Duke Power Company
McGuire Nuclear Station
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

(704)875-4000

DUKE POWER

April 8, 1993

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: McGuire Nuclear Station Unit 2
Docket No. 50-370
Licensee Event Report 370/93-02

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report 370/93-02 concerning a Unit 2 Manual Reactor Trip. This report is being submitted in accordance with 10 CFR 50.73 (a) (2) (iv). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

TCMcMeekin

T.C. McMeekin

TLP/bcb

Attachment

xc: Mr. S.D. Ebneter
Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta St., NW, Suite 2900
Atlanta, GA 30323

INPO Records Center
Suite 1500
1100 Circle 75 Parkway
Atlanta, GA 30339

Mr. Tim Reed
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Mr. P.K. Van Doorn
NRC Resident Inspector
McGuire Nuclear Station

JE22

LICENSEE EVENT REPORT (LER)

FACILITY NAME(1)	DOCKET NUMBER(2)	PAGE(3)
McGuire Nuclear Station, Unit 2	05000 370	1 OF 13

TITLE(4) A Unit 2 Manual Reactor Trip was Initiated Due to an Equipment Failure.

EVENT DATE(5)			LER NUMBER(6)			REPORT DATE(7)			OTHER FACILITIES INVOLVED(8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
										05000
03	09	93	93	02	0	04	09	93		05000

OPERATING MODE(9)	1	THIS REPORT IS SUBMITTED PURSUANT TO REQUIREMENTS OF 10CFR (Check one or more of the following)(11)							
			20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
POWER LEVEL(10)	100%		20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
			20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER (Specify in Abstract below and in Text)
			20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
			20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER(12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
Terry L. Pedersen, Manager, Safety Review Group	704	875-4487

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT(13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
F	CF	BE	M430	Y						

SUPPLEMENTAL REPORT EXPECTED(14)		EXPECTED SUBMISSION DATE(15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines (16)

A Unit 2 manual Reactor Trip was initiated by Operations Control Room personnel after receipt of alarms indicating a low and decreasing feedwater (CF) flow to Steam Generator (SG) 2B. Control Room personnel assumed manual control of valve 2CF-23, Steam Generator 2B Main Feedwater Control Valve, and attempted to increase CF flow to SG 2B. The attempt was unsuccessful. As the SG water level approached 45 percent, the Senior Reactor Operator instructed the Reactor Operator at the Controls to manually exercise Reactor Trip Breakers A and B. The Reactor was tripped at 0416 on March 9, 1993. The manual Reactor trip was followed by an automatic Turbine Trip. Operations personnel immediately began recovery efforts. Unit 2 was operating in Mode 1 (Power Operation), at 100 percent power at the time of the event. An investigation revealed that the inner metallic bellows in the valve positioner relay for valve 2CF-23 failed as a result of fatigue cracking, causing the valve to close resulting in a loss of CF flow to SG 2B. Corrective actions included the replacement of all Unit 2 CF valve positioner relay bellows with relays which incorporate viton elastomer diaphragms as opposed to metallic bellows.

EVALUATION:

Background

The Main Feedwater (CF) system [EIIS:SJ] supplies feedwater at the required temperature, pressure and flow rate to the Steam Generators (SGs) [EIIS:SG] to maintain the proper water levels with respect to Reactor [EIIS:RCT] power output and Turbine [EIIS:TRB] steam requirements.

Valve 2CF-23, Steam Generator 2B Main Feedwater Control Valve, is one of four (one per SG) air diaphragm SG CF Control Valves [EIIS:FCV]. In the automatic mode of operation, the valve is regulated by the Feedwater Control system using feedwater flow, steam generator water level, and steam flow as control parameters. In the manual mode of operation, the valve is controlled by the Control Room [EIIS:NA] Operator with the manual loader provided on the control board [EIIS:MCBD].

The valve positioner on the Feedwater Control Valves is used to position the valve dependent on the requirements of the Feedwater Control system.

Moore Products, Model 72P315 valve positioners are currently in use at this facility. This valve positioner uses a metal bellows type relay to compare the actual valve position to the required position based on monitored plant parameters. If an error exists, the relay will change its output to reposition the valve to the correct position.

Description of Event

On March 9, 1993, at approximately 0416, with Unit 2 operating in Mode 1 (Power Operation), at 100 percent power, and all systems in automatic, Operations (OPS) Control Room personnel received annunciator [EIIS:ANN] alarm [EIIS:ALM] 2AD-4 C2 "SG B Flow Mismatch Lo CF Flow". The Reactor Operator at the Controls (ROATC) noted that SG 2B feedwater flow was rapidly decreasing. The ROATC immediately placed the valve controller in manual and attempted to open the valve. The demand indicator for valve 2CF-23 reflected a full open position; however, CF flow to SG 2B continued to decrease. OPS Control Room personnel received alarms at 2AD-4 B2, "SG 2B Level Deviation" and 2AD-4 E2, "SG 2B Lo Level Alert". The Senior Reactor Operator (SRO) instructed the ROATC to trip the Reactor if the SG level approached 45 percent. As the SG level approached the 45 percent level, the ROATC manually exercised the Reactor Trip Breakers [EIIS:52] at 0416. The manual Reactor Trip was followed by an automatic Turbine Trip. OPS Control Room personnel immediately implemented procedures EP/2/A/5000/01, Reactor Trip Or Safety Injection, and EP/2/A/5000/1.3, Reactor Trip. The Motor Driven Auxiliary Feedwater (CA) pumps automatically started as required, at 0416:02. All plant parameters were essentially stable within 30 minutes after the trip. Steam Generator 2B level deviated

significantly 30 minutes after the trip due to valve 2CF-23 closure; however, stable no load conditions for SG 2B were established within 2 hours. Notification was made to the NRC by the Shift Supervisor at 0527, on March 9, 1993, in accordance with procedure RP/0/A/5700/10, NRC Immediate Notification Requirements.

Instrument and Electrical (IAE) personnel investigating the malfunction of valve 2CF-23 reported that they heard and felt air escaping from the valve positioner manifold. Unit 2 had experienced a similar manual Reactor Trip on February 22, 1993. Based on the detection of escaping control air and the previous Reactor Trip, IAE personnel removed the positioner relay bellows and forwarded it to the Duke Power Analytical and Predictive Technologies Metallurgical Laboratory for failure analysis (Metallurgical Analysis Report attached).

Emergency work orders 93018840, 93018841,93018699 and 93018842 were generated to remove the bellows on all Unit 2 CF Control valves and replace them with Moore Products, model 721P315 bellows which incorporate viton elastomer diaphragms as opposed to metallic bellows. The necessary components were acquired from Catawba Nuclear Station and installed in the CF system on March 9, 1993. The unit returned to Mode 1 operation at 0915 on March 10, 1993.

Conclusion

A cause of Equipment Failure has been assigned to this event because the control air sent to the valve positioner relay was escaping through a fatigue crack in the bellows wall thereby allowing valve 2CF-23 to drift closed.

The Root Cause investigation associated with valve 2CF-23 revealed that IAE personnel, who inspected valve 2CF-23 in the field shortly after the trip, noted air escaping from the area between the positioner case and the bellows assembly. This was indicative of a leaking bellows. Since the fail safe position of the positioner is closed on a loss of signal air, the leak by of the bellows was seen by the pneumatic relay as a "close" signal. Because the bellows was unable to pressurize, the required 15 psi "open" signal could not be attained and the valve continued to close. Upon initiation of the manual Reactor Trip and subsequent CF Isolation, valve 2CF-23 moved to its fully closed position.

A review of the Operating Experience Program (OEP) Database for the twenty-four months prior to this event revealed two Reactor Trips that resulted from failed CF valve controllers. License Event Report (LER) 370/92-04 was generated in response to a Unit 2 Reactor Trip that occurred when the positioner linkage for 2CF-20, Steam Generator 2C Main CF Control Valve, failed. LER 370/93-01 investigated a Unit 2 Reactor Trip that occurred on February 22, 1993. The unit was manually tripped when valve 2CF-20 failed to maintain its desired position causing a low CF flow to SG 2C.

The corrective actions formulated for LER 370/92-04 were specific to that event; however, the corrective actions drafted for LER 370/93-01 would have likely precluded the current event, but insufficient time between the two trips prevented full implementation of the corrective actions. Based on the search data, the event addressed in this LER is determined to be recurring.

This event is Nuclear Plant Reliability Data System (NPRDS) reportable due to the fatigue cracking of the bellows in the positioner for valve 2CF-23.

There were no personnel injuries, radiation overexposures, or uncontrolled releases of radioactive material resulting from this event.

CORRECTIVE ACTIONS:

Immediate: Operations personnel implemented procedures EP/2/A/5000/01 and EP/2/A/5000/1.3 and stabilized the unit in Mode 3 (Hot Standby).

Subsequent:

1) Emergency work orders 93018840, 93018841, 93018842, and 93018699 were initiated by OPS personnel to replace all Unit 2 CF Control valve positioner metallic bellows (Moore Products, Model 72P315) with Moore Products, Model 721P315 viton elastomer diaphragms.

2) IAE personnel replaced all Unit 2 CF valve positioners as described above.

3) Personnel working with the Secondary Valve Reliability Team have accelerated their review of past performance problems associated with the CF valves to identify probable causes (see LER 370/93-01). The findings and resulting corrective actions stemming from this team will be tracked under PIP 2M-93-0147.

Planned: The CF Control Valve pneumatic relay incorporating metallic bellows currently in use in McGuire Unit 1, will be replaced with Moore Products viton elastomer diaphragms during EOC8.

SAFETY ANALYSIS:

During this event, the valve positioner associated with valve 2CF-23 failed to maintain its desired position causing a decrease in SG 2B feedwater flow resulting in a decrease in SG 2B level, requiring valve 2CF-23 to open. Valve 2CF-23 did not open on demand and the control circuitry continued to send an "open" signal to the positioner in an effort to maintain feedwater flow. The valve failed to respond properly because the control instrument air sent to the valve positioner was escaping though the crack in the bellows on the pneumatic relay. The control circuitry performed as designed. As the level in SG 2B approached the 40 percent setpoint, the Reactor Trip Breakers were manually exercised in order to avert a challenge to the plant safety systems. This event is bounded by Turbine Trip found in Chapter 15 of the McGuire Final Safety Analysis Report.

Primary and Secondary system no-load conditions necessary to achieve a safe shutdown were attained within 2 hours after the Reactor trip. This event presented no hazard to the integrity of the plant; therefore, the health and safety of the public were not affected by this event.

ADDITIONAL INFORMATION:

Sequence of Events:

PTR - Post Trip Report
SSL - Unit 2 Shift Supervisors Logbook
WO - Work Order
ER - Unit 2 Events Recorder
PR - Personnel Recollection
OAC - Operator Aid Computer

Date	Time	Event
3/9/93	~0416	Operations Control Room personnel received SG 2B Flow mismatch Lo CF Flow alarm. (PR) Operations Control Room personnel received SG 2B Level Deviation alarm. (PR)
3/9/93	~0416	The ROATC assumed manual control of 2CF-23. (PR)
3/9/93	0416:00.743	The ROATC opened Unit 2 Reactor Trip Breaker A. (PTR) (ER)
3/9/93	0416:00.748	The ROATC opened Unit 2 Reactor Trip Breaker B. (PTR) (ER)
3/9/93	0416:00.856	Unit 2 Turbine tripped automatically following manual Reactor Trip. (PTR) (ER)
3/9/93	0416:58.51	Unit 2 experienced a Feedwater Isolation. (PTR) (ER)
3/9/93	0416:02	MDAFWP A automatically started. (PTR)
3/9/93	0416:03	MDAFWP B automatically started. (PTR)
3/9/93	------	Operations personnel generated repair work orders. (WO)
3/9/93	0527	Notification was made to the NRC by the Shift Supervisor. (PR) (SSL)

Sequence of Events, Cont.

Date	Time	Event
3/9/93	-------	Instrument and Electrical personnel replaced all Unit 2 CF regulating valve positioner metallic bellows (Moore Products, Model 72P315) with Moore Products, Model 721P315 viton elastomer diaphragms. (WO)
3/10/93	0915	Unit 2 entered Mode 1. (SSL)

APPLIED SCIENCE CENTER

Metallurgical Analysis Report

Sample No.: 1446 **Station:** McGuire **Unit:** 2
Requestor/Dept.: Jeff Miller - MNS
Principal Investigator: Sue Anderson
Submitted To: Jeff Miller **Date:** 3/11/93
cc: Donna Keck - NGD Nuclear Services
John Washam - MNS Safety Review

Equipment Description:
Bellows-type relay from Moore valve positioner on 2CF23

Background Information:
Feedwater regulation valve 2CF23 failed closed, leading to a manual unit trip. The relay in the valve positioner was determined to have failed. This relay had been replaced two weeks earlier, when the corresponding relay from 2CF20 failed and caused a unit trip. A failure analysis was requested.

Description/Macro-Examination:
A slight amount of white powdery material was smeared around the control air port, but the port was not partially blocked. The stamp on the manifold block was "MP 29". No foreign material was heard inside the relay when shaken.

Prior to disassembling the relay to examine the internal bellows, a vertical crack was noticed in the shield (Figure 1). The crack was 0.6" in length, with the first 0.2" near the solder joint filled with solder (Figure 2). The first half of the crack coincided with a scratch mark, several of which were found on the shield of every relay examined.

The shield and outer bellows were cut off through the solder joints and around the top cap. No cracking was seen in the outer bellows. The inner bellows was cracked below the second convolution from the bottom, the same location as in Report #1436 (Figure 3). The crack covered ~140° of the circumference.

The interior of the outer bellows and the base plate around the inner bellows were generally clean, not covered with white powder or excess solder flux as before. A few patches of white powder were seen, apparently flux which had spilled and did not melt during soldering. The inner bellows was generally clean also, with a few small splotchy brown to green-blue stains. None were obviously located at the center of the crack (Figure 4).

Fractography:
The bellows crack was pulled apart to examine the fracture morphology. Fatigue striations were identified both near the crack tip and near the center of the fracture (Figure 5). The striation spacing was very fine, typical of vibration or high-cycle loading. Crack propagation was from exterior to interior.

The crack in the shield was broken open and also examined in the scanning electron microscope. The morphology of the visible portion (not covered by solder) was entirely intergranular.

Metallography:
A cross-section was taken through the center of the bellows crack. No secondary or parallel cracking was seen around the fracture (Figure 6). The crack path was transgranular.

The shield fracture was sectioned through both the solder-covered region and near the crack tip. The crack path was confirmed to be intergranular, and minor secondary cracking was seen.

Chemistry/Mechanical Testing:
No bulk quantitative or EDS qualitative chemical analysis was performed on the bellows or the patches of white deposit.

Conclusions:

The crack in the inner bellows of the 2CF23 valve positioner relay was attributed to fatigue cracking.

No stress corrosion cracking was observed around the origin area of the bellows crack. Stress corrosion cracks appeared to have initiated the fatigue crack which failed the 2CF20 bellows earlier. Geometrical and metallurgical stress concentrations may have been sufficient to initiate a fatigue crack in the short service time under vibrational stress.

No excessive solder flux was present, as was discovered in the relay from 2CF20. The solder joints were relatively clean. A small amount of white powder, probably unmelted flux, was present and was drawn into the control air port.

The vertical crack in the outer shield was attributed to liquid-metal embrittlement of the brass by the lead-tin solder. This crack occurred during manufacture, as solder was present in the crack. Brasses are susceptible to LME by lead and tin alloys even at very low stresses. The scratch which the crack followed was a forming artifact and likely a site of residual stress in the shield. The presence of the crack did not affect the operation of the relay.

This failure is the latest in a series of incidents in the valve positioner relays involving fatigue cracking. Continued use of a device of this design should result in more failures under the alternating stresses being imposed. A relay design using a viton elastomer diaphragm has been substituted for the metal bellows design. In the case that these replacement devices might also be sensitive to fatigue damage, the source of the alternating stresses should be pursued and corrected.

If the Metallurgy Lab can be of further assistance, please call me at (704) 875-5326.

Approved by: S Anderson Date: 3-16-93

Reviewed by: CR Jaye 3-17-93

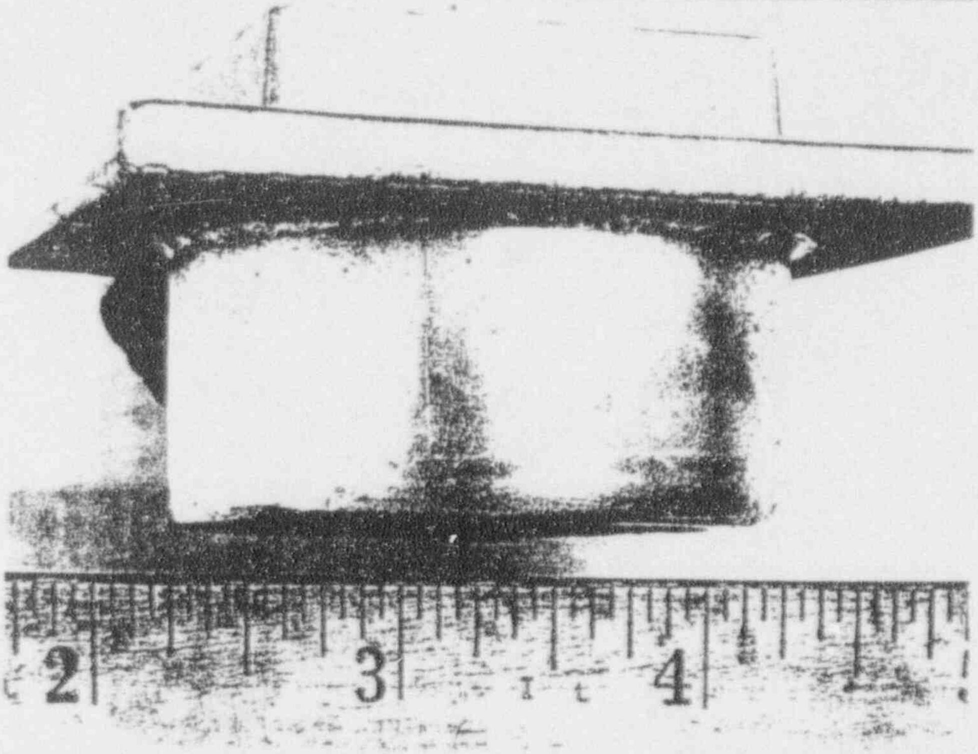

Figure 1 Vertical crack in outer shield of relay, as received. Portion of crack near solder joint is filled with solder. Ma-2513.

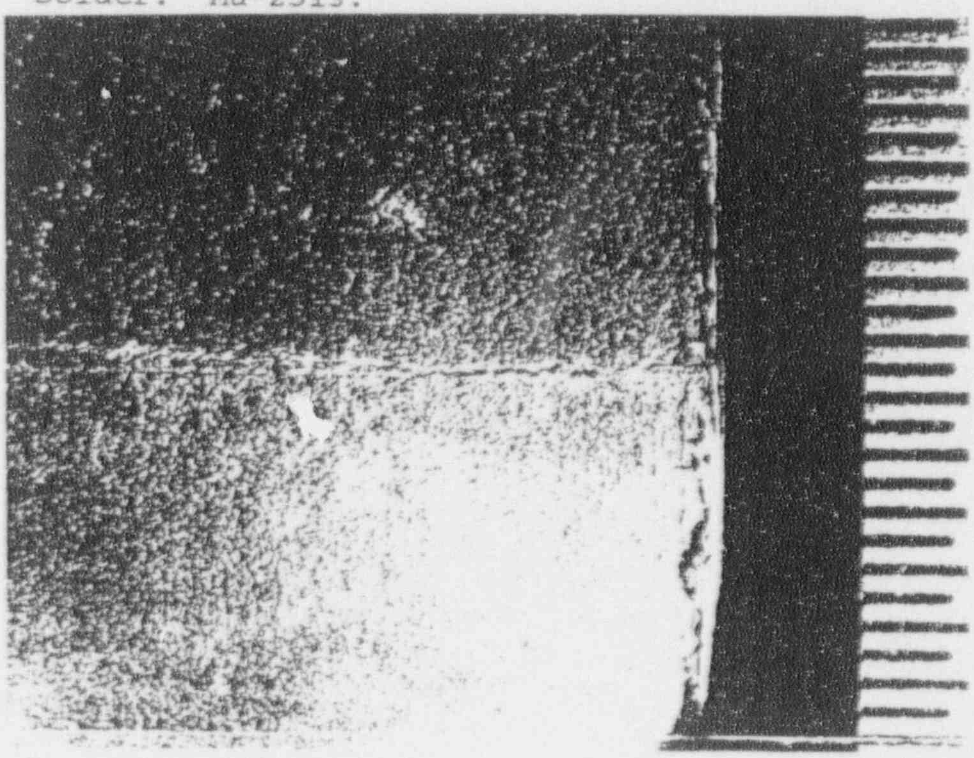

Figure 2 Detail of crack in shield after shield was removed. Solder is visible in crack. Scratch mark which crack followed is visible at left. Gauge - 1/64". Ma-2514.

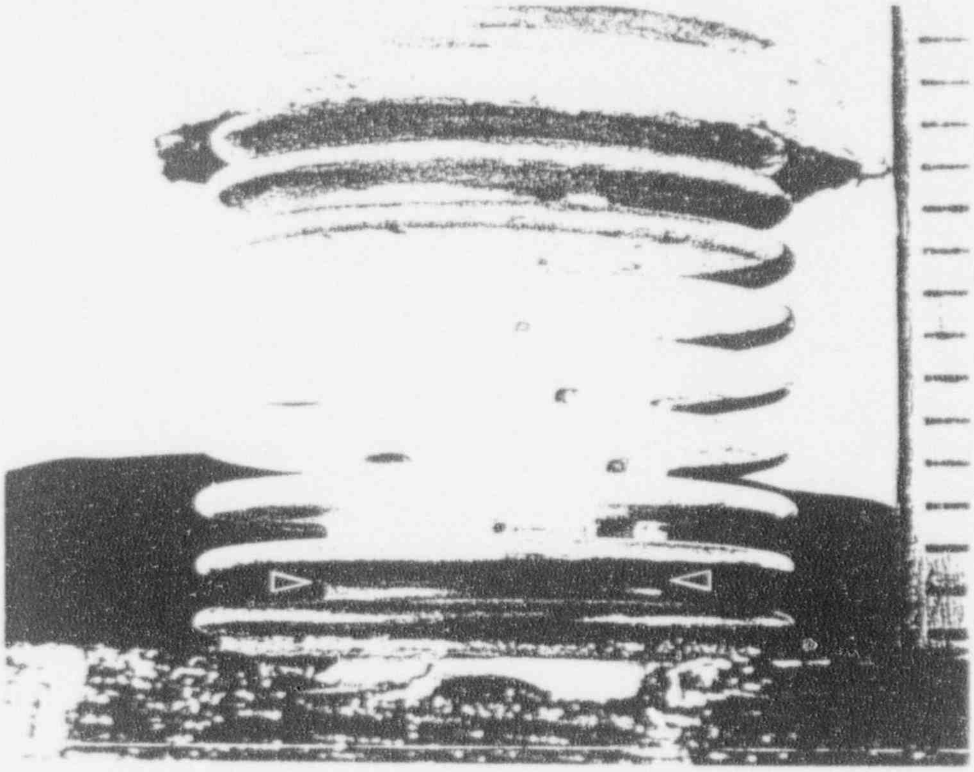

Figure 3 Inner bellows revealed by sectioning. Crack is visible under second convolution from bottom (arrows). Gauge = 1/16". Ma-2515.

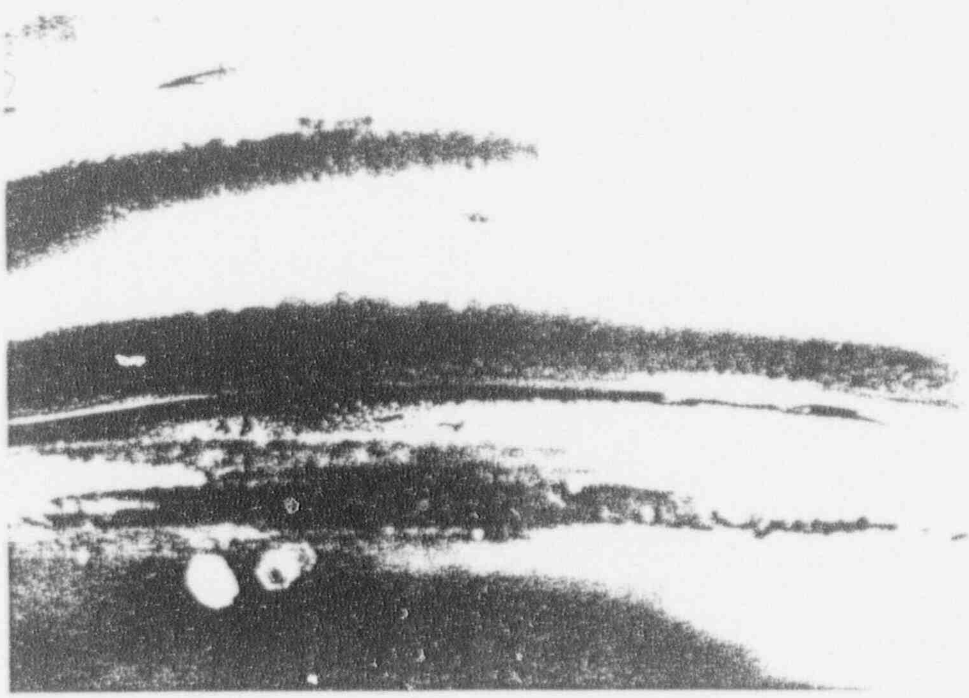

Figure 4 Detail of crack from Figure 3. Light stains are present but are not located near the center of the crack. (~9X) Ma-2516.

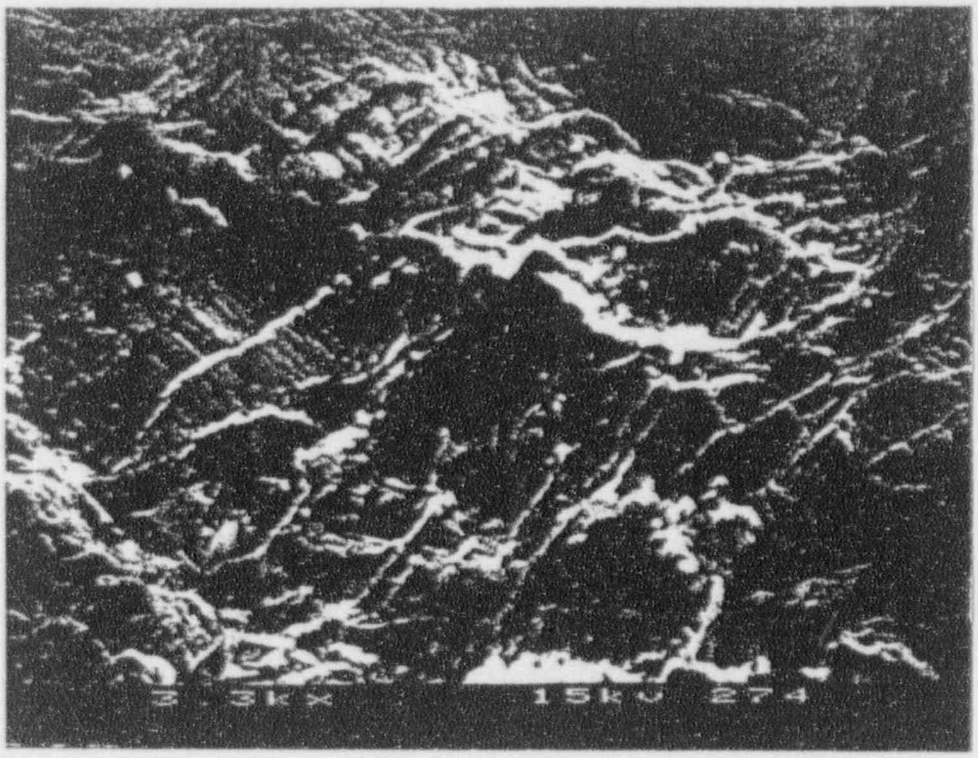

Figure 5 SEM photo of fatigue striations near center of bellows crack. Crack traveled from OD to ID (upper right to lower left in photo). (3300X) S-1904.

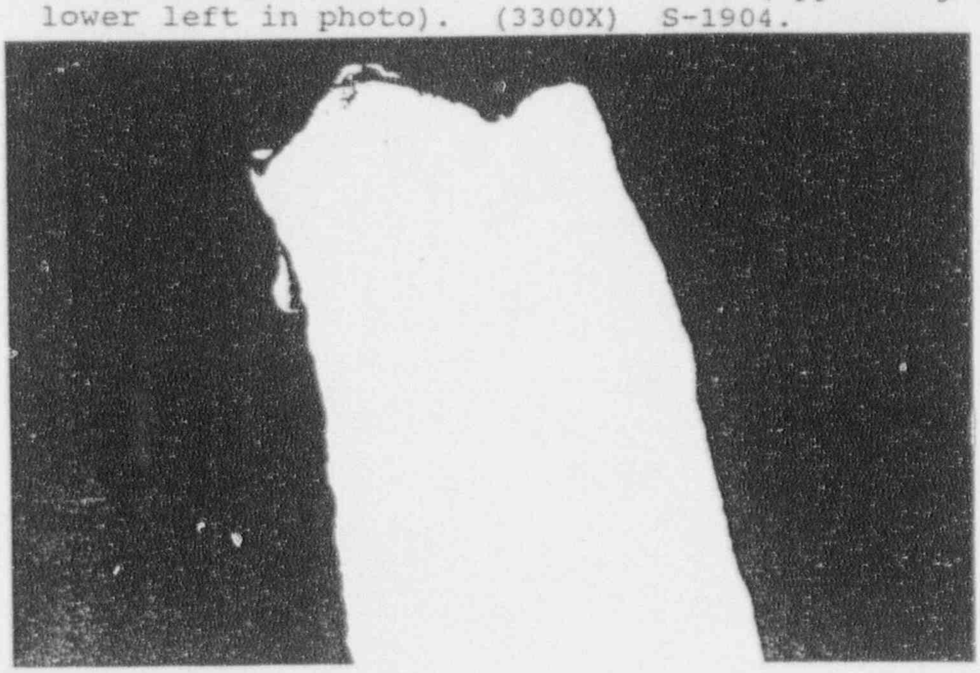

Figure 6 Cross-section through bellows, near center of crack. No parallel cracking visible near main fracture (top); minor 2ª crack is mechanical damage. (500X) Mi-2637.

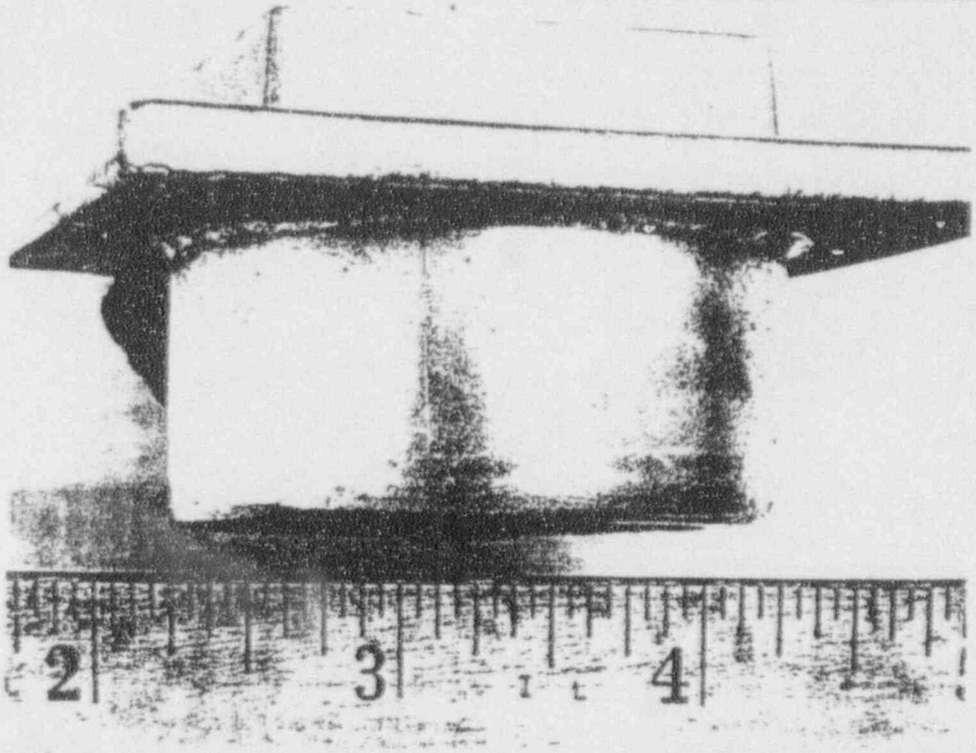

Figure 1 Vertical crack in outer shield of relay, as received. Portion of crack near solder joint is filled with solder. Ma-2513.

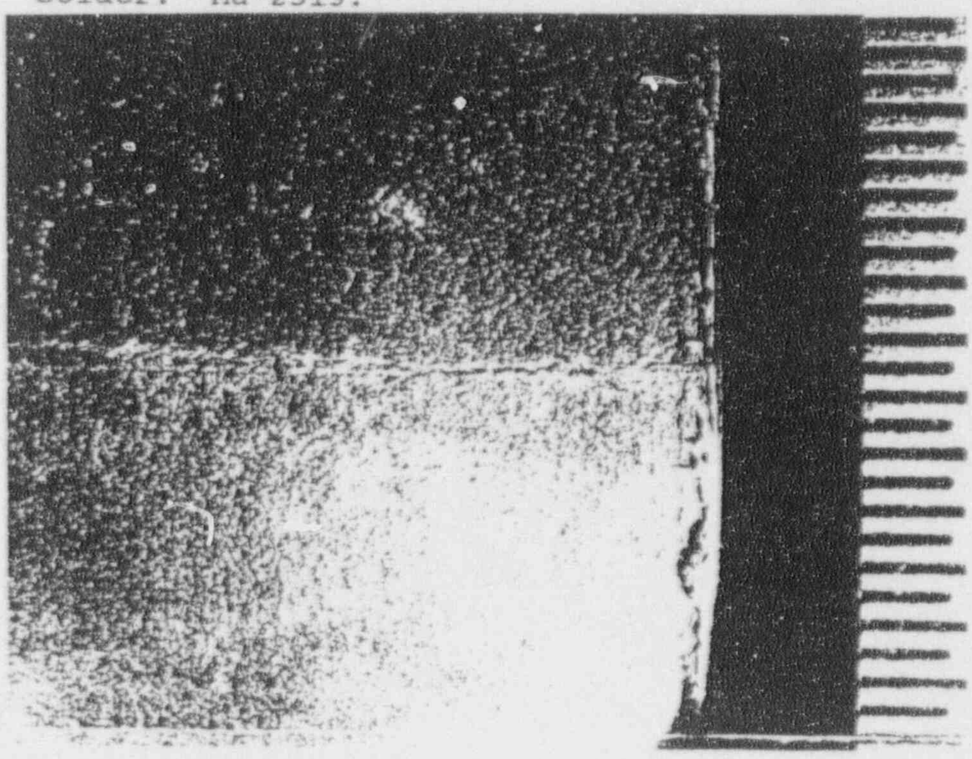

Figure 2 Detail of crack in shield after shield was removed. Solder is visible in crack. Scratch mark which crack followed is visible at left. Gauge - 1/64". Ma-2514.

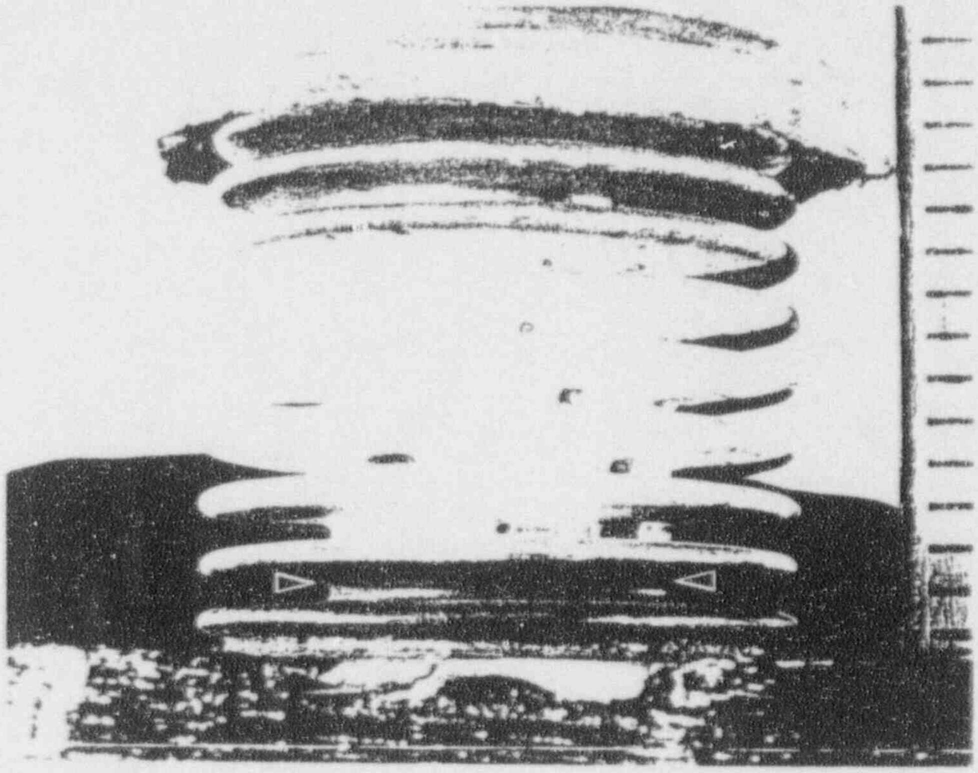

Figure 3 Inner bellows revealed by sectioning. Crack is visible under second convolution from bottom (arrows). Gauge = 1/16". Ma-2515.

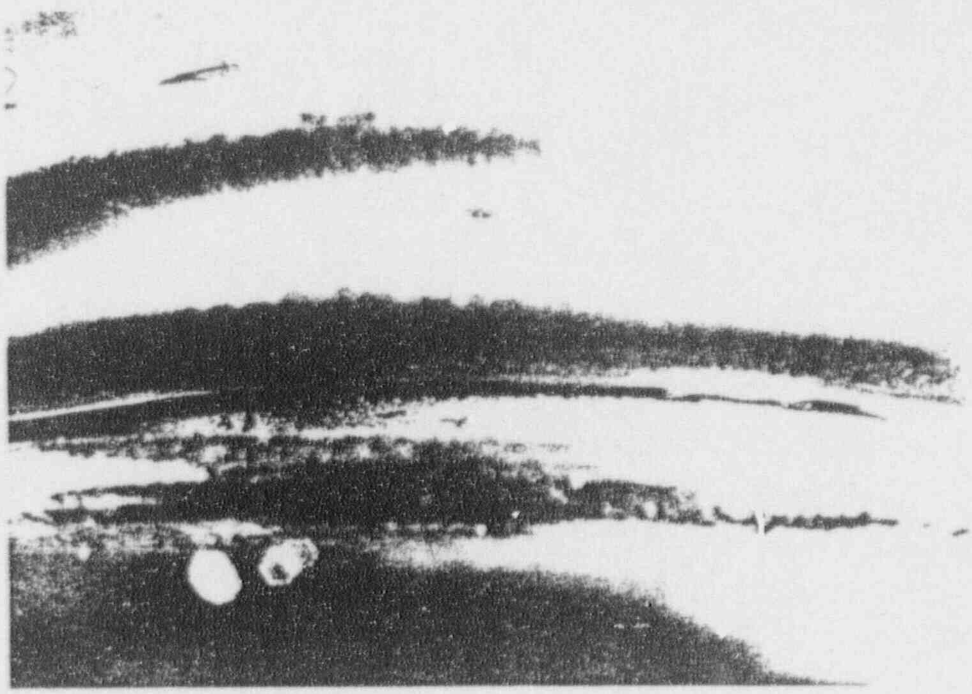

Figure 4 Detail of crack from Figure 3. Light stains are present but are not located near the center of the crack. (~9X) Ma-2516.

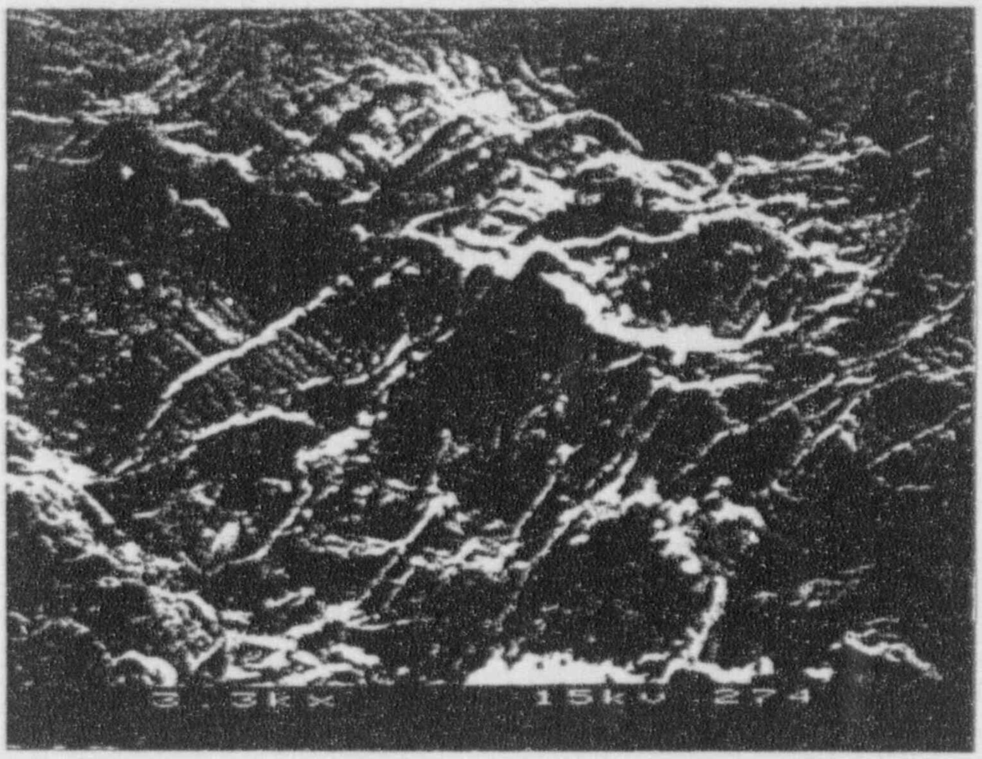

Figure 5 SEM photo of fatigue striations near center of bellows crack. Crack traveled from OD to ID (upper right to lower left in photo). (3300X) S-1904.

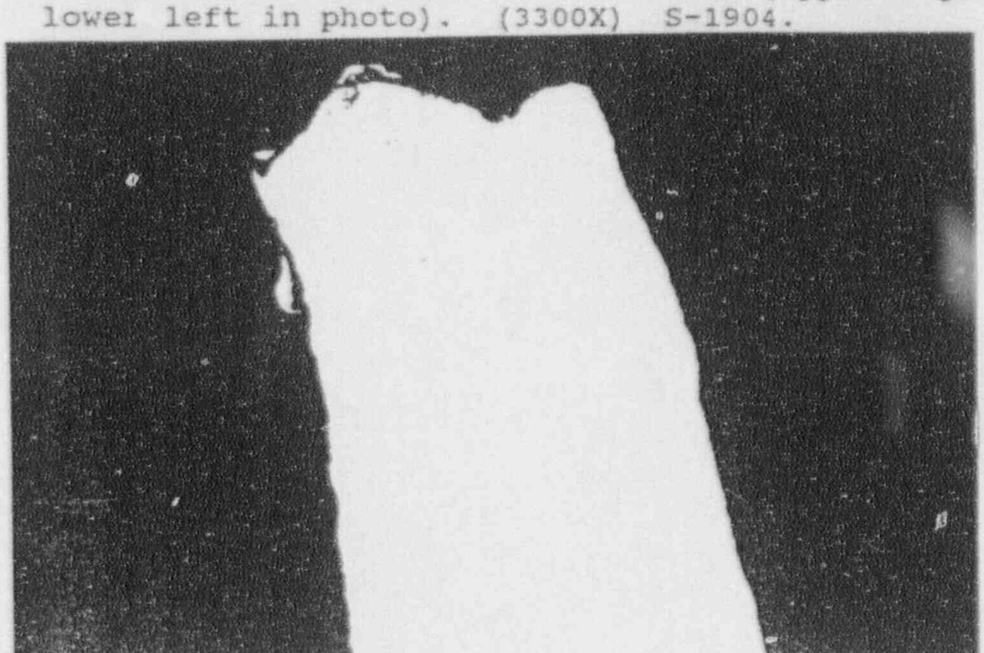

Figure 6 Cross-section through bellows, near center of crack. No parallel cracking visible near main fracture (top); minor 2° crack is mechanical damage. (500X) Mi-2637.