APR 2 9 1993

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
APRIL 21, 1993 - BRIEFING 93-15

On April 21, 1993, we conducted an Operating Reactors Events Briefing (93-15) to inform senior managers from offices of the Commission, RES, SECY, AEOD, ACRS, OE, NRR, and regional offices of selected events that occurred since our last briefing on April 14, 1993. Enclosure 1 lists the attendees. Enclosure 2 presents the significant elements of the discussed events.

Enclosure 3 contains reactor scram statistics for the week ending April 18, 1993. Two significant events were identified for input into the NRC performance indicator program (Enclosure 4).

/s/

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

Enclosures: As stated

cc w/attachments:
See next page

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1. RETURN TO REGULATORY CENTRAL FILES ROOM

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| | | | | |
|---|---|---|---|---|
| KGray | | RD | EG | AEChaffee |
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cc:

T. Murley, NRR (12G18)
F. Miraglia, NRR (12G18)
F. Gillespie, NRR (12G18)
J. Partlow, NRR (12G18)
S. Varga, NRR (14E4)
J. Calvo, NRR (14A4)
G. Lainas, NRR (14H3)
J. Roe, NRR (13E4)
J. Zwolinski, NRR (13H24)
M. Virgilio, NRR (13E4)
W. Russell, NRR (12G18)
J. Richardson, NRR (7D26)
A. Thadani, 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)
Acting Director, DSP/AEOD (MN-9112)
L. Spessard, AEOD (MN-3701)
K. Brockman, AEOD (MN-3206)
S. Rubin, AEOD (MN [illegible]9)
M. Harper, AEOD (M[illegible]12)
G. Grant, EDO (17G21)
R. Newlin, GPA (2G5)
E. Beckjord, RES (NLS-007)
A. Bates, SECY (16G15)
G. Rammling, OCM (16G15)
T. Martin, Region I
W. Kane, Region I
C. Hehl, Region I
S. Ebneter, Region II
E. Merschoff, Region II
S. Vias, Region II
J. Martin, Region III
E. Greenman, Region III
J. Milhoan, Region IV
B. Beach, Region IV
B. Faulkenberry, Region V
K. Perkins, Region V

L. Engle (PDII-2)
H. Berkow (PDII-2)
D. Hood (PDII-3)
D. Matthews (PDII-3)

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

ENCLOSURE 1

LIST OF ATTENDEES

OPERATING REACTORS EVENTS FULL BRIEFING (93-15)

APRIL 21, 1993

| NAME | OFFICE | NAME | OFFICE |
|---|---|---|---|
| A. CHAFFEE | NRR | V. HODGE | NRR |
| R. BENEDICT | NRR | T. LE | NR |
| T. GREENE | NRR | C. JACKSON | N |
| K. GRAY | NRR | D. MATTHEWS | NRR |
| R. DENNIG | NRR | J. ROE | NRR |
| E. GOODWIN | NRR | G. ZECH | NRR |
| T. KOSHY | NRR | J. PARTLOW | NRR |
| R. ECKENRODE | NRR | D. HOOD | NRR |
| S. LONG | NRR | P. ENG | NRR |
| M. MARKLEY | NRR | M. LEACH | OCM/IS |
| C. BEARDSLEE | NRR | P. NORIAN | RES |
| W. KOO | NRR | V. BENAROYA | AEOD |
| J. ZARZUELA | NRR | D. COE | ACRS |
| L. LOIS | NRR | W. TROSKOSKI | OE |
| H. BERKOW | NRR | K. HART | SECY |
| L. ENGLE | NRR | J. SCARBOROUGH | OCM/KR |

EPHONE ATTENDANCE
AT ROLL CALL)

<u>Regions</u>
Region I
Region II
Region III
Region IV
Region V

<u>Resident Inspectors</u>
M. Lesser (North Anna)
B. Bonser (Vogtle)

<u>IIT/AIT Team Leaders</u>

<u>Misc.</u>

ENCLOSURE 2

# OPERATING REACTORS EVENTS BRIEFING 93-15

LOCATION: 10 B11, WHITE FLINT
WEDNESDAY, APRIL 21, 1993, 11:00 A.M.

| | |
|---|---|
| NORTH ANNA, UNIT 2 | SCRAM WITH SAFETY FEATURE BYPASS |
| VOGTLE, UNIT 1 | INADVERTENT SAFETY INJECTION WHILE SHUT DOWN WITH THE REACTOR COOLANT SYSTEM SOLID |
| VARIOUS PLANTS | PRIMARY WATER STRESS CORROSION CRACKING |

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

# NORTH ANNA, UNIT 2
# SCRAM WITH SAFETY FEATURE BYPASS
# APRIL 16, 1993

## PROBLEM

REACTOR OPERATOR (RO) BYPASSED A SAFETY FEATURE FOLLOWING A REACTOR TRIP.

## CAUSE

OPERATOR SECURED AUXILIARY FEEDWATER (AFW) PUMPS TO "PULL-TO-LOCK" (PTL) RATHER THAN "AUTO".

## SAFETY SIGNIFICANCE

AFW PUMPS WOULD NOT HAVE STARTED AUTOMATICALLY IF CALLED UPON BY SAFETY ACTUATION.

## DISCUSSION

- TURBINE TRIP/REACTOR TRIP OCCURRED FROM 100% POWER. MAIN FEEDWATER REGULATING VALVES CLOSED. AUXILIARY FEEDWATER PUMPS STARTED AUTOMATICALLY WHEN STEAM GENERATOR WATER LEVELS DROPPED.

- IN ABOUT FIVE MINUTES, THE FULL AFW FLOW (1400 GPM) COOLED THE REACTOR COOLANT SYSTEM (RCS) TO 547°F. AFW SHOULD THEN HAVE BEEN THROTTLED TO MAINTAIN 547°F, BUT IT WASN'T.

| | | | |
|---|---|---|---|
| CONTACT: | R. BENEDICT, NRR/OEAB | AIT: | NO |
| REFERENCE: | 10 CFR 50.72 #25403 | SIGEVENT: | TBD |

- ABOUT FOUR MINUTES LATER, OPERATORS NOTICED RCS TEMPERATURE WAS 540°F, AND SECURED AFW PUMPS IN PTL TO REDUCE OVER-COOLING.

- MAIN FEEDWATER REGULATING VALVE BYPASSES WERE USED TO BRING STEAM GENERATOR WATER LEVELS UP. WARMER MAIN FEEDWATER HELPED RAISE RCS TEMPERATURE TO 547°F.

- ABOUT 18 MINUTES AFTER REACTOR TRIP, SRO NOTICED THE PTL POSITION OF THE AFW PUMP SWITCHES, AND THE REACTOR OPERATOR TURNED THEM BACK TO AUTO.

- CONTRIBUTING FACTORS:
  - INADEQUATE COMMAND/CONTROL/COMMUNICATIONS BETWEEN RO AND SRO ON CONTROLLING THE COOLDOWN.

  - APPARENT LACK OF SENSITIVITY ON BYPASSING OF SAFETY FUNCTION.

FOLLOWUP

- NRC INFORMATION NOTICES HAVE DESCRIBED PREVIOUS EVENTS (LASALLE, CRYSTAL RIVER, FOREIGN REACTOR) DURING WHICH SAFETY FEATURE BYPASS HAS OCCURRED.

- WAS THIS AN INDIVIDUAL LAPSE OR A MANIFESTATION OF A GENERAL BREAKDOWN IN ATTITUDE TOWARD SAFETY FEATURE BYPASS?

- WHY WASN'T "PTL" RECOGNIZED AS SOON AS IT OCCURRED?

- REGION II, ASSISTED BY AEOD, IS CONDUCTING A SPECIAL INSPECTION OF THE HUMAN FACTORS ASPECTS OF THE EVENT.

# VOGTLE, UNIT 1
# INADVERTENT SAFETY INJECTION WHILE SHUT DOWN WITH THE REACTOR COOLANT SYSTEM SOLID
# APRIL 18, 1993

## PROBLEM

A SAFETY INJECTION (SI) INITIATION WHILE SHUT DOWN CREATED A POTENTIAL COLD OVERPRESSURIZATION SITUATION.

## CAUSE

CAUSE OF SI WAS PROCEDURE DEFICIENCY WHILE PERFORMING TESTING OF SOLID STATE PROTECTION SYSTEM (SSPS).

## SAFETY SIGNIFICANCE

OVERPRESSURIZATION WHILE SHUT DOWN CAN POTENTIALLY CAUSE BRITTLE FRACTURE. ALSO BECAUSE EMERGENCY CORE COOLING SYSTEM (ECCS) ALTERNATE MINI-FLOW SYSTEM CAN DIVERT SOME ECCS FLOW, THERE IS A CONCERN OF HOW THIS EVENT IS RELATED TO THE SHEARON HARRIS EVENT.

## SEQUENCE OF EVENTS

INITIAL CONDITIONS:

- PLANT IN COLD SHUTDOWN.
- REACTOR COOLANT SYSTEM (RCS) SOLID.
- REACTOR COOLANT PUMP SWEEPS WERE IN PROGRESS.
- RCS PRESSURE AT 345 PSI.
- RCS TEMPERATURE AT 120°F.
- BUS "BY1B" DEENERGIZED.
- DELTA T CHANNEL CALIBRATION ON TRAIN A.

| | | | |
|---|---|---|---|
| CONTACT: | T. GREENE, NRR/OEAB | AIT: | NO |
| REFERENCE: | 10 CFR 50.72 #25411 | SIGEVENT: | TBD |

- "BY1B" OUTAGE AND DELTA T CALIBRATION CAUSED 2 OF 3 SSPS PRESSURIZER PRESSURE BISTABLES TO BE IN A TRIPPED CONDITION.

- AT 7:56 A.M., TRAIN A SSPS WAS TAKEN OUT OF "TEST" TO "OPERATE" TO PERFORM SLAVE RELAY TESTING. SI SUBSEQUENTLY INITIATED ON PRESSURIZER LOW PRESSURE AND LOW STEAMLINE PRESSURE.

- CENTRIFUGAL CHARGING PUMP (CCP) A, WHICH WAS PREVIOUSLY RUNNING, REALIGNED TO INJECT INTO THE CORE.

- DIESEL GENERATOR STARTED BUT DID NOT LOAD.

- CCP A INJECTED FOR APPROXIMATELY 15 SECONDS BEFORE BEING ISOLATED.

- RCS PRESSURE ROSE TO 450 PSI, AT WHICH POINT THE RESIDUAL HEAT REMOVAL (RHR) SUCTION RELIEF VALVES LIFTED TO RELIEVE RCS PRESSURE.

- UPON ISOLATION OF CCP A INJECTION VALVES, CCP A ALTERNATE MINIMUM FLOW (AMF) VALVE, PSV-8501A BELLOWS BLEW OUT LEAKING WATER INTO CCP A ROOM.

DISCUSSION

- BOTH SI PUMPS WERE TAGGED OUT AS REQUIRED BY TECHNICAL SPECIFICATIONS.

- RHR SUCTION RELIEF VALVES ARE SIZED TO RELIEVE OUTPUT OF BOTH CCPs AND THE POSITIVE DISPLACEMENT CHARGING PUMP.

- LICENSEE HAS HAD BELLOWS CRACKING PREVIOUSLY UPON INITIATION OF AMF SYSTEM. BELIEVES THAT THIS DOES NOT IMPACT OPERABILITY OF SYSTEM.

- LICENSEE HAS BEEN CHECKING SETPOINT OF ONE AMF RELIEF VALVE EACH OUTAGE, HAS NOTED NO SIGNIFICANT SETPOINT DRIFT.

- THERE HAVE BEEN SIMILAR EVENTS AT OTHER PLANTS.

FOLLOWUP

- LICENSEE WILL WALK DOWN RHR RELIEF LINE AND AMF LINE FOR EVIDENCE OF WATER HAMMER.

- LICENSEE WILL INSPECT RELIEF VALVES FOR SIGNS OF CHATTER AND SETPOINT DRIFT.

- OEAB PLANS TO DETERMINE SHUTDOWN RISK ASSOCIATED WITH THIS EVENT.

- NRC WILL REVIEW LICENSEE'S SUBMITTAL CONCERNING AMF OPERABILITY IN LIGHT OF THIS EVENT.

## PWSCC OF INCONEL 600 AT PWRs

- PWSCC Identified to Commission as an Emerging Issue in 1989

    - US Experience includes Pressurizer Instruments Nozzles and Heater Sleeves

    - Foreign Experience includes Instrument Nozzles and Control Rod Drive Penetrations

    - Steam generator Tubes and Plugs also Affected

- Factors Contributing to Cracking

    - Susceptible Material

      Lower Temperature Solution Heat Treatment

    - Environment

      PWR Operating Temperatures

      Crevices

    - Stress

      Cold Work

      Residual Stress

      Pressure

PWSCC OF INCONEL 600 AT PWRs

INDUSTRY EXPERIENCE

## DOMESTIC PLANTS

### CE Plants

#### San Onofre Unit 3

(2 yrs.) • 4 Pressurizer instrument nozzle
(1986)
(1-leaked)

(8 yrs.) • 3 Pressurizer instrument nozzles (1992)
( [illegible]eaked)

#### St. Lucie Unit 2

(4 yrs.) • 4 pressurizer instrument nozzles (1987)
(1-leaked)

(4 yrs.) • 4 pressurizer instrument nozzles (1993)
(4-leaked)

#### Calvert Cliff Unit 2

(12 yrs.) • 1 pressurizer instrument nozzle
(1989)
(1-leaked)

(12 yrs.) • 20 pressurizer heater sleeves
(1989)
(1-leaked)

Palo Verde Unit 1

(6 yrs.) • 3 pressurizer instrument nozzles (1992)
(1-leaked)

San Onofre Unit 2

(7 yrs.) • 2 pressurizer instrument nozzles (1992)
(2-leaked)

B&W PLANTS

ANO-1

(16 yrs.) • 1 pressurizer instrument nozzle
(1990)

(1-leaked)

FOREIGN PLANTS

French PWRs

1300 MW Units

(1 yr.) • Numerous Pressurizer Instrument Nozzles (1989)
(mechanically rolled into vessel)

• 9 units inspected for CRD penetrations (1992)
5 units found indications in 9 penetrations.

900 MW Units

(10 yrs.) • Bugey 3
3 CRD penetrations (1991)
(1-leaked)

• Bugey 4
8 CRD penetrations (1992)

- 9 units inspected for CRD penetrations
  8 units found indications in 27 penetrations.

Swedish PWRs

- Ringhals 2

  5 CRD penetrations (1992)

- 3 units inspected for CRD penetrations
  2 units found indications in 8 penetrations.

Switzerland

- 2 units inspected for CRD penetrations
  1 unit found indications in 2 penetrations.

Japan

- 11 units inspected for CRD penetrations
  No indication was found.

Belgium

- 1 unit inspected for CRD penetrations
  No indication was found.

## INDUSTRY WIDE ACTIVITIES

## CONCERNING PWSCC OF INCONEL 600

- Meeting Materials Sub-group - WOG (1/7/92), CEOG (3/25/92) and BWOG (5/12/92) concerning their programs for PWSCC of Inconel 600 particularly regarding the French experience in CRD penetrations

- Meeting with all PWR owners groups coordinated by NUMARC on 8/92, 11/92 and 3//93

- Reaction to NRC's Information Notice 90-10, CEOG and BWOG have initiated programs

  - To identify all Inconel 600 applications in the primary systems and review its

  - Materials (grade and form, chemistry and mechanical properties);

  - Welding, fabrication, and installation processes;

  - Operating condition (temperature, stresses and water chemistry) and

  - Other relevant parameters.

- WOG has initiated a similar program for Inconel 600 applications in the reactor vessels.

- Other program activities included development of advanced inspection techniques, repair techniques, and replacement materials and others.

- CEOG recommended visual examinations of pressurizer heater sleeves penetration once every 1100 days (about every 2 cycles)

- All OGs have ongoing programs to monitor the French CRD penetration experiences and to determine its relevancy to the respective OG's plants.

- EPRI is coordinating the long term Inconel 600 R&D efforts, which are sponsored jointly by W, B&W and CE Owners Groups.

- NUMARC is coordinating industry activities with NRC

- On-going major industry efforts: (CRD Penetrations)

    - Safety evaluation pertaining to potential PWSCC in U.S. plants
    - Flaw acceptance criteria
    - NDE (UT and ECT) demonstration program(EPRI)

- Pilot inspection of U.S. PWR plants starting in spring 1994 (CRD Penetrations)

## STAFF VIEW OF SAFETY SIGNIFICANCE

- PWSCC cracking of Inconel 600 has a minimal safety impact because all reported cracking has been short and axially oriented.

- Potential exists for cracking in a large number of CRD housing penetrations

- Concern about corrosion from boric acid deposits on head, early leak detection is important since corrosion rates may be high

- Unique stress distribution could cause circumferential cracking

- Prudence suggests an orderly inspection program

  - Defense in Depth

  - GDC 14 - Low Probability of RCPB leakage

REACTOR SCRAM

Reporting Period: 04/12/93 to 04/18/93

| DATE | PLANT & UNIT | POWER | TYPE | CAUSE | COMPLICATIONS | YTD ABOVE 15% | YTD BELOW 15% | YTD TOTAL |
|---|---|---|---|---|---|---|---|---|
| 04/13/93 | NINE MILE POINT 1 | 0 | SA | Equipment Failure | NO | 1 | 1 | 2 |
| 04/16/93 | NORTH ANNA 2 | 100 | SA | Equipment Failure | NO | 1 | 0 | 1 |

Note: 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
04/18/93

| SCRAM CAUSE | NUMBER OF SCRAMS | 1993 WEEKLY AVERAGE (YTD) | 1992 WEEKLY AVERAGE | 1991* WEEKLY AVERAGE | 1990* WEEKLY AVERAGE | 1989* WEEKLY AVERAGE |
|---|---|---|---|---|---|---|
| POWER GREATER THAN OR EQUAL TO 15% | | | | | | |
| EQUIPMENT FAILURE | 1 | 2.3 | 2.6 | 2.9 | 3.4 | 3.1 |
| DESIGN/INSTALLATION ERROR* | 0 | 0.1 | - | - | - | - |
| OPERATING ERROR* | 0 | 0.4 | 0.2 | 0.6 | 0.5 | 1.0 |
| MAINTENANCE ERROR* | 0 | 0.5 | 0.4 | - | - | - |
| EXTERNAL* | 0 | 0.2 | - | - | - | - |
| OTHER* | 0 | 0.0 | 0.2 | - | - | 0.1 |
| Subtotal | 1 | 3.5 | 3.4 | 3.5 | 3.9 | 4.2 |
| POWER LESS THAN 15% | | | | | | |
| EQUIPMENT FAILURE* | 1 | 0.3 | 0.4 | 0.3 | 0.4 | 0.3 |
| DESIGN/INSTALLATION ERROR* | 0 | 0.0 | - | - | - | - |
| OPERATING ERROR* | 0 | 0.1 | 0.1 | 0.2 | 0.1 | 0.3 |
| MAINTENANCE ERROR* | 0 | 0.0 | 0.1 | - | - | - |
| EXTERNAL* | 0 | 0.0 | - | - | - | - |
| OTHER* | 0 | 0.0 | 0.1 | - | - | - |
| Subtotal | 1 | 0.4 | 0.7 | 0.5 | 0.5 | 0.6 |
| TOTAL | 2 | 3.9 | 4.1 | 4.0 | 4.4 | 4.8 |

| SCRAM TYPE | NO. OF SCRAMS | 1993 WEEKLY AVERAGE (YTD) | 1992 WEEKLY AVERAGE | 1991 WEEKLY AVERAGE | 1990 WEEKLY AVERAGE | 1989 WEEKLY AVERAGE |
|---|---|---|---|---|---|---|
| TOTAL AUTOMATIC SCRAMS | 2 | 2.9 | 3.1 | 3.3 | 3.2 | 3.9 |
| TOTAL MANUAL SCRAMS | 0 | 1.1 | 1.0 | 0.7 | 1.2 | 0.9 |

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

3. SA = Scram Automatic; SM = Scram Manual

## 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 --(YTD 04/18/93)-- 61

ENCLOSURE 4

# OPERATING REACTOR PLANTS SIGNIFICANT EVENTS

SORT> Event Date
QUERY> Event Type SIG & Close Out Date >= 04/20/93 & Close Out Date <= 04/20/93 & Event Type = "SIG"

| PLANT & UNIT | DATE OF EVENT | 50.72 NUMBER | DESCRIPTION OF EVENT | SIGNIFICANCE | OR BRIEFING | PRESENTER | CLOSEOUT RECORD |
|---|---|---|---|---|---|---|---|
| SOUTH TEXAS 1,2 | 02/03/93 | 25008 | SCRAM WITH COMPLICATIONS (PARTIAL LOSS OF AFW) ON UNIT 2; MALFUNCTIONS OF TURBINE DRIVEN AFW DURING SURVEILLANCE ON UNIT 1. | Unexpected Plant Performance | 93-05 | FIELDS M. | EFR 93-022 |
| THREE MILE ISLAND 1 | 02/07/93 | 25035 | PLANT SECURITY BREACH AND DECLARATION OF SITE AREA EMERGENCY. | OTHER - PROTECTED AREA BOUNDARY | 93-05 | BENNER E. | EFR 93-015 |