MAY 27 1987

MEMORANDUM FOR: Charles E. Rossi, Director
Division of Operational Events Assessment

FROM: Wayne Lanning, Acting Chief
Events Assessment Branch
Division of Operational Events Assessment

SUBJECT: SUMMARY OF THE OPERATING REACTORS EVENTS MEETING ON MAY 12, 1987 - MEETING 87-14

On Tuesday, May 12, 1987, an Operating Reactors Events meeting (87-14) was held to brief senior managers from NRR, RES, AEOD and Regional Offices on events which occurred since our last meeting on May 5, 1987. The list of attendees is included as Enclosure 1.

The events discussed and the significant elements of these events are presented in Enclosure 2. Enclosure 3 provides a summary of those presented events that will be input to NRC's performance indicator program as significant events.

Original Signed By:

Wayne Lanning, Acting Chief
Events Assessment Branch
Division of Operational Events Assessment

Enclosures:
As stated

cc w/Encl.:
See Next Page

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5/18/87	5/18/87	5/21/87	5/27/87

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IDAR-5-1
OPERATING
EXPERIENCE

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

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Wayne Lanning, Acting Chief
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Division of Operational Events Assessment

Enclosures:
As stated

cc w/Encl.:
See Next Page

cc: T. Murley
J. Sniezek
R. Starostecki
J. Taylor
E. Jordan
E. Beckjord
W. Russell, Reg. I
J. Nelson Grace, Reg. II
B. Davis, Reg. III
R. D. Martin, Reg. IV
J. B. Martin, Reg. V
S. Varga
D. Crutchfield
R. Boger
G. Lainas
G. Holahan
F. Schroeder
L. Shao
J. Partlow
B. Grimes
F. Congel
H. Miller
E. Weiss
S. Black
T. Martin, EDO
F. Miraglia
E. Merschoff
R. Hernan
J. Norris
R. Samworth
H. Schierling
C. Trammel
S. McNeil
M. Slosson
J. Neighbors
P. Sears
G. Gears
H. Pastis

ENCLOSURE 1

LIST OF ATTENDEES

OPERATING REACTORS EVENTS BRIEFING (87-14)

MAY 12, 1987

NAME	DIVISION	NAME	DIVISION
R. Starostecki	NRR	R. Bosnak	RES
G. Holahan	NRR	F. Miraglia	NRR
A. Thadani	NRR	B. Clayton	OEDO
E. Rossi	NRR	J. Heltemes	AEOD
S. Rubin	AEOD	T. Speis	RES
C. Berlinger	NRR	L. Shao	NRR
J. Richardson	NRR	L. Spessard	AEOD
B. Grimes	NRR	T. Novak	AEOD
B. Boger	NRR	G. Murphy	OPNL
M. Slosson	NRR	R. Hernan	NRR
W. Beach	NRR	R. Scholl	NRR
D. Basdekas	RES		

OPERATING REACTORS EVENTS BRIEFING 87-14

MAY 12, 1987

ZION 1	MSIV OPENING/INADVERTENT SAFETY INJECTION
WNP-2	WATER IN STEAM LINE
DIABLO CANYON	LOSS OF RHR (UPDATE)
CALVERT CLIFFS 2	CRACKED PIPE IN SHUTDOWN COOLING SYSTEM

OTHER EVENTS OF INTEREST

INDIAN POINT 2/3	SINGLE FAILURE IN AUX FEEDWATER SYSTEM
MAINE YANKEE	POTENTIAL FAILURE OF CCW FOR EDG
BROWNS FERRY 1/3	MULTIPLE CHECK VALVE FAILURES
OCONEE	HEAT EXCHANGER FOULING (UPDATE)

ZION 1 - MSIV OPENING/INADVERTENT SAFETY INJECTION

PROBLEM FAILURE TO FOLLOW PROCEDURES RESULTED IN SUDDEN OPENING OF MSIVs AND INADVERTENT SAFETY INJECTION

CAUSE MISCOMMUNICATION

SIGNIFICANCE

- LOST AUTOMATIC/REMOTE STEAM LINE ISOLATION CAPABILITY WHILE IN HOT SHUTDOWN

DISCUSSION

- ON APRIL 30, 1987 UNIT IN HOT S/D FOR REPAIR ON TURBINE GENERATOR; MSIVs CLOSED
- MSIVs ARE HYDRAULICALLY CONTROLLED AND SOLENOIDS MUST BE ENEPGIZED TO CLOSE MSIVs
- SOLENOID CANNOT BE KEPT ENERGIZED MORE THAN 7 DAYS PER YEAR TO MAINTAIN COMPLIANCE WITH EQ CRITERIA
- WHEN MSIVs NEED TO BE CLOSED FOR A LONG TIME, LICENSEE OPENS BREAKER TO THE ELECTRIC MOTOR OPERATING THE HYDRAULIC PUMP BEFORE DEENERGIZING THE SOLENOIDS TO THE DRAIN VALVES
- DUE TO COMMUNICATION ERROR BETWEEN SHIFT CONTROL ROOM ENGINEER AND EQUIPMENT OPERATOR, THE TRIP SOLENOID FUSES WERE PULLED BEFORE OPENING BREAKER TO MOTOR
- DELTA P SIGNAL BETWEEN S/G LOOPS ACTUATED SI

FOLLOWUP

- LICENSEE HAS INSTALLED SHIELDS AND WARNING LAPELS ON FUSE BOXES

CONTACT: D. OUDINOT

WNP-2 WATER IN STEAMLINE

PROBLEM SCRAM WITH COMPLICATIONS

CAUSE OPERATOR ERROR

SIGNIFICANCE

- WATER IN STEAMLINES
- OPERATION DIFFICULTY DUE TO CONTAINMENT ISOLATION

DISCUSSION

- FAILED FUSE CAUSED MAXIMUM FEEDWATER DEMAND: FW PUMPS TRIPPED ON LOW SUCTION PRESSURE
- ANTICIPATORY MANUAL SCRAM: RV LEVEL ? INITIATED HPCS, RCIC, CONTAINMENT ISOLATION, ATWS RECIRC PUMP TRIPS, AND SERVICE WATER SYSTEM (LOAD SHEDDING)
- IMPROPER FEEDWATER ALIGNMENT CONTRIBUTED TO OVERFILL OF RV (+17, 34, 169 MINS AFTER SCRAM)
- HPCS & RCIC ISOLATED ON HIGH LEVEL; PRESSURE/WATER RELIEF THRU SRVS
- OPERATORS PREOCCUPIED WITH REESTABLISHING CONTAINMENT COOLING
- STEAMLINES FLOODED PREVIOUSLY (LER 86-25) BY CONDENSATE BOOSTER PUMPS
- A-47 EVALUATED OVERFILL

FOLLOWUP

- ISOLATION OF CONTAINMENT COOLING FUNCTION
- ISOLATION OF FEEDWATER SYSTEM ON HIGH RV LEVEL
- REPORTING REQUIREMENTS (ESF ACTUATION)
- OPERATOR TRAINING/PROCEDURES
- HYDRODYNAMIC EFFECTS (WATER HAMMER, WATER IN SRV, THERMAL EFFECTS)

DIABLO CANYON LOSS OF RHR

FOLLOW UP ACTIONS

- OWNERS GROUP REGULATORY RESPONSE GROUPS (3) HAVE BEEN INFORMED OF THE EVENT AND NRC CONCERNS. RESPONSES DUE 5/11/87

- PLANTS PLANNING TO SHUTDOWN DURING THE NEXT SIX (6) WEEKS HAVE BEEN IDENTIFIED

- PROJECT MANAGERS WILL BE BRIEFED AND THEY WILL BRIEF RESIDENT INSPECTORS AND LICENSEES DURING WEEK OF 5/11/87

- DOG AND PONY SHOWS IN REGIONS (EXCEPT REGION V), REGIONS I & III WITHIN 2 WEEKS

- INFORMATION NOTICE BEING PREPARED FOR 5/15/87. TARGET - ISSUE BY 5/29/87

- A 50.54(F) LETTER BEING PREPARED FOR ISSUANCE 5/22/87

CALVERT CLIFFS 2 - CRACKED PIPE IN SHUTDOWN COOLING SYSTEM

PROBLEM LEAK FROM A RELIEF LINE IN THE SHUTDOWN COOLING SYSTEM (SDCS)

CAUSE CRACKED PIPE

SIGNIFICANCE THE LEAK AFFECTS BOTH TRAINS OF THE SDCS

DISCUSSION

- THE UNIT HAS BEEN SHUT DOWN SINCE 3/14
- THE LEAK RATE WAS 0.3 GPM
- A LEAK HAD OCCURRED AT THE SAME LOCATION EARLIER IN THE PRESENT OUTAGE, WAS REPAIRED, AND A SUPPORT HAD BEEN ADDED TO REDUCE VIBRATION
- TO REPAIR THE LEAK, DECAY HEAT REMOVAL WAS PROVIDED BY ALIGNING A CONTAINMENT SPRAY PUMP THROUGH A SHUTDOWN HEAT EXCHANGER TO THE SUCTION SIDE OF AN HPSI PUMP
- THE CRACK APPARENTLY STARTED IN THE HEAT AFFECTED ZONE OF A WELD, WAS DIRECTED AT A 45 DEGREE ANGLE FROM THE WELD, AND EXTENDED 1 INCH
- THE PIPE MATERIAL IS 1/2 INCH B-16 SA 182

FOLLOWUP

- EAB WILL OBTAIN ADDITIONAL INFORMATION

INDIAN POINT 2 & 3 - POTENTIAL SINGLE FAILURE IN AUXILIARY FEEDWATER SYSTEM

PROBLEM POTENTIAL SINGLE FAILURE IN THE INDIAN POINT 2 & 3 (IP2, IP3) AUXILIARY FEEDWATER (AFW) SYSTEMS CAUSES LOSS OF BOTH MOTOR DRIVEN AFW PUMPS (MDAFWP).

CAUSE RELAY FAILURE PREVENTS AUTO START OF MDAFWP

SIGNIFICANCE

- UNANALYZED SCENARIO PLACES UNITS OUTSIDE FSAR ACCIDENT ANALYSES
- POSSIBLE GENERIC IMPLICATIONS

DISCUSSION

- QA AUDIT OF MAIN FEEDWATER (MFW) AND AFW SYSTEMS AT IP2 SURFACED POTENTIAL SINGLE FAILURE.
- TWO SERIES REDUNDANT RELAYS, ENERGIZED TO OPEN ON SI, BLOCK AUTO START SIGNAL TO BOTH AFW MOTOR DRIVEN PUMPS FACILITATING THEIR LOADING BY SAFEGUARDS SEQUENCER. THESE RELAYS ARE PART OF AFW SYSTEM'S ORIGINAL DESIGN.
- FAILURE OF EITHER RELAY (REQUIRING CONTINUOUS HOT SHORT) WOULD INHIBIT A VALID AUTO START SIGNAL TO BOTH PUMPS ON LOW LOW STEAM GENERATOR LEVEL OR LOSS OF MFW.
- FAILURE WOULD NOT AFFECT MANUAL START OF PUMP MOTORS.
- AFW TURBINE DRIVEN PUMP NOT AFFECTED. (IP2 FSAR ACCIDENT ANALYSIS TAKES NO CREDIT FOR TURBINE DRIVEN PUMP).
- SIMILAR, THOUGH NOT IDENTICAL PROBLEM, IDENTIFIED AT IP3. IF BFP RELAY FAILS, HALF OF SIGNAL REQUIRED TO START BOTH AFW PUMPS ON SG LOW LOW LEVEL OR LOSS OF MFW WILL NOT DEVELOP. MANUAL START AND INITIATION ON SI SIGNAL WILL NOT BE IMPACTED.
- LATEST INFORMATION INDICATES IP3 MAY NOT ALTER AFW LOGIC DESIGN. LICENSEE IS EVALUATING.

FOLLOWUP

- EAB PRELIMINARY SURVEY HAS NOT IDENTIFIED OTHER PLANTS WITH PROBLEM.
- IP2 LICENSEE HAS MADE CIRCUIT MODIFICATIONS.
- RECOMMEND FOLLOWUP FOR IP3.
- REGION I HAS LEAD RESPONSIBILITY FOR REVIEW AND HAS PREPARED A DRAFT INFORMATION NOTICE FOR HQ REVIEW.
- WESTINGHOUSE IS LOOKING AT REPORTABILITY UNDER PART 21, 50.59, AND 50.55E.

MAINE YANKEE - POTENTIAL FAILURE OF CCW TO EMERGENCY DIESEL GENERATORS (EDGs)

PROBLEM A POTENTIAL BREAK IN A NON-SAFETY GRADE FIRE WATER LINE COUPLED WITH LOSS OF OFFSITE POWER COULD CAUSE LOSS OF EDG COOLING.

CAUSE SINGLE AIR OPERATED VALVES WHICH FAIL OPEN ON LOSS OF AIR SEPARATE CCW FROM THE FIRE WATER LINES.

SIGNIFICANCE POTENTIAL FOR LOSS OF EMERGENCY AC POWER.

DISCUSSION

- LICENSEE DISCOVERED A POTENTIAL FAILURE MODE FOR BOTH TRAINS OF CCW TO THE EDG COOLERS. EITHER THE CCW OR THE FIRE WATER COOLING (FWC) SYSTEM CAN SUPPLY THE COOLERS. THE FWC SYSTEM IS NORMALLY ISOLATED FROM THE COOLERS BY AIR OPERATED VALVES WHICH FAIL OPEN. LOSS OF SERVICE AIR COUPLED WITH A BREAK IN THE FWC PIPING COULD CAUSE ALL CCW TO BOTH EDG COOLERS TO BE LOST.
- A RELATED PROBLEM WAS DISCOVERED AT MAINE YANKEE ON JUNE 25, 1985 AS PART OF A SAFETY SYSTEM DESIGN REVIEW.
- A SIMILAR PROBLEM WAS DISCOVERED FOR HADDAM NECK (VALVES BLOCKED OPEN PLUS PRA STUDY) NOVEMBER 1, 1985.
- A DESIGN DEFICIENCY IN EMERGENCY DIESEL OPERATION WAS DISCOVERED AT H.B. ROBINSON (EVENT 8550) MAY 1, 1987.

FOLLOWUP

- THE LICENSEE IS EVALUATING PERMANENT CORRECTIVE ACTION TO BE COMPLETED BEFORE STARTUP (BLOCKING FWC ISOLATION VALVES CLOSED).
- NRR ACTIONS RESULTING FROM AEOD CASE STUDY (C701)
  - LETTER TO RES REQUESTING REVISION TO GI-43 AND REPRIORITIZATION
  - INFORMATION NOTICE TO ENSURE BROAD DISSEMINATION OF CASE STUDY FINDINGS

BROWNS FERRY - 1 AND 3
MULTIPLE CHECK VALVE FAILURES IN EECWS

PROBLEM 24 CHECK VALVES FAILED IN OPEN POSITION IN THE EMERGENCY EQUIPMENT COOLING WATER SYSTEM

CAUSE CORROSION OF CARBON STEEL CHECK VALVES

SIGNIFICANCE

- COULD AFFECT LPCS ROOM COOLERS FOR UNITS 1 AND 3, RHR SEAL AND RHR ROOM COOLERS FOR UNIT 3 DURING DBA
- CHECK VALVE FAILURES CAUSE LOSS OF SEPARABILITY OF EECWS TRAINS
- UNKNOWN HOW LONG CONDITION EXISTED

DISCUSSION

- EECW SYSTEM SERVICES EDGs, RHR, LPCS, CONTROL ROOM - AC, OTHERS
- EECWS KEPT RUNNING DURING NORMAL OPERATION BUT ONLY REQUIRED TO OPERATE DURING DBA
- EECWS COMPRISED OF TWO "INDEPENDENT" LOOPS WHICH FEED COMMON SUPPLY HEADERS TO EQUIPMENT AND ROOM COOLERS
- CHECK VALVES PROVIDED TO PREVENT CROSS FLOW FROM ONE LOOP TO THE OTHER
- DURING FIRST TIME SURVEILLANCE TO MEET ASME SECTION XI WHICH REQUIRES INTERNAL INSPECTION, 24 CHECK VALVES FOUND FAILED OPEN (8 VALVES AFFECT UNIT 1 SERVICE, 16 VALVES AFFECT UNIT 3 SERVICE)
- FAILURE OF EECWS PUMPS IN ONE TRAIN COMBINED WITH CHECK VALVES FAILED OPEN COULD CAUSE LOSS OF COOLING TO AFFECTED EQUIPMENT

FOLLOWUP

- LICENSEE PLANS TO INSTALL STAINLESS STEEL VALVES AS REPLACEMENTS
- LICENSEE WILL INVESTIGATE THE POSSIBILITY OF USING EXISTING TEST DATA TO CHECK FOR REVERSE FLOW THROUGH CHECK VALVES
- LICENSEE IS EVALUATING OTHER SYSTEM FOR SIMILAR PROBLEM(S)

OCONEE HEAT EXCHANGER FOULING

PROBLEM: REDUCED HEAT TRANSFER CAPABILITY IN LOW PRESSURE SERVICE WATER, LOW PRESSURE INJECTION AND REACTOR BUILDING COOLING SYSTEMS

CAUSE: LAKE SEDIMENT DEPOSITED IN HEAT EXCHANGERS

SIGNIFICANCE: SAFETY EQUIPMENT INCAPABLE OF PERFORMING DESIGN FUNCTION

REQUESTED ACTIONS:

- DETERMINE WHAT OTHER PLANTS ARE DOING ABOUT HEAT EXCHANGER FOULING
- WHAT ACTIONS HAS INPO TAKEN
- WHAT ACTIONS HAS NRC STAFF TAKEN
  - A) GENERIC COMMUNICATIONS
  - B) TEMPORARY INSTRUCTIONS

FINDINGS:

- STAFF ISSUED BULLETIN 81-03
- STAFF ISSUED IN 81-21 AND IN 86-96
- INPO ISSUED SOER 84-01, RECOMMENDED
  - A) ONE (1) DESIGN CHANGE

B) THREE (3) PROCEDURAL CHANGES

C) ONE (1) INSPECTION

- STAFF ISSUED TEMPORARY INSTRUCTION 2515/77 REGARDING INDUSTPY RESPONSE TO INPO RECOMMENDATIONS IN SOER 84-01

- HEAT EXCHANGER FOULING IS A SUBJECT OF GI-51; TO BE CONCLUDED BY PES WITHIN SIX (6) MONTHS

- MEMO FROM R BAER TO E JORDAN, 3/2/87 REPORTED THAT 50% OF PLANTS HAVE IMPLEMENTED INPO RECOMMENDATIONS (TI 2515/77 RESPONSE)

- INPO SENT EVALUATORS TO 68 SITES (95 PLANTS) 85-90% HAVE SATISFACTORILY IMPLEMENTED RECOMMENDATIONS

- INPO DESIGN CHANGE - INSTALLATION OF INSTRUMENTATION TO MONITOR ADEQUACY OF HEAT TRANSFER CAPABILITY (FLOW AND TEMPERATURE, HEAT BALANCE)

- ONLY SEVEN (7) PLANTS REMAIN TO IMPLEMENT DESIGN MODIFICATION

CONCLUSIONS

1) NEED FOR HEAT BALANCE MEASUREMENT RATHER THAN JUST FLOW WELL UNDERSTOOD

2) NO FURTHER GENERIC COMMUNICATION NEEDED AT THIS TIME

3) WHEN GI-51 CONCLUDED RECONSIDER NEED FOR FURTHER ACTION

VOGTLE 1 SCRAM SUMMARY - 3/19/87 THROUGH 5/6/87

DATE	EVENT	POWER	CAUSE	REMARKS	NOTES
87/03/19	08099	3%	FLUX RATE	GAIN IMPROPERLY SET	
87/03/20	08114	3%	S6 LEVEL	LO LO - UNDERFED BY OPERATOR	
87/03/21	08127	3%	S6 LEVEL	LO LO - OSCILLATION CAUSED BY STEAM DUMPS	1
87/03/23	08138	4%	S6 LEVEL	LO LO - AFW CONTROL ERROR BY OPERATOR	2
87/03/24	08146	7%	S6 LEVEL	LO LO - MALFUNCTION OF STEAM DUMP?	3
87/03/27	08178	11%	S6 LEVEL	LO LO - HOT WELL LEVEL INSTRUMENT FAILURE	4
87/04/05	08278	25%	S6 LEVEL	HI HI/LO LO - CAUSED BY TURBINE TRIP	5
87/04/10	08338	30%	S6 LEVEL	HI HI/LO LO - CAUSED BY STUCK CHECK VALVE	6
87/04/11	08345	21%	S6 LEVEL	HI HI/LO LO - TRANSFER BYPASS TO MAIN FEED REG	7
87/04/29	08524	75%	OPDT/OTDT	2/4 LOGIC - 1 TRIP FOR SURV, 1 TRIP UNKNOWN	8
87/05/04	08572	76%	OPDT/OTDT	2/4 LOGIC - 1 TRIP FOR SURV, 1 TRIP UNKNOWN	8
87/05/09	08618	80%	S6 LEVEL	LO LO - OVERSHOOT ON LOAD DEMAND TEST	9

1. STEAM DUMP VALVES OSCILLATED IN AUTOMATIC AND WERE PLACED IN MANUAL. SUBSEQUENT OPERATOR ACTIONS CAUSED LEVEL OSCILLATIONS.
2. MANUAL CONTROL BY OPERATOR WAS NOT ADEQUATE.
3. APPARENT MALFUNCTION OF STEAM DUMP VALVE.
4. INSTRUMENT FAILED HIGH. WATER WAS DIVERTED FROM HOT WELL TO CONDENSATE STORAGE TANK. CONDENSATE PUMPS CAVITATED. MAIN FEED PUMP TRIPPED.
5. TURBINE TRIP CAUSED STEAM GENERATOR TO FILL TO HI HI, WHICH CAUSED MAIN FEED PUMP TRIP.
6. WITH ONE MAIN FEED OPERATING, DISCHARGE VALVE FOR OTHER PUMP WAS OPENED. THE CHECK VALVE FOR THIS PUMP WAS STUCK OPEN, DIVERTING FLOW FROM THE STEAM GENERATORS AND CAUSING A LOW LEVEL ALARM. OPERATOR INCREASED FLOW AND HI HI LEVEL TRIPPED PUMP.
7. MAIN FEED REG VALVE MALFUNCTIONED.
8. 1 BISTABLE TRIPPED FOR SURVEILLANCE, 1 TRIPPED FOR UNKNOWN CAUSE.
9. LOAD DEMAND TEST WAS INTENDED TO BE 65% TO 75%. IT WAS ACTUALLY 65% TO 80%.

REACTOR SCRAM SUMMARY
WEEK ENDING 05/10/87

I. PLANT SPECIFIC DATA

DATE	SITE	UNIT	POWER	RPS	CAUSE	COMPLI-CATIONS	YTD ABOVE 15%	YTD BELOW 15%	YTD TOTAL
05/04/87	BYRON	2	88	A	EQUIP/ELECT	NO	3	3	6
05/04/87	VOGTLE	1	76	A	UNKNOWN	NO	5	6	11
05/06/87	CATAWBA	2	100	M	EQUIPMENT	NO	4	0	4
05/06/87	CLINTON	1	17	M	EQUIPMENT	NO	1	1	2
05/07/87	MILLSTONE	3	100	A	EQUIPMENT	NO	4	2	6
05/08/87	CATAWBA	2	65	M	EQUIPMENT	YES	5	0	5
05/09/87	VOGTLE	1	80	A	EQUIPMENT	NO	6	6	12

SUMMARY OF COMPLICATIONS

SITE	UNIT	COMPLICATIONS
CATAWBA	2	B SG SAMPLE VALVE FAILED TO FULLY CLOSE ON ISOLATION SIGNAL

II. COMPARISON OF WEEKLY STATISTICS WITH INDUSTRY AVERAGES

SCRAMS FOR WEEK ENDING
05/10/87

SCRAM CAUSE	POWER	NUMBER OF SCRAMS (5)	1987 WEEKLY AVERAGE YTD	1986 WEEKLY AVERAGE (3)(4)	1985 WEEKLY AVERAGE (8)(9)
** POWER >15%					
EQUIP. RELATED	>15%	6	3.8	4.3	5.4
PERS. RELATED(6)	>15%	0	1.6	1.8	2.0
OTHER(7)	>15%	1	1.2	0.4	0.6
** Subtotal **		7	6.6	6.5	8.0
** POWER <15%					
EQUIP. RELATED	<15%	0	1.1	1.4	1.3
PERS. RELATED	<15%	0	0.8	0.8	0.9
OTHER	<15%	0	0.4	0.2	0.2
** Subtotal **		0	2.3	2.4	2.4
*** Total ***		7	8.9	8.9	10.4

MANUAL VS AUTO SCRAMS

TYPE	NUMBER OF SCRAMS	1987 WEEKLY AVERAGE YTD	1986 WEEKLY AVERAGE	1985 WEEKLY AVERAGE
MANUAL SCRAMS	3	1.6	1.0	1.0
AUTOMATIC SCRAMS	4	7.3	7.9	9.4

NOTES

1. PLANT SPECIFIC DATA BASED ON INITIAL REVIEW OF 50.72 REPORTS FOR THE WEEK OF INTEREST. PERIOD IS MIDNIGHT SUNDAY THROUGH MIDNIGHT SUNDAY SCRAMS ARE DEFINED AS REACTOR PROTECTIVE ACTUATIONS WHICH RESULT IN ROD MOTION, AND EXCLUDE PLANNED TESTS OR SCRAMS AS PART OF PLANNED SHUTDOWN IN ACCORDANCE WITH A PLANT PROCEDURE.

2. RECOVERY COMPLICATED BY EQUIPMENT FAILURES OR PERSONNEL ERRORS UNRELATED TO CAUSE OF SCRAM.

3. 1986 INFORMATION DERIVED FROM ORAS STUDY OF UNPLANNED REACTOR TRIPS IN 1986. WEEKLY DATA DETERMINED BY TAKING TOTAL TRIPS IN A GIVEN CATEGORY AND DIVIDING BY 52 WEEKS/YEAR.

4. IN 1986, THERE WERE AN ESTIMATED TOTAL OF 461 AUTOMATIC AND MANUAL UNPLANNED REACTOR TRIPS AT 104 REACTORS (HOLDING OPERATING LICENSES). THIS YIELDS AN AVERAGE RATE OF 4.4 TRIPS PER REACTOR PER YEAR AND AN AVERAGE RATE OF 8.8 TRIPS PER WEEK FOR ALL REACTORS.

5. BASED ON 107 REACTORS HOLDING AN OPERATING LICENSE.

6. PERSONNEL RELATED PROBLEMS INCLUDE HUMAN ERROR, PROCEDURAL DEFICIENCIES, AND MANUAL STEAM GENERATOR LEVEL CONTROL PROBLEMS.

7. "OTHER" INCLUDES AUTOMATIC SCRAMS ATTRIBUTED TO ENVIRONMENTAL CAUSES (LIGHTNING), SYSTEM DESIGN, OR UNKNOWN CAUSE.

8. 1985 INFORMATION DERIVED FROM AN ORAS STUDY OF UNPLANNED REACTOR TRIPS IN 1985. WEEKLY DATA DETERMINED BY TAKING TOTAL TRIPS IN A GIVEN CATEGORY AND DIVIDING BY 52 WEEKS/YEAR.

9. IN 1985, THERE WERE AN ESTIMATED TOTAL OF 541 AUTOMATIC AND MANUAL UNPLANNED REACTOR TRIPS AT 93 REACTORS (HOLDING FULL POWER LICENSES). THIS YIELDS AN AVERAGE RATE OF 5.8 TRIPS PER REACTOR YEAR AND AN AVERAGE RATE OF 10.4 TRIPS PER WEEK FOR ALL REACTORS.

ENCLOSURE 3

SIGNIFICANT EVENTS FREQUENCY PERFORMANCE INDICATOR No. 3

PLANT NAME	EVENT DATE	EVENT DESCRIPTION	QTR CAUSE
CALVERT CLIFFS 2	05/07/87	CRACK DISCOVERED IN LPSI PIPING, SAME LOCATION AS RECENTLY REPAIRED CRACK . BOTH TRAINS OF S/D COOLING LOST	2 MAINTENANCE
MNP 2	03/22/87	MANUAL SCRAM COMPLICATED BY WATER IN STEAM LINES	1 PERSONNEL ERROR
ZION 1	04/30/87	FAILURE TO FOLLOW PROCEDURES RESULTED IN OPENING OF MSIVs AND INADVERTENT SAFETY INJECTION	1 PERSONNEL ERROR