March 7, 1994

MEMORANDUM FOR: Brian K. Grimes, Director
Division of Operating Reactor Support

FROM: Alfred E. Chaffee, Chief
Events Assessment Branch
Division of Operating Reactor Support

SUBJECT: OPERATING REACTORS EVENTS BRIEFING
MARCH 2, 1994 - BRIEFING 94-09

On March 2, 1994, we conducted an Operating Reactors Events Briefing (94-09) to inform senior managers from offices of the Commission, AEOD, EDO, NRR, NMSS, and regional offices of selected events that occurred since our last briefing on February 23, 1994. Enclosure 1 lists the attendees. Enclosure 2 presents the significant elements of the discussed events.

Enclosure 3 contains reactor scram statistics for weeks ending February 20, 1994 and February 27, 1994. No significant events were identified for input into the NRC Performance Indicator Program.

[original signed by]

Alfred E. Chaffee, Chief
Events Assessment Branch
Division of Operating
Reactor Support

TO: NUDOCS

Enclosures: As stated

cc w/enclosures:
See next page

9403170244 940307
PDR ORG NRRB
PDR

DISTRIBUTION:
Central Files KGray
PDR JCarter
LKilgore, SECY NFields
EAB R/F

EAB/DORS	EAB/DORS	EAB/DORS	EAB/DORS
KGray	JCarter	NFields	EGoodwin
03/04/94	03/4/94	03/4/94	03/2/94
EAB/DORS	EAB/DORS		
RDennig	AChaffee		
03/7/94	03/7/94		

003020

OFFICIAL RECORD COPY
DOCUMENT NAME: G:\KAG\ORTRANS

O+M-6-Meetings
x SBR-5 Facility License

150083

RETURN TO REGULATORY CENTRAL FILES

cc:

W. Russell, NRR (12G18)
F. Miraglia, NRR (12G18)
F. Gillespie, NRR (12G18)
Acting ADPR, NRR (12G18)
S. Varga, NRR (14E4)
J. Calvo, NRR (14A4)
G. Lainas, NRR (14H3)
J. Roe, NRR (13E4)
J. Zwolinski, NRR (13H24)
E. Adensam, NRR (13E4)
A. Thadani, NRR (12G18)
M. Hodges (Acting), NRR (7D26)
M. Virgilio, NRR (8E2)
S. Rosenberg, NRR (10E4)
C. Rossi, NRR (9A2)
B. Boger, NRR (10H3)
F. Congel, NRR (10E2)
D. Crutchfield, NRR (11H21)
W. Travers, NRR (11B19)
D. Coe, ACRS (P-315)
E. Jordan, AEOD (MN-3701)
G. Holahan, AEOD (MN-9112)
L. Spessard, AEOD (MN-3701)
K. Brockman, AEOD (MN-3206)
S. Rubin, AEOD (MN-5219)
M. Harper, AEOD (MN-9112)
W. Bateman, EDO (17G21)
F. Ingram, PA (2G5)
E. Beckjord, RES (NLS-007)
A. Bates, SECY (16G15)
T. Martin, Region I
R. Cooper, Region I
S. Ebneter, Region II
E. Merschoff, Region II
S. Vias, Region II
J. Martin, Region III
E. Greenman, Region III
L. Callan, Region IV
A. Beach, Region IV
K. Perkins, Region V
S. Richards, Region V
P. Erickson (ONDD)
S. Weiss (ONDD)
A. DeAgazio (PDI-4)
J. Stolz (PDI-4)

bcc: Mr. Sam Newton, Manager
Fvents Analysis Department
Institute of Nuclear Power Operations
700 Galleria Parkway
Atlanta, GA 30339-5957

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 7, 1994

MEMORANDUM FOR: Brian K. Grimes, Director
Division of Operating Reactor Support

FROM: Alfred E. Chaffee, Chief
Events Assessment Branch
Division of Operating Reactor Support

SUBJECT: OPERATING REACTORS EVENTS BRIEFING
MARCH 2, 1994 - BRIEFING 94-09

On March 2, 1994, we conducted an Operating Reactors Events Briefing (94-09) to inform senior managers from offices of the Commission, AEOD, EDO, NRR, NMSS, and regional offices of selected events that occurred since our last briefing on February 23, 1994. Enclosure 1 lists the attendees. Enclosure 2 presents the significant elements of the discussed events.

Enclosure 3 contains reactor scram statistics for weeks ending February 20, 1994 and February 27, 1994. No significant events were identified for input into the NRC Performance Indicator Program.

Alfred E. Chaffee

Alfred E. Chaffee, Chief
Events Assessment Branch
Division of Operating
Reactor Support

Enclosures: As stated

cc w/enclosures:
See next page

ENCLOSURE 1

LIST OF ATTENDEES

OPERATING REACTORS EVENTS FULL BRIEFING (94-09)

MARCH 2, 1994

NAME	OFFICE	NAME	OFFICE
A. CHAFFEE	NRR	P. ERICKSON	NRR
J. CARTER	NRR	R. JONES	NRR
N. FIELDS	NRR	C. GRIMES	NRR
R. DENNIG	NRR	F. MIRAGLIA	NRR
T. KOSHY	NRR	L. REYES	NRR
T. YAMADA	NRR	C. ROSSI	NRR
A. BYRDSONG	NRR	C. THOMAS	NRR
J. STOLZ	NRR	S. BROWN	NMSS
S. ROSENBERG	NRR	L. BELL	NMSS
M. CARUSO	NRR	J. AUSTIN	NMSS
S. VARGA	NRR	J. GREEVES	NMSS
T. DUNNING	NRR	A. VIETTI-COOK	OCM/IS
D. O'NEAL	NRR	B. HOLIAN	OEDO
R. DUDLEY	NRR	G. HOLAHAN	AEOD

TELEPHONE ATTENDANCE
(AT ROLL CALL)

Regions
Region I
Region II
Region III
Region IV
Region V

Resident Inspectors

IIT/AIT Team Leaders

Misc.
J. McCormick-Barger

ENCLOSURE 2

NOTE -- ROOM 8 B11

OPERATING REACTORS EVENTS BRIEFING 94-09

LOCATION: 8 B11, WHITE FLINT
WEDNESDAY, MARCH 2, 1994 11:00 A.M.

DRESDEN, UNIT 1	COLD WEATHER IMPACT ON DECOMMISSIONED REACTOR (UPDATE)
SEABROOK, UNIT 1	GENERIC ISSUE - POTENTIAL OPERATION WITH INADEQUATE SECONDARY RELIEF CAPACITY

PRESENTED BY: EVENTS ASSESSMENT BRANCH
DIVISION OF OPERATING REACTOR
SUPPORT, NRR

DRESDEN, UNIT 1
COLD WEATHER IMPACT ON
DECOMMISSIONED REACTOR (UPDATE)
JANUARY 24, 1994

PROBLEM
APPROXIMATELY 55,000 GALLONS OF WATER WAS SPILLED FROM BREAKS IN SERVICE WATER SYSTEM TO UNHEATED CONTAINMENT. FUEL POOL TRANSFER SYSTEM WAS ALSO LOCATED INSIDE CONTAINMENT AND SUSCEPTIBLE TO FREEZING.

CAUSE
INADEQUATE LICENSEE REVIEW PERFORMED PRIOR TO REMOVING HEAT FROM CONTAINMENT AND EXISTENCE OF SEVERE COLD WEATHER WHICH RESULTED FREEZING OF WATER FILLED PIPING AND FAILURE OF COMPONENTS.

SAFETY SIGNIFICANCE
FAILURE OF 42" FUEL TRANSFER TUBE COULD RAPIDLY DRAIN FUEL POOL TO A LEVEL SEVERAL FEET BELOW TOP OF STORED FUEL BUNDLES. DOSE RATES AT FUEL POOL RAIL WOULD HAVE BEEN ABOUT 733 REM/HR AND SIGNIFICANT SCATTER DOSE WITHIN SITE BOUNDARY.

LICENSEE IMMEDIATE ACTIONS
- PERFORMED A VISUAL INSPECTION OF TUBE AND ISOLATION VALVES--NO DAMAGE OBSERVED.

CONTACTS: J. CARTER, NRR/DORS/EAB AIT: NO
J. MCCORMICK-BARGER, RIII
REFERENCE: MORNING REPORT DATED 01/28/94 SIGEVENT: TBD

- TOOK CONTACT TEMPERATURE MEASUREMENTS OF TRANSFER TUBE ABOVE AND BELOW ISOLATION VALVES--34 F ABOVE VALVES AND 64 F BELOW VALVES.

- INSTALLED FUEL POOL TRANSFER GATES.

- PLACED ELECTRIC HEATER CLOSE TO TUBE.
  - INSTALLED HEAT SENSORS ON TUBE
  - READINGS TAKEN EVERY SHIFT

- FORMED A 13 PERSON INVESTIGATIVE TEAM TO REVIEW THE EVENT.

<u>NRC IMMEDIATE ACTIONS</u>

- ISSUED A CONFIRMATORY ACTION LETTER ON FEBRUARY 1, 1994.

- ESTABLISHED A SPECIAL INSPECTION TEAM COMPRISED OF RIII, NRR, AND NMSS STAFF TO ASSESS EVENT.

- CALLED OTHER DECOMMISSIONED SITES WITH FUEL TO INFORM THEM OF EVENT.

<u>RESULTS OF SPECIAL TEAM INSPECTION</u>

- THE TEAM EXITED ON FEBRUARY 18, 1994.

- IDENTIFIED PIPING ASSOCIATED WITH THE ABANDONED ORIGINAL POOL CLEANUP AND COOLING SYSTEMS THAT REPRESENTED A SIPHON THREAT.
  - VERIFIED ISOLATION VALVES WERE CLOSED
  - HOLE DRILLED IN LINE

- REVIEWED DESIGN OF NEW CLEANUP SYSTEM.

- FOUND FUEL POOL WATER CHEMISTRY TO BE POOR.
  - Cs 137 WAS HIGH (10-2 UCI/CC)

- FUEL TRANSFER SYSTEM APPEARED TO BE UNDAMAGED. LICENSEE'S ENGINEERING EVALUATION INDICATED THAT TUBE BYPASS LINE COULD HAVE FROZEN. UT EXAMINATION IDENTIFIED THAT THERE WAS WATER OVER VALVE AND GAS BUBBLE UNDER VALVE. LICENSEE IS PLANNING TO SEND REMOTE DEVICE THROUGH TRANSFER TUNNEL TO SAMPLE GAS BUBBLE.

- LICENSEE HAD NO LEAK DETECTION SYSTEM OR POOL WATER INVENTORY PROGRAM.

- IDENTIFIED A NUMBER OF WATER LINES THAT ARE PIPED TO CONTAINMENT.

- CONTAINMENT VENTILATION SYSTEM HAD BEEN SHUTDOWN FOR SEVERAL YEARS.

- FUEL POOL BUILDING VENTILATION SYSTEM DID NOT HAVE CAPACITY STATED IN THE ODCM.
  - ONLY 2000 CFM VERSES 5200 CFM STATED

- BECAUSE OF PREVIOUS FREEZE DAMAGE TO CONTAINMENT HEATERS, LICENSEE DISCONTINUED HEATING CONTAINMENT.

- EMERGENCY PROCEDURES (EP) HAD PROVISIONS TO ADDRESS A FUEL DRAIN-DOWN EVENT.

- OVERSIGHT OF FACILITY HAD NOT BEEN ADEQUATE.

- AUDITS WERE MINIMAL, AUDITS WERE PERFORMED ON A SITE BASES AND FOCUSED ON OPERATING UNITS.

- UNIT 2/3 CONTROL ROOM HAS 3 FEET OF CONCRETE SHIELDING.
  - EVENT WOULD NOT HAVE AFFECTED CONTROL ROOM FUNCTIONS

- SPING AIR MONITOR NOT INSTALLED IN FUEL POOL BUILDING.

- DURING INSPECTION, LICENSEE ASSIGNED A FULL TIME PROJECT MANAGER TO UNIT 1 AND PLANNED TO ASSIGN ADDITIONAL STAFF AS NEEDED--A WRITTEN DESCRIPTION OF NEW ORGANIZATION WAS REQUESTED.

- LICENSEE'S INVESTIGATIVE TEAM EFFORTS WERE OUTSTANDING. IDENTIFIED MANY WEAKNESSES IN THEIR MANAGEMENT OF FACILITY, TRAINING, ENGINEERING AND LICENSING SUPPORT, AND STAFF ATTITUDE THAT UNIT 1 CAN'T CAUSE A SAFETY PROBLEM.

<u>FUTURE ACTIONS</u>

- NRC TEAM COMPRISING OF RIII, NRR, AND NMSS STAFF WILL MEET WEEK OF MARCH 7 TO REVIEW NRC LICENSING AND INSPECTION PROGRAMS TO DEVELOP INTERNAL LESSONS LEARNED RECOMMENDATIONS FROM THIS EVENT.

SEABROOK, UNIT 1
GENERIC ISSUE - POTENTIAL OPERATION WITH INADEQUATE SECONDARY RELIEF CAPACITY
SEPTEMBER 1992

PROBLEM

POTENTIAL OVERPRESSURIZATION OF THE MAIN STEAM SYSTEM.

CAUSE

DEFICIENCY IN THE WESTINGHOUSE BASIS FOR THE OPERABLE MAIN STEAM SAFETY VALVES VERSUS APPLICABLE POWER IN PERCENT OF RATED POWER. THIS INFORMATION IS REFLECTED IN WESTINGHOUSE STANDARD TECHNICAL SPECIFICATIONS TABLE 3.7-1.

SAFETY SIGNIFICANCE

OVERPRESSURIZATION COULD CAUSE MAIN STEAM SYSTEM PRESSURE TO INCREASE BEYOND 110 PERCENT, THEREBY EXCEEDING THE DESIGN BASES.

DESCRIPTION OF EVENTS

- IN SEPTEMBER 1992, THE SEABROOK UNIT 1 LICENSEE CONDUCTED A TEST OF THE SETPOINTS OF THEIR MAIN STEAM SAFETY VALVES (MSSVs) IN ACCORDANCE WITH THE TECHNICAL SPECIFICATIONS (T/S) REQUIREMENTS AND THE ASME BOILER AND PRESSURE VESSEL CODE, SECTION XI.

CONTACT:	N. FIELDS, NRR/DORS/EAB	AIT: NO
REFERENCES:	10 CFR 50.72 #26803, AND WESTINGHOUSE NSAL 94-01	SIGEVENT: TBD

- THE SETPOINTS OF EACH OF THE 20 MSSVs (5 VALVES PER STEAM GENERATOR) WERE TESTED USING A FURMANITE TREVITEST DEVICE. THIS DEVICE RESTRICTS THE MSSV FROM RELIEVING FULLY. THUS, THE MSSV IS INOPERABLE WHILE IT IS BEING TESTED.

- IN ACCORDANCE WITH TECH. SPEC. TABLE 3.7-1. THE NEUTRON HIGH FLUX TRIP SETPOINT WAS REDUCED TO 87% WHILE OPERATING WITH A MAXIMUM OF 1 MSSV INOPERABLE.

- ON JANUARY 20, 1994, WESTINGHOUSE ISSUED NUCLEAR SAFETY ADVISORY LETTER (NSAL) 94-01, "OPERATION AT REDUCED POWER LEVELS WITH INOPERABLE MSSVs." THIS LETTER STATED THAT PLANT OPERATION AT POWER LEVELS REDUCED IN ACCORDANCE WITH THE REQUIREMENTS OF TECH. SPEC. TABLE 3.7-1 MAY NOT BE CONSERVATIVE.

- LICENSEE PERFORMED A CALCULATION IN ACCORDANCE WITH THE RECOMMENDATIONS OF NSAL 94-01 AND ESTABLISHED A VALUE OF 60% POWER FOR THE HIGH FLUX TRIP SETPOINT WHILE OPERATING WITH A MAXIMUM OF 1 INOPERABLE MSSV.

- IN RETROSPECT, DURING THE SEPTEMBER 1992 MSSV SETPOINT TEST, THE HIGH FLUX TRIP SETPOINTS USED WERE NOT CONSERVATIVE RESULTING IN PLANT OPERATION IN AN UNANALYZED CONDITION AND A CONDITION OUTSIDE THE DESIGN BASIS OF THE PLANT.

- T/S TABLE 3.7-1 ASSUMED A LESS CONSERVATIVE LINEAR FUNCTION TO CALCULATE MAXIMUM POWER LEVEL RELATIVE TO AVAILABLE MSSV RELIEF CAPACITY. THE LINEAR FUNCTION IS IDENTIFIED IN THE BASES SECTION FOR THE T/S TABLE AND IS PROVIDED HERE AS ATTACHMENT 1.

- FOR A LOSS-OF-LOAD/TURBINE TRIP (LOL/TT) AT REDUCED POWER LEVELS, IF MAIN FEEDWATER IS LOST, A REACTOR TRIP IS NECESSARY TO PREVENT OVERPRESSURIZATION OF THE SECONDARY SIDE.

- AT HIGH POWER LEVELS A REACTOR TRIP WOULD OCCUR EARLY IN THE LOL/TT TRANSIENT FROM HIGH PRESSURIZER PRESSURE OR OVERTEMPERATURE DELTA T (ANALYZED TRANSIENT).

- HOWEVER, AT THE LOWER POWER LEVELS AT WHICH MSSV TESTING TAKES PLACE, A REACTOR TRIP MAY NOT OCCUR AS EARLY IN THE TRANSIENT, RESULTING IN A LONGER PERIOD DURING WHICH PRIMARY HEAT IS TRANSFERRED TO THE SECONDARY. REACTOR EVENTUALLY TRIPS ON SG LEVEL. BUT THIS MAY NOT OCCUR BEFORE SECONDARY PRESSURE EXCEEDS 110% OF DESIGN PRESSURE IF ONE OR MORE MSSVs ARE INOPERABLE IN ACCORDANCE WITH T/S TABLE 3.7-1.

- NSAL 94-01 RECOMMENDED A MORE CONSERVATIVE METHODOLOGY FOR CORRECTLY ADJUSTING THE HIGH FLUX TRIP SETPOINTS. THIS METHODOLOGY IS PROVIDED IN ATTACHMENT 2.

- WESTINGHOUSE IDENTIFIED A LIST OF AFFECTED U.S. PLANTS (ATTACHMENT 3).

FOLLOWUP

- WESTINGHOUSE HAS FORMALLY NOTIFIED THE AFFECTED PLANTS.
- TECHNICAL SPECIFICATIONS BRANCH IS EXAMINING POTENTIAL CHANGE TO WESTINGHOUSE STANDARD TECHNICAL SPECIFICATIONS.
- VENDOR BRANCH HAS INCLUDED THIS ISSUE IN THE PART 21 DATABASE.
- REACTOR SYSTEMS BRANCH HAS LEAD FOR LONG-TERM FOLLOWUP.
- NRR WILL ISSUE AN INFORMATION NOTICE.

ATTACHMENT 1

LINEAR FUNCTION IDENTIFIED IN THE BASES SECTION FOR T/S TABLE 3.7-1

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

SP = REDUCED REACTOR TRIP SETPOINT IN % OF RATED THERMAL POWER

V = MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES PER STEAM LINE

X = TOTAL RELIEVING CAPACITY OF ALL SAFETY VALVES PER STEAM LINE IN LBM/HR

Y = MAXIMUM RELIEVING CAPACITY OF ANY ONE SAFETY VALVE IN LBM/HR

(109) = POWER RANGE NEUTRON FLUX-HIGH TRIP SETPOINT FOR ALL LOOPS IN OPERATION

ATTACHMENT 2

ALGORITHM FOR USE IN DEFINING THE REVISED T/S TABLE 3.7-1 HIGH FLUX TRIP SETPOINT VALUES

$$H_I\ \phi = \frac{100}{Q} \times \frac{w_s\ h_{fg}\ N}{K}$$

$H_I\ \phi$ = SAFETY ANALYSIS POWER RANGE HIGH NEUTRON FLUX SETPOINT, %

Q = NOMINAL NSSS POWER RATING OF THE PLANT (INCLUDING REACTOR COOLANT PUMP HEAT), MW_T

K = CONVERSION FACTOR, 947.82 (BTU/SEC)/MW_T

w_s = MINIMUM TOTAL STEAM FLOW RATE CAPABILITY OF THE OPERABLE MSSVs ON ANY ONE STEAM GENERATOR AT THE HIGHEST MSSV OPENING PRESSURE INCLUDING TOLERANCE AND ACCUMULATION, AS APPROPRIATE, IN LB/SEC. FOR EXAMPLE, IF THE MAXIMUM NUMBER OF INOPERABLE MSSVs ON ANY ONE STEAM GENERATOR IS ONE, THEN w_s SHOULD BE A SUMMATION OF THE CAPACITY OF THE OPERABLE MSSVs AT THE HIGHEST OPERABLE MSSV OPERATING PRESSURE, EXCLUDING THE HIGHEST CAPACITY MSSV. IF THE MAXIMUM NUMBER OF INOPERABLE MSSVs PER STEAM GENERATOR IS THREE THEN w_s SHOULD BE A SUMMATION OF THE CAPACITY OF THE OPERABLE MSSVs AT THE HIGHEST OPERABLE MSSV OPERATING PRESSURE, EXCLUDING THE THREE HIGHEST CAPACITY MSSVs.

H_{FG} = HEAT OF VAPORIZATION FOR STEAM AT THE HIGHEST MSSV OPENING PRESSURE INCLUDING TOLERANCE AND ACCUMULATION, AS APPROPRIATE, BTU/LBM.

N = NUMBER OF LOOPS IN PLANT

ATTACHMENT 3

AFFECTED U.S PLANTS

D.C. COOK 1 & 2
J.M. FARLEY 1 & 2
BYRON 1 & 2
BRAIDWOOD 1 & 2
V.C. SUMMER 1
ZION 1 & 2
SHEARON HARRIS 1
W.B. MCGUIRE 1 & 2
CATAWBA 1 & 2
BEAVER VALLEY 1 & 2
TURKEY POINT 3 & 4
VOGTLE 1 & 2
INDIAN POINT 2 & 3
SEABROOK 1
MILLSTONE 3
DIABLO CANYON 1 & 2
WOLF CREEK
CALLAWAY 1
COMANCHE PEAK 1 & 2
SOUTH TEXAS 1 & 2
SEQUOYAH 1 & 2
NORTH ANNA 1 & 2
WATTS BAR 1 & 2
SALEM 1 & 2

REACTOR SCRAM

Reporting Period: 02/14/94 to 02/20/94

DATE	PLANT & UNIT	POWER	TYPE	CAUSE	COMPLICATIONS	YTD ABOVE 15%	YTD BELOW 15%	YTD TOTAL
02/14/94	SOUTH TEXAS 1	0	SM	Design or Installati	NO	0	1	1

REACTOR SCRAM

Reporting Period: 02/21/94 to 02/27/94

DATE	PLANT & UNIT	POWER	TYPE	CAUSE	COMPLICATIONS	YTD ABOVE 15%	YTD BELOW 15%	YTD TOTAL
02/21/94	COOK 2	60	SA	Maintenance Error	NO	1	0	1
02/26/94	OCONEE 1	100	SA	Equipment Failure	NO	1	0	1

te: Year To Date (YTD) Totals Include Events Within The Calendar Year Indicated By The End Date Of The Specified Reporting Period

COMPARISON OF WEEKLY SCRAM STATISTICS WITH INDUSTRY AVERAGES

PERIOD ENDING
02/20/94

SCRAM CAUSE	NUMBER OF SCRAMS	1994 WEEKLY AVERAGE (YTD)	1993 WEEKLY AVERAGE	1992 WEEKLY AVERAGE	1991* WEEKLY AVERAGE	1990* WEEKLY AVERAGE
POWER GREATER THAN OR EQUAL TO 15%						
EQUIPMENT FAILURE*	0	1.6	1.8	2.6	2.9	3.4
DESIGN/INSTALLATION ERROR*	0	0.0	-	-	-	-
OPERATING ERROR*	0	0.0	0.3	0.2	0.6	0.5
MAINTENANCE ERROR*	0	0.5	0.5	0.4	-	-
EXTERNAL*	0	0.0	0.1	-	-	-
OTHER*	0	0.0	-	0.2	-	-
Subtotal	0	2.1	2.7	3.4	3.5	3.9
POWER LESS THAN 15%						
EQUIPMENT FAILURE*	0	0.3	0.4	0.4	0.3	0.4
DESIGN/INSTALLATION ERROR*	1	0.1	-	-	-	-
OPERATING ERROR*	0	0.1	0.1	0.1	0.2	0.1
MAINTENANCE ERROR*	0	0.0	-	0.1	-	-
EXTERNAL*	0	0.0	-	-	-	-
OTHER*	0	0.0	-	0.1	-	-
Subtotal	1	0.5	0.5	0.7	0.5	0.5
TOTAL	1	2.6	3.2	4.1	4.0	4.4

SCRAM TYPE	NO. OF SCRAMS	1994 WEEKLY AVERAGE (YTD)	1993 WEEKLY AVERAGE	1992 WEEKLY AVERAGE	1991 WEEKLY AVERAGE	1990 WEEKLY AVERAGE
TOTAL AUTOMATIC SCRAMS	0	1.9	2.4	3.1	3.3	3.2
TOTAL MANUAL SCRAMS	1	0.8	0.9	1.0	0.7	1.2

TOTALS MAY DIFFER BECAUSE OF ROUNDING OFF

* Detailed breakdown not in database for 1991 and earlier

- EXTERNAL cause included in EQUIPMENT FAILURE
- MAINTENANCE ERROR and DESIGN/INSTALLATION ERROR causes included in OPERATING ERROR
- OTHER cause included in EQUIPMENT FAILURE 1991 and 1990

COMPARISON OF WEEKLY SCRAM STATISTICS WITH INDUSTRY AVERAGES

PERIOD ENDING
02/27/94

SCRAM CAUSE	NUMBER OF SCRAMS	1994 WEEKLY AVERAGE (YTD)	1993 WEEKLY AVERAGE	1992 WEEKLY AVERAGE	1991* WEEKLY AVERAGE	1990* WEEKLY AVERAGE
POWER GREATER THAN OR EQUAL TO 15%						
EQUIPMENT FAILURE*	1	1.6	1.8	2.6	2.9	3.4
DESIGN/INSTALLATION ERROR*	0	0.0	-	-	-	-
OPERATING ERROR*	0	0.0	0.3	0.2	0.6	0.5
MAINTENANCE ERROR*	1	0.6	0.5	0.4	-	-
EXTERNAL*	0	0.0	0.1	-	-	-
OTHER*	0	0.0	-	0.2	-	-
Subtotal	2	2.2	2.7	3.4	3.5	3.9
POWER LESS THAN 15%						
EQUIPMENT FAILURE*	0	0.2	0.4	0.4	0.3	0.4
DESIGN/INSTALLATION ERROR*	0	0.1	-	-	-	-
OPERATING ERROR*	0	0.1	0.1	0.1	0.2	0.1
MAINTENANCE ERROR*	0	0.0	-	0.1	-	-
EXTERNAL*	0	0.0	-	-	-	-
OTHER*	0	0.0	-	0.1	-	-
Subtotal	0	0.4	0.5	0.7	0.5	0.5
TOTAL	2	2.6	3.2	4.1	4.0	4.4

SCRAM TYPE	NO. OF SCRAMS	1994 WEEKLY AVERAGE (YTD)	1993 WEEKLY AVERAGE	1992 WEEKLY AVERAGE	1991 WEEKLY AVERAGE	1990 WEEKLY AVERAGE
TOTAL AUTOMATIC SCRAMS	2	1.9	2.4	3.1	3.3	3.2
TOTAL MANUAL SCRAMS	0	0.7	0.9	1.0	0.7	1.2

TOTALS MAY DIFFER BECAUSE OF ROUNDING OFF

* Detailed breakdown not in database for 1991 and earlier

- EXTERNAL cause included in EQUIPMENT FAILURE
- MAINTENANCE ERROR and DESIGN/INSTALLATION ERROR causes included in OPERATING ERROR
- OTHER cause included in EQUIPMENT FAILURE 1991 and 1990

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. THERE ARE 111 REACTORS HOLDING AN OPERATING LICENSE.

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

3. COMPLICATIONS: RECOVERY COMPLICATED BY EQUIPMENT FAILURES OR PERSONNEL ERRORS UNRELATED TO CAUSE OF SCRAM.

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

OEAB SCRAM DATA

Manual and Automatic Scrams for 1987 ------------------ 435
Manual and Automatic Scrams for 1988 ------------------ 291
Manual and Automatic Scrams for 1989 ------------------ 252
Manual and Automatic Scrams for 1990 ------------------ 226
Manual and Automatic Scrams for 1991 ------------------ 206
Manual and Automatic Scrams for 1992 ------------------ 212
Manual and Automatic Scrams for 1993 ------------------ 176
Manual and Automatic Scrams for 1994 --(YTD 02/27/94)-- 22