SENT TO TRANSFER THE DISTRIBUTION PANEL BACK TO THE PREFERRED POWER SUPPLY. IMPROPER OPERATOR ACTION AND PROCEDURE INADEQUACIES RESULTED IN A LOSS OF THE EMERGENCY POWER SUPPLY WITH THE SUBSEQUENT LOSS OF CONTROL POWER TO THE TOXIC GAS MONITOR ACTUATION RELAYS. THIS CAUSED AN AUTOMATIC ACTUATION OF THE CONTROL ROOM VENTILATION SYSTEM TO THE RECIRCULATION MODE. THE CORRECTIVE ACTIONS WHICH ARE BEING TAKEN INCLUDE RETRAINING THE OPERATORS AND REVISING THE OPERATING PROCEDURES.

[247] SOUTH TEXAS 1 DOCKET 50-498 LER 88-004
LOOSE OR CORRODED TOXIC GAS MONITOR COMPUTER BOARD ELECTRICAL CONNECTION RESULTING IN AN ESF ACTUATION.
EVENT DATE: 011088 REPORT DATE: 020988 NSSS: WE TYPE: PWR
VENDOR: FOXBORO CO., THE

(NSIC 208281) AT APPROXIMATELY 0506 HOURS ON JANUARY 10, 1988 WITH UNIT 1 IN MODE 5, A COMPUTER BOARD ELECTRICAL CONNECTION FAILURE IN ONE OF THE TOXIC GAS MONITORS CAUSED THE CONTROL ROOM VENTILATION SYSTEM TO AUTOMATICALLY SHIFT INTO THE RECIRCULATION MODE. THIS EVENT COULD HAVE BEEN CAUSED BY A LOOSE OR CORRODED CONNECTION BETWEEN A COMPUTER BOARD AND THE ANALYZER CARD CONNECTOR WHICH CAUSED THE TOXIC GAS ANALYZER TO RESPOND TO WHAT THE LOGIC SENSED AS A PROGRAMMABLE READ ONLY MEMORY (PROM) COMPUTER CHIP FAILURE. CORRECTIVE ACTIONS INCLUDE INSPECTING AND CLEANING THE RANDOM ACCESS MEMORY PRINTED WIRING ASSEMBLY BOARDS AND CAGES. IN ADDITION AN ENGINEERING REVIEW OF THE COMPUTER BOARD MOUNTINGS WILL BE PERFORMED.

[248] SOUTH TEXAS 1 DOCKET 50-498 LER 88-005
INADEQUATE SURVEILLANCE PERFORMED ON A CONTROL ROOM INTAKE AIR RADIOACTIVITY MONITOR.
EVENT DATE: 011088 REPORT DATE: 020988 NSSS: WE TYPE: PWR

(NSIC 208282) AT APPROXIMATELY 1148 HOURS ON JANUARY 10, 1988, WITH UNIT 1 IN MODE 5, A LICENSED REACTOR OPERATOR NOTICED DURING PREPARATION OF MODES 5 AND 6 OPERATOR LOGS THAT CONTROL ROOM INTAKE AIR RADIOACTIVITY MONITOR READINGS WERE CONSTANT AT 6.10 E-05 UCI/CC. A LOG REVIEW REVEALED THAT OPERATORS HAD BEEN RECORDING THE HI-ALARM SETPOINT RATHER THAN ACTUAL GASEOUS ACTIVITY FOR THE CHANNEL CHECK SINCE JANUARY 8, 1988. THE RADIOACTIVITY MONITOR WAS DISPLAYING THE HI-ALARM SETPOINT BECAUSE THE MONTHLY SURVEILLANCE PROCEDURE DID NOT ENSURE THAT THE MONITOR DISPLAY WAS RETURNED TO NORMAL. THIS RESULTED IN MISSING THE CHANNEL CHECK SURVEILLANCE SPECIFIED IN TECHNICAL SPECIFICATIONS WHICH ARE REQUIRED TO BE PERFORMED EVERY 12 HOURS. THE ROOT CAUSES OF THIS EVENT WERE PROCEDURE ERROR AND OPERATOR ERROR. TO PREVENT RECURRENCE OF THE EVENT, PROCEDURES WILL BE REVISED AND ADDITIONAL GUIDANCE WILL BE GIVEN TO OPERATORS ON HOW TO RECORD AND EVALUATE ACTIVITY READINGS.

[249] SOUTH TEXAS 1 DOCKET 50-498 LER 88-006
INADEQUATE SURVEILLANCE TESTING OF MASTER RELAYS.
EVENT DATE: 011288 REPORT DATE: 021188 NSSS: WE TYPE: PWR

(NSIC 208283) ON JANUARY 12, 1988, WITH THE UNIT IN MODE 5, A REVIEW OF A REFUELING WATER STORAGE TANK LEVEL ANALOG CHANNEL OPERATIONAL TEST (ACOT) SURVEILLANCE PROCEDURE, DETERMINED THAT THE SOLID STATE PROTECTION SYSTEM ACTUATION TRAIN MASTER RELAYS WHICH INITIATE AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP DURING A SAFETY INJECTION WERE NOT ADEQUATELY TESTED. THE MASTER RELAY ACTUATED DURING THE TEST BUT WAS NOT VERIFIED. THE SURVEILLANCE TEST IS REQUIRED BY TECHNICAL SPECIFICATIONS IN PLANT OPERATING MODES 1 THROUGH 4. THE CAUSE OF THIS EVENT WAS AN INADEQUATE TECHNICAL DEVELOPMENT OF A SURVEILLANCE PROCEDURE WHICH RESULTED IN A SURVEILLANCE DEFICIENCY. THREE AFFECTED PROCEDURES WERE IMMEDIATELY CORRECTED AND PERFORMED TO VERIFY OPERABILITY OF THE AUTOMATIC SWITCHOVER FUNCTION IN ALL THREE ACTUATION TRAINS. TO PREVENT RECURRENCE, SURVEILLANCE TEST PROCEDURES ARE BEING REVIEWED TO ENSURE THAT THE PROCEDURES ADEQUATELY ADDRESS THE TECHNICAL SPECIFICATION REQUIREMENTS FROM SENSOR TO ACTUATED DEVICE AND TO ENSURE THAT THE APPROPRIATE FUNCTIONS OF SYSTEMS AND COMPONENTS ARE TESTED AND WILL PERFORM AS DESIGNED.

[250] SOUTH TEXAS 1 DOCKET 50-498 LER 88-007
INCORRECT FORMULA IN A HVAC SURVEILLANCE PROCEDURE.
EVENT DATE: 011588 REPORT DATE: 021688 NSSS: WE TYPE: PWR

(NSIC 208284) ON JANUARY 15, 1988, WHILE THE PLANT WAS IN MODE 5, PRIOR TO INITIAL CRITICALITY, A SURVEILLANCE PROCEDURE REVIEW IDENTIFIED THAT INCORRECTLY APPLIED AIR DENSITY CORRECTION FACTORS WERE USED IN AIRFLOW CALCULATIONS. THE AFFECTED TEST DATA WAS REVIEWED AND IT WAS DETERMINED THAT DURING SOME TESTS,

AIRFLOW RATES WERE SLIGHTLY IN EXCESS OF PRESCRIBED TOLERANCES. THE CAUSE OF THIS EVENT WAS AN ERROR IN DEVELOPMENT AND A WEAKNESS IN THE INDEPENDENT REVIEW PROCESS FOR THE HEATING, VENTILATION AND AIR CONDITIONING (HVAC) FILTER TEST PROCEDURES. TO CORRECT THIS PROCEDURE INADEQUACY THE SURVEILLANCE PROCEDURES ARE BEING REVISED. ALSO THE PREOPERATIONAL TEST AND SURVEILLANCE PROCEDURES WHICH WERE USED TO SATISFY SURVEILLANCE REQUIREMENTS HAVE BEEN REVIEWED TO IDENTIFY OUT-OF-SPECIFICATION FLOWRATES. WHERE PREVIOUS DATA WAS OUTSIDE OF ACCEPTANCE CRITERIA, APPLICABLE TESTS HAVE BEEN REPERFORMED. ADDITIONALLY, A SECOND, INDEPENDENT, TECHNICAL REVIEW OF PROCEDURE REVISIONS AND NEW PROCEDURES IS NOW BEING PERFORMED.

[251] SOUTH TEXAS 1 DOCKET 50-498 LER 88-008
SAFETY-RELATED ELECTRICAL CABLE SPLICES INCORRECTLY INSTALLED.
EVENT DATE: 012088 REPORT DATE: 021988 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SOUTH TEXAS 2 (PWR)
VENDOR: RAYCHEM CORP.

(NSIC 208368) AT APPROXIMATELY 1330 HOURS ON JANUARY 20, 1988, WITH UNIT 1 IN MODE 5, THE SHIFT SUPERVISOR WAS NOTIFIED THAT A NUMBER OF INCORRECTLY INSTALLED SAFETY-RELATED ELECTRICAL CABLE SPLICES WERE FOUND IN UNIT 1 WHICH COULD HAVE HAD A SAFETY IMPACT. SITE PROCEDURES WHICH REQUIRED THE SPLICES TO BE INSTALLED AND INSPECTED TO THE CORRECT CABLE SPLICE DETAIL SHEET WERE NOT FOLLOWED. THIS CONDITION RESULTED IN A SINGLE CAUSE WHICH AFFECTED, AMONG OTHER EQUIPMENT, TWO (2) INDEPENDENT TRAINS OF THE RESIDUAL HEAT REMOVAL SYSTEM. THESE DEFICIENT SPLICES HAVE BEEN REWORKED PENDING FINAL VERIFICATION AND PERSONNEL HAVE BEEN REINSTRUCTED ON THE INSTALLATION OF THE SUBJECT SPLICES. TO PREVENT RECURRENCE OF THE EVENT, THE PROCEDURE HAS BEEN CHANGED REQUIRING MORE DETAILED WORK INSTRUCTIONS AND INSPECTIONS REGARDING THE INSTALLATION OF THE SPLICES. AN INVESTIGATION OF THE UNIT 1 RAYCHEM SPLICE PROGRAM IS CURRENTLY BEING PERFORMED AND WILL IDENTIFY THE ULTIMATE ROOT CAUSE AND CORRECTIVE ACTIONS.

[252] SOUTH TEXAS 1 DOCKET 50-498 LER 88-009
UNANTICIPATED SAFETY INJECTION SIGNAL FROM SOLID STATE PROTECTION SYSTEM RESULTING FROM PROCEDURAL DEFICIENCY.
EVENT DATE: 012388 REPORT DATE: 021988 NSSS: WE TYPE: PWR

(NSIC 208369) AT APPROXIMATELY 0144 HOURS ON JANUARY 23, 1988, WITH UNIT 1 IN MODE 5, AN UNANTICIPATED SAFETY INJECTION SIGNAL OCCURRED FROM SOLID STATE PROTECTION SYSTEM ACTUATION TRAIN A. ALL ACTUATED EQUIPMENT OPERATED AS REQUIRED AND NO OTHER PLANT TRANSIENTS RESULTED FROM THIS ACTUATION. THE EVENT WAS INITIATED WHEN A SURVEILLANCE PROCEDURE, AS MODIFIED BY A FIELD CHANGE, WAS PERFORMED WHICH HAD A STEP MISSING FROM THE SEQUENCE. TRAIN B HAD BEEN SUCCESSFULLY TESTED PREVIOUSLY USING THE SAME PROCEDURE, AS MODIFIED BY A FIELD CHANGE, WITHOUT INITIATING A SAFETY INJECTION SIGNAL. THE TECHNICIAN, BASED ON EXPERIENCE, TOOK THE ACTION THE PROCEDURE NEEDED TO PREVENT AN ESF ACTUATION. THE ROOT CAUSE OF THE EVENT WAS AN INADEQUATE SURVEILLANCE PROCEDURE. TO PREVENT RECURRENCE OF THE EVENT, TEST PROCEDURES HAVE BEEN REVISED, TRAINING WILL BE CONDUCTED ON PROCEDURE CONTROL, AND THE SURVEILLANCE PROCEDURE REVIEW PROCESS IS BEING CHANGED.

[253] SOUTH TEXAS 1 DOCKET 50-498 LER 88-011
NON-PERFORMANCE OF SCHEDULED TEST FOR ESSENTIAL CHILLED WATER PUMP AS A RESULT OF A LOST TEST PACKAGE.
EVENT DATE: 012788 REPORT DATE: 022688 NSSS: WE TYPE: PWR

(NSIC 208479) ON JANUARY 27, 1988, WHILE THE PLANT WAS IN MODE 4, PRIOR TO INITIAL CRITICALITY, THE SURVEILLANCE TEST COORDINATOR FOUND THAT A SCHEDULED SURVEILLANCE TEST ON ESSENTIAL CHILLED WATER PUMP 11B WAS NOT PERFORMED. THE

SURVEILLANCE TEST WAS RUN ON ESSENTIAL CHILLED WATER PUMP 11B WITHIN EIGHT HOURS AND ALL MEASURED PARAMETERS WERE IN THE ACCEPTABLE RANGE FOR PUMP OPERABILITY. THE CAUSE OF THE MISSED SURVEILLANCE WAS DETERMINED TO BE INADEQUATE CONTROL OF SURVEILLANCE TEST PACKAGES, SINCE THE TEST PACKAGE FOR PUMP 11B APPARENTLY WAS LOST DURING ROUTING BETWEEN THE SURVEILLANCE PROGRAM SCHEDULER AND THE MAIN CONTROL ROOM. THERE WERE NO ADVERSE SAFETY OR RADIOLOGICAL CONSEQUENCES AS A RESULT OF THIS EVENT SINCE THE PLANT HAD NOT ACHIEVED INITIAL CRITICALITY AND NO RADIOACTIVITY HAD BEEN PRODUCED. TO PREVENT RECURRENCE OF THE EVENT SURVEILLANCE PROGRAM PROCEDURES ARE BEING REVISED TO ADD MORE APPROPRIATE CONTROLS.

[254] SOUTH TEXAS 1 DOCKET 50-498 LER 88-012
FAILURE TO FULLY IMPLEMENT TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 012988 REPORT DATE: 022988 NSSS: WE TYPE: PWR

(NSIC 208467) AT APPROXIMATELY 1145 HOURS ON JANUARY 29, 1988, WITH UNIT 1 IN MODE 4, PRIOR TO INITIAL CRITICALITY, AN INCONSISTENCY WAS IDENTIFIED DURING A REVIEW OF REVISION 2 OF THE STPEGS UNIT 1 PUMP AND VALVE INSERVICE TEST (IST) PLAN. A LIMITING VALUE FOR STROKE TIME ON VALVE CV-FCV-0205 WAS FOUND TO BE OMITTED. UPON REVIEW OF THE AFFECTED SURVEILLANCE PROCEDURE IT WAS LEARNED THAT ALTHOUGH THE VALVE WAS REQUIRED TO BE STROKED OPEN AND CLOSED, MEASUREMENT AND RECORDING OF THE CORRESPONDING OPEN STROKE TIME WAS NOT PERFORMED. A TEMPORARY PROCEDURE WAS GENERATED AND PERFORMED TO RETURN THE VALVE TO FULL SERVICE. THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE ADMINISTRATIVE CONTROLS ON THE PROCESSES USED TO REVISE THE IST PLAN AND CORRESPONDING SURVEILLANCE PROCEDURES. THE IST PLAN REVIEW WAS COMPLETED WITH NO FURTHER DISCREPANCIES NOTED; A SUBSEQUENT REVIEW OF AFFECTED SURVEILLANCE PROCEDURES IDENTIFIED NO ADDITIONAL DISCREPANCIES. TO PRECLUDE SIMILAR OCCURRENCES THE CONTROLLING DOCUMENT, PROCEDURE OPGP03-ZE-0021, INSERVICE TESTING PROGRAM FOR VALVES, IS BEING REVISED TO INCLUDE ADDITIONAL CONTROLS. ADDITIONALLY, A SECOND, INDEPENDENT TECHNICAL REVIEW OF PROCEDURE REVISIONS AND NEW PROCEDURES IS NOW REQUIRED TO BE PERFORMED.

[255] SOUTH TEXAS 1 DOCKET 50-498 LER 88-013
FAILURE TO TEST RCS LOW FLOW TIMERS.
EVENT DATE: 020488 REPORT DATE: 030488 NSSS: WE TYPE: PWR
VENDOR: AGASTAT RELAY CO.

(NSIC 208468) ON FEBRUARY 4, 1988 AT APPROXIMATELY 1349 HOURS WITH UNIT 1 IN MODE 3 AND PRIOR TO INITIAL CRITICALITY, A SYSTEM ENGINEER IDENTIFIED TWO (2) TIME DELAY RELAYS IN THE SOLID STATE PROTECTION SYSTEM (SSPS) WHICH HAD NOT BEEN TESTED UNDER THE SURVEILLANCE TEST PROGRAM. SINCE THESE RELAYS ARE REQUIRED TO BE DEMONSTRATED OPERABLE BY THE TECHNICAL SPECIFICATIONS THEY WERE DECLARED INOPERABLE AND LIMITING CONDITION FOR OPERATION (LCO) STATEMENT 3.0.3 WAS ENTERED UNTIL TESTING WAS PERFORMED. THESE RELAYS ARE INCLUDED IN THE TECHNICAL SPECIFICATIONS BECAUSE THEY ARE SAFETY-RELATED AND PERFORM A DIVERSE ENGINEERED SAFETY FEATURE FUNCTION; HOWEVER, THESE RELAYS ARE NOT PART OF A PRIMARY REACTOR PROTECTION SCHEME AND ARE NOT TAKEN CREDIT FOR IN ANY FSAR CHAPTER 15 ACCIDENT ANALYSIS, THEREFORE, THERE WOULD BE NO SAFETY SIGNIFICANCE SHOULD THEY FAIL TO FUNCTION. ACTIONS TO PREVENT RECURRENCE INCLUDE A REVIEW OF OTHER SURVEILLANC PROCEDURES FOR SIMILAR OMISSIONS, REVISION TO PLANT PROCEDURES ON SURVEILLANCE TEST PROCEDURE PREPARATION AND REVIEW, AND REVISION OF SURVEILLANCE PROCEDURES TO INCLUDE LOW FLOW TIMER TESTING.

[256] ST. LUCIE 1 DOCKET 50-335 LER 87-012 REV 01
UPDATE ON LOSS OF COMPONENT COOLING WATER REDUNDANCY CAUSED BY CROSSTIE VALVES BEING IN THE OPEN POSITION DUE TO PERSONNEL ERROR.
EVENT DATE: 060987 REPORT DATE: 020988 NSSS: CE TYPE: PWR

(NSIC 208254) ON JUNE 19, 1987, ST. LUCIE UNIT #1 WAS IN MODE 1, 100% POWER, AND AT STEADY STATE CONDITIONS. ALL CONTROL STATIONS WERE IN NORMAL OPERATING MODE. AT 0124 HOURS, THE REACTOR CONTROL OPERATOR (RCO) WAS PERFORMING A MONTHLY PUMP SURVEILLANCE RUN FOR THE 1B COMPONENT COOLING WATER (CCW) PUMP. DURING THIS SURVEILLANCE RUN, IT WAS DISCOVERED THAT THE 1A AND 1B CCW HEAT EXCHANGER OUTLET CROSS-TIE VALVES WERE IN THE OPEN POSITION. THE POSITION FOR THESE VALVES IS NORMALLY CLOSED. THE OPERATOR IMMEDIATELY PERFORMED A VALVE ALIGNMENT VERIFICATION AND CLOSED THE VALVES. THE ROOT CAUSE OF THE EVENT WAS A COGNITIVE PERSONNEL ERROR BY UTILITY LICENSED OPERATORS WHO FAILED TO HAVE PROPER ADMINISTRATIVE CONTROL ON THESE VALVES. THE IMMEDIATE CORRECTIVE ACTION WAS TO CLOSE THE VALVES. THE EVENT WAS TERMINATED AT 0130 HOURS. NO OTHER SYSTEM MALFUNCTION RESULTED FROM THIS EVENT. THIS EVENT IS REPORTABLE UNDER THE CODE OF FEDERAL REGULATIONS 10 CFR 50.73 (A)(2)(I)(B), "ANY EVENT OR CONDITION PROHIBITED BY PLANT'S TECHNICAL SPECIFICATIONS".

[257] ST. LUCIE 1 DOCKET 50-335 LER 87-013 REV 01
UPDATE ON REACTOR TRIP ON HIGH PRESSURE DUE TO TURBINE RUNBACK CAUSED BY LOSS OF 1B SGFP.
EVENT DATE: 061487 REPORT DATE: 021988 NSSS: CE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 208378) ON JUNE 14, 1987, WHILE OPERATING AT 100% POWER (MODE 1), ST. LUCIE UNIT 1 EXPERIENCED A MAIN TURBINE RUNBACK TO 60% POWER DUE TO THE LOSS OF THE 1B STEAM GENERATOR FEED PUMP (SGFP). THE UTILITY LICENSED OPERATORS RESTARTED THE SGFP AND MANUALLY INSERTED CONTROL RODS TO REDUCE THE PRIMARY/SECONDARY POWER MISMATCH. THE UNIT AUTOMATICALLY TRIPPED ON HIGH PRESSURE APPROXIMATELY 22 SECONDS INTO THE EVENT. THE TRIP WAS UNCOMPLICATED AND THE UNIT WAS QUICKLY STABILIZED IN HOT STANDBY. THE CAUSE OF THE 1B SGFP TRIP HAS NOT BEEN CONCLUSIVELY IDENTIFIED; HOWEVER, A FAILURE OF THE DIFFERENTIAL CURRENT RELAY, TYPE SA-1, IN THE 1B SGFP IS BELIEVED TO HAVE BEEN THE CAUSE. AN UPGRADE HAS BEEN INSTALLED IN ALL UNIT #2 AND IN ONE UNIT #1 TYPE SA-1 RELAYS. THE REMAINING UPGRADES FOR UNIT #1 TYPE SA-1 RELAYS ARE ON ORDER. THE INTENTION OF THE MAIN TURBINE RUNBACK IS TO REDUCE THE TURBINE STEAM LOAD IN RESPONSE TO THE REDUCED FEEDWATER FLOW AND SUBSEQUENT DECREASE IN S/G INVENTORIES. IT IS NOT REQUIRED FOR REACTOR SAFETY. THE REACTOR PROTECTION SYSTEM WILL ASSURE REACTOR SAFETY BY AUTOMATICALLY TRIPPING THE UNIT AND THE MAIN TURBINE ON LOW S/G LEVEL OR HIGH PRESSURE AS IT DID IN THIS EVENT. THIS WILL ASSURE SUFFICIENT S/G INVENTORIES TO PROVIDE AN ADEQUATE PRIMARY HEAT SINK.

[258] SURRY 1 DOCKET 50-280 LER 87-024 REV 01
UPDATE ON REACTOR TRIP ON LOW RCS FLOW DUE TO REACTOR COOLANT PUMP TRIP.
EVENT DATE: 092087 REPORT DATE: 021788 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 208305) ON SEPTEMBER 20, 1987 AT 2028 HOURS, WITH UNIT 1 AT 100% POWER, A LOW REACTOR COOLANT SYSTEM (RCS) FLOW REACTOR TRIP OCCURRED WHEN THE 'B' REACTOR COOLANT PUMP (RCP) (EIIS-AB P) TRIPPED. APPROXIMATELY 35 SECONDS AFTER THE REACTOR TRIP, A HIGH STEAM FLOW WITH LOW RCS TAVG SAFETY INJECTION OCCURRED. OPERATORS PERFORMED THE APPROPRIATE EMERGENCY AND FUNCTION RESTORATION PROCEDURES AND QUICKLY STABILIZED THE UNIT. THE 'B' REACTOR COOLANT PUMP BREAKER TRIPPED ON INSTANTANEOUS GROUND FAULT. INSPECTION OF THE MOTOR LEADS REVEALED A COMPLETE SEPARATION OF THE 'A' PHASE MAIN LOAD CONNECTION BUS BAR. AN ENGINEERING EVALUATION HAS CONCLUDED THAT THE FAILURE WAS CAUSED BY VIBRATION OF THE UNSUPPORTED LENGTH OF FEEDER CABLE LEADS WHICH FATIGUED AND CRACKED THE COPPER AT THE KNEE OF THE 90 DEGREE BEND IN THE BUS BAR. THE FAILED BUS BAR WAS REPLACED. THE OTHER BUS BARS ON A, B AND C RCPS WERE VISUALLY INSPECTED AND MEGGERED TO VERIFY THEIR INTEGRITY. A CABLE RESTRAINT WILL BE INSTALLED IN THE MOTOR TERMINATION BOX TO SECURE THE FEEDER CABLE LEADS TO PREVENT VIBRATION. THE HIGH STEAM FLOW SAFETY INJECTION INITIATION WAS DETERMINED TO BE SPURIOUS.

[259] SURRY 1 DOCKET 50-280 LER 88-001
CONTAINMENT ISOLATION VALVE NOT PROPERLY LOCKED UNDER ADMINISTRATIVE CONTROL DUE TO HUMAN ERROR.
EVENT DATE: 013188 REPORT DATE: 030188 NSSS: WE TYPE: PWR

(NSIC 208405) ON JANUARY 31, 1988 AT 0055 HOURS, UNIT 1 WAS AT 100% POWER. DURING THE NORMAL ROUNDS OF THE AUXILIARY BUILDING OPERATOR, IT WAS OBSERVED THAT THE LOCK AND CHAIN WERE NOT PROPERLY INSTALLED ON THE SAFETY INJECTION (EIIS BQ) PIPING FLUSH AND FILL LINE VALVE (EIIS-ISV) (1-SI-150). TECHNICAL SPECIFICATIONS 1.0.H.1 REQUIRE THAT ALL NON-AUTOMATIC CONTAINMENT (EIIS-NH) ISOLATION VALVES BE LOCKED CLOSED AND UNDER ADMINISTRATIVE CONTROL. 1-SI-150 WAS OBSERVED TO BE IN THE CLOSED POSITION, HOWEVER, IT WAS NOT LOCKED UNDER ADMINISTRATIVE CONTROL. THE VALVE WAS VERIFIED TO BE IN THE CLOSED POSITION. THE CHAIN WAS SECURED ON THE VALVE AND VERIFIED TO BE LOCKED CLOSED. A HUMAN PERFORMANCE EVALUATION SYSTEM INVESTIGATION WAS CONDUCTED, HOWEVER, IT WAS NOT ABLE TO DETERMINE THE EXACT TIME OR CAUSE OF WHY THE VALVE WAS NOT LOCKED. THE STATION ADMINISTRATIVE CONTROL PROCESS WILL BE STRENGTHENED TO ENHANCE CONTROL OF ADMINISTRATIVE LOCKS AND KEYS. OPERATORS WILL BE INSTRUCTED ON THE PROPER LOCKING OF T HANDLE VALVES.

[260] SURRY 1 DOCKET 50-280 LER 88-002
INOPERABLE INDIVIDUAL ROD POSITION INDICATORS DUE TO INSTRUMENT DRIFT.
EVENT DATE: 020688 REPORT DATE: 022988 NSSS: WE TYPE: PWR

(NSIC 208406) ON FEBRUARY 6, 1988 AT 1450 HOURS, WITH UNIT 1 AT 70% POWER, FOLLOWING THE PERFORMANCE OF THE TURBINE STOP VALVE/GOVERNOR VALVE FREEDOM TEST, PT-29.1, THE UNIT 1 INDIVIDUAL ROD POSITION INDICATORS (IRPIS) (EIIS-ZI) FOR CONTROL BANK 'D', RODS H-2, P-8, F-10, K-10 AND K-6 INDICATED MORE THAN 12 STEPS FROM THE CONTROL ROD GROUP D' DEMAND COUNTER POSITION (EIIS-CTR). THE IRPIS WERE DECLARED INOPERABLE IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.12.E. THE IRPIS WERE CALIBRATED AND RETURNED TO SERVICE AT 1547 HOURS. IRPI ROD POSITION INDICATION IS KNOWN TO BE A GENERIC WESTINGHOUSE PWR CONCERN. THE INSTRUMENT DRIFT APPEARS TO BE AFFECTED BY REACTOR COOLANT TEMPERATURE.

[261] SURRY 2 DOCKET 50-281 LER 88-001
IMPROPER ADMINISTRATIVE CONTROL OF CONTAINMENT ISOLATION VALVES DUE TO PERSONNEL ERROR.
EVENT DATE: 012788 REPORT DATE: 022688 NSSS: WE TYPE: PWR
VENDOR: HOKE, INC.

(NSIC 208453) ON JANUARY 27, 1988, WITH UNIT 2 AT 100% POWER, IT WAS DETERMINED AT 1944 HOURS THAT LEAKAGE THROUGH CONTAINMENT ISOLATION VALVES TV-SS-201A AND TV-SS-201B (PRESSURIZER VAPOR SPACE SAMPLE) (EIIS-JM ISV) WAS GREATER THAN THE ASME SECTION XI SPECIFICATION. THE VALVES WERE DECLARED INOPERABLE AT THAT TIME. THE VALVES WERE MAINTAINED CLOSED AND PLACED UNDER ADMINISTRATIVE CONTROL AT 2241 HOURS BY LIFTING A LEAD ON TV-SS-201A (A SOLENOID OPERATED VALVE) AND BY LIFTING A LEAD AND ISOLATING INSTRUMENT AIR TO TV-SS-201B (AN AIR OPERATED VALVE). HOWEVER, ON FEBRUARY 2, 1988 AT 2225 HOURS, IT WAS DISCOVERED THAT THE WRONG LEADS HAD BEEN LIFTED FOR THE TRIP VALVES, AND THAT TV-SS-201A HAD NOT BEEN PROPERLY ADMINISTRATIVELY CONTROLLED. THE CORRECT LEADS WERE LIFTED AT 2324 HOURS, AND THE TRIP VALVES WERE VERIFIED TO BE PROPERLY CONTROLLED. ELECTRICIANS HAVE BEEN INSTRUCTED AS TO WHICH ARE THE PROPER LEADS TO LIFT TO DISABLE THESE VALVES.

[262] SURRY 2 DOCKET 50-281 LER 88-002
INOPERABLE CONTAINMENT ISOLATION VALVES DUE TO EXCESSIVE LEAKAGE.
EVENT DATE: 020288 REPORT DATE: 030388 NSSS: WE TYPE: PWR
VENDOR: VALCOR ENGINEERING CORP.

(NSIC 208454) ON FEBRUARY 2, 1988, UNIT 2 WAS AT 100% POWER. AT 1715 HOURS, IT WAS DETERMINED THAT THE INDIVIDUAL VALVE LEAKAGE THROUGH THE REACTOR COOLANT COLD LEG SAMPLE ISOLATION VALVES (EIIS-JM ISV) (TV-SS-202A AND TV-SS-202B) WAS GREATER THAN THE ASME SECTION XI SPECIFICATION. THEY WERE THEN DECLARED INOPERABLE. TECHNICAL SPECIFICATIONS REQUIRE ADMINISTRATIVE CONTROL FOR INOPERABLE AUTOMATIC CONTAINMENT ISOLATION VALVES. THE VALVES WERE MAINTAINED CLOSED AND PLACED UNDER ADMINISTRATIVE CONTROL AT 2244 HOURS BY LIFTING A LEAD ON BOTH SOLENOID VALVES. THE CAUSE OF THE LEAKAGE HAS NOT BEEN DETERMINED. A FAILURE ANALYSIS WILL BE PERFORMED DURING THE NEXT REFUELING OUTAGE. THE VALVES WILL BE REPAIRED OR REPLACED AS NECESSARY BASED UPON THE RESULTS OF THE FAILURE ANALYSIS.

[263] SUSQUEHANNA 1 DOCKET 50-387 LER 87-003 REV 01
UPDATE ON DIESEL GENERATOR TRIP ALARM INVESTIGATION NECESSITATED DECLARING ONE DIESEL GENERATOR INOPERABLE WHEN A SECOND DIESEL GENERATOR WAS ALREADY INOPERABLE.
EVENT DATE: 012787 REPORT DATE: 022988 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)
VENDOR: RILEY COMPANY, THE - PANALARM DIVISION

(NSIC 208310) ON 1/27/87 AT 0758 HOURS WITH THE "B" DIESEL GENERATOR OUT OF SERVICE FOR MAINTENANCE, THE "A" DIESEL GENERATOR TRIP ALARM WAS RECEIVED IN THE CONTROL ROOM. NO ALARMS WERE LIT AT THE "A" DIESEL GENERATOR LOCAL PANEL; THIS PANEL WOULD NOT RESPOND TO AN ALARM TEST. THE "A" DIESEL GENERATOR WAS DECLARED INOPERABLE AT THIS TIME. AN INVESTIGATION DETERMINED THAT THE PROBLEM ON THE "A" D/G WAS CAUSED BY FAILURE OF AN INPUT CAPACITOR ON AN ALARM FLASHER CARD. IT WAS CONCLUDED THAT THE INPUT CAPACITOR HAD REACHED THE NORMAL END OF LIFE EXPECTED FOR THIS DEVICE. THIS EVENT WAS DETERMINED TO BE REPORTABLE PER 10CFR50.73 (A)(2)(V), IN THAT TWO DIESEL GENERATORS WERE DECLARED INOPERABLE, ONE BEING OUT OF SERVICE FOR MAINTENANCE AND ANOTHER PLACED IN LOCAL CONTROL TO INVESTIGATE RECEIPT OF A DIESEL GENERATOR TRIP ALARM IN THE CONTROL ROOM. THE SAFETY ANALYSIS FOR SUSQUEHANNA REQUIRES THREE OPERABLE D/G'S AND UNDER THE ABOVE CIRCUMSTANCES ONLY TWO DIESELS WERE OPERABLE. EXCEPT DURING THE BRIEF PERIOD WHEN PLACED IN LOCAL CONTROL, THE "A" D/G WOULD STILL HAVE PERFORMED ITS SAFETY FUNCTION WHICH WOULD NOT BE AFFECTED BY THE FAILED COMPONENTS. MAINTENANCE WORK WAS STOPPED ON THE "B" D/G AND IT WAS RETURNED TO OPERABILITY AT 1600 HOURS ON THE DAY OF THE EVENT.

[264] SUSQUEHANNA 1 DOCKET 50-387 LER 88-003
MIS-SCHEDULING OF SURVEILLANCE PROCEDURE RESULTS IN OPERATIONS PROHIBITED BY TECHNICAL SPECIFICATIONS.
EVENT DATE: 121087 REPORT DATE: 021088 NSSS: GE TYPE: BWR

(NSIC 208271) FOLLOWING THE MARCH, 1985, COMPLETION OF THE QUARTERLY CALIBRATION OF THE SOURCE RANGE MONITORS, INSTRUMENT AND CONTROL (I&C) PERSONNEL CHANGED THE SURVEILLANCE FREQUENCY TO SEMI-ANNUALLY. I&C BELIEVED THAT AN AMENDMENT TO THE PLANT'S TECHNICAL SPECIFICATIONS HAD BEEN APPROVED BY THE COMMISSION CHANGING THE SURVEILLANCE FREQUENCY FROM QUARTERLY TO SEMI-ANNUALLY. THEY WERE MISTAKEN. THIS ACTION RESULTED IN THE PLANT OPERATING IN A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS DURING THE FOLLOWING TIME PERIODS: FROM 02/17/86 1130 HRS TO 04/07/86 1155 HRS; FROM 09/22/86 0610 HRS TO 09/23/86 2334 HRS; FROM 07/11/87 1300 HRS TO 07/12/87 0450 HRS; AND FROM 09/14/87 2043 HRS TO 10/12/87 0945 HRS UPON DISCOVERY OF THE ERROR ON DECEMBER 10, 1987, I&C PERSONNEL CHANGED THE FREQUENCY BACK TO QUARTERLY. PLANT PERSONNEL ARE CONTINUING THE INVESTIGATION AND WILL PROVIDE ANY FURTHER CORRECTIVE ACTIONS IN AN UPDATE. A REVIEW OF THE SURVEILLANCES COMPLETED AFTER THE ABOVE TIME PERIODS (I.E., IN APRIL 1986, SEPTEMBER 1986 AND OCTOBER 1987) INDICATE THAT THE "AS FOUND" SETPOINTS WERE WITHIN TECHNICAL SPECIFICATION LIMITS. AS SUCH, DURING THESE PERIODS THE SRM CONTROL ROD BLOCK INSTRUMENTATION WAS FUNCTIONAL AND CAPABLE OF PERFORMING ITS DESIGN FUNCTION.

[265] SUSQUEHANNA 1 DOCKET 50-387 LER 88-0[illegible]
HIGH PRESSURE COOLANT INJECTION STEAM SUPPLY VALVE ISOLATION.
EVENT DATE: 011188 REPORT DATE: 021088 NSSS: GE TYPE: BWR

(NSIC 208270) AT 1655 HOURS ON JANUARY 11, 1988 WITH UNIT 1 OPERATING IN CONDITION 1 AT 100% POWER, AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION OCCURRED WHEN THE HIGH PRESSURE COOLANT INJECTION (HPCI) STEAM SUPPLY OUTBOARD ISOLATION VALVE, HV155F003, WENT CLOSED UNEXPECTEDLY. TECHNICIANS HAD SUCCESSFULLY COMPLETED THE 31 DAY EQUIPMENT ROOM CHANNELS SURVEILLANCE TEST ON EQUIPMENT AREA TEMPERATURE MODULE TSH-E41-1N600B. UPON COMPLETION, THEY HEARD A HUMMING NOISE FROM THE MODULE. AFTER THEY REMOVED A METER USED TO INVESTIGATE THIS ABNORMALITY, THE MODULE ACTUATED CAUSING THE VALVE TO CLOSE. OPERATIONS IMMEDIATELY RESET THE HPCI ISOLATION LOGIC AND RE-OPENED HV155F003. THIS EVENT WAS DETERMINED TO BE REPORTABLE PER 10CFR50.73 (A)(2)(IV), IN THAT IT COMPRISED AN UNPLANNED ESF ACTUATION. THERE WAS NO MEASURABLE HAZARD PRESENTED TO THE PLANT BY THE HPCI SYSTEM BEING UNAVAILABLE FOR ONLY A FEW MINUTES WITH THE REMAINING EMERGENCY CORE COOLING SYSTEMS AVAILABLE TO RESPOND IN THEIR DESIGNED MANNER. CORRECTIVE ACTIONS INCLUDED RESETTING THE HPCI ISOLATION LOGIC, REOPENING HV155F003, AND ULTIMATELY REPLACING THE RILEY MODULE. THIS EVENT WILL ALSO BE REVIEWED WITH INSTRUMENTATION AND CONTROLS PERSONNEL.

[266] SUSQUEHANNA 1 DOCKET 50-387 LER 88-004
REACTOR BUILDING VENTILATION BOUNDARY DOOR BLOCKED OPEN.
EVENT DATE: 020388 REPORT DATE: 030488 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)

(NSIC 208460) ON 2/3/88, WITH BOTH UNIT 1 AND UNIT 2 OPERATING AT 100% POWER, IT WAS DISCOVERED THAT UTILITY WORK GROUP PERSONNEL HAD PROPPED OPEN A DOOR APPROXIMATELY 3 INCHES TO ALLOW FOR THE PASSAGE OF A TEMPORARY POWER CABLE ON 2/2/88 AND 2/3/88. THIS DOOR IS A BOUNDARY BETWEEN REACTOR BUILDING VENTILATION ZONE I AND A NO ZONE (ESSENTIALLY OUTSIDE ATMOSPHERE), WHEN THE RAILROAD ACCESS BAY IS ALIGNED TO ZONE I. PROPPING THIS DOOR OPEN RESULTS IN AN ALIGNMENT THAT IS PROHIBITED BY THE TECH SPEC AND SECONDARY CONTAINMENT LEAKAGE RATES WHICH COULD HAVE EXCEEDED THE AUTHORIZED LIMITS IN THE EVENT OF A ZONE I AND III OR A ZONE I, II AND III ISOLATION, WITH A SINGLE COMPONENT (DAMPER) FAILURE. THIS INCIDENT WAS CAUSED BY PERSONNEL ERROR. THE WORK GROUP PERSONNEL FAILED TO OBSERVE A CAUTION TAG ON THE DOOR WHICH STATED THAT THE DOOR COULD BE OPENED FOR PERSONNEL ACCESS ONLY. TRAINING WAS PROVIDED FOR THE APPLICABLE WORK GROUP. ADDITIONALLY, THE SUBJECT WORK GROUP WILL ISSUE A PLANNING DIRECTIVE TO ENSURE THAT BOUNDARY DOORS WILL NOT BE BLOCKED OPEN DURING WORK INVOLVING EQUIPMENT TRANSFERS BETWEEN UNIT 1 AND UNIT 2. SINCE IT IS POSTULATED THAT STANDBY GAS TREATMENT SYSTEM DRAWDOWN TIMES MAY HAVE BEEN AFFECTED, FURTHER ENGINEERING EVALUATION IS BEING PERFORMED TO DETERMINE WHAT EFFECTS THIS INCIDENT MAY HAVE HAD ON OFFSITE DOSE PREDICTIONS, HAD AN ACCIDENT OCCURRED.

[267] SUSQUEHANNA 2 DOCKET 50-388 LER 87-012 REV 01
UPDATE ON AUXILIARY BOILER ARC-OVER CAUSES PRIMARY CONTAINMENT ISOLATION VALVE CLOSURE.
EVENT DATE: 102887 REPORT DATE: 030388 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 1 (BWR)

(NSIC 208399) AT 0830 ON OCTOBER 28, 1987, A SPURIOUS ENGINEERED SAFETY FEATURE ACTUATION OCCURRED ON UNIT 2. WITH UNIT 1 IN REFUELING AND UNIT 2 IN NORMAL POWER OPERATIONS AT 100% POWER, AUXILIARY BOILER "A" EXPERIENCED AN INTERNAL ELECTRICAL ARC-OVER WHICH CAUSED AN OVERCURRENT TRIP OF ITS 13.8KV SUPPLY BREAKER. THIS CAUSED A TRANSIENT ON THE STARTUP BUS 10 WHICH RESULTED IN CLOSURE OF UNIT 2 PRIMARY CONTAINMENT ISOLATION VALVES ASSOCIATED WITH THE CONTAINMENT ATMOSPHERE CONTROL SYSTEM AND VARIOUS OTHER MINOR SYSTEM PERTURBATIONS AND ALARMS ON UNIT 2 AND UNIT 1. UNIT 2 REACTOR POWER REMAINED CONSTANT THROUGHOUT THE

OCCURRENCE AND RECOVERY. ALL AFFECTED SYSTEMS WERE PROMPTLY RETURNED TO NORMAL OPERATION. ROOT CAUSE OF THE EVENT WAS COGNITIVE PERSONNEL ERROR. THE UTILITY NONLICENSED OPERATOR INVOLVED DID NOT ADHERE TO THE AUXILIARY BOILER SYSTEM OPERATING PROCEDURE. THE OPERATOR DID NOT ESTABLISH PROPER BOILER PRESSURE AND FEEDWATER CONDUCTIVITY PRIOR TO SUPPLYING HIGH VOLTAGE TO THE BOILER AS REQUIRED BY THE SYSTEM'S OPERATING PROCEDURE. ALL OPERATORS WILL RECEIVE TRAINING ON THE IMPORTANCE OF ADHERING TO THE AUXILIARY BOILER SYSTEM OPERATING PROCEDURE AS WELL AS OTHER SYSTEM OPERATING PROCEDURES. INVESTIGATION INTO THE ADEQUACY OF AUXILIARY BOILER SYSTEM DESIGN AND OPERATING PROCEDURES IS CONTINUING AMONG OPERATIONS AND ENGINEERING PERSONNEL.

[268] THREE MILE ISLAND 1 DOCKET 50-289 LER 87-008 REV 02
UPDATE ON REACTOR TRIP FROM TURBINE TRIP DUE TO HIGH MOISTURE SEPARATOR LEVEL.
EVENT DATE: 091687 REPORT DATE: 022688 NSSS: BW TYPE: PWR
VENDOR: FISHER CONTROLS CO.

(NSIC 208395) TMI-1 WAS OPERATING AT 100% POWER WITH THE INTEGRATED CONTROL SYSTEM (ICS) IN FULL AUTOMATIC NO SYSTEMS OR COMPONENTS WERE INOPERABLE THAT AFFECTED THE POST TRIP RESPONSE. AT 1704 HOURS ON 9/16/87, THE TURBINE TRIPPED ON HIGH MOISTURE SEPARATOR LEVEL FOLLOWED BY A REACTOR TRIP. THE REACTOR PROTECTION SYSTEM (RPS) FUNCTIONED AS DESIGNED. THE PLANT POST TRIP RESPONSE WAS NORMAL EXCEPT FOR A SLIGHT FEEDWATER REFEED TO THE STEAM GENERATOR. NO OTHER SAFETY SYSTEMS WERE ACTUATED. THE ROOT CAUSE WAS THE FAILURE OF THE NON-SAFETY RELATED LEVEL CONTROLLER LC77E. LC77E CONTROLS THE HIGH LEVEL DUMP VALVE MO-V-2E AND DID NOT OPEN THE VALVE FAR ENOUGH TO MAINTAIN WATER LEVEL. SIMILAR REACTOR TRIPS RESULTING FROM HIGH MOISTURE SEPARATOR LEVEL OCCURRED ON 1/4/86 AND AND 5/9/75. CORRECTIVE AND PREVENTATIVE ACTIONS BEING TAKEN ARE: HIGH LEVEL TRIP TIME DELAY RELAY WAS REPLACED; DUMP VALVE LEVEL CONTROLLER (LC77E) WAS REPAIRED; ALL 6 HIGH LEVEL TRIP SWITCHES WERE TESTED PRIOR TO STARTUP; ALL 6 HIGH LEVEL DUMP VALVES WERE TESTED PRIOR TO STARTUP; ALL 6 HIGH LEVEL ALARM SWITCHES WERE TESTED PRIOR TO STARTUP; LEVELS OF ALL 6 MOISTURE SEPARATORS WERE MONITORED DURING STARTUP; THE MOISTURE SEPARATOR DRAIN SYSTEM IS BEING EVALUATED FOR IMPROVEMENTS; AND THE PREVENTIVE MAINTENANCE PERFORMED ON THE HIGH LEVEL DUMP VALVE CONTROLLERS AND ALARM SWITCHES WILL BE IMPROVED.

[269] THREE MILE ISLAND 2 DOCKET 50-320 LER 88-001
LIFT OF HEAVY LOAD OVER THE REACTOR VESSEL ABOVE THE MAXIMUM ALLOWABLE WEIGHT.
EVENT DATE: 010688 REPORT DATE: 020488 NSSS: BW TYPE: PWR

(NSIC 208387) AT APPROXIMATELY 1520 HOURS ON JANUARY 6, 1988, THE CORE BORE MACHINE WAS REINSTALLED ON THE DEFUELING WORK PLATFORM OVER THE TMI-2 REACTOR VESSEL (SEE FIGURES 1 AND 1A). HOWEVER, THE DRILL INDEXING PLATFORM STRUCTURE (DIPS), WHICH SUPPORTS THE CORE BORE MACHINE WAS ORIENTED 180 DEGREES FROM ITS CORRECT POSITION. THIS EVOLUTION WAS BEING PERFORMED IN ACCORDANCE WITH UNIT WORK INSTRUCTION (UWI) 4730-3100-87-C1544. THE REFERENCED UWI PERMITTED A LIFT HEIGHT OF 340'-6" OVER A SPECIFIED LOAD PATH. HOWEVER, DUE TO THE MISORIENTATION OF THE DIPS, THE CORE BORE MACHINE WAS TRANSPORTED OVER A LOAD PATH WHOSE LIFT HEIGHT ELEVATION WAS LIMITED TO 339' PURSUANT TO A NRC-APPROVED SAFETY EVALUATION REPORT. PER THE ACTION STATEMENT OF TECH SPEC 3.10.1, THIS EVENT REPORT IS BEING SUBMITTED AS A SPECIAL REPORT PURSUANT TO SPEC 6.9.2. UPON DISCOVERY OF THIS EVENT ON JANUARY 7, 1988, AN INFORMAL CRITIQUE WAS HELD TO DETERMINE THE CAUSE OF THE EVENT AND THE CORRECTIVE ACTIONS REQUIRED. FOLLOWING THE DETERMINATION THAT THE DIPS HAD BEEN INVERSELY INSTALLED, THE CORE BORE MACHINE AND SUPPORT STRUCTURES WERE REMOVED AND INSTALLED IN THE CORRECT POSITION. THE REFERENCED UWI HAS BEEN REVISED TO PROVIDE GUIDANCE FOR VERIFICATION OF THE PROPER ORIENTATION OF THE DIPS PRIOR TO INSTALLATION OF THE CORE BORE MACHINE.

[270] THREE MILE ISLAND 2 DOCKET 50-320 LER 88-002
FAILURE TO COMPLY WITH TECH SPEC 3.3.3.8 BY NOT STARTING A FIREWATCH WHEN AUXILIARY BUILDING VENTILATION IS SECURED AND FIRE DETECTORS ARE INOPERABLE.
EVENT DATE: 011288 REPORT DATE: 021188 NSSS: BW TYPE: PWR

(NSIC 208386) AT 1330 HOURS ON JANUARY 7, 1988, A TMI-2 CONTROL ROOM OPERATOR (CRO) RAISED THE ISSUE OF WHETHER THE UNIT NEEDED TO BE PLACED IN THE ACTION STATEMENT FOR TECH SPEC 3.3.3.8 WHEN THE VENTILATION WAS SECURED FOR THE CONTROL ROOM. TECH SPEC ACTION STATEMENT FOR SECTION 3.3.3.8 REQUIRES A 1 HOUR FIREWATCH IF THE PRIMARY AND ALTERNATE FIRE DETECTION INSTRUMENTATION IS INOPERABLE AND RESTORATION OF THE DETECTORS TO AN OPERABLE STATUS WITHIN 14 DAYS. IT WAS DISCOVERED THAT THE CONTROL ROOM ALTERNATE FIRE DETECTION INSTRUMENTATION IS AN AREA MOUNTED INSTRUMENT AND IS NOT RENDERED INOPERABLE WHEN CONTROL ROOM VENTILATION IS SECURED. THUS, THE ACTION STATEMENT FOR TECH SPEC 3.3.3.8 DID NOT APPLY. THE CRO'S INITIAL QUESTION CONCERNING TECH SPEC 3.3.3.8 WAS APPLIED TO ALL BUILDINGS IN UNIT 2. AT 1015 HOURS ON JANUARY 12, 1988, IT WAS DETERMINED THAT THE AUXILIARY BUILDING FIRE DETECTION INSTRUMENTATION, BOTH PRIMARY AND ALTERNATE, WOULD BE INOPERABLE WHEN THE BUILDING VENTILATION WAS SECURED. THE AUXILIARY BUILDING FIRE DETECTION SYSTEM UTILIZED DUCT MOUNTED INSTRUMENTATION FOR BOTH PRIMARY AND ALTERNATE DETECTORS. REVIEW OF RECORDS HAS SHOWN THAT ON AT LEAST ONE INSTANCE IN THE PAST, THE VENTILATION HAS BEEN SECURED AND NO FIREWATCH INITIATED.

[271] THREE MILE ISLAND 2 DOCKET 50-320 LER 88-003
INOPERABILITY OF THE TMI-2 WIND SPEED INDICATOR DUE TO INCLEMENT WEATHER CONDITIONS.
EVENT DATE: 012688 REPORT DATE: 022288 NSSS: BW TYPE: PWR
VENDOR: TAYLOR INSTRUMENT COMPANIES
TELEDYNE-GEOTECH

(NSIC 208385) AT 1515 HOURS ON JANUARY 26, 1988, THE WIND SPEED RECORDER IN THE TMI-2 CONTROL ROOM WAS DECLARED INOPERABLE DUE TO INCLEMENT WEATHER CONDITIONS. THE ACCUMULATION OF SNOW IN THE 100' ELEVATION ANEMOMETER CUPS LOCATED ON THE METEOROLOGICAL TOWER IMPEDED THE OPERATION OF THE ANEMOMETER. THE TMI METEOROLOGICAL TOWER WIND SPEED INDICATOR AVERAGED APPROXIMATELY 50% LESS THAN THE NATIONAL WEATHER SERVICE (NWS) WIND SPEED INDICATION. THE NWS HAS SENSORS LOCATED AT THE HARRISBURG INTERNATIONAL AIRPORT WHICH IS IN CLOSE PROXIMITY TO TMI. THE INCLEMENT WEATHER CONDITIONS PRECLUDED THE ABILITY TO MANUALLY CLEAR THE SNOW IN THE ANEMOMETER CUPS. DUE TO THE INOPERABLE WIND SPEED INDICATOR, THE UNIT WAS PLACED INTO THE ACTION OF TECH SPEC 3.3.3.4 WHICH HAS A 8-HOUR TIMECLOCK. AT 2315 HOURS ON JANUARY 26, 1988, THE TIMECLOCK WAS EXCEEDED, THUS, THIS EVENT IS REPORTABLE PURSUANT TO 10 CFR 50.73(A)(2)(I)(B). THE WEATHER CONDITIONS IMPROVED THE NEXT DAY AND THE WIND SPEED RECORDER WAS RESTORED TO AN OPERABLE STATUS AT 1245 HOURS ON JANUARY 27, 1988. THE ROOT CAUSE OF THIS EVENT WAS ADVERSE WEATHER CONDITIONS WHICH IMPAIRED THE OPERATION OF THE ANEMOMETERS, THUS, RESULTING IN AN ABNORMAL WIND SPEED INDICATION.

[272] TROJAN DOCKET 50-344 LER 87-027 REV 01
UPDATE ON BORON INJECTION TANK RELIEF VALVE LEAK GREATER THAN FSAR ASSUMED LIMITS.
EVENT DATE: 092787 REPORT DATE: 021188 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE & GAGE CO.

(NSIC 208255) ON SEPTEMBER 27, 1987, BORON INJECTION TANK (BIT) OUTLET RELIEF VALVE PSV-8852 WAS EXHIBITING SEAT LEAKAGE OF 207 CUBIC CENTIMETERS (CC)/MINUTE (12,420 CC/HOUR). THE VALVE DISCHARGE WAS DIRECTED TO AN OPEN FLOOR DRAIN IN THE AUXILIARY BUILDING. VALVE PSV-8852 IS LOCATED IN THE EMERGENCY CORE COOLING SYSTEM FLOW PATH AND THE LEAKAGE EXCEEDED THE 1580 CC/HOUR ASSUMED IN THE FINAL SAFETY ANALYSIS REPORT FOR SYSTEMS OUTSIDE CONTAINMENT WHICH COULD CONTAIN RADIOACTIVE WATER FOLLOWING A DESIGN BASIS ACCIDENT (DBA). THIS LEAKAGE COULD

HAVE RESULTED IN THYROID DOSES TO CONTROL ROOM OPERATORS FOLLOWING A DBA EXCEEDING THOSE STATED IN THE FSAR. THE CAUSE OF THE EVENT WAS LEAKAGE PAST THE SEAT OF VALVE PSV-8852. A CONTRIBUTING CAUSE WAS THAT THE DISCHARGE OF PSV-8852 WAS DIRECTED TO AN OPEN FLOOR DRAIN INSTEAD OF A CLOSED SYSTEM. THE PLANT WAS SHUT DOWN AND PSV-8852 WAS REMOVED AND REPLACED WITH A BLIND FLANGE. A REVIEW FOR OTHER LEAK-PATHS IN SYSTEMS OUTSIDE CONTAINMENT WHICH COULD CONTAIN RADIOACTIVE WATER FOLLOWING A DBA WAS PERFORMED. NO SIMILAR LEAK PATHS WERE IDENTIFIED. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[273] TROJAN DOCKET 50-344 LER 87-034 REV 01
UPDATE ON OFFSITE POWER SOURCES NOT DEMONSTRATED OPERABLE PER TECHNICAL SPECIFICATION DUE TO PERSONNEL ERROR.
EVENT DATE: 110287 REPORT DATE: 021188 NSSS: WE TYPE: PWR

(NSIC 208290) ON NOVEMBER 2, 1987, UPON REMOVAL OF THE "A" (EDG) FROM SERVICE FOR ROUTINE MAINTENANCE, THE OFFSITE POWER SOURCES WERE NOT DEMONSTRATED OPERABLE WITHIN ONE HOUR AS REQUIRED BY TECH SPEC 3.8.1.1. THE OFFSITE POWER SOURCES WERE DEMONSTRATED OPERABLE 2.5 HOURS AFTER THE "A" EDG BECAME INOPERABLE. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THE PERSONNEL INVOLVED DID NOT COMPLY WITH THE TECH SPECS NOR PROCEDURES. THE IMMEDIATE CORRECTIVE ACTION WAS TO DEMONSTRATE OPERABILITY OF THE OFFSITE POWER SOURCES. PERIODIC OPERATING TEST (POT) 21-2, "ENGINEERED SAFETY FEATURES AND OFFSITE POWER AVAILABILITY," WAS REVIEWED AND IT WAS CONFIRMED THAT THIS POT SPECIFICALLY REQUIRES THAT THE OFFSITE POWER SOURCES BE DEMONSTRATED OPERABLE WHEN AN EDG IS REMOVED FROM SERVICE. PERSONNEL INVOLVED WERE COUNSELED ON THE REQUIREMENTS OF THE TECH SPECS AND PROCEDURES. A REVISION TO THE OPERATIONS MANUAL HAS BEEN ISSUED THAT REQUIRES CONTROL ROOM OPERATORS TO MAINTAIN A COPY OF SCHEDULED SURVEILLANCES FOR THEIR SHIFT ON THEIR DESK. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[274] TROJAN DOCKET 50-344 LER 88-001
REACTOR TRIP DUE TO FAILED OVERPOWER DELTA TEMPERATURE CHANNEL.
EVENT DATE: 010888 REPORT DATE: 020588 NSSS: WE TYPE: PWR
VENDOR: HAGAN CONTROLS

(NSIC 208194) ON JANUARY 8, 1988, THE PLAN WAS OPERATING AT 100% POWER AT NORMAL OPERATING TEMPERATURE AND PRESSURE. AT 1506, A REACTOR TRIP OCCURRED DUE TO A SPURIOUS OVERPRESSURE DELTA TEMPERATURE (OPDT) TRIP SIGNAL ON ONE CHANNEL WHICH OCCURRED WHILE ANOTHER OPDT CHANNEL WAS IN A TRIPPED STATE WHILE SURVEILLANCE TESTING. AT 1518, SOURCE RANGE NEUTRON DETECTOR N-32 WAS ENERGIZED, BUT NO INDICATION WAS RECEIVED ON THIS CHANNEL OF NUCLEAR INSTRUMENTATION. THE CAUSE OF THE OPDT CHANNEL'S SPURIOUS TRIP SIGNAL WAS FOUND TO BE A FAILED TRANSISTOR IN THE CHANNEL'S LEAD-LAG AMPLIFIER (TY-412D). THE SOURCE RANGE CHANNEL'S FAILURE TO INDICATE WAS DUE TO A REDUCTION IN CHANNEL SIGNAL STRENGTH WHICH APPARENTLY RESULTED FROM FORMATION OF AN OXIDE FILM ON THE CABLE CONNECTOR TO THE INSTRUMENT'S PRE-AMPLIFIER. THE FAULTY TRANSISTOR IN THE OPDT AMPLIFIER WAS REPLACED. THE CONNECTOR TO THE SOURCE RANGE INSTRUMENT'S PRE-AMPLIFIER WAS CLEANED. THIS EVEN HAD NO IMPACT ON PUBLIC HEALTH AND SAFETY.

[275] TURKEY POINT 3 DOCKET 50-250 LER 87-034
MANUAL REACTOR TRIP FROM 70% REACTOR POWER DUE TO LOSS OF TURBINE GENERATOR ELECTRICAL LOAD.
EVENT DATE: 122987 REPORT DATE: 012888 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 207955) ON DECEMBER 29, 1987, UNIT 3 EXPERIENCED A REACTOR TRIP FROM APPROXIMATELY 70% REACTOR POWER. THE REACTOR CONTROL OPERATOR (RCO) BEGAN A LOAD INCREASE ON UNIT 3 FROM 50% POWER TO 75% POWER AT A RAMP RATE OF APPROXIMATELY 3% PER HOUR. VARIOUS SECONDARY ALARMS CAME IN FOR TURBINE OVERSPEED PROTECTION, AND

FOR GENERATOR ANTI-MOTORING TRIP. THE RCO IMMEDIATELY NOTICED THAT THE TURBINE GENERATOR ELECTRICAL LOAD HAD DROPPED TO APPROXIMATELY 24 MEGAWATTS (MW) ELECTRICAL AND THE LOAD LIMIT ON THE TURBINE GENERATOR WAS INDICATING APPROXIMATELY 1 POUND PER SQUARE INCH (PSI) WITH NO APPRECIABLE DECREASE IN REACTOR POWER. AT THIS TIME THE PLANT SUPERVISOR - NUCLEAR (PSN) AUTHORIZED A MANUAL REACTOR TRIP WHICH WAS COMPLETED BY THE RCO. THE REACTOR TRIP OCCURRED AS DESIGNED AND THE PLANT WAS STABILIZED IN MODE 3 (HOT STANDBY). AN EVENT RESPONSE TEAM (ERT) WAS FORMED TO REVIEW THIS EVENT TO ASSIST IN DETERMINING THE ROOT CAUSE AND CORRECTIVE ACTIONS. THE ROOT CAUSE FOR THE LOSS OF TURBINE LOAD WAS STUCK CONTACTS ON THE OVERSPEED CONTROLLER 20% UNDERPOWER RELAY. THE RELAY CONTACTS WERE FREED AND CLEANED. THE INPUTS TO THE RELAY WERE SATISFACTORILY CALIBRATED. UPON COMPLETION OF THE POST TRIP REVIEW AND APPROPRIATE MAINTENANCE, THE UNIT WAS RETURNED TO SERVICE ON DECEMBER 31, 1987.

[276] TURKEY POINT 3 DOCKET 50-250 LER 88-001
TURBINE RUNBACK DUE TO DROPPED CONTROL ROD AND SUBSEQUENT MANUAL SUBCRITICAL REACTOR TRIP WHEN ADDITIONAL CONTROL RODS DROPPED INTO THE CORE.
EVENT DATE: 011388 REPORT DATE: 021288 NSSS: WE TYPE: PWR

(NSIC 208260) ON JANUARY 13, 1988, WHILE AT 100% POWER, UNIT 3 EXPERIENCED A TURBINE RUNBACK DUE TO A DROPPED CONTROL ROD ASSEMBLY. THE OPERATORS COMMENCED OPERATING PROCEDURE (OP) 1604.1, FULL LENGTH RCC - PERIODIC EXERCISE. STEP 8.4.1 WAS INITIATED TO STEP SHUTDOWN BANK A CONTROL RODS, WHEN ROD CONTROL CLUSTER (RCC) N-9 DROPPED INTO THE CORE. THIS ENERGIZED THE ROD BOTTOM BISTABLE FOR N-9 INITIATING A ROD POSITION INDICATION (RPI) TURBINE RUNBACK TO APPROXIMATELY 70% POWER. OFF NORMAL OPERATING PROCEDURES WERE CONSULTED TO STABILIZE REACTOR POWER. AN ATTEMPT WAS MADE TO RETRIEVE THE RCC WHICH WAS UNSUCCESSFUL. THE UNIT WAS SHUTDOWN TO FACILITATE REPAIRS. DURING THE SHUTDOWN SEQUENCE TWO MORE RCCS DROPPED AND A MANUAL REACTOR TRIP WAS INITIATED. THE UNIT WAS STABILIZED IN MODE 3. AN INVESTIGATION INTO THE ROOT CAUSE OF THE FIRST DROPPED CONTROL ROD DETERMINED THAT A PIN ON THE CONNECTOR FOR THE MOVABLE GRIPPER COIL ON THE REACTOR HEAD HAD BEEN PUSHED OUT OF ENGAGEMENT. THEREFORE WHEN THE RCC WAS MOVED, THE MOVABLE GRIPPER COIL DID NOT ENGAGE AND THE RCC DROPPED INTO THE CORE. THE OTHER TWO DROPPED RCCS WERE DUE TO PERSONNEL ERROR IN THAT A TECHNICIAN REMOVED THE POWER FUSES TO N-9 AND DID NOT IDENTIFY THAT THE POWER FUSE TO THE MOVABLE GRIPPER COIL WAS COMMON TO TWO OTHER RCCS.

[277] TURKEY POINT 3 DOCKET 50-250 LER 88-002
REACTOR COOLANT SYSTEM PRESSURE DECREASE DUE TO MALFUNCTIONING PRESSURIZER SPRAY VALVE CAUSES PARTIAL ACCUMULATOR DISCHARGE.
EVENT DATE: 011588 REPORT DATE: 022288 NSSS: WE TYPE: PWR
VENDOR: HAGAN CONTROLS

(NSIC 208318) ON 1/15/88, UNIT 3 WAS BEING COOLED DOWN AND DEPRESSURIZED. PRESSURIZER SPRAY VALVE PCV-3-455B WAS IDENTIFIED AS HAVING ERRATIC OPERATION ON 1/11/88, AND A PLANT WORK ORDER TO CORRECT THE PROBLEM WAS ISSUED. WITH THE REACTOR COOLANT SYSTEM (RCS) TEMPERATURE AT 400 DEGREES F AND PRESSURE AT APPROXIMATELY 950 PSIG, IT WAS DECIDED TO SLOW THE COOLDOWN RATE FROM 90 DEGREES F PER HOUR TO ABOUT 20 DEGREES PER HOUR, DUE TO THE SHORTLY UPCOMING SHIFT TURNOVER. AT 0650, AS THE COOLDOWN RATE WAS BEING DECREASED, THE PRESSURIZER LEVEL STARTED TO INCREASE. AT THIS POINT, THE REACTOR CONTROL OPERATOR (RCO) SECURED CHARGING PUMP 3A. AN ADJUSTMENT WAS MADE TO VALVE PCV-3-455B AT 0650 IN ORDER TO DECREASE RCS PRESSURE. BY 0730, SHORTLY AFTER TURNOVER, RCS PRESSURE DECREASED TO 625 PSIG AND THE ACCUMULATORS STARTED TO INJECT. UPON NOTICING THE RCS PRESSURE DROP, AN RCO IMMEDIATELY CLOSED PCV-3- 455B AND TERMINATED THE RCS PRESSURE DECREASE. APPROXIMATELY 65 GALLONS OF WATER WERE INJECTED INTO THE RCS. THE PRIMARY CAUSE OF THE EVENT WAS A DEFECTIVE CONTROLLER FOR VALVE PCV-3-455B. CONTRIBUTING CAUSES WERE THE RCO'S FAILURE TO NOTE THE DECREASING RCS PRESSURE IN TIME TO TAKE PROMPT CORRECTIVE ACTION, AND A MISCOMMUNICATION BETWEEN THE

ONCOMING AND THE OFFGOING RCO'S. THE CONTROLLER FOR VALVE PCV-3-455B WAS REPLACED.

[278] TURKEY POINT 4 DOCKET 50-251 LER 86-015 REV 01
UPDATE ON AUXILIARY FEEDWATER SYSTEM TRAIN DECLARED OUT OF SERVICE DUE TO FAILED VALVES.
EVENT DATE: 082186 REPORT DATE: 022588 NSSS: WE TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 208303) ON 8/20/86, WITH UNIT 4 IN MODE 2 TECH SPEC 3.0.1 WAS ENTERED DUE TO VALVE MOV-4-1404 FAILING TO CLOSE DURING TESTING PER PREOPERATIONAL PROCEDURE (POP) 0800.111, "AFW STEAM SUPPLY REPLACEMENT VALVE TEST-UNIT 4." THEREFORE, PER PROCEDURE 4-OP-075, "AUXILIARY FEEDWATER SYSTEM," ONE OF TWO TRAINS OF AFW TO UNIT 4 WAS NOT OPERABLE. AFTER MOV-4-1404 WAS DECLARED OOS, THE VALVE WAS RETURNED TO SERVICE PRIOR TO THE REACTOR HAVING TO BE PLACED IN HOT STANDBY. ON 8/21, WITH UNIT 4 IN MODE 2, TS 3.0.1 WAS ENTERED AGAIN BECAUSE DURING ADDITIONAL TESTING PER POP 0800.111, VALVE 10-4-383 FAILED TO SEAT PROPERLY. A COOLDOWN OF UNIT 4 TO BELOW 350 DEGREES FAHRENHEIT WAS INITIATED. FOLLOWING COOLDOWN, VALVE 10-4-383 WAS REPLACED. AFTER COMPLETION OF ADDITIONAL MAINTENANCE, UNIT 4 RETURNED TO CRITICALITY ON 8/23 AND VALVE 10-4-383 WAS SUCCESSFULLY TESTED PER POP 0800.111. VALVE MOV-4-1404 FAILED TO CLOSE DUE TO A WIRE BUNDLE BEING PRESSED UP AGAINST THE TORQUE SWITCH CLOSE CONTACT SPRING, KEEPING THE CONTACT OPEN. THE WIRE BUNDLE WAS RELOCATED AND THE VALVE WAS SUCCESSFULLY STROKED. THE CHECK VALVE WAS RETURNED TO THE MANUFACTURER AND IT WAS DETERMINED THAT THE VALVE FAILED BECAUSE OF A MISALIGNMENT OF THE DISC AND SEAT. THE OTHER CHECK VALVES THAT WERE REPLACED WERE SATISFACTORILY TESTED AS PER POP 0800.111 AND DID NOT REQUIRE ANY ADJUSTMENT.

[279] VERMONT YANKEE DOCKET 50-271 LER 87-019 REV 01
UPDATE ON MISSED SURVEILLANCES OF LPCI REACTOR VESSEL SHROUD LEVEL PERMISSIVE AND SECONDARY CONTAINMENT QUARTERLY VALVE EXERCISE DUE TO PERSONNEL ERROR.
EVENT DATE: 121587 REPORT DATE: 021288 NSSS: GE TYPE: BWR

(NSIC 208288) ON 12/15/87, DURING NORMAL OPERATION AT 99% POWER, IT WAS DISCOVERED THAT SURVEILLANCE TESTING OF THE LOW PRESSURE COOLANT INJECTION (LPCI) REACTOR VESSEL SHROUD LEVEL PERMISSIVE FOR CONTAINMENT SPRAY (EIIS IDENTIFIER = BM) HAD NOT BEEN COMPLETED AS SPECIFIED BY TECH SPEC TABLE 4.2.1. THE MISSED SURVEILLANCE TESTS ARE ATTRIBUTED TO ERRORS MADE IN UPDATING THE MONTHLY SURVEILLANCE TESTING SCHEDULES. SUBSEQUENT INSPECTION AND TESTING WAS COMPLETED AND HAD REMAINED OPERABLE AT ALL TIMES. THE SURVEILLANCE TEST PROGRAM WAS REVIEWED, AND ALL ASSOCIATED PERSONNEL HAVE BEEN INSTRUCTED ON THE IMPORTANCE OF ACCURATE REVISION PROCESSES. DURING THE SUBSEQUENT CORRECTIVE ACTION REVIEW OF THE SURVEILLANCE TEST TRACKING SYSTEM, ON JANUARY 14, 1988, A TRANSCRIPTION ERROR WAS DISCOVERED. THIS ERROR PRECIPITATED THE OMISSION OF THE SECONDARY CONTAINMENT QUARTERLY VALVE EXERCISE (EIIS IDENTIFIER = BD) TEST FOR THE FIRST QUARTER OF 1988. FURTHER INVESTIGATION REVEALED THAT THE TEST HAD BEEN INCLUDED IN THE SECOND, THIRD, AND FOURTH QUARTER SCHEDULE. THIS OMISSION LED TO THE MISSED COMPLETION OF THE TEST. UPON DISCOVERY, THE TEST WAS IMMEDIATELY COMPLETED, ON JANUARY 14, 1988, WITH NO DEFICIENCIES NOTED. FURTHER REVIEW OF THE SURVEILLANCE TEST TRACKING SYSTEM REVEALED NO OTHER PROBLEMS.

[280] VOGTLE 1 DOCKET 50-424 LER 88-001
MALFUNCTION OF A REACTOR COOLANT PUMP PROTECTION RELAY CAUSES REACTOR TRIP.
EVENT DATE: 011788 REPORT DATE: 021688 NSSS: SS TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 208358) ON JANUARY 17, 1988, AT 1902 CST, THE REACTOR TRIPPED FROM 100% OF RATED THERMAL POWER DUE TO A LOW FLOW TRIP SIGNAL. THE LOW FLOW TRIP SIGNAL

OCCURRED WHEN A NON-SAFETY RELATED SYSTEM PROTECTION RELAY, TYPE K-10, TRIPPED THE REACTOR COOLANT PUMP (RCP) #2. THE KD-10 RELAY IS USED TO PROVIDE FOR FAST CLEARING DURING STALLED ROTOR OR PROLONGED STARTING CONDITIONS AND ALSO PROVIDES REDUNDANT INSTANTANEOUS FAULT DETECTION. IT WAS DETERMINED THAT THE KD-10 RELAY (1NAB06-221) MALFUNCTIONED, SINCE A TRIP CONDITION DID NOT ACTUALLY EXIST. THE RELAY WAS REPLACED AND A CALIBRATION TEST WAS PERFORMED. A DESIGN MODIFICATION HAS BEEN INSTALLED TO ADD SUPERVISORY FAULT DETECTION IN ADDITION TO THE EXISTING KD-10 RELAY SCHEME. FURTHER TESTING FOR THE KD-10 RELAY THAT MALFUNCTIONED IS SCHEDULED TO BE COMPLETED BY MARCH 1988.

[281] VOGTLE 1 DOCKET 50-424 LER 88-002
PERSONNEL ERROR DURING DIESEL TESTING CAUSES A VIOLATION OF A TECHNICAL SPECIFICATION.
EVENT DATE: 012188 REPORT DATE: 022288 NSSS: SS TYPE: PWR

(NSIC 208359) ON JANUARY 21, 1988, PLANT EQUIPMENT OPERATORS WERE CONDUCTING MOISTURE CHECKS ON DIESEL GENERATOR (DG) 1B PER PLANT PROCEDURE 13145-1, "DIESEL GENERATORS". WHEN THE MOISTURE CHECKS ARE PERFORMED, THE DG MODE SWITCH IS PLACED IN THE "MAINTENANCE" POSITION. DURING THIS PERIOD, THE DG IS NOT AVAILABLE FOR STANDBY SERVICE AND IS CONSIDERED INOPERABLE. AT APPROXIMATELY 0935 CST, THE DG BARRING DEVICE FAILED. BY 1124 CST, IT WAS DECIDED TO POSTPONE THE MOISTURE CHECKS PENDING REPAIR OF THE BARRING DEVICE, AND THE DG WAS RETURNED TO STANDBY SERVICE. AT THIS TIME, PLANT PERSONNEL REALIZED THAT THE OPERABILITY OF THE REQUIRED A.C. OFFSITE POWER SOURCES HAD NOT BEEN DEMONSTRATED AS REQUIRED BY TECHNICAL SPECIFICATION (TS) ACTION STATEMENT 3.8.1.1.B. BY 1224 CST, OPERABILITY OF THE REQUIRED OFFSITE POWER SOURCES HAD BEEN DEMONSTRATED. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR IN FAILING TO PERFORM THE REQUIRED TS ACTION STATEMENT WITH THE DIESEL GENERATOR INOPERABLE FOR OVER 1 HOUR. THE APPROPRIATE PERSONNEL WERE COUNSELED REGARDING THE IMPORTANCE OF TS COMPLIANCE. ALSO, OTHER PERSONNEL RESPONSIBLE FOR PLANT OPERATION WERE ADVISED OF THIS EVENT AND OF THE CORRECT ACTIONS TO TAKE TO PREVENT RECURRENCE.

[282] WATERFORD 3 DOCKET 50-382 LER 87-026 REV 01
UPDATE ON CONTAINMENT ELECTRIC PENETRATION BACKUP PROTECTION INOPERABLE DUE TO INADEQUATE CONSTRUCTION DOCUMENTATION.
EVENT DATE: 112387 REPORT DATE: 022588 NSSS: CE TYPE: PWR

(NSIC 208309) AT 1500 HOURS ON 11/23/87, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% POWER WHEN OPERATIONS PERSONNEL DISCOVERED THAT CHEMICAL AND VOLUME CONTROL (CVC) VALVES 218A AND 218B HAD FAILED TO MEET THE CONTAINMENT PENETRATION BACKUP OVERCURRENT PROTECTION OPERABILITY REQUIREMENT OF TECH SPEC (TS) 3.8.4.1. AT 1415 HOURS ON 1/26/88, WHILE IN HOT STANDBY, A FOLLOWUP TO CORRECTIVE ACTIONS INITIATED FROM THE FIRST EVENT RESULTED IN THE DISCOVERY OF A CONTROL ELEMENT DRIVE MECHANISM (CEDM) COOLING VALVE LIMIT SWITCH CABLE PENETRATION WHICH DID NOT HAVE THE REQUIRED BACKUP OVERCURRENT PROTECTION. IN BOTH EVENTS, THE PLANT OPERATED IN A CONDITION PROHIBITED BY TS SINCE INITIAL PLANT STARTUP. THE ROOT CAUSE OF THESE EVENTS WAS COGNITIVE PERSONNEL ERROR BY ARCHITECT ENGINEERS. DISCREPANCIES IN DESIGN DOCUMENTATION RESULTED IN THE INCORRECT CIRCUIT CONFIGURATIONS. CVC AND CEDM CABLES WERE PROPERLY CONNECTED AND DECLARED OPERABLE WITHIN APPROXIMATELY 4 HOURS AND 24 HOURS, RESPECTIVELY. A REVIEW OF MAINTENANCE HISTORY RECORDS INDICATES THERE WERE NO EQUIPMENT PROBLEMS WHICH WOULD HAVE CHALLENGED THE OVERCURRENT PROTECTION OF THE BREAKERS. THE BREAKERS WERE PROPERLY WIRED AND OPERABLE AND THE EQUIPMENT WOULD HAVE FUNCTIONED AS DESIGNED.

[283] WATERFORD 3 DOCKET 50-382 LER 88-001
REACTOR TRIP FROM OUT-OF-RANGE ASI DUE TO INADEQUATE PROCEDURES.
EVENT DATE: 010188 REPORT DATE: 020188 NSSS: CE TYPE: PWR

(NSIC 208199) AT 1327 HOURS ON JANUARY 1, 1988, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT APPROXIMATELY 13% POWER WHEN THE REACTOR TRIPPED DUE TO OUT-OF-RANGE AXIAL SHAPE INDEX (ASI). ASI, A MEASURE OF CORE POWER DISTRIBUTION, CAUSES AN AUXILIARY TRIP OF THE CORE PROTECTION CALCULATORS (CPCS) IF ASI IS OUT OF ITS ALLOWED BAND AND POWER IS GREATER THAN APPROXIMATELY 17%. THIS CONDITION OCCURS DURING A STARTUP CONDUCTED BEFORE MOST OF THE XENON PRODUCED DURING PREVIOUS OPERATION HAS DECAYED. THE ENABLING OF THIS TRIP IS BASED ON RAW EXCORE DETECTOR SIGNALS, BUT OPERATORS WERE NOT PREVIOUSLY AWARE OF THIS FACT. OPERATORS WERE MAINTAINING POWER AT APPROXIMATELY 13 TO 15% BY COMPENSATED POWER INDICATIONS WHEN THE REACTOR TRIPPED AFTER A SHORT WITHDRAWAL OF CONTROL ELEMENT ASSEMBLIES. THIS EVENT IS REPORTABLE AS AN AUTOMATIC PROTECTIVE SYSTEM ACTUATION. THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE PROCEDURES SINCE SUFFICIENT GUIDANCE IS NOT PROVIDED FOR ASI CONTROL DURING REACTOR STARTUP. INFORMATION ON CPC MONITORED ASI AT LOW POWERS HAS BEEN DISSEMINATED. PROCEDURES HAVE BEEN REVISED SO THAT REACTOR POWER SIGNALS USED BY CPCS TO DETERMINE ASI WILL BE MONITORED, AND ALLOWABLE OPERATING MARGINS TO THE ASI AND POWER LIMITS PROVIDED. SINCE ALL PROTECTIVE FEATURES FUNCTIONED AS DESIGNED, THERE WAS NO THREAT TO THE HEALTH AND SAFETY OF THE PUBLIC OR PLANT PERSONNEL.

[284] WATERFORD 3 DOCKET 50-382 LER 88-002
REACTOR MANUALLY TRIPPED WHEN REACTOR COOLANT PUMP UPPER THRUST BEARING OVERHEATED DUE TO A CLOGGED LUBE OIL STRAINER.
EVENT DATE: 012688 REPORT DATE: 022588 NSSS: CE TYPE: PWR
VENDOR: ELLIOTT CO.

(NSIC 208343) AT 1019 HOURS ON 1/26/88 WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS AT 86% POWER WHEN THE REACTOR WAS MANUALLY TRIPPED DUE TO HIGH TEMPERATURE ON REACTOR COOLANT PUMP (RCP) 2B UPPER THRUST BEARING (UTB). THE UTB TEMPERATURE HAD BEEN SLOWLY RISING FOR 3 DAYS. AT 0730 HOURS IT WAS NOTED THAT THE TEMPERATURE WAS NEAR THE LIMIT AND A SHUTDOWN WAS ANTICIPATED LATER IN THE DAY. AT 1000 HOURS TEMPERATURE BEGAN TO INCREASE MARKEDLY AND A RAPID SHUTDOWN WAS COMMENCED. AT 1019 HOURS THE UTB TEMPERATURE EXCEEDED THE PREDETERMINED LIMIT AND THE REACTOR WAS MANUALLY TRIPPED. STEAM GENERATOR LEVELS DROPPED LOW ENOUGH TO ACTUATE THE EMERGENCY FEEDWATER (EFW) SYSTEM BUT NOT LOW ENOUGH TO INITIATE EFW FLOW. RCP 2B WAS SECURED AFTER THE REACTOR TRIP. ALL AUTOMATIC SYSTEMS AND PROTECTIVE FEATURES FUNCTIONED AS DESIGNED. THE ROOT CAUSE OF THIS EVENT WAS A CLOGGED OIL STRAINER IN THE UTB LUBE OIL SYSTEM. RCP 2B UTB OVERHEATING DUE TO A CLOGGED STRAINER HAS NOW RESULTED IN THREE FORCED OUTAGES. DURING THE RECOVERY FROM THIS EVENT THE SMALL MESH STRAINER WAS REPLACED WITH A LARGER MESH STRAINER. THERE WAS NO INDICATION OF DAMAGE TO RCP 2B OR ITS MOTOR. SINCE THE REACTOR WAS MANUALLY TRIPPED TO AVOID DAMAGE TO PLANT EQUIPMENT, THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT.

[285] WOLF CREEK 1 DOCKET 50-482 LER 87-056
INADEQUATE CONTROL RESULTS IN LOSS OF LICENSED MATERIAL.
EVENT DATE: 121887 REPORT DATE: 011588 NSSS: WE TYPE: PWR
VENDOR: MONSANTO RESEARCH CORP.

(NSIC 208291) ON DECEMBER 15, 1987, DURING THE PERFORMANCE OF AN INVENTORY OF SEALED SOURCES, ONE 400 MICROCURIE STRONTIUM-90/YTTRIUM-90 SOURCE COULD NOT BE LOCATED. FOLLOWING UNSUCCESSFUL EFFORTS TO LOCATE THE SOURCE, ON DECEMBER 18, 1987, AT APPROXIMATELY 0855 CST, THE SOURCE WAS DECLARED LOST. ADDITIONAL SEARCHES HAVE BEEN UNSUCCESSFUL IN LOCATING THE LOST SOURCE. IT IS BELIEVED THAT THE SOURCE WAS LOST DURING USE. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INADEQUATE CONTROL OF LICENSED MATERIAL, INCLUDING INADEQUATE PROCEDURES AND

FAILURE OF HEALTH PHYSICS (HP) PERSONNEL TO FOLLOW PROCEDURES. THE DESIGN OF THE FAN-SHAPED APPARATUS IN WHICH THE SOURCE WAS HOUSED, IS BELIEVED TO HAVE BEEN A CONTRIBUTORY CAUSE OF THE EVENT. THE FAN-SHAPED APPARATUSES HAVE BEEN REMOVED FROM SERVICE IN AN EFFORT TO PREVENT RECURRENCE. THE PROCEDURE GOVERNING THE CONTROL OF RADIOACTIVE MATERIALS HAS BEEN ENHANCED, REFERENCING THE PROCEDURE, "LOSS OF A RADIOACTIVE SOURCE," AND REQUIRING THE USE OF THE SOURCE ISSUE LOG. THE HP TECHNICIAN WHO FAILED TO FOLLOW THE APPROPRIATE HP PROCEDURE HAS BEEN REPRIMANDED.

[286] WOLF CREEK 1 DOCKET 50-482 LER 88-001
RADIATION MONITOR SPIKE CAUSES ENGINEERED SAFETY FEATURES ACTUATION DUE TO UNKNOWN CAUSES.
EVENT DATE: 012088 REPORT DATE: 021588 NSSS: WE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 366) AT APPROXIMATELY 1829 CST ON JANUARY 20, 1988, A FUEL BUILDING VENTILATION ISOLATION SIGNAL AND A CONTROL ROOM VENTILATION ISOLATION SIGNAL WERE RECEIVED WHEN THE GAS DETECTOR SIGNAL SPIKED HIGH ON FUEL BUILDING EXHAUST RADIATION MONITOR GG-RE-27. ALL REQUIRED ENGINEERED SAFETY FEATURES EQUIPMENT PERFORMED PROPERLY. FOLLOWING A REVIEW OF THE INFORMATION AVAILABLE, INTERVIEWS WITH PERSONNEL INVOLVED, AND AN EVENT RECREATION, A DEFINITIVE ROOT CAUSE COULD NOT BE DETERMINED. NO PROBLEMS WERE FOUND WITH MONITOR GG-RE-27 AND IT HAS EXPERIENCED NO FURTHER PROBLEMS. THEREFORE, NO CORRECTIVE ACTIONS ARE PLANNED AT THIS TIME.

[287] WOLF CREEK 1 DOCKET 50-482 LER 88-002
PROBABLE TRANSIENT IN POWER SUPPLY FOR RADIATION MONITOR CAUSES ENGINEERED SAFETY FEATURES ACTUATION.
EVENT DATE: 012488 REPORT DATE: 021888 NSSS: WE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 208367) ON JANUARY 24, 1988, AT APPROXIMATELY 0523 CST, A CONTAINMENT PURGE ISOLATION SIGNAL AND A CONTROL ROOM VENTILATION ISOLATION SIGNAL WERE RECEIVED FROM CONTAINMENT ATMOSPHERE RADIATION MONITOR GT-RE-32. ALL ENGINEERED SAFETY FEATURES EQUIPMENT REQUIRED TO OPERATE RESPONDED PROPERLY. REVIEWS OF THE INFORMATION AVAILABLE, INTERVIEWS WITH PERSONNEL INVOLVED IN THE EVENT, AND SUBSEQUENT TROUBLESHOOTING ACTIVITIES HAVE BEEN UNSUCCESSFUL IN DETERMINING A ROOT CAUSE FOR THIS EVENT. THEREFORE, NO CORRECTIVE ACTIONS ARE PLANNED AT THIS TIME.

[288] WOLF CREEK 1 DOCKET 50-482 LER 88-003
ENGINNEERED SAFETY FEATURES ACTUATION ON CONTROL ROOM VENTILATION ISOLATION SIGNAL CAUSED BY SPURIOUS SPIKE ON CHLORINE MONITOR.
EVENT DATE: 021588 REPORT DATE: 022688 NSSS: WE TYPE: PWR
VENDOR: M D A SCIENTIFIC, INC.

(NSIC 208478) ON FEBRUARY 15, 1988, AT 2130 CST, A CONTROL ROOM VENTILATION ISOLATION SIGNAL (CRVIS) OCCURRED DUE TO CHLORINE MONITOR GK-AITS-3 INDICATING HIGH CHLORINE LEVEL IN THE OUTSIDE AIR MAKEUP TO THE CONTROL BUILDING HEATING, VENTILATING AND AIR CONDITIONING SYSTEM. UPON RECEIPT OF THE CRVIS, ALL ENGINEERED SAFETY FEATURES EQUIPMENT REQUIRED TO OPERATE RESPONDED PROPERLY. DURING THIS EVENT, THE PLANT WAS IN MODE 3, HOT STANDBY, WITH THE REACTOR COOLANT SYSTEM PRESSURE AT APPROXIMATELY 2235 POUNDS PER SQUARE INCH AND A REACTOR COOLANT AVERAGE TEMPERATURE OF APPROXIMATELY 557 DEGREES FAHRENHEIT. NO CHLORINE WAS PRESENT AS EVIDENCED BY NORMAL READINGS ON THE REDUNDANT CHLORINE MONITOR. THE CONTROL BUILDING HEATING, VENTILATING AND AIR CONDITIONING SYSTEM WAS RETURNED TO A NORMAL CONFIGURATION AT 2 20 CST, FEBRUARY 15, 1988, AND THE AFFECTED MONITOR WAS PLACED IN BYPASS FOR TROUBLESHOOTING. DURING SUBSEQUENT

TROUBLESHOOTING BY INSTRUMENTATION AND CONTROLS PERSONNEL A ROOT CAUSE FOR THE SPIKE COULD NOT BE DETERMINED, AND IT IS THEREFORE CONSIDERED TO BE AN ISOLATED SPURIOUS SPIKE.

[289] WPPSS 2 DOCKET 50-397 LER 88-001
REACTOR PROTECTIVE SYSTEM AUTOMATIC ACTUATION DURING PLANT SHUTDOWN DUE TO INADEQUATE PROCEDURE.
EVENT DATE: 011888 REPORT DATE: 021588 NSSS: GE TYPE: BWR
VENDOR: BAILEY INSTRUMENT CO., INC.

(NSIC 208272) ON 1/18/88 AT 1354 HOURS, FOLLOWING A PLANT SHUTDOWN, AN AUTOMATIC RPS ACTUATION OCCURRED DUE TO AN ACTUAL LOW RPV WATER LEVEL CONDITION. WHILE THE PLANT WAS BEING MAINTAINED IN THE HOT SHUTDOWN CONDITION WITH THE MAIN STEAM ISOLATION VALVES CLOSED, A MAIN STEAM SRV WAS OPENED TO PRECLUDE APPROACHING THE AUTOMATIC REACTOR HIGH PRESSURE RPS ACTUATION SETPOINT OF 1037 PSIG. WHILE THE SRV WAS OPEN, RPV WATER LEVEL SWELL EFFECT CAUSED LEVEL TO INCREASE ABOVE THE RPV HIGH WATER LEVEL SETPOINT WHICH RESULTED IN THE AUTOMATIC SHUTDOWN OF THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM. THIS SYSTEM WAS BEING OPERATED TO SUPPLY WATER TO THE RPV AND ASSIST IN CONTROLLING RPV PRESSURE. WHEN THE RCIC SYSTEM SHUT DOWN, THE DECISION WAS MADE TO DEPRESSURIZE TO 875 PSIG WITH AN SRV. AFTER REACHING THE DESIRED PRESSURE, THE SRV WAS CLOSED, CAUSING RPV LEVEL TO DROP FROM +40 INCHES TO -5 INCHES. THE AUTOMATIC LOW RPV LEVEL RPS ACTUATION OCCURRED AS LEVEL DECREASED THROUGH THE +13 INCH SETPOINT. DURING THE EVENT, A NARROW RANGE RPV LEVEL RECORDER. PEN STUCK AT THE HIGH END OF THE SCALE FOR APPROXIMATELY ONE MINUTE. THE ROOT CAUSE OF THE EVENT WAS PROCEDURAL INADEQUACY. PLANT PROCEDURES DID NOT CONTAIN SUFFICIENT INFORMATION TO ADEQUATELY ADDRESS THE DEGREE OF DIFFICULTY INVOLVED IN CONTROLLING REACTOR PRESSURE WITH THE PLANT IN HOT SHUTDOWN.

[290] WPPSS 2 DOCKET 50-397 LER 88-002
RCIC PUMP SUCTION LINE OUTSIDE THE PLANT DESIGN BASIS DUE TO A DESIGN ERROR BY THE PLANT ARCHITECT/ENGINEER.
EVENT DATE: 012288 REPORT DATE: 021988 NSSS: GE TYPE: BWR

(NSIC 208344) ON JANUARY 22, 1988, DURING AN ENGINEERING REVIEW OF MOTOR-OPERATED VALVES, A WNP-2 PLANT DESIGN ERROR WAS DISCOVERED. THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM PUMP SUCTION LINE AS INSTALLED AND OPERATED DID NOT MEET CONTAINMENT ISOLATION AND SINGLE FAILURE DESIGN CRITERIA. IN ADDITION TO BEING REPORTABLE PER 10CFR50.73, THIS CONDITION HAS ALSO BEEN DETERMINED TO BE REPORTABLE PER 10CFR, PART 21. THE PORTION OF THE RCIC SYSTEM IN QUESTION IS SHOWN IN FIGURE 1. FAILURE OF THE NON-SEISMICALLY QUALIFIED PORTION OF THE PIPE BETWEEN THE CONDENSATE SUPPLY VALVE (RCIC-V-10) AND THE CONDENSATE STORAGE TANK (CST), COUPLED WITH AN ASSUMED SINGLE FAILURE OF RCIC-V-10 TO CLOSE WOULD RESULT IN AN UNMONITORED EFFLUENT PATH TO THE ENVIRONMENT. THUS, THE CONTAINMENT ISOLATION AND SINGLE FAILURE PROTECTION REQUIREMENTS OF 10CFR50, APPENDIX A, GENERAL DESIGN CRITERION (GDC) 54 AND STANDARD REVIEW PLAN (SRP), SECTION 6.2.4 ARE NOT SATISFIED. THE CAUSE OF THIS EVENT IS UNKNOWN. THE CONDITION DEVELOPED FROM DESIGN CHANGES MADE TO THE RCIC SYSTEM BY BURNS & ROE INC. IN 1981. IT APPEARS THAT BURNS & ROE, IN ITS REVIEW OF THE SAFETY IMPACT OF THIS DESIGN CHANGE, OVERLOOKED THE REQUIREMENTS OF DC 54 AND THE SRP.

[291] ZION 1 DOCKET 50-295 LER 88-001
MISSED BATTERY TEMPERATURE SURVEILLANCE.
EVENT DATE: 010688 REPORT DATE: 020588 NSSS: WE TYPE: PWR

(NSIC 208221) ON JANUARY 6, 1988, IT WAS DISCOVERED BY THE PLANT SURVEILLANCE COORDINATOR THAT FOR THE WEEKLY STATION BATTERY SURVEILLANCE PERFORMED ON DECEMBER 27, 1987, TEMPERATURES HAD BEEN RECORDED FOR SOME BUT NOT ALL OF THE

CELLS REQUIRED BY TECH SPECS. AT THE TIME OF THE SURVEILLANCE, TWO OF THE THREE THERMOMETERS NORMALLY IN THE BATTERY HAD BEEN REMOVED FOR QUALITY ASSURANCE CERTIFICATION. A MEMO TO THE OPERATING DEPARTMENT ON OCTOBER 23, 1987 STATED THAT FOR CELLS WITHOUT THERMOMETERS, ONE CERTIFIED THERMOMETER COULD BE USED TO MEASURE ALL REQUIRED CELLS. THE INDIVIDUAL WHO PERFORMED THE SURVEILLANCE IS NOT NORMALLY RESPONSIBLE TO PERFORM THIS SURVEILLANCE, AND WAS UNFAMILIAR WITH THE CONTENTS OF THIS MEMO, AND THEREFORE DID NOT MEASURE THE TEMPERATURE OF CELLS WHICH DID NOT HAVE THERMOMETERS. THE LICENSED SHIFT SUPERVISOR WHO APPROVED THE SURVEILLANCE DID SO BASED ON A NOTE IN THE SURVEILLANCE WHICH STATED THAT THE BATTERY IS OPERABLE IF THE VOLTAGE IS SATISFACTORY. THE SURVEILLANCE WILL BE CHANGED TO REFLECT THE ACTUAL TECH SPEC REQUIREMENTS. CERTIFIED THERMOMETERS HAVE BEEN INSTALLED IN THE REQUIRED CELLS.

[292] ZION 2 DOCKET 50-304 LER 88-001
NUCLEAR STATION OPERATOR FAILURE TO LOG DELTA I WITH NUCLEAR INSTRUMENTATION CHANNEL IN TEST.
EVENT DATE: 010888 REPORT DATE: 020588 NSSS: WE TYPE: PWR

(NSIC 208224) THE DELTA-I LOGGING PROGRAM ON THE UNIT 2 PROCESS COMPUTER WAS AUTOMATICALLY DEACTIVATED AT 2014 JANUARY 8, 1988 DURING CHANNEL FUNCTIONAL TESTS ON POWER RANGE CHANNEL 41 AND DELTA-I WAS NOT MANUALLY LOGGED UNTIL 2300. TECH SPEC 4.2.2.A.5 REQUIRES THAT DELTA-I BE MANUALLY LOGGED UNTIL THE ACKNOWLEDGED THE ALARM INDICATIONS AND BEGAN TO MANUALLY LOG DELTA-I. THE DELTA-I PROGRAM WAS RESTORED AT 2330 ON JANUARY 8, 1988, TERMINATING THE NEED FOR THE MANUAL LOG. THE MISSED SURVEILLANCE OF LOGGING DELTA-I IS A RESULT OF PERSONNEL ERROR RESULTING FROM THE NSO'S MISCONCEPTION THAT THE CHANNEL TEST PREEMPTED THE REQUIREMENT TO MANUALLY LOG DELTA-I. UNIT 2 WAS AT 99.4% POWER AND THE DELTA-I VALUES STAYED WITHIN THE OPERATING BAND THROUGHOUT THE INCIDENT.

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.

SYSTEM INDEX

SYSTEM INDEX

SYSTEM INDEX

KEYWORD INDEX

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

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

VENDOR CODE INDEX

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202 — U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1 REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)
NUREG/CR-2000, Vol. 7, No.4
ORNL/NSIC-200

2 TITLE AND SUBTITLE
Licensee Event Report (LER) Compilation for month of April 1988

3 LEAVE BLANK

4 DATE REPORT COMPLETED
MONTH: April YEAR: 1988

5 AUTHOR(S)

6 DATE REPORT ISSUED
MONTH: May YEAR: 1988

7 PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Oak Ridge National Laboratory
Nuclear Operations Analysis Center
Oak Ridge, TN 37831

8 PROJECT/TASK/WORK UNIT NUMBER

9 FIN OR GRANT NUMBER
FIN A9135

10 SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Office for Analysis and Evaluation of Operational Data
U.S. Nuclear Regulatory Commission
Washington, DC 20555

11a TYPE OF REPORT
Monthly Report

b PERIOD COVERED (Inclusive dates)
April 1988

12 SUPPLEMENTARY NOTES

13 ABSTRACT (200 words or less)

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one-month period identified on the cover of the document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting for revisions to those events occurring prior to 1984 are described in NRC Regulatory Guide 1.16 and NUREG-1061, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports. For those events occurring on and after January 1, 1984, LERs are being submitted in accordance with the revised rule contained in Title 10 Part 50.73 of the Code of Federal Regulations (10 CFR 50.73 - Licensee Event Report System) which was published in the Federal Register (Vol. 48, No. 144) on July 26, 1983. NUREG-1022, Licensee Event Report System - Description of Systems and Guidelines for Reporting, provides supporting guidance and information on the revised LER rule. The LER summaries in this report are arranged alphabetically by facility name and then chronologically by event date for each facility. Component, system, keyword, and component vendor indexes follow the summaries. Vendors are those identified by the utility when the LER form is initiated; the keywords for the component, system, and general keyword indexes are assigned by the computer using correlation tables from the Sequence Coding and Search System.

14 DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS
licensee event report
sequence coding and search system
vendors

b. IDENTIFIERS/OPEN-ENDED TERMS

15 AVAILABILITY STATEMENT
Unlimited

16 SECURITY CLASSIFICATION
(This page) Unclassified
(This report) Unclassified

17 NUMBER OF PAGES

18 PRICE

* U.S. GOVERNMENT PRINTING OFFICE:1988-202-292:80071

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

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

FIRST CLASS MAIL
POSTAGE & FEES PAID
USNRC

PERMIT No. G-67

120555078877 1 1AN1CV1NJ11M1
US NRC-OARM-ADM
DIV OF PUB SVCS
POLICY & PUB MGT BR-PDR NUREG
W-537
WASHINGTON DC 20555