AUG 0 8 1988

MEMORANDUM FOR: Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

FROM: Wayne Lanning, Chief
Events Assessment Branch
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

SUBJECT: THE OPERATING REACTORS EVENTS MEETING
August 2, 1988 - MEETING 88-31

On August 2, 1988 an Operating Reactors Events meeting (88-31) was held to brief senior managers from NRR, OSP, AEOD, Commission Staff, and Regional Offices on events which occurred since our last meeting on July 26, 1988. 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 presents one event suggested for long-term followup and a summary of reactor scrams. One significant event was identified for input to NRC's Performance Indicator Program.

Wayne Lanning, Chief
Events Assessment Branch
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc w/Enclo.:
See Next Page

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cc:
T. Murley, 12G-18
F. Miraglia, 12G-18
E. Jordan, AEOD
E. Beckjord, NL-007
W. Russell, RI
B. Davis, RIII
J. N. Grace, RII
R. D. Martin, RIV
J. B. Martin, RV
W. Kane, RI
L. Reyes, RII
E. Greenman, RIII
J. Callan, RIV
D. Kirsch, RV
S. Varga, 14E-4
D. Crutchfield, 13A-2
B. Boger, 14A-2
G. Holahan, 13H-4
G. Lainas, 14H-3
L. Shao, 8E-2
J. Partlow, 7D-24
B. Grimes, 9A-2
F. Congel, 10E-4
E. Weiss, AEOD
T. Martin, 12G-18
J. Guttmann, SECY
A. Thadani, 7E-4
S. Rubin, AEOD
R. Barrett, 10E-2
J. Sniezek, 12G-18
J. Forsyth, INPO
E. Sylvester, 14A-18
E. Adensam, 14H-12
D. Dilanni, 13H-20
M. Virgilio, 13H-14
G. Dick, 13G-16
J. Calvo, 13D-17

ENCLOSURE 1

LIST OF ATTENDEES

OPERATING REACTORS EVENTS BRIEFING (88-31)

August 2, 1988

NAME	ORGANIZATION	NAME	ORGANIZATION
R. Scholl	NRR/DOEA	M.Virgilio	NRR/DRSP
W. Troskoski	DEDRO	F. Paultiz	OSP/TVA
T.P. Gwynn	OCM/LZ	M.S. Callahan	GPA/CA
J. Guttmann	SECY	C. Harbuck	NRR/DRSP
J. Kelly	OSP/TVA	M.L. Reardon	NRR/DOEA
E.A. Reeves	NRR/PD2-1	B. Buckley	NRR/PD2-1
T. Greene	NRR/DOEA	H. Berkow	NRR/PD2-3
S. Varga	NRR/DRP	D. Dilanni	NRR/PD3-1
G. Holahan	NRR/DRSP	L. Rubenstein	NRR/DRSP
J. Calvo	NRR/DRSP	W. Minners	RES/DRPS
D. Crutchfield	NRR/DRSP	D.C. Fischer	NRR/DOEA
C.E.Rossi	NRR/DOEA	B. Boger	NRR/ADRI
J.E. Rosenthal	AEOD	C. Haughney	NRR/DRIS
F. Miraglia	NRR/ADP		

ENCLOSURE 2

OPERATING REACTORS EVENTS BRIEFING 88-31
EVENTS ASSESSMENT BRANCH
LOCATION: 12-B-11 WHITE FLINT

TUESDAY, AUGUST 2, 1988, 11:00 A.M.

PRAIRIE ISLAND 1 & 2	REACTOR PROTECTIVE SYSTEM CIRCUIT DESIGN DEFICIENCY
ALMARAZ UNIT 1	STRESS CORROSION CRACKING STEAM GENERATOR TUBES (UPDATE)
BRUNSWICK UNIT 2	ASCO PRESSURE SWITCHES
ARKANSAS UNIT 2	REACTOR COOLANT PUMP SEAL LEAK

PRAIRIE ISLAND UNITS 1 & 2
REACTOR PROTECTIVE SYSTEM CIRCUIT DESIGN DEFICIENCY
JULY 26, 1988

PROBLEM
OVERPOWER DELTA TEMPERATURE (OPΔT) AND OVER TEMPERATURE DELTA TEMPERATURE (OTΔT) TRIP SETPOINTS DO NOT COMPENSATE PROPERLY FOR HIGH AXIAL FLUX DIFFERENCE (AFD).

CAUSE
A DESIGN FLAW PREVENTS THE NEWER CIRCUIT MODULES FROM FUNCTIONING AS ORIGINALLY INTENDED.

SAFETY SIGNIFICANCE
CRITICAL HEAT FLUX LIMITS COULD BE EXCEEDED IF AFD IS NOT ADEQUATELY COMPENSATED FOR IN RPS VARIABLE TRIP SETPOINTS.

DISCUSSION
- O UNIT 1 WAS IN COASTDOWN AT END OF CORE LIFE-84% POWER.
- O UNIT 2 WAS AT 100% POWER.
- O UNIT 1 LOWERED POWER IN ORDER TO AFFECT MINOR REPAIR TO A RCP. AFTER THE REPAIRS WERE COMPLETED, REACTOR POWER WAS TEMPORARILY LIMITED TO 48% DUE TO XENON BUILDUP.
- O WHILE WAITING FOR XENON DECAY/BURNOUT, THE CORE DEVELOPED AN AXIAL FLUX DIFFERENCE (AFD).
- O AS THE AFD INCREASED THE OPERATORS EXPECTED TO OBSERVE A DECREASE IN THE SETPOINTS FOR OPΔT AND OTΔT INSTRUMENTS.
- O 3 OUT OF 4 INSTRUMENT CHANNELS RESPONDED AS EXPECTED. THE ANOMALOUS CHANNEL HAD A FOXBORO 62H STYLE "C" CONTROLLER INSTALLED (THE THREE OPERABLE CHANNELS HAD STYLE "B" CONTROLLERS).
- O TESTING ON UNIT 2 REVEALED THAT ALL STYLE "C" MODULES FAILED TO RESPOND PROPERLY (3 OUT OF 4 CHANNELS).
- O THE LICENSEE ENTERED THE APPROPRIATE LCO.
- O UNIT 1 WAS SHUTDOWN AND UNIT 2 WAS RAMPED DOWN. UNIT 1'S STYLE "B" MODULES WERE TRANSFERRED TO THE UNIT 2 INSTRUMENT CHANNELS SO THAT UNIT 2 COULD RESUME FULL POWER OPERATION.

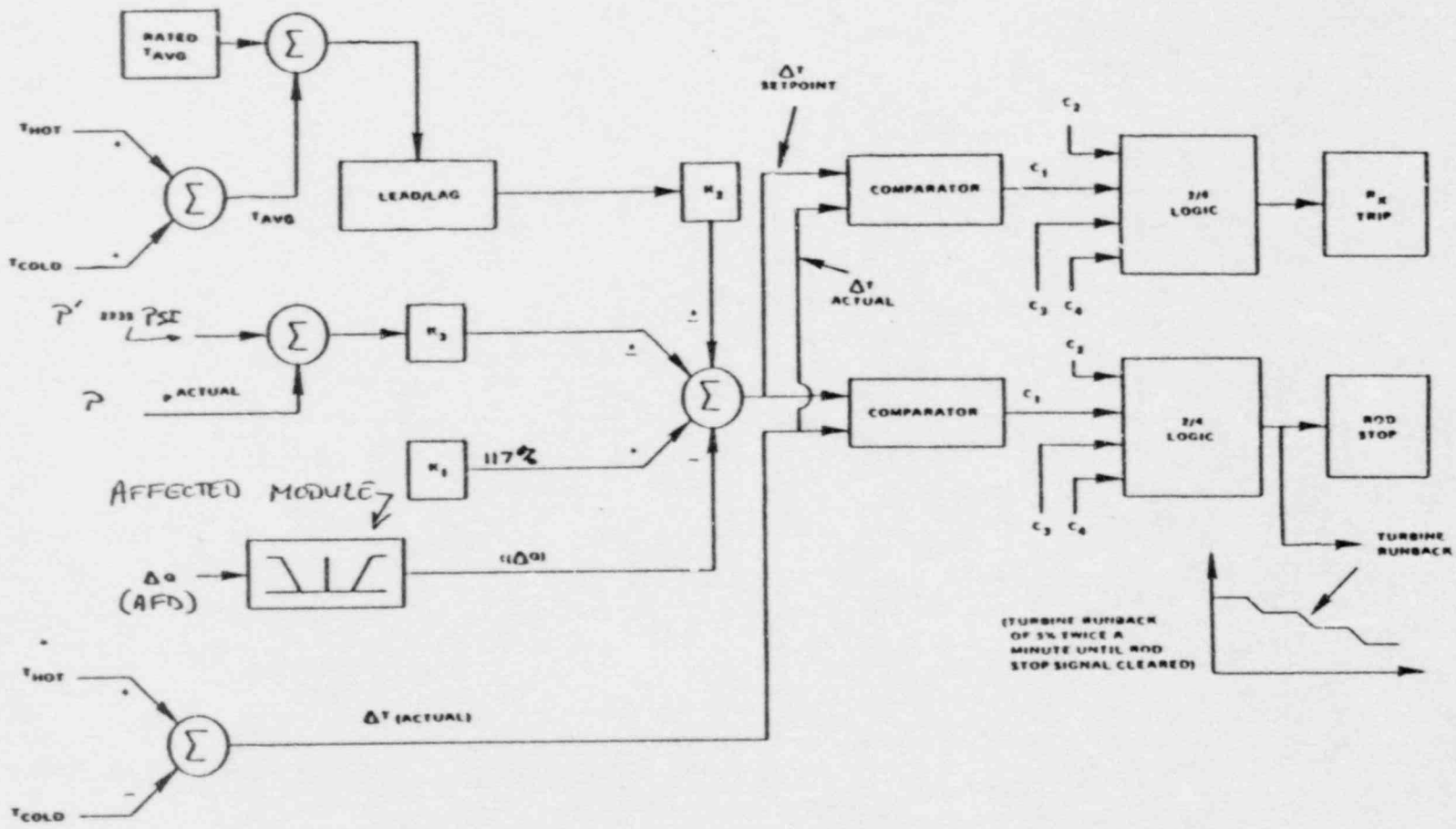
CONTACT: R. KARSCH
REFERENCE: 50.72 # 12973

- O WESTINGHOUSE HAD PREVIOUSLY ADVISED THE LICENSEE THAT STYLE "C" MODULES WERE AN ACCEPTABLE REPLACEMENT FOR STYLE "B" MODULES.
- O THIS PROBLEM OCCURRED AT KEWAUNEE IN 1974 AND WAS SOLVED BY A SIMPLE MODIFICATION TO THE MODULE'S WIRING. THIS CHANGE WAS NOT COORDINATED WITH EITHER THE NRC OR WESTINGHOUSE.
- O PRAIRIE ISLAND MODIFIED THEIR STYLE "C" MODULES AND RESTARTED UNIT 1 FOR THE REMAINDER OF THEIR COASTDOWN. RIII MONITORED MODIFICATION AND POST-MAINTENANCE TEST.
- O WESTINGHOUSE WAS NOTIFIED BY THE LICENSEE. WESTINGHOUSE WILL INFORM THE AFFECTED PLANTS TO TAKE CORRECTIVE ACTION (MAY BE AS FEW AS 3 OR AS MANY AS 6 PLANTS).
- O THE PRAIRIE ISLAND LICENSEE HAS INDICATED THAT THEY WILL ISSUE A PART 21 NOTICE AND NOTIFY INPO.

FOLLOWUP

POSSIBLE INFORMATION NOTICE.

$$\text{TRIP S.P.} = \Delta T_0 \left[K_1 - K_2 (T_{AVG} - T') \frac{1 + \tau_1 S}{1 + \tau_2 S} + K_3 (P - P') - f(\Delta q) \right]$$



Overtemperature ΔT Channel Block Diagram

PRAIRIE ISLAND UNITS 1 AND 2

ALMARAZ UNIT 1
STRESS CORROSION CRACKING SG-TUBES
JULY 13, 1988

PROBLEM
PRIMARY TO THE SECONDARY SIDE LEAK WAS DETECTED IN STEAM GENERATOR "A"; RAPID STRESS CORROSION CRACKING IN SG TUBES HAS BEEN FOUND.

CAUSE
CAUSTIC STRESS CORROSION CRACKING.

DISCUSSION

- THE UNIT IS A 900 MWE, W-TYPE, 3-LOOP PWR IN SPAIN.
- STEAM GENERATOR TYPE D3, MANUFACTURED IN PENSACOLA (USA).
- DURING REFUELING OUTAGE IN FEBRUARY 1988, 60% OF STEAM GENERATOR TUBES WERE INSPECTED AND NOTHING ABNORMAL WAS DETECTED.
- NEW FULL FLOW CONDENSATE POLISHING SYSTEM PUT INTO OPERATION THIS CYCLE.
- BETWEEN APRIL AND JULY 1988, THE CONCENTRATION OF SODIUM IONS WAS ABOVE 20 PPB AND REACHED 60 PPB.
- WESTINGHOUSE RECOMMENDS AN UPPER LIMIT OF 20 PPB FOR SODIUM CONCENTRATION.
- SODIUM ATTRIBUTED TO POLISHING SYSTEM (SODIUM HYDROXIDE IS USED TO CLEAN RESINS, SOME Na^+ REMAINED IN THE DEMINERALIZERS AND LATER FLUSHED OUT WHENEVER THE DEMINERALIZERS WERE PLACED IN SERVICE).
- ON JULY 13, 1988, A PRIMARY TO SECONDARY LEAK WAS DETECTED (1 GPM) AND PLANT WAS SHUTDOWN.
- IMMEDIATE INSPECTION OF ALL SG TUBES WAS INITIATED USING STANDARD BOBBIN PROBE-EDDY CURRENT AND ROTATING BOBBIN PROBE (FOR TUBES WITH INDICATION OF DEGRADATION).
- THE RESULTS OF THE INSPECTION TO DATE ARE AS FOLLOWS:

CONTACT: R. CID AND J. GUILLEN

STEAM GENERATOR	DEFECTIVE TUBES
"A"	106
"B"	235
"C"	160

DEFECTIVE: THOSE TUBES FOUND WITH GREATER THAN 20% TUBE WALL DEGRADATION.

- TWO TUBE SAMPLES WERE REMOVED FROM THE "A" SG (THE LEAKING TUBE AND ANOTHER WITH 90% TUBE WALL DEGRADATION).
- LEAKING TUBE
  - 1 CRACK, AXIALY ORIENTED, 1" LENGTH
  - 3/8" ABOVE TUBE SUPPORT PLATE
  - CRUD WAS DETECTED AROUND THE TUBE, 3" HEIGHT
- SECOND TUBE
  - 3 CRACKS, AXIALY ORIENTED,
  - LOCATED JUST AT THE SUPPORT PLATE
  - CRUD WAS NOT DETECTED
  - ONE SAMPLE HAS BEEN SENT TO W (PITTSBURGH) AND THE OTHER TO CIEMAT (SPAIN) FOR INDEPENDENT TESTING.
- THE CHARACTERISTIC OF THE DEFECTS FOUND ARE AS FOLLOWS:
  - ALL CRACKS ARE AXIALLY ORIENTED (1-5 CRACKS).
  - MOST ARE LOCATED AT THE FIRST SUPPORT PLATE, HOT LEG.
  - THE CRACKS ORIGINATED ON THE SECONDARY SIDE.
  - MOST OF THE CRACKS (90%) HAVE A DEPTH BETWEEN 60% AND 80% OF THE TUBE WALL.
- THE CAUSE OF THE CRACKING IS ATTRIBUTED TO LARGE CONCENTRATION Na^+ IN THE SECONDARY COOLANT.
  - THE Na^+ IS BELIEVED TO HAVE BEEN DEPOSITED IN THE HARDENED POROUS CRUD AND IN THE GAP BETWEEN THE TUBES AND THE SUPPORT PLATE.
- POLISHING SYSTEM TO REMAIN OFF-LINE UNTIL PROBLEM FULLY UNDERSTOOD.

<u>FOLLOWUP</u>

- CORRECTIVE ACTION IS TO PLUG THOSE TUBES FOUND WITH GREATER THAN 20% TUBE WALL DEGRADATION (501 TUBES).
- WESTINGHOUSE AND CIEMAT WILL PROVIDE METALLURGICAL REPORT ON THEIR EXAMINATION IN TWO WEEKS TO CONFIRM ROOT CAUSE.
- ACCIDENT ANALYSES WILL BE REANALYZED.

BRUNSWICK UNIT 2
ASCO PRESSURE SWITCHES
JULY 25, 1988

PROBLEM
POTENTIAL GENERIC PROBLEM WITH ASCO PRESSURE SWITCHES MODEL TG13A42.

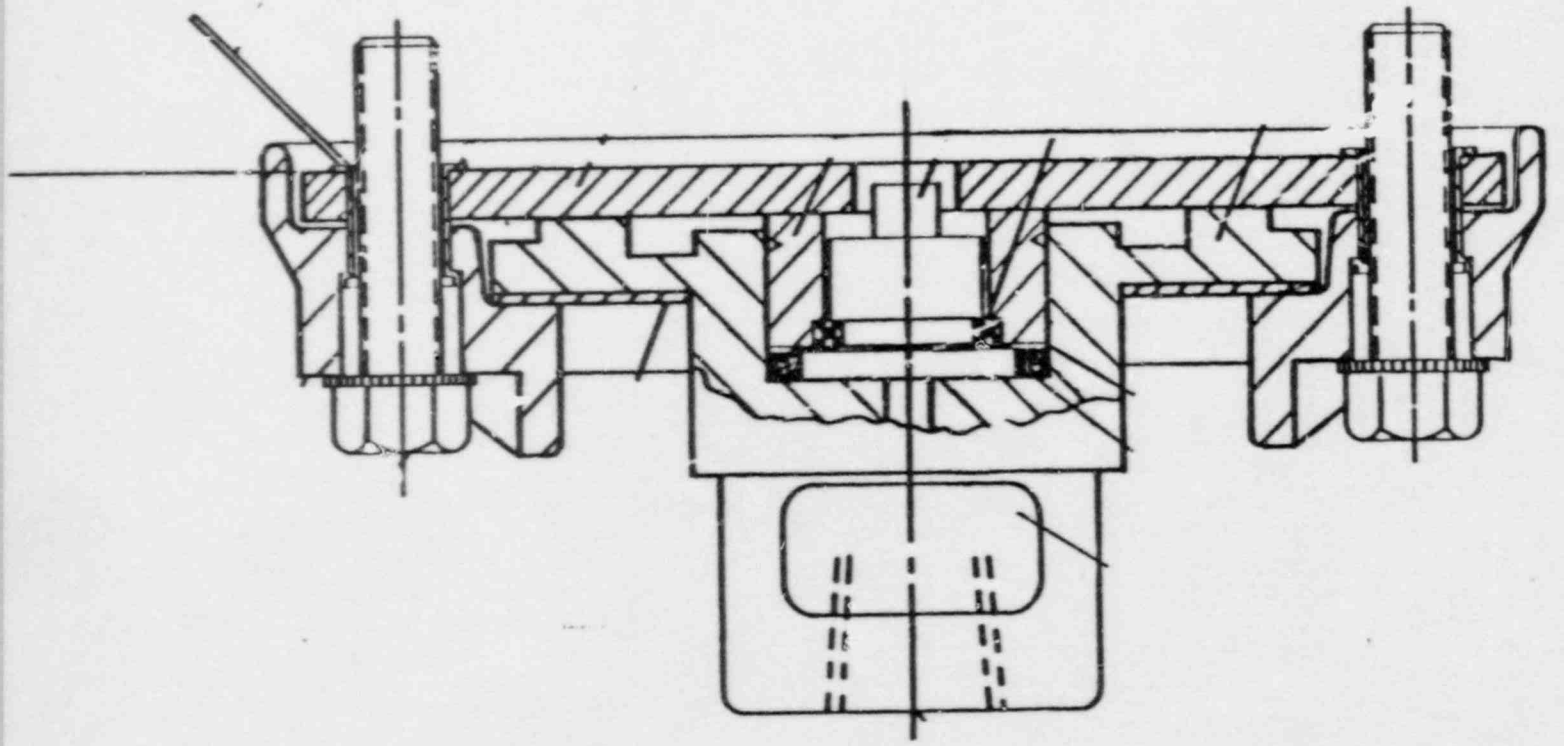
CAUSE
UNKNOWN

SAFETY SIGNIFICANCE
SWITCHES USED IN SAFETY SYSTEM (HPCS AND RCIC).

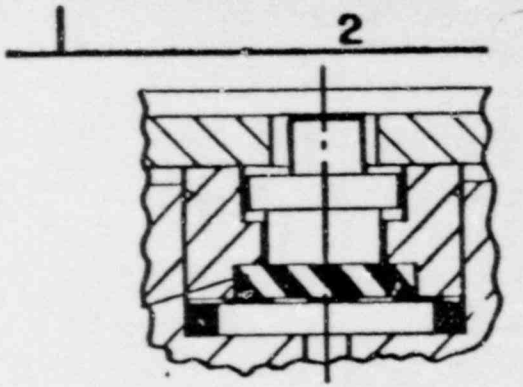
DISCUSSION
- o DISCOVERED WHEN RHR SHUTDOWN COOLING COULD NOT BE ESTABLISHED (INBOARD SUCTION VALVE WOULD NOT OPEN).
- o FOLLOWING CYCLING SWITCH FUNCTIONED PROPERLY.
- o ABOUT 44 SWITCHES PER UNIT INVOLVED OF WHICH 10 SWITCHES ARE SAFETY-RELATED.
- o LICENSEE REQUIRES BY TECH. SPEC. TO CHECK SAFETY SYSTEM ONCE PER MONTH.
- o LICENSEE PLANNING TO DISASSEMBLE SWITCHES TO DETERMINE FAILURE MODE.
- o UNKNOWN IF SIMILAR TO PROBLEM AT YANKEE-ROWE (DEC. '87) MODEL TL10A22 AND TM10A22.

FOLLOWUP
- o REGION II AND RVIB (VENDOR BRANCH) CONTINUING TO FOLLOW.
- o EAB TO DETERMINE IF AN INFORMATION NOTICE IS WARRANTED.

CONTACT: T. GREENE
REFERENCE: MORNING REPORT 07/26/88 AND 50.72 # 12955

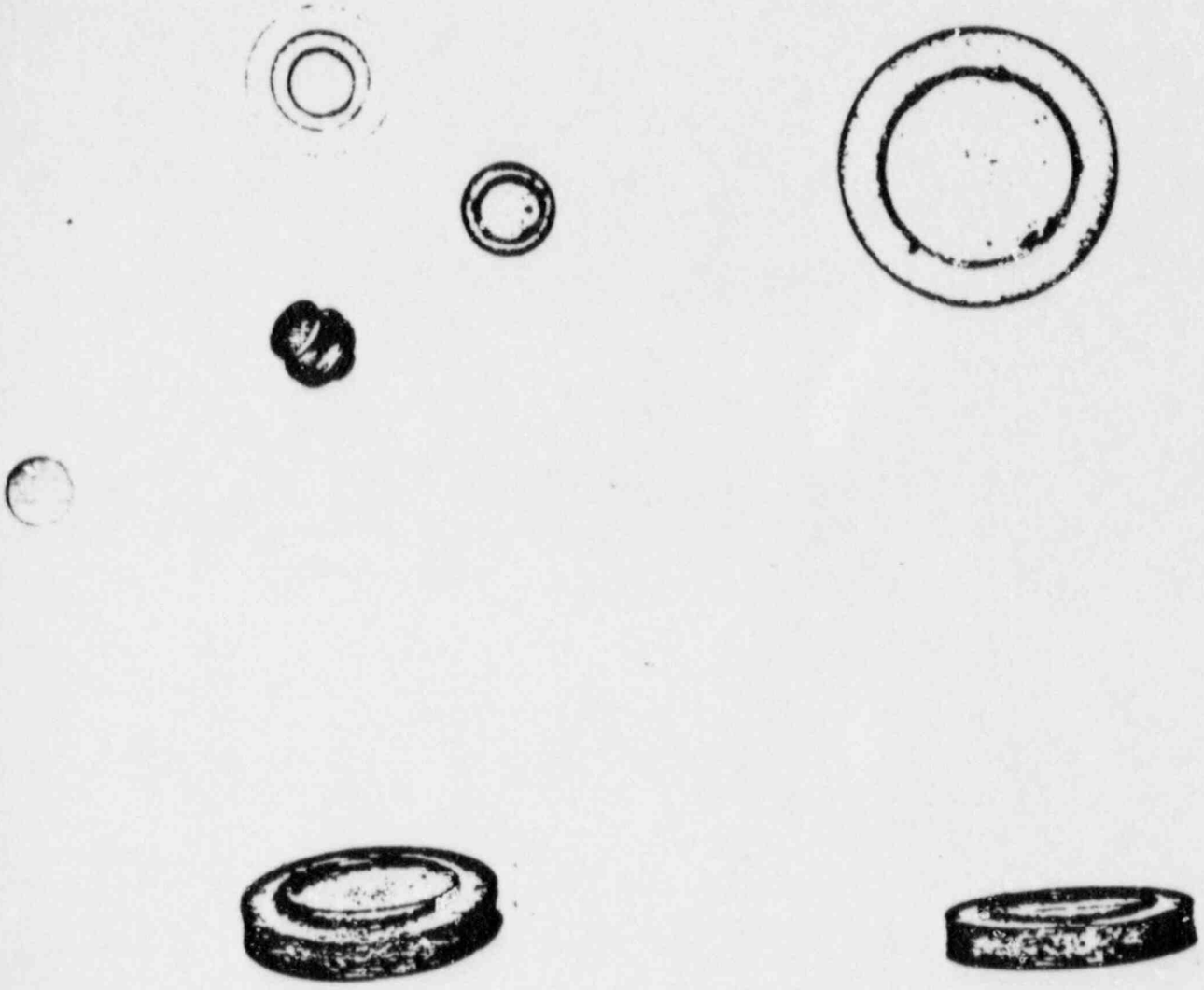


TYPICAL ASCO PRESSURE SWITCH AT YANKEE ROWE



VIEW SHOWING TL10A CONSTRUCTION

DISK DEFORMATION RESULTING FROM QUALIFICATION TEST



ARKANSAS UNIT 2
REACTOR COOLANT PUMP SEAL LEAK
AUGUST 1, 1988

PROBLEM

LEAKAGE OF REACTOR COOLANT PUMP SEAL PACKAGE.

CAUSE

MIDDLE SEAL PRESSURE SENSING LINE SHEARED.

SAFETY SIGNIFICANCE

REACTOR COOLANT LOSS TO CONTAINMENT.

DISCUSSION

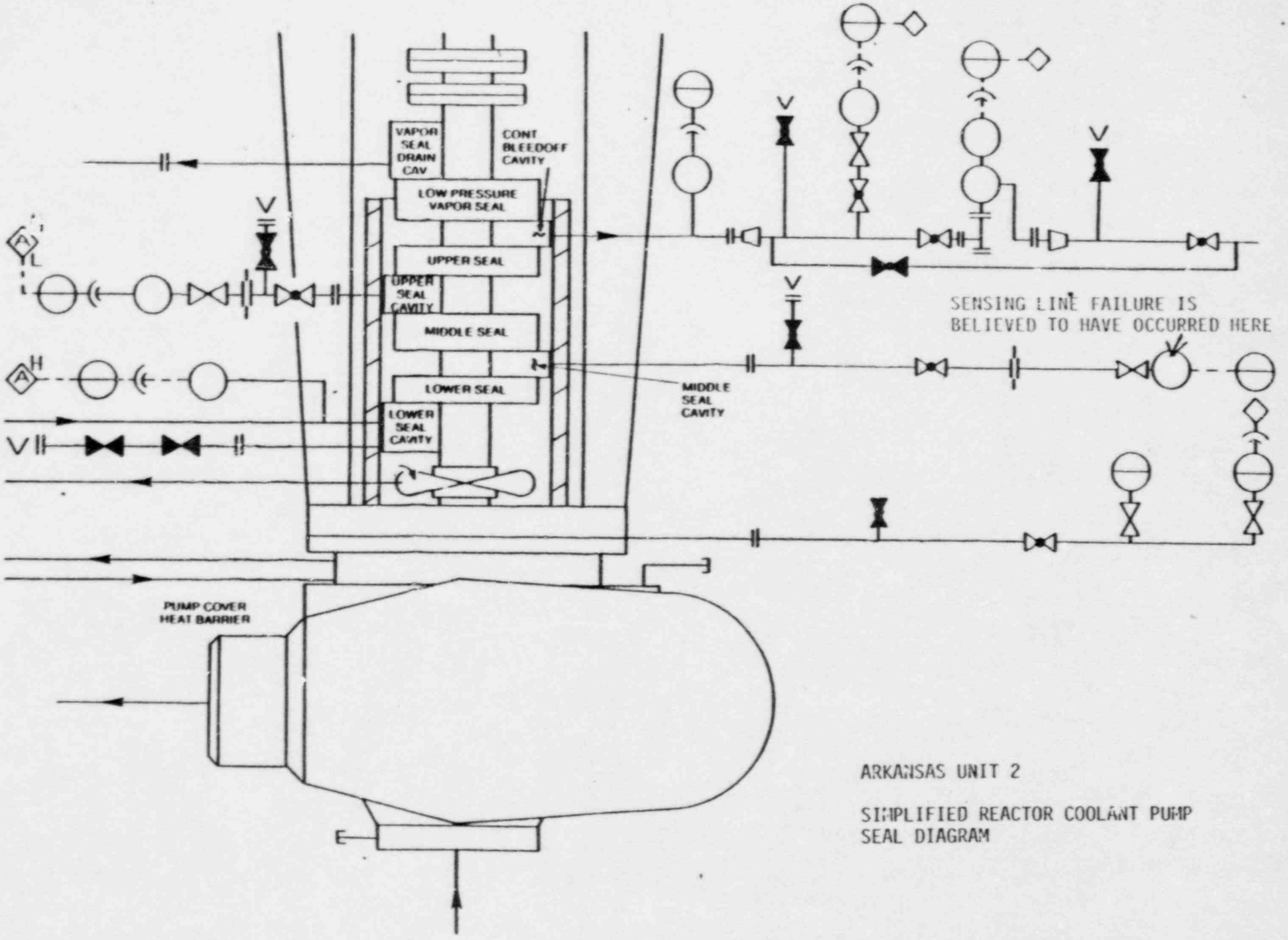
- ON AUGUST 1, 1988 AT 16:50 CDT HIGH/LOW CONTROLLED BLEEDOFF ALARM ON "A" RCP.
- DECREASING PRESSURIZER LEVEL, INCREASED SUMP LEVEL, INCREASED CHARGING FLOW, HIGH CONTAINMENT RADIATION.
- REACTOR MANUAL TRIP AT 17:00 CDT.
- LEAK RATE INCREASED FROM APPROXIMATELY 2 GPM TO 35 GPM BEFORE COOLDOWN.
- UNUSUAL EVENT DECLARED BY LICENSEE AT 17:25 CDT BASED ON LEAK RATE GREATER THAN 10 GPM.
- BY 19:28 CDT REACTOR COOLED TO 532^{O}F, DEPRESSURIZED TO 1550 PSIG LEAK INCREASED TO 40 PSIG.
- LEAK RATE DECREASED BY CONTINUED RCS COOLING AND DEPRESSURIZATION.
- CONTAINMENT ENTRY BY 00:38 CDT AUGUST 2, 1988.
- BROKEN SENSING LINE APPEARS TO HAVE CAUSED FAILURE OF MIDDLE, UPPER AND VAPOR SEALS FROM LACK OF FLOW.
- LOWER SEAL APPEARS TO HAVE FAILED FROM HIGH ΔP.
- RC PUMPS MANUFACTURED BY BYRON JACKSON, CONTROLLED LEAKAGE DESIGN WITHOUT SEAL INJECTION.

FOLLOWUP

LICENSEE AND NRR CONTINUING TO INVESTIGATE CAUSE OF PUMP SEAL FAILURE.

CONTACT: W. JENSEN

REFERENCE: 50.72 #S 13055 AND 13056



ARKANSAS UNIT 2

SIMPLIFIED REACTOR COOLANT PUMP SEAL DIAGRAM

ENCLOSURE 3

2

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SUGGESTED LONGTERM FOLLOWUP

DATE OF EVENT	PLANT NAME AND UNIT	SIGNIFICANT EVENT	INITIAL FOLLOWUP ASSIGNMENT	SUGGESTED RESOLUTION	SUGGEST TRANSFER TO:	EXPECTED COMPLETION DATE
05/22/88	DIABLO CANYON 1	.F.	INNER PURGE ISOLATION VALVES FAILED. CAUSE OF THE FAILURE? IS IT GENERIC?	PLANT STYSTEM BRANCH (NRR/PSLB) AND RV TO EVALUATE REVERSED PURGE VALVES. CONTACTS: W. ROOD, NRR/PD5; J. PULSIPER, NRR/PSLB;K. NAIDU, NRR/RVID; R. KARSCH, NRR/DOEA, OEAB. PM OBTAINED TWO TACs: TAC # 68344 IS ON DIABLO CANYON UNIT 1 AND TAC # 68345 IS ON DIABLO CANYON UNIT 2.	NRR/SPLB & RV	/ /

REACTOR SCRAM SUMMARY
WEEK ENDING 07/31/88

I. PLANT SPECIFIC DATA

DATE	SITE	UNIT	POWER	SIGNAL	CAUSE	COMPLI-CATIONS	YTD ABOVE 15%	YTD BELOW 15%	YTD TOTAL
07/26/88	SUMMER	1	100	A	PERSONNEL	NO	3	0	3
07/27/88	BEAVER VALLEY	2	100	A	PERSONNEL	NO	3	0	3
07/30/88	VOGTLE	1	100	A	EQUIPMENT	NO	6	0	6
07/30/88	SALEM	2	80	A	EQUIPMENT	NO	4	1	5
07/31/88	MCGUIRE	2	100	M	PERSONNEL	NO	2	0	2
07/31/88	VOGTLE	1	16	A	EQUIPMENT	NO	7	0	7

No. 1
/88

PERFORMANCE INDICATORS SIGNIFICANT EVENTS

PLANT NAME	EVENT DATE	EVENT DESCRIPTION	OTR SIGNIFICANCE
ARKANSAS 2	08/01/88	REACTOR COOLANT PUMP SEAL LEAKAGE.	0 UNEXPECTED PLANT RESPONSE TO A SET OF CONDITIONS