DEC 2 4 1980

50-206

MEMORANDUM FOR: Carlyle Michelson, Director
Office for Analysis and Evaluation
of Operational Data

FROM: Earl J. Brown
Office for Analysis and Evaluation
of Operational Data

SUBJECT: INTERNAL APPURTENANCES IN LWR's

An AEOD study was recently initiated as a follow-up to previous ACRS concerns about degradation of internal appurtenances in LWR piping during normal plant operation and the potential effects when degraded appurtenances are subject to accident loads. The primary concerns are that accident loads could result in failure of devices already degraded during normal service such that mitigation of the accident would be adversely affected or that continued degradation of these devices during normal service may in itself lead to an accident. The purpose of this memorandum is to provide a brief background summary of the initial ACRS inquiry and identify additional events where appurtenances were degraded during normal operation so that the information will be available to the responsible program office as appropriate.

BACKGROUND

The initial inquiry was raised in your letter dated June 13, 1979 (See enclosure 1) to ACRS members. The subject was failure of a feedwater flow straightener as San Onofre Unit 1 as reported in LER 78-013. NRR staff subsequently discussed this with ACRS as part of the general subject of seismic or blowdown capability of small components within the reactor coolant system. A specific NRR staff response to the ACRS on the San Onofre LER 78-013 was to have been handled by letter after the meeting, but there is apparently no documentation of a response.

EVENTS ASSOCIATED WITH INTERNAL APPURTENANCES

A. San Onofre, Unit 1

Several events involving feedwater flow straighteners have occurred. the sequence and actions were:

LER 78-009, August 1978 - The flow straightener in loop "B" feedwater line dislodged from its normal location and became lodged against the downstream orifice plate. The straighteners in loops "B" and "A" were removed and replaced.

*LER 78-013, November 1978 - The flow straightener in loop "C" feedwater line dislodged from its normal location and became lodged against the

*This was the LER that promp ad the ACRS questions.

THIS DOCUMENT CONTAINS POOR QUALITY PAGES

downstream orifice plate. A new straightener was ordered and was to be installed at the next cold shutdown of sufficient duration. (See enclosure 2)

LER 79-002, April 1979 - The flow straightener in loop "B" feedwater line dislodged (2nd time) from its normal locations and became lodged against the downstream orifice plate. Straighteners in loops "B" (2nd time) and "C" were replaced.

LER-80-004, January 1980 - Segments of the flow straightener in loop "B" were dislodged (3rd time) and found against the downstream orifice plate. The loop "B" flow straightener was replaced (3rd time) but the replacement was fabricated from stainless steel rather than the original carbon steel. The flow straighteners in loops "A" and "C" were scheduled for modification during the April 1980 refueling outage. (See enclosure 3)

B. Takahama 1 and Ikata 1

It was reported on May 26, 1980 (enclosure 4) that cracking of flow straightening vanes on the inlet side of primary coolant pump piping had been discovered on Takahama 1 (a PWR supplied by Westinghouse) and Ikata 1 (a PWR supplied by Mitsubishi Heavy Industries). They plan to remove the flow straighteners.

C. Dresden 3

LER 77-021, June, 1977; LER 78-005, March 1978; LER 78-037, September 1978; and LER 79-032, November 1979 report leaks in the reactor feed pump mini flow line in the vicinity of a restricting orifice. The problem had occurred at Dresden 2 and 3 since 1972 which led to sections of pipe being replaced in 1975 and these replacement pipes are those that began leaking in 1977 as reported by these four LERs.

D. Dresden 2, 1972

A main steamline flow restrictor failed on September 7, 1972 and became lodged in the inboard main steam isolation valve. Failure was caused by vibration induced fatigue.

E. Turkey Point - 4, April 1980

LER 80-006 reports that auxiliary feedwater pump "A" failed to deliver the required flow rate during performance of inservice testing. The cau[illegible] was thought to be either control circuit calibration and/or a flow restriction in the pump discharge valve.

F. Beaver Valley,1; 1979

LER 79-36 reports that the LHSI pump failed to develop required recirculation flow. Pieces of a 11/2 inchplastic fire hose nozzle were discovered in the recirculation check valve.

OFFICE						
SURNAME						
DATE						

G. Duane Arnold, 1979

LER 79-005 reports that both emergency service water systems were inoperable because of plugged water strainers. Apparent cause is improper design application.

H. Crystal River, February 1978

LER 78-017 reports that loose parts from the Burnable Poison Rod Assembly damaged a steam generator above the tube sheet. Debris was found in both the upper and lower head of the once through steam generator.

I. St. Lucie, April 1978

Some incore instrument thimbles separated from the upper guide structure.

J. Three Mile Island 1, March 1976, and Arkansas Nuclear One, Unit 1, March 1976

Surveillance Specimen holder tubes have damaged and missing parts as a result of flow induced vibration.

K. Chooz Sena & Trin Vercellese, Mid 1976

Steam generator tube sheet was damaged by parts broken loose in the reactor pressure vessel.

L. Kewaunee 1, 1975

During startup, all auxiliary feedwater pumps exhibited reduced flow resulting from resin beads plugging the strainer to each pump. Fine mesh strainers were removed.

DISCUSSION AND RECOMMENDATIONS

The events cited in the previous section illustrate that several different types of problems have occurred. Some general problem areas are damage and dislodgement of relatively large devices such as flow straighteners, vanes, and restrictors; cracked piping as a result of complex flow conditions at flow orifices; some type of debris restricting flow through valves and strainers; and objects breaking loose to damage large equipment such as steam generators. Also, some events are essentially repeat occurrences such as the flow straightener problem at San Onofre #1 which has developed twice since the initial ACRS inquiry about LER 78-013 and the Dresden #3 leaking mini flow pipes in the vicinity of restricting orifices. It should be noted that these examples were gathered from different sources or search techniques and illustrate the occurrence of degradation during normal operation that raises some safety questions about potential operation when subjected to normally more severe loading under accident conditions. Since reporting of these events may be done under many categories such as instrumentation, flow blockage, specific systems, or specific components, it could be a difficult task to develop a comprehensive event list of occurrences of degradation of internal appurtenances.

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DATE ▶						

It is recommended that appropriate program offices should give consideration to the following:

1. Review the flow straightener problems and proposed corrections for the events at San Onofre Unit 1.

2. Initiate a review to identify: a) the use and location of internal appurtenances, b) the design methods, c) load definition for such components, d) the effect of failure with respect to potential for causing an accident or impeding mitigation of an accident, and e) the occurrence of operational problems to date. The review should be relatively broad and include such devices as flow orifices, flow restrictors, flow straightening vanes, thermocouple wells, flow scoops, diffusers, and thermal sleeves.

Original Signed by
Earl J. Brown

Earl J. Brown
Office for Analysis and Evaluation
of Operational Data

Enclosures:
1. Letter dated June 13, 1979 to ACRS discussing feedwater flow straightener.
2. Letter dated December 20, 1978 and LER 78-013 from San Onofre #1.
3. Letter dated February 13, 1980 and LER 80-004 from San Onofre #1.
4. Letter reporting straightener vane cracking on Takahama 1 and kata 1.

cc:w/enclosures:
J. P. Knight, NRR
R. J. Bosnak, NRR
J. R. Rajan, NRR
W. R. Rutherford, IE
J. C. McKinley, ACRS

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E. Brown, AEOD

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EBrown:bm
12/23/80

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

Enclosure 1

June 13, 1979

TO: ACRS MEMBERS

FROM: C. Michelson, ACRS Consultant

SUBJECT: Failure of Feedwater Flow Straightner at San Onofre Nuclear Station, Unit 1

LER 78-013 for the subject incident brings into focus a related concern which needs to be addressed for all piping systems containing internal appurtenances such as flow straightening devices. The concern is that such devices might have to be designed to withstand blowdown flow loadings in addition to normal loadings.

Consideration should be given to the possible consequences of an internal appurtenance being dislodged by the blowdown flow. Of particular concern would be the case of a straightening vane which might become lodged in a valve whose closure is an essential mitigating step following an accident. If such a consequential failure is unacceptable, the supporting attachment scheme for the appurtenance must be designed for the blowdown loadings including the degradation effects of cyclic fatigue. It should also be recognized that blowdown flow might be in either direction. This could affect the severity of loading for nonsymmetrical arrangements.

When considering the consequential effects of an internal appurtenance failure, consideration should also be given to possible steam generator tube damage. Of particular concern would be a primary or secondary side piping failure whose blowdown consequences might include the generation of internal debris due to one or more consequential failures. This could lead to multiple steam generator tube failures including, perhaps, both steam generators. The present design basis for PWR plants does not take into account the possibility of such a combined primary/secondary side blowdown into containment or the possibility of a blowdown outside of containment if the debris should also prevent isolation valve closure.

There are a number of internal appurtenances which need to be examined from this viewpoint. Typical examples are flow orfices, flow elements, thermocouple wells, straightening vanes, flow scoops, flow tubes, diffusers, and thermal sleeves. The plant Safety Analysis Report drawings may not clearly identify such appurtenances for review. It might be interesting to find out how this problem is being handled by the NRC.

dupe of 7910180473 2pp

In my opinion, this LER is a good example of the type of safety question which might be uncovered by a proper examination of plant failures during normal operation. Equipment weaknesses revealed during the generally less severe loading conditions of normal operation could be important indicators of how the equipment and systems might be expected to perform under potentially more severe accident loading conditions during which such consequential failures might not be acceptable.

Enclosure 2

Southern California Edison Company

P. O. BOX 800

2244 WALNUT GROVE AVENUE

ROSEMEAD, CALIFORNIA 91770

T. HEAD, JR

December 20, 1978

.S. Nuclear Regulatory Commission
egion V
Suite 202, Walnut Creek Plaza
1990 North California Boulevard
Walnut Creek, California 94596

Attention: Mr. R. H. Engelken, Director

Docket No. 50-206
San Onofre Unit 1

Dear Sir:

In accordance with the reporting requirements of Section 6.9.2.b(2) of ppendix A to the San Onofre Unit 1 Provisional Operating License, the following information and attached Licensee Event Report are submitted. This letter escribes a reportable occurrence involving the feedwater system.

On November 25, 1978, the indicated feedwater flow to "C" steam generator increased approximately seventy-five percent. Actual "C" steam generator feedwater flow was determined to be less than the indicated value through ...ination of steam generator level, steam flow and other plant parameters s a result of the inoperability of one of the three steam/feedwater flow mismatch channels, continuous operator surveillance was initiated in accordance with Technical Specification 3.5, Table 3.5.1. The feedwater control to "C" steam generator was maintained on automatic and the level stabilized at the normal value.

Radiographic examination of the "C" feedwater piping indicates that the flow straightener upstream of the feedwater flow orifice plate dislodged rom its normal location and moved downstream where it lodged against the rifice plate. This resulted in an increase in feedwater flow indication he flow transmitter span was adjusted to return indicated and recorded alues of feedwater flow to "C" steam generator to their actual values. This as the effect of returning the steam feedwater mismatch channel to operative tatus.

dupe of
7901160285
2pp

By letter dated August 29, 1978, we reported a similar occurrence involving the flow straightener in the "B" steam generator feedwater piping. As indicated in that letter, the "B" flow straightener was removed and replaced during the recently completed refueling outage. Additionally, the "A" loop flow straightener was removed and replaced. The "C" loop flow straightener was not replaced since at that time only two replacement flow straighteners were available from the manufacturer. The " " loop was radiographed, however, and the flow straightener appeared to be securely fastened.

A replacement for the loop "C" flow straightener, along with a spare, has been ordered from the manufacturer and will be installed at the next cold shutdown of sufficient duration. At that time an improved fastening method, such as the used on loops "A" and "B" will be employed [illegible] these modifications will preclude occurrences of this type.

There was no degradation of plant safety during this incident. Two of the three steam/feedwater flow mismatch channels remained fully operational meeting the minimum requirements of Technical Specification Table

If you should require additional information concerning this occurrence, please contact me.

Sincerely,

Attachment: Licensee Event Report No. 78-01[illegible]

cc: Director, Office of Inspection and Enforcement (30)
Director, Office of Management Information and Program Control

LICENSEE EVENT REPORT

CONTROL BLOCK (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

01 C A S O S 1 (2) 0 0 - 0 0 0 0 0 - 0 0 (3) 4 1 1 1 1 (4) (5)

CON'T

01 REPORT SOURCE L (6) 0 5 0 0 0 2 0 6 (7) 1 1 2 5 7 8 (8) (9)

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

02 During normal plant operation the [illegible] feedwater flow meter exhibited a sudden

03 increase in indication resulting in a loss of one channel of the steam flow/

04 feedflow mismatch trip. Continuous operator surveillance was initiated in

05 accordance with T.S. 3.5 Table 3.5.1.

06

07

08

09 SYSTEM CODE C H (11) CAUSE CODE E (12) CAUSE SUBCODE B (13) COMPONENT CODE I N S T R U (14) E (15) Z (16)

(17) LER/RO REPORT NUMBER EVENT YEAR 7 8 - SEQUENTIAL REPORT NO. 0 1 3 / 0 3 L -

ACTION TAKEN E (18) FUTURE ACTION A (19) EFFECT ON PLANT Z (20) SHUTDOWN METHOD Z (21) 0 0 0 0 (22) ATTACHMENT SUBMITTED Y (23) N (24) N (25) Z (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

10 X-Rays of the feedwater piping indicate that the flow straightener became

11 dislodged from its normal location and moved downstream where it became lodged

12 against the feedwater flow orifice plate.

13

14

15 FACILITY STATUS E (28) % POWER 1 0 0 (29) OTHER STATUS (30) NA METHOD OF DISCOVERY A (31) Operator Observation

16 ACTIVITY CONTENT RELEASED OF RELEASE Z (33) Z (34) AMOUNT OF ACTIVITY (35) NA LOCATION OF RELEASE NA

17 PERSONNEL EXPOSURES NUMBER 0 0 0 (37) TYPE Z (38) DESCRIPTION (39) NA

18 PERSONNEL INJURIES NUMBER 0 0 0 (40) DESCRIPTION (41) NA

19 LOSS OF OR DAMAGE TO FACILITY (43) TYPE Z (42) DESCRIPTION NA

7811G [illegible]

20 PUBLICITY ISSUED N (44) DESCRIPTION (45) NA

Enclosure 3

Southern California Edison Company

P. O. BOX 800
2244 WALNUT GROVE AVENUE
ROSEMEAD, CALIFORNIA 91770

February 13, 1980

U. S. Nuclear Regulatory Commission
Region V
Suite 202, Walnut Creek Plaza
1990 North California Boulevard
Walnut Creek, California 94596

Attention: Mr. R. H. Engelken, Director

Docket No. 50-206
San Onofre - Unit 1

Dear Sir:

This letter constitutes a reportable occurrence involving the feedwater system. Submittal is in accordance with the reporting requirements stipulated in Section 6.9.2(b) of Appendix A to the Provisional Operating License DPR-13.

On January 19, 1980 the indicated feedwater flow to "B" steam generator increased approximately fifteen percent. Actual "B" steam generator feedwater flow was determined to be less than the indicated value through examination of steam generator level, steam flow and other plant parameters. As a result of the inoperability of one of the three steam/feedwater flow mismatch channels, continuous operator surveillance was initiated in accordance with Technical Specification 3.5, Table 3.5.1. The feedwater control to "B" steam generator was maintained on automatic and the level stabilized at the normal value.

Radiographic examination of the "B" feedwater piping indicated that the flow straightener upstream of the feedwater flow orifice plate was missing segments of various tubes. The tube segments had apparently lodged against the orifice plate causing the increase in feedwater flow indication. The flow transmitter span was adjusted to return indicated and recorded values of feedwater flow to "B" steam generator to their actual values. This had the effect of returning the steam/feedwater flow mismatch channel to operative status. Similar incidents involving the flow straighteners at San Onofre were reported in LER's 78-009, 79-013 and 79-002.

The "B" flow straightener was removed during the January 26, 1980 plant outage. An examination of the damaged straightener indicated that the carbon steel tubing was experiencing excessive erosion.

dupe of
8002190494
2pp

A new straightener fabricated from stainless steel tubing with a modified weld attachment scheme was installed on the "B" loop. The improved erosion resistance and structural strength of the stainless steel material should prevent it from segmenting. The "A" and "C" feedwater flow straighteners will be similarly modified during the April 1980 refueling outage.

At the time of the feedwater piping disassembly, no flow straightener segments were found near the orifice plate. However, segments were removed from the flow control valve located downstream of the orifice and in the bypass regulator. The valves and piping were thoroughly inspected and returned to service.

There was no degradation of plant safety during this incident. Two of the three steam/feedwater flow mismatch channels remained fully operational meeting the minimum requirements of Technical Specification 3.5, Table 3.5.1.

If you should require additional information concerning this occurrence please contact me.

Sincerely,

H. L. Ottoson
Manager, Nuclear Generation

Attachment: Licensee Event Report No. 80-004

Director, Nuclear Reactor Regulation (30)
Director, Office of Management Information & Program Control (3)
Director, Nuclear Safety Analysis Center (1)

LICENSEE EVENT REPORT

CONTROL BLOCK: 1 (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

C A S O S 1 (2) 0 0 - 0 0 0 0 0 - 0 0 (3) 4 1 1 1 1 (4)

LICENSEE CODE — LICENSE NUMBER — LICENSE TYPE

REPORT SOURCE: L (6) 0 5 0 0 0 2 0 6 (7) 0 1 1 9 8 0 (8) 0 2 1 3 8 0

DOCKET NUMBER — EVENT DATE — REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

During normal operation the "B" feedwater flow indication increased approximately 15% above the actual flow. Previous similar events were reported with LER's 78-009, 78-013 and 79-002. Continuous operator surveillance was initiated in accordance with T.S. 3.5, Table 3.5.1. Redundant systems were available.

SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP SUBCODE	VALVE SUBCODE
C H (11)	E (12)	D (13)	I N S T R U (14)	E (15)	Z (16)

LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	8 0	—	0 0 4	/	0 3	T	—	0

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
E (18)	F (19)	Z (20)	Z (21)	0 0 0 0	Y (23)	Y (24)	N (25)	0 0 1 2 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

Radiographs of the feedwater piping indicated that segments of the flow straightener were missing. The "B" straightener was replaced with an improved flow straightener.

FACILITY STATUS: E (28) % POWER: 1 0 0 (29) OTHER STATUS (30): NA

METHOD OF DISCOVERY: A (31) DISCOVERY DESCRIPTION (32): Operator observation

ACTIVITY CONTENT — RELEASED OF RELEASE: Z (33) Z (34) AMOUNT OF ACTIVITY (35): NA LOCATION OF RELEASE (36): NA

PERSONNEL EXPOSURES — NUMBER: 0 0 0 (37) TYPE: Z (38) DESCRIPTION (39): NA

PERSONNEL INJURIES — NUMBER: 0 0 0 (40) DESCRIPTION (41): NA

LOSS OF OR DAMAGE TO FACILITY (43) — TYPE (42) DESCRIPTION: NA

8002100

PUBLICITY — ISSUED: N (44) DESCRIPTION (45): NA

NRC USE ONLY

Enclosure 4

UNCLASSIFIED

Department of State

INCOMING TELEGRAM

0241

ACTION NRC-02

INFO OCT-01 EA-12 ADS-00 OES-09 DOE-17 INR-10 /051 W
------------------076127 010319Z /14

R 300600Z MAY 80
FM AMEMBASSY TOKYO
TO SECSTATE WASHDC 8852

UNCLAS TOKYO 09526

PLEASE PASS TO NRC FOR J. LAFLEUR, IP

E.O. 12065: N/A
TAGS: TECH, ENRG, JA
SUBJECT: STATUS OF PERIODIC EXAMINATIONS ON NUCLEAR POWER PLANTS IN JAPAN

1. AGENCY OF NATURAL RESOURCES & ENERGY (ANRE), ON MAY 26, 1980, ISSUED INTERIM ANNOUNCEMENT OF THE STATUS OF PERIODIC MAINTENANCE EXAMINATIONS ON KANSAI EPCO'S TAKAHAMA-1 (PWR 826 MWE), OHI-1 (PWR 1,175 E), AND SHIKOKU EPCO'S IKATA-1 (PWR 565 MWE) WHICH HAVE BEEN SHUTDOWN FOR INSPECTIONS. FOLLOWING ARE FINDINGS OF ANRE:

- A. ON TAKAHAMA-1 AND IKATA-1

SUPERSONIC INSPECTIONS ON STRAIGHTENING VANES, LOCATED INSIDE OF THE INLET SIDE OF PRIMARY COOLANT PUMP PIPING, INDICATED POSSIBLE CRACKING. SURFACE INSPECTIONS ON THE VANES REVEALED THERE WERE SEVERAL CRACKS ON THE VANES. ANRE BELIEVES HIGH CYCLE RESONANCE VIBRATIONS DEVELOPED, BY WHIRLPOOL AT THE TIPS OF THE VANES ARE THE CAUSE OF THESE FATIGUE CRACKINGS. ANRE INSTRUCTED KANSAI AND SHIKOKU OPERATORS TO REMOVE CRACKED VANES AS TESTS HAVE VERIFIED THE REACTORS CAN GO BACK TO NORMAL OPERATION WITHOUT VANES IN THE PIPING.

- B. ON OHI-1

VISUAL INSPECTIONS ON FUEL ASSEMBLIES, USING SUB-MERGED TELEVISION CAMERA, REVEALED 31 DAMAGED COIL SPRINGS FOR RETAINING CONTENTS (BURNABLE POISONS, NEUTRON SOURCES, PLUGING DEVICES). VISUAL INSPECTIONS ON CRACKED SURFACE OF SPRINGS HAVE INDICATED SOME "STRIATION PATTERNS" WHICH ARE CHARACTERISTIC OF HIGH CYCLE FATIGUE OF MATERIALS. ANRE INSTRUCTED KANSAI TO REPLACE ALL DAMAGED SPRINGS WITH NEW ONES, ALTHOUGH ANALYTICAL TESTS CONFIRMED THAT EVEN WITH DAMAGED SPRINGS THERE WILL BE NO RISK OF LOSS OF CONTENTS FROM FUEL ASSEMBLIES DURING NORMAL REACTOR OPERATION.

2. FURTHER DETAILS WILL BE CABLED TO NRC BY HAYAKAWA OF NSB/STA. MANSFIELD

UNCLASSIFIED