50-275/323-OLA-2
I-MFP-57

MFP Exhibit 57
8/18/93 Dollie Feigel Reporter

NCR DC2-93-EM-N014 Rev. 00
DRAFT: May 13, 1993

'93 OCT 28 P6:30

MANAGEMENT SUMMARY

During performance of STP P-6B, 2-FCV-37 failed to close on demand from the control room. A design basis function of this valve is to provide containment isolation.

The NCR was initiated on March 15, 1993 (discovery date).

Root Cause: Procedure deficiency in that Electrical Maintenance Procedure MP E-53.10J, Revision 1 (dated 12/18/89), "Limitorque SMB-00 and SB-00 Valve Operator Maintenance", did not have sufficient detail to ensure that the quad rings were properly installed after limitorque operator disassembly.

Corrective Action: Electrical Maintenance Procedure MP E-53.10M, Revision 0, "Limitorque SMB-00 and SB-00 Valve Operator Maintenance", was issued on January 22, 1993. This maintenance procedure contains detailed steps and a composite assembly drawing for re-assembly of the limitorque operators. Therefore, there is no need for additional corrective action.

This draft dated May 13, 1993 reflects the final NCR per the TRG meeting held on May 13, 1993.

NUCLEAR REGULATORY COMMISSION
Docket No. 50-275-OLA Official Exh. No. MFP 57
In the matter of PACIFIC GAS AND ELECTRIC CO
Staff ____ IDENTIFIED ✓
Applicant ____ RECEIVED ✓
Intervenor ✓ REJECTED ____
Contg Off ____
Contractor ____ DATE 8-18-93
Other ____ Witness ____
Reporter Dollie Feigel

NCR DC2-93-EM-N014
AFW PP 21 STEAM SUPPLY, 2-FCV-37, FAILURE TO CLOSE

I. Plant Conditions

Unit 2 was in Mode 1 (Power Operation) at 100% power.

II. Description of Event

A. Summary:

During performance of STP P-6B, "Routine Surveillance Test of Turbine-Driven Auxiliary Feedwater Pump", Step 10.26.5, flow control valve 2-FCV-37 was taken to the closed position. The position indicating lights indicated mid-position (both red and green lights illuminated). Locally, the valve was determined to still be in the "open" position. 2-FCV-37 is the Unit 2 isolation valve for off-steam lead number two from the main steam line and is located outside in the pipe rack. Valve closure was stopped by the closing torque limit switch. The breaker (52-2H-30) for the motor operator on 2-FCV-37 did not trip open when the attempt was made to cycle the valve.

On March 15, 1993 NES Engineering determined that with the corrosion on the upper bearing, combined with the degraded stem lubrication, the ability of the valve to close with full flow differential pressure (DP) conditions is suspect.

B. Background:

Technical Specification 3.7.1.2 requires for Modes 1, 2 and 3: "At least three steam generator auxiliary feedwater pumps and associated flow paths shall be operable with: (a) Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate vital busses, and (b) One steam turbine-driven auxiliary feedwater pump capable of being powered from two OPERABLE and redundant steam supply sources."

PSRC TS 3.7.1.2b Interpretation 89-04, dated 5/4/89, requests: "The LCO requires that the Turbine Driven Auxiliary Feedwater pump must be capable of being supplied from an OPERABLE steam supply system, but provides no guidance on what is necessary for the steam supply system to be OPERABLE." The PSRC Interpretation clarifies: "For the Turbine Driven Auxiliary Feedwater Pump steam supply system to be OPERABLE, the following must be met.

1. FCVs 37 and 38 must be OPERABLE <u>and</u> open.
2. Check Valves MS-5166 and MS-5167 must be Operable.
3. Steam Traps TRP 104, 105, and 106 must be Operable or bypassed to ensure adequate line condensate removal."

Technical Specification 3.7.1.2 ACTION requires: "With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours."

C. Event Description:

On January 31, 1993 during performance of STP P-6B, Step 10.26.5, flow control valve 2-FCV-37 was taken to the closed position. The position indicating lights indicated mid-position (both red and green lights illuminated). Locally, the valve was determined to still be in the "open" position.

Work Order C0110207 was initiated to investigate the failure of the valve to close on demand. The valve was stroked manually, but experienced difficult operation during a small portion of the closing stroke from the full open position. The valve was then stroked electrically and the torque switch was observed to be "bouncing". On the third valve stroke, the torque switch opened and the valve stopped at approximately ninety-five percent open. When the stem cover was removed, a large amount of "Lubriplate" lubricant was found pooled in the stem nut/lock nut depression. Inspection of the stem showed that the stem

lubrication was marginal. The valve stem was lubricated with "Lubriplate" and successfully stroked manually four times. The valve also was stroked electrically with no indication of torque switch bounce. Grease samples were obtained, and both the limit switch grease and the actuator grease showed signs of separation and evidence of weeping into the limit switch compartment was found (Note: this is not unusual). A subsequent test of the grease sample confirmed the grease in the actuator to be Type EP-0, per design. The immediate root cause was determined to be a sticking valve stem. The valve was returned to service on February 1, 1993. A work order was prepared to inspect 2-FCV-37 further during the fifth refueling outage (2R5).

On February 4, 1993 a partial internal actuator inspection was performed. After removal of the stem cover and the limit switch compartment cover, and the spring cartridge cap cover, visual inspection found nothing that could have caused the actuator to fail to close. A detailed component inspection was planned for 2R5. [REF: W/O C0110396]

On February 5, 1993 Quality Evaluation Q0010397 was initiated to address potential quality concerns related to this event.

On February 17, 1993, votes testing was completed and no problems were noted. [REF: W/O C0110455]

On March 9, 1993 a manual load cell test was performed. The as-found thrust was acceptable. [REF: W/O C0109271]

On March 12, 1993, detailed internal inspection identified significant particulates, water and corrosion. The upper bearing had visible corrosion. Preliminary analysis of the grease sample showed that the grease had foreign material present (i.e. dirt, rust, metal shavings, etc...). Engineering analysis determined that the January 31, 1993 condition of 2-FCV-37, with the bearing corrosion and stem grease degradation, may have resulted in inability to meet its Generic Letter

(GL) 89-10 mispositioning closing thrust requirement. STP P-6B, which 2-FCV-37 failed, tests at approximately 2000 pounds thrust. GL 89-10 mispositioning thrust requirement is approximately 6500 pounds thrust. [REF: W/O C0109271]

On March 15, 1993 NES Engineering determined that the ability of the 2-FCV-37 to close with full flow differential pressure (DP) was suspect prior to January 31, 1993 with the buildup of corrosion on the upper bearing combined with the degraded stem lubrication. (Reference A0292330, Eval 09).

However, the successful manual load test performed March 9, 1993, demonstrated that the GL 89-10 thrust requirement probably could be met with the as-found bearing corrosion.

D. Inoperable Structures, Components, or Systems that Contributed to the Event:

None.

E. Dates and Approximate Times for Major Occurrences:

1.	April 16, 1990	FCV-37 Overhauled under W/O R0058494
2.	January 31, 1993	FCV-37 failed to close during STP P-6B. TS 3.7.1.2 entered. A watch was established to meet the requirements of TS 3.6.3 for containment isolation.
3.	February 01, 1993	Valve stem cleaned and lubricated, FCV-37 returned to service.
4.	February 5, 1993	Q0010397 initiated.

5.	March 9, 1993	GL 89-10 thrust requirement met during manual load cell test.
6.	March 12, 1993	Internal inspection found upper bearing corroded.
7.	March 15, 1993	Engineering determined that prior to 1/31/93, Valve 2-FCV-37 may have been unable to close under full flow DP conditions, and was potentially reportable.

F. Other Systems or Secondary Functions Affected:

None.

G. Method of Discovery:

PG&E plant personnel, during the performance of a scheduled STP P-6B, "Routine Surveillance Test of Turbine-Driven Auxiliary Feedwater Pump", identified the problem.

H. Operator Actions:

2-FCV-37 was declared inoperable and TS 3.7.1.2 and TS 3.6.3 were entered. 2-FCV-37, after corrective maintenance, was declared operable prior to exceeding the LCO action statements.

I. Safety System Responses:

None.

III. Cause of the Event

A. Immediate Cause:

Valve 2-FCV-37 was not capable of closing under full flow differential pressure conditions due to corrosion of the upper bearing.

B. Root Cause:

Procedure deficiency in that Electrical Maintenance Procedure MP E-53.10J, Revision 1 (dated 12/18/89), "Limitorque SMB-00 and SB-00 Valve Operator Maintenance", did not have sufficient detail to ensure that the quad rings were properly installed after limitorque operator disassembly.

C. Contributory Cause:

None.

IV. Analysis of the Event

A. Safety Analysis:

The Auxiliary Feedwater System (AFWS) serves as a backup supply of feedwater to the secondary side of the steam generators when the main feedwater system is not available, thereby maintaining the heat sink capabilities of the steam generators (SGs). As an engineered safety feature (ESF) system, the AFWS is directly relied upon to prevent core damage and system overpressurization (release of reactor coolant through the pressurizer power operated relief valves or pressurizer safeties) in the event of transients such as a loss of normal feedwater or a secondary system pipe rupture, and to provide a means for plant cooldown following any plant transient.

Auxiliary feed pumps (two 4kV motor-driven pumps and one steam turbine-driven pump) are provided and designed to ensure complete reactor decay heat removal under all conditions including loss of power and loss of the normal heat sink (the condenser circulating water), while maintaining minimum water levels within the steam generators. The design basis for the AFWS is to ensure that the minimum required flow (440 gpm) will be delivered to the minimum number of SGs (two SGs), within one minute, during any design bases event.

The reactor plant conditions that impose safety-related performance requirements on the AFWS are as follows: (1) Loss of main feedwater transient, (2) Secondary system pipe ruptures, (3) Loss of all ac power (station blackout), (4) Loss-of-coolant accident (LOCA), and (5) Cooldown.

The steam turbine-driven AFW pump supply train consists of a full-capacity turbine-driven pump which is sized to provide a minimum flow of 930 gpm. The turbine-driven pump is powered by steam supplied through two full capacity redundant lines taken from two of the four main steam lines upstream of the main steam isolation valves. The redundant supply lines ensure continued availability of steam to the turbine in the case of a faulted SG or ruptured main steam line associated with one of the supply lines. However, there are specific requirements for component availability and status for the steam supply to the turbine-driven pump to be considered operable.

An operable steam supply system for the AFWS is defined as follows: (Modes 1, 2, and 3)

1. Two steam supply lines each with an operable remote manual motor-operated isolation valve in the open position (FCV-37 & FCV-38),
2. An operable check valve in each of these lines,
3. Associated steam traps must be operable or bypassed to ensure adequate condensate removal, and
4. A single operable motor-operated flow control valve on the common turbine steam supply line must be capable of automatic actuation from closed to open for all applicable ESF and manual actuation signals (FCV-95).

For Modes 4, 5, and 6 the AFWS is not required by Technical Specification to be operational.

The two steam supply lines to the turbine-driven pump each contain a check valve and a normally open motor-operated flow control valve (FCV-37 and FCV-38). The check valves are provided for passive backflow isolation of either steam supply

line in the event of a main steam line break, allowing continued turbine-driven pump operation using the unaffected steam supply line. The two steam lines join together into a common line prior to entering the turbine-driver. This common line contains a normally-closed 125V DC powered motor operated valve (FCV-95). During normal plant operation, the AFWS steam supply lines are pressurized up to this flow control valve.

FCV-37 and FCV-38 are remote manual containment isolation valves for the main steam system to the AFW turbine-driven pump, and are Design Class I, "Group D". To meet the "Group D" containment piping isolation classification, a single manually operated stop valve is required. This requirement can either be met through remote manual operation from the control room, or by local manual manipulation if the valve motor or controls have failed. These valves are designed to remain open during operation and close on demand against 1150 psi maximum differential pressure. These valves fail "as-is" upon a loss of either control or actuator power supply. The valve operators for FCV-37 and 38 are Instrument Class IA since the valves may be called upon for remote manual operation to isolate a faulted steam generator if required.

None of these reactor plant conditions described above (Loss of MFW, secondary pipe ruptures, station blackout, LOCA, and cooldown) is affected by the failure of 2-FCV-37 to close on demand since this valve is normally open and fails "as-is". For these reactor conditions, 2-FCV-37 needs to remain open so as not to impact availability of the turbine-driven AFW pump.

For the rupture of a main steam line, FSAR Section 15.4.2.1 states that the MSIV's fully close within ten seconds. Furthermore, for any break, in any location, no more than one steam generator would blow down even if one of the MSIV's fails to close. This implies that a break in the line common to FCV-95, and FCV-37/38, in conjunction with a the degraded condition of FCV-37 and a single failure FCV-95 could result in simultaneous

blowdown of two steam generators. However, an analysis by Westinghouse has determined that these valves are not required to isolate a break downstream of these valves and upstream of FCV-95. This is because such a line break does not initiate a plant trip and therefore, main feedwater can be used to support continued plant operation until the line break can be isolated manually (Reference EOI-8018 and 8062). Furthermore, engineering has reviewed the associated piping stress calculations and verified that the most probable break point is just upstream of FCV-95, as assumed in the Westinghouse analysis.

The only other safety-related function associated with FCV-37 is to help mitigate the consequences of a steam generator tube rupture (SGTR) accident. The Westinghouse SGTR analysis assumes FCV-95 is used to stop the steam driven AFW turbine driven pump to avoid overfill. It is assumed that in the event that FCV-95 fails to close, flow to the steam generator would be stopped by an unspecified means (i.e., by closure of the turbine driven pump LCVs, closure of FCV-37 and FCV-38, or by tripping closed FCV-152).

Emergency Procedure E-3 addresses operator action during a SGTR. However, discovery of the rupture is assumed to be preceded by a reactor trip, caused by the difference in temperature between the hot leg and the cold leg (over temperature Delta-T), which is covered by Procedure E-0. Step 14 of E-0 directs operator attention to the AFW system status. Initially, the operator establishes that the system is delivering at least 470 gpm. If operation of the AFW system cannot be established, the operator is referred to EP FR-H.1 "Response to Loss of Secondary Cooling." If the flow requirements are met, the operator's attention is focused toward maintaining level in the steam generator by placing the motor driven AFW pumps in auto which controls automatically on level and manually operating the turbine driven (TD) AFW pump (step 14.b) to maintain level.

In step 14.c (EP E-0), when the SG level reaches 8% in 3 of the 4 steam generators, the operator is directed to stop the TD AFW pump, otherwise control of level continues by manual operation of the TD AFW pump LCVs. Once the decision is made to stop the TD AFW pump, the operator has the option of closing FCV-95, closing FCVs-37 and 38, closing the TD pump LCVs (which can be closed by remote manual action), or by tripping FCV-152 (the turbine trip valve) by local manual action. Procedure E-0 does not specify which of these options to use. As noted above, the Westinghouse analysis assumes FCV-95 is closed. This assumption is consistent with current operator practice.

Should FCV-95 fail to close, the Westinghouse analysis assumed the operator would stop flow to the steam generators within two minutes. The means of accomplishing this within the two minute limit was not specified. Current operator practice is to remotely close both FCV-37 and FCV-38. Should either of these valves fail to close by remote manual operation, the TD AFW LCVs are closed by remote manual operation, which can be accomplished within the allotted two minutes. Local manual closure of FCV-37 and 38, as well as local tripping of FCV-152, would require longer than two minutes.

Because closure of the TD AFW pump LCVs would prevent steam generator overfill, there would be adequate time to permit either local manual closure of FCV-37/38 or FCV-152 to stop the TD AFW pump. Therefore, remote manual operation of FCV-37 and FCV-38 is not required and local manual operation is an acceptable minimum design bases for these valves for the SG overfill event.

An additional design function of FCV-37 and FCV-38 is to mitigate a radiological release from a SGTR through isolation of the ruptured SG. The operators would proceed through E-0 following the reactor trip to step 26 of Emergency Procedure E-0 which refers the operator to Emergency Procedure E-3 "Steam Generator Tube Rupture" if secondary side radiation is abnormal. At this point, having

assumed FCV-95 has failed to close, there is no requirement to assume an additional failure of the 10% dump valve for the faulted steam generator. Likewise, if the 10% dump has failed, there is no requirement to assume FCV-95 has failed.

If the 10% dump valve has failed to close, the safety analysis allows 30 minutes for the plant operator to manually close it. The AFW turbine exhaust to atmosphere is isolated by closing FCV-95.

If FCV-95 has failed to close, the 10% dump valve is assumed to have been closed. The flow to the AFW turbine is less than the amount of steam that would be released through a stuck open 10% dump valve. Therefore, in excess of 30 minutes is available to isolate the flow from the AFW turbine. This is ample time for the plant operator to request local manual closure of FCV-152 or FCV-37 and FCV-38 per E-3 step 3c. Therefore, the stuck open 10% steam dump dose release bounds a release through the turbine driven AFW pump.

Thus, the feature of remote manual (electrical) closure of FCV-37 and 38 is not required to mitigate the radiological release due to a SGTR.

Therefore, since these administrative controls (EP E-0 & EP E-3) are already in place, the ability to mitigate the consequences of an accident is not increased outside the existing plant design basis.

The loss of remote manual (electrical) closure capability of FCV-37 will not stop the turbine-driven AFW pump from performing its safety function. This valve is an isolation valve on one of two steam supplies to the turbine-driven pump. Other than maintaining pressure boundary integrity, the valve serves no actual safety-related function relative to the operation of the AFW pump.

Therefore, this event did not adversely affect the health and safety of the public.

B. Reportability:

1. Reviewed under QAP-15.B and determined to be non-conforming in accordance with Section 2.1.2.

2. Reviewed under 10 CFR 50.72 and 10 CFR 50.73 per NUREG 1022 and determined to be not reportable.

   The safety function of FCV-37 was researched to determine whether remote electrical closure is required for accident mitigation. Based on the "Group D" containment isolation classification (remote manual closure) and the steps in Emergency Procedures E-0 (close FCV-95) and E-3 (local closure of FCV-152 if FCV-37 cannot be remotely operated), there is no design basis requirement for electrical closure.

3. Reviewed under 10 CFR Part 21 and determined that this problem will not require a 10 CFR 21 report, since (a) it is being evaluated under 10 CFR 50.72/73, and (b) it does not involve defects in vendor-supplied services/spare parts in stock.

4. This problem will not be reported via an INPO Nuclear Network entry.

5. Reviewed under 10 CFR 50.9 and determined to be not reportable since this event does not have a significant implication for public health and safety or common defense and security.

6. Reviewed under the criteria of AP C-29 requiring the issue and approval of an OE and determined that an OE is not required.

V. <u>Corrective Actions</u>

A. Immediate Corrective Actions:

1. The valve actuator stem was manually and electrically stroked multiple times, after which the valve operated smoothly.

2. The actuator was electrically stroked several times. No signs of torque switch chatter were noted during operation. 2-FCV-37 returned to service.

3. Quality Evaluation Q0010397 initiated to track the problem investigation and resolution. An as-found inspection and diagnostic testing were planned for performance during 2R5.

4. Stem lube inspections (and lubricate as necessary) of FCV-37/38 & 95 temporarily increased to a quarterly frequency. [REF: 2-b]

5. Stem covers of 2-FCV-438/439 removed and actuators inspected, since these valves are also located in the pipe rack. 2-FCV-438 was acceptable; heavy, flaky, rust and some standing water were found in the upper section of the stem cover area for 2-FCV-439, however no water was found in the gear box and grease samples appear normal. [Reference 2d and 5e]

B. Investigative Actions:

Votes diagnostic testing was performed on 2-FCV-37 during the next STP P-6B performance. This votes test found no unusual characteristics, and the trace was similar to the DP test performed during 2R4. Reference Work Order C00110455.

1. Review the overhaul records for valves 1-FCV-37, 1-FCV-438, 1-FCV-439, 2-FCV-37, 2-FCV-438 and 2-FCV-439 to provide assurance that the proper quad rings were staged and used for valve reassembly.

   RESPONSIBILITY: C. Shortt
   DEPARTMENT: PGEM
   Tracking AR: A0298496, AE #01
   STATUS: Return

2. Inspect 1-FCV-37, 1-FCV-438, and 1-FCV-439 for signs of water intrusion, grease degradation, and corrosion.

   RESPONSIBILITY: M. Frauenheim
   DEPARTMENT: PGMB
   Tracking AR: A0298496, AE #02
   STATUS: Return

3. Perform a root cause analysis. Problem statement, as agreed by the TRG is as follows: "Valve 2-FCV-37 outside of its design basis prior to January 31, 1993." Subsequent investigation determined that 2-FCV-37 was not outside of design basis. This AE tracks root cause analysis performance only.

   RESPONSIBILITY: C. Shortt
   DEPARTMENT: PGEM
   Tracking AR: A0298496, AE #03
   STATUS: Return

4. Investigate and evaluate the condition of the grease for the upper bearing and the gear box for 2-FCV-37.

   RESPONSIBILITY: M. Frauenheim
   DEPARTMENT: PGMP
   Tracking AR: A0298496, AE #04
   STATUS: Return

5. Determine, if possible, how the wrong quad ring was utilized during the reassembly of valve 2-FCV-37 after its overhaul in 2R3.

   RESPONSIBILITY: M. Frauenheim
   DEPARTMENT: PGMB
   Tracking AR: A0298496, AE #05
   STATUS: Return

6. Determine if the apprentice, during 2R3, was qualified to work on limitorque operators.

   RESPONSIBILITY: C. Shortt
   DEPARTMENT: PGEM
   Tracking AR: A0298496, AE #06
   STATUS: Return

7. Document basis for non-reportability.

   RESPONSIBILITY: K. Riches
   DEPARTMENT: PTRC
   Tracking AR: A0298496, AE #07
   STATUS: Return

C. Corrective Actions to Prevent Recurrence:

Electrical Maintenance Procedure MP E-53.10M, Revision 0, "Limitorque SMB-00 and SB-00 Valve Operator Maintenance", was issued on January 22, 1993. This maintenance procedure contains detailed steps and a composite assembly drawing for re-assembly of the limitorque operators. No further corrective actions are required.

MP E-53.10M will supersede MP E-53.10J when MP E-53.10J is rescinded. Rescission of MP E-53.10J is tracked on A0298496, AE #08.

D. Prudent Actions (not required for NCR closure)

1. Provide protective covers for the FCV's (2-FCV-37, 438 & 439) located outside in the pipe rack area.

   RESPONSIBILITY: M. Frauenheim
   Tracking AR: A0304019

2. Design basis operability following a failed STP: If there is any doubt about the cause of a MOV failing to meet STP requirements, what is the best way to document the consequences of failure and subsequent acceptability to return to service. Please investigate and implement appropriate actions. Consideration of this issue should also be given to components other than MOV's.

   RESPONSIBILITY: H. Phillips ECD: 4/27/94
   Tracking AR: A0305252

3. Revise DCM S-3B, "Auxiliary Feedwater System," to include an additional safety function for the turbine-driven auxiliary feedwater pump steam supply line. Specifically, this line

must be isolated following a SGTR to preclude an offsite radioactive release if SG-2 or SG-3 has the ruptured tube(s). Please include the allowable time for isolation of this line following a SGTR. As applicable, also revise DCM T-15, "Radiation Protection" to include this information.

RESPONSIBILITY: C. Rhodes ECD: 4/27/94
Tracking AR: A0305259

VI. Additional Information

A. Failed Components:

FCV-37, motor operated flow control valve operator for the AFW pump steam Supply.

Manufacturer: Limitorque
Model No.: SMB-00

B. Previous Similar Events:

NCR DC2-89-OP-N009, "AFW Pump 2-1 Inoperable Due to FCV-37 Being Shut"; this is the NCR that resulted in initiation of Tech Spec interpretation 89-04. On January 17, 1989, action b. of Tech Spec (TS) 3.7.1.2, "Auxiliary Feedwater System", was exceeded when both the steam driven auxiliary feedwater (AFW) pump 2-1 and motor driven AFW pump 2-3 were inoperable for greater than 6 hours with the unit in Mode 1. Earlier, AFW pump 2-3 was removed from service to allow maintenance on level control valve LCV-115. AFW pump 2-1 was made inoperable by removal from service of one steam supply to the pump when FCV-37, a steam supply isolation valve, was shut to allow maintenance on the valve motor operator. The senior licensed operator had concluded that this activity did not render AFW pump 2-1 inoperable based on a review of the applicable surveillance test procedures. The procedures implied that the pump is operable if the pump can maintain full speed and flow with only one operable steam supply. The total time that AFW pump 2-1 was out of service was less than the action statement time of TS 3.7.1.2 action a.

The immediate cause of this event is that the senior licensed operator who evaluated the operability concerns associated with taking FCV-37 out of service incorrectly concluded that this action would not render AFW pump 2-1 inoperable. The root cause of this event was a lack of understanding by plant personnel of the design basis operability requirements of the steam driven AFW pump. This lack of understanding was due to inadequate guidance in plant procedures which were used to determine the pump operability. Applicable plant procedures did not reflect the requirement to have both turbine steam supply paths operable. This requirement is specified in the Westinghouse Steam Systems Design Manual (WCAP-7451) and was not incorporated in applicable plant design bases documentation.

When FCV-37 is closed and AFW pump 2-1 is not declared inoperable, a feedwater line break associated with a failure of AFW pump 2-2 causes the AFW system to be incapable of delivering the design AFW flow. If this scenario would have occurred, Emergency Operating Procedure (EP) F-0, "Critical Safety Function Status Trees", would direct operators to EP FR-H.1, "Response to Loss of Secondary Heat Sink." This procedure would instruct the operators to restore at least 460 gpm of feedwater flow to the SGs by performing local manual valve alignments as necessary to achieve the minimum flow requirements. The operators would open the closed steam supply to the steam driven AFW pump and/or cross-tie the motor driven AFW pumps to establish the required feedwater flow conditions. Westinghouse has performed a feedline break evaluation for the past operation with one valve closed on one line of the steam supply to the turbine auxiliary feedwater pump. The following assumptions were made in this analysis:

One of the two parallel valves to the auxiliary feedwater pump turbine is closed for maintenance.

The main feedline break occurs in the steam generator feeding the operating steam supply, closing off the only other path for steam to

t: turbine driven pump. The turbine driven pump is, therefore, disabled.

A single failure of the motor driven pump which is not associated with the faulted SG occurs.

Ten minutes after reactor trip, operator action supplies water to one intact SG by isolating the feedline break. Estimated flow to the intact SG is 325 gpm.

Thirty minutes after reactor trip, operator action increases the auxiliary feedwater to 440 gpm. This additional feedwater is fed to at least two steam generators.

This analysis was performed for with power and without power cases. The results were shown to be within FSAR limits by showing that no boiling occurred in the hot leg of the RCS and that the pressurizer did not fill.

Corrective Actions to Prevent Recurrence include; (1) A revision to the FSAR Update will be made to clearly state both main steam supply valves must be open for the steam driven AFW pump to be operable, and (2) PG&E will evaluate the need for a technical specification change to clarify that both steam leads are necessary in order for the steam driven AFW pump to be operable.

The corrective actions implemented due to this NCR resulted in proper operator actions to declare FCV-37, conservatively, inoperable and investigate the closing problem with the motor operator.

C. Operating Experience Review:

1. NPRDS:

   Not applicable.

2. NRC Information Notices, Bulletins, Generic Letters:

   None.

3. INPO SOERs and SERs:

None.

D. Trend Code:

Responsible department EM, and cause code B2.

E. Corrective Action Tracking:

1. The tracking action request is A0298496.

2. Are the corrective actions outage related?

NO.

F. Footnotes and Special Comments:

1. Even though FCV-37 is not environmentally qualified, the generic question was asked: "What is the impact on EQ Limitorque operators should the quad rings be accidentally left out?". For FCV-37, since it is located outside in the pipe rack, and it was exposed to large amounts of rainfall, there is a high probability of flooding in the actuator housing if missing the upper quad rings. For the EQ Limitorque's, the protective enclosures or buildings in which they are located, protect the actuators from inclement weather (heavy rain fall). In addition, the EQ Limitorque's are all maintained at a high level of assurance for proper functionality. Therefore, if the quad rings were missing form an EQ Limitorque, the probability of failure is negligible.
Reference: AR A0292330, Eval 08.

2. Condensation from 10% steam dump valve, PCV-20, has been seen to splash onto FCV-37.

G. References:

1. Technical Specification 3/4.7.1.2 and TS PSRC Interpretation 89-04.

2. Related Action Requests.

   a. A0292530; Initiating Action Request. Also refer to associated QE Q0010397.

   b. A0295133; Increase stem lube inspection frequency for FCV-37/38/95.

   c. A0297566; Inspect FCV-438 & 439 for water contamination (also located in pipe rack).

   d. A0298244; Water found above housing cover of 2-FCV-439.

3. Licensee Event Report (LER) or other reporting reference.

4. STP P-6B, Rev. 26; "Routine Surveillance Test of Turbine-Driven Auxiliary Feedwater Pump."

5. Problem Investigation Work Orders, completion remarks, as applicable.

   a. W/O C0110207. "MS-2-FCV-37: Investigate Failure to Operate."
   b. W/O C0110396. "2-FCV-37: Perform Internal Visual Inspection".
   c. W/O C0110455. "MS-2-FCV-37: Perform Votes Test".
   d. W/O C0109271, 2R5 detailed inspection.
   e. W/O C0112151, 2R5 FCV-439 inspection

6. Component Data and Component History (PIMS).

7. R0058494 dated 4/16/90 and version of Procedure MP E-53.10J, Revision 1 dated 12/18/89, that was in effect during the 2R3 actuator overhaul.

8. Procedure MP E-53.10M, "Limitorque SMB-00 and SB-00 Valve Operator Maintenance," Revision 0. dated 1/22/93; Components Reassembly instructions.

9. DCM S-3B, Revision No. 1; Auxiliary Feedwater System. Applicable sections only.

10. Applicable sections of the FSAR Update

    a. Section 6.5, "Auxiliary Feedwater System"
    b. Section 10.3, "Main Steam System"
    c. Section 15.4, "Accident Analyses, Condition IV - Limiting Faults"

11. NCR DC2-89-OP-N009, "AFW Pump 2-1 Inoperable Due to FCV-37 Being Shut".

12. Herguth Laboratories grease analysis reports.

13. Chron 207103; Memorandum from NES to Electrical Maintenance regarding the design basis for FCV-37 and FCV-38.

H. TRG Meeting Minutes:

On March 23, 1993, the TRG convened and considered the following:

1. Investigative actions 1-5 were assigned.
2. The chronology of the event, as presented in this write-up was discussed.
3. Problem statement development

The upper bearing had visible corrosion. The preliminary analysis of the grease sample showed that the grease had material present (i.e. dirt, rust, metal shavings, etc...). Engineering analysis had determined that the 1/31/93 condition, with the bearing corrosion and stem grease degradation, would not be able to meet its Generic Letter 89-10 design basis function. This is reportable under 50.73. STP P-6B, which 2-FCV-37 failed, tests at approximately 2000 pounds thrust. GL 89-10 mispositioning thrust requirement is approximately 6500 pounds thrust.

TRG PROBLEM STATEMENT: Valve 2-FCV-37 was outside of its design basis prior to January 31, 1993.

NOTE: Subsequent investigation indicates that the valve is not outside its design basis, but was not operable for a period of time, assumed to be greater than 72 hours, prior to January 31, 1993. In-operability of 2-FCV-37 also makes the turbine-driven auxiliary feedwater pump inoperable. The limiting conditions of operation for Tech Spec 3.7.1.2 were exceeded.

The motor breaker for FCV-37 did not trip open when the attempt was made to cycle the valve during the performance of STP P-6B.

In addition, the TRG agreed that the issue, What should DCPP do in the future when a MOV fails a STP, needs to be addressed. For example, what actions and associated actions need to be taken if a MOV fails its STP, corrective maintenance is performed, and then the valve passes the STP? The issue is that the design basis operability needs to be addressed in a timely manner (i.e. the right questions need to be asked). One suggestion is that a POA could be used.

On March 30, 1993, the TRG convened and considered the following:

The root cause was discussed, and the following preliminary root cause was presented: Procedure deficiency in that MP E-53.10J did not have enough detail to ensure that the quad rings were installed during valve assembly.

Investigative Action #6 was assigned.

Reportability was discussed. Can this event be assumed to occur at the time of STP performance, or since (1) engineering determined that the corrosion present would prevent the valve from closing during full flow differential pressure, (2) the corrosion is not a short term failure mechanism, and (3) the valve was last overhauled during 2R3 when the quad rings were left out; there is sufficient evidence to determine that the

valve was inoperable for at least 72 hours prior to the performance of the STP.

The TRG will reconvene on April 9 to discuss contributory causes and associated corrective actions.

On April 2 and again on April 9, 1993 the TRG convened and considered the following:

1. Design Basis of FCV-37; there is no indication that remote closure of FCV-37 is required for operability of the steam-driven AFW pump. Preliminary presentation of this clarification to the PSRC indicates that the PSRC concurs with this, but is concerned whether FCV is required for any other accident mitigation scenarios, i.e. SGTR. PSRC requested further investigation.

2. Discussion as to whether there is firm evidence that FCV was inoperable prior to the performance of STP P-6B on 1/31/93. System Engineering presented the following information to indicate that "firm" evidence does not exist to justify assuming outside of the guidelines of NUREG 1022, Supplement 1 that the failure did not occur at the time of discovery:

* FCV-37 is only required to be open for steam driven AFW pump functionality. This is substantiated by STP V-3R6 not testing the remote operation of the valve from the control room. The as-found condition of FCV-37 on 1/31/93, the valve was capable of being opened electrically, and therefore, the steam-driven AFW pump was never inoperable.

* GL 89-10 test for FCV-37 was successfully completed during 2R4.

* STP P-6B was successfully completed just 14 days prior to the failure of FCV-37 on 1/31/93.

* The Unit trip on 1/30/93 resulted FCV-37 being exposed to steam dump condensation moisture, and therefore, this is the most probable cause of the moisture that resulted in final corrosion inoperability of FCV-37.

* Corrosion is a steady degradation, not a step degradation. This is analogous to a pump performance degradation and associated STP performance. NUREG 1022 allows first STP performance failure as the discovery date and does not require plotting a time-line backwards to try and pinpoint the exact point of failure.

* Since the quad ring is not required for valve operation, this can be considered as a random failure during the STP performance.

* On March 9, 1993 the valve was tested to determine the as-found thrust of the operator. The thrust was sufficient to meet the design requirements of FCV-37, including GL 89-10 mispositioning thrust valves, even with the corrosion present.

NOTE: A conference call held on 4/13/93 between Engineering, Electrical Maintenance, Quality Assurance, System Engineering, and Regulatory Compliance determined that there is no design basis requirement for remote operation of FCV-37 since the emergency operating procedures adequately prevent an additional release, for a SGTR with a stuck open 10% steam dump, path through the turbine-driven AFW pump. Therefore, this event does not appear to be reportable.

April 9, 1993 TRG: Preliminary results of the grease analysis was discussed. Two grease samples from 2-FCV-37 were analyzed:

1. Grease sample from the main gearbox (should be Exxon Nebula EP-0). Visual inspection shows the sample to match the

color of EP-0, but some water and dark swirls were evident, indicating possible contamination. Lab results confirm the grease to be EP-0. Lab results also indicate high silicon, iron and sodium levels indicating water and dirt contamination. The grease had softened to the lower range of acceptability, but would still be usable as a lubricant.

2. Grease sample from the upper roller bearing area. Sample was completely black with excessive amounts of large metal particles and water. The sample was abnormal and unacceptable for use as a lubricant.

The TRG will reconvene on April 22 to discuss contributory causes and associated corrective actions.

On April 22, 1993 the TRG re-convened and considered the following:

The grease sample preliminary analysis from the April 9, 1993 TRG was finalized. The gearbox grease was acceptable. The upper roller bearing grease was confirmed to NOT be Lubriplate nor Nebula EP-0, and the grease was unacceptable for use as a lubricant.

An investigative action to document the basis for non-reportability was agreed to. This AE will be used to track PSRC approval of the FCV-37 operability requirements (i.e. via a revision to TS Interpretation 89-04).

Design basis operability following a failed STP was discussed. It was agreed that better documentation of investigations is required. A prudent action was agreed to for the following issue: If there is any doubt about the cause of a MOV failing to meet STP requirements, what is the best way to document the consequences of failure and subsequent acceptability to return to service.

The need to revise DCM S-3B, "Auxiliary Feedwater System," and possibly DCM T-15, "Radiation Protection" to include an additional safety function for the turbine-driven auxiliary feedwater pump steam supply line. Specifically, this line must be isolated following a SGTR to preclude an offsite radioactive release if SG-2 or SG-3 has the ruptured tube(s). Please include the allowable time for isolation of this line following a SGTR.

A NCR closure date of 7/13/93 was established.

The TRG will re-convene on 5/13/93 for review of the NCR writeup and final signoff.

On May 13, 1993 the TRG re-convened and considered the following:

The safety analysis, as revised based on NES engineering input, was reviewed. See above.

The TRG concurred that the NCR does not need to track PSRC review of a rescission to TS Interpretation 89-04. This is reflected in the response to Investigative Action #7 (A0298496, AE #7).

I. Remarks:

None.