50-275/323-OLA-2 I-MFP-192

MFP Exhibit 192
8/23/93
Dollie Feigel
164388

Pacific Gas and Electric Company

77 Beale Street
San Francisco, CA 94106
415/973-4684
TWX 910-372-6587

James D. Shiffer
Senior Vice President and
General Manager
Nuclear Power Generation

January 25, 1991

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DIABLO CANYON POWER PLANT REGULATORY COMPLIANCE
TLG DDM MTH LEE JAC JRH RIJ SDW JCN PGD MLO
JAN 3 1 1991

PG&E Letter No. DCL-91-013

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

Re: Docket No. 50-275, OL-DPR-80
Diablo Canyon Unit 1
Licensee Event Report 1-90-015-01
ESF Actuation, P-14 (High-High Steam Generator Water Level),
Due to Feedwater Regulating and Bypass Valves Leakage

Gentlemen:

Pursuant to 10 CFR 50.73(a)(2)(iv), PG&E is submitting the enclosed supplemental Licensee Event Report (LER) regarding a high-high steam generator water level signal which initiated a P-14 signal, an engineered safety feature (ESF) actuation.

NRC Region V's letter to PG&E, dated January 18, 1991, requested that PG&E provide clarification on LER 1-90-015 Revision 0, submitted to the NRC in PG&E letter DCL-91-003, dated January 7, 1991. The enclosed LER 1-90-015 Revision 1 addresses the NRC's request and clarifies the cause of the event, analysis of the event, and corrective actions to prevent recurrence. Margin bars indicate the revised LER sections.

Sincerely,

J. D. Shiffer

cc: A. P. Hodgdon
J. B. Martin
P. J. Morrill
P. P. Narbut
H. Rood
CPUC
Diablo Distribution
INPO

NUCLEAR REGULATORY COMMISSION
Docket No. 50-275-OLA Official Exh. No. MFP-192
In the matter of PACIFIC GAS and ELECTRIC Co
Staff ____ IDENTIFIED ✓
Applicant ____ RECEIVED ✓
Intervenor ✓ REJECTED ____
Cont'g Off'r ____
Contractor Ann Riley & Assocs DATE 8-23-93
Other ____ Witness ____
Reporter Dollie Feigel

DC1-90-OP-N083

Enclosure

5190S/0085K/JHA/2246

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PFS
Corresp. Control Center

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LICENSEE EVENT REPORT (LER)

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FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4): UNIT 1 ESF ACTUATION, P-14 (HIGH-HIGH STEAM GENERATOR LEVEL), DUE TO FEEDWATER VALVE LEAKAGE

EVENT DATE (5) MON	DAY	YR	LER NUMBER (6) YR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MON	DAY	YR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER (S)
12	08	90	90	- 0 1 5 -	0 1	01	25	91		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 2

POWER LEVEL (10): 0 0 2

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (11)

X 10 CFR 50.73(a)(2)(iv)

OTHER - (Specify in Abstract below and in text, NRC Form 366A)

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER AREA CODE	
MARTIN T. HUG, REGULATORY COMPLIANCE SENIOR ENGINEER	805	545-4005

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (if yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR

ABSTRACT (16)

On December 8, 1990, at 0031 hours PST, with Unit 1 in Mode 2 (Startup) at approximately 2 percent reactor power, steam generator (SG) 1-3 level exceeded the 67 percent narrow range level setpoint initiating a P-14 signal, an engineered safety feature (ESF) actuation.

The root cause of the event is leakage through feedwater regulating and bypass valves FW-1-FCV-530 and FW-1-FCV-1530. Leakage past feedwater check valve FW-1-531 and feedwater recirculation control valve FW-1-FCV-420 contributed to this event. Other contributing causes included plant management's failure to recognize the consequences of the combined leakage of the feedwater check and regulating valves, and to provide direction for either a timely repair of the feedwater valves or an adequate strategy for operators to enable them to startup the unit with leaking feedwater valves.

Operating procedure OP L-3 was revised to include steps to check for main feedwater regulating and bypass valve leakage during startup, and to provide guidance for taking action to deal with any leakage identified. OP L-3 was also revised to require isolating FW-1-FCV-420 when it is not in service. FW-1-FCV-420 was isolated and will be repaired during the next refueling outage. Corrective maintenance was performed on FW-1-FCV-530, FW-1-FCV-1530, and FW-1-531 during a Unit 1 forced outage. Further maintenance will be performed on FW-1-FCV-530 during the next refueling outage. A memorandum will be issued to plant staff regarding this event. Operators were informed to obtain management concurrence on recovery plans.

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I. Plant Conditions

Unit 1 was in Mode 2 (Startup) at approximately 2 percent reactor power. The plant was in the process of starting up from a reactor trip which occurred on December 5, 1990.

II. Description of Event

A. Event:

On December 8, 1990, at 0031 hours PST, with Unit 1 in Mode 2 (Startup) at 2 percent reactor power, steam generator (SB)(SG) 1-3 exceeded the 67 percent narrow range level setpoint, which initiated a P-14 signal. This engineered safety feature (ESF) actuation occurred during the transfer of the feedwater flow controls (SJ)(FC) from auxiliary feedwater to automatic main feedwater level control and resulted in main feedwater isolation, main feedwater pump (MFP) 1-2 (SJ)(P) trip, and main turbine (TA)(TRB) trip.

Figure 1 provides a schematic of the main feedwater system.

Three days earlier, on December 5, 1990, Unit 1 had experienced a reactor trip. During recovery from that trip, backleakage was experienced through SG 1-3 inlet feedwater check valve FW-1-531 (SJ)(V). Operations management believed this leakage to be within the capability of the operators control since similar leakage had been observed during previous startups. Therefore, valve corrective actions were deferred until the next Unit 1 refueling outage.

On December 7, 1990, at approximately 2300 hours, the control room operators had just completed the procedural steps to place MFP 1-2 in service with pump speed adjusted for a feedwater/main steam (SJ)/(SB) differential pressure of approximately zero psid. Main feedwater isolation valve FW-1-FCV-440 (SJ)(FCV) was open. Feedwater regulating and bypass valves (SJ)(FCV), FW-1-FCV-530 and FW-1-FCV-1530, respectively, indicated zero demand with the feedwater controller in automatic.

On December 8, 1990, shortly after 0000 hours, the control room operators observed that SG 1-3 level was decreasing, with indicated auxiliary feedwater (AFW)(BA) flow to SG 1-3 greater than flow to the other three SGs. This problem was initially diagnosed by the operators as backleakage through feedwater check valve FW-1-531 because, as discussed above, backleakage was experienced during recovery from the December 5, 1990 reactor trip. Backleakage was also experienced during previous startups when low differential pressures existed across the check valve.

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To increase SG 1-3 level, operators increased speed on MFP 1-2 to prevent the backleakage and feed SG 1-3 from the main feedwater system. The level in SG 1-3 rapidly increased beyond its normal no-load level setpoint. The control room operators attempted to close feedwater regulating and bypass valves FW-1-FCV-530 and FW-1-FCV-1530 by manually decreasing valve demand, even though the demand was at zero. This action appeared to terminate the initial steam generator level increase, so the decision was made to proceed with the Unit 1 startup.

The main feedwater controls (SJ)(LIK) were placed in automatic and the level in SG 1-3, as indicated on the narrow and wide range level instrumentation (SJ)(LI), immediately began to increase. Wide range level appeared to stabilize shortly into the transient. The control room operators concluded that the narrow range level increase was caused by SG swell and since wide range level did not show an increasing trend, the reactor power was slowly increased. Reactor coolant system (RCS)(AB) temperature was increasing and the steam dumps (SB)(V) opened. The narrow range level increased to 67 percent, causing the P-14 actuation. The Operations crew on shift determined that the P-14 actuation was caused by steam generator overfeeding due to leakage past feedwater regulating valve FW-1-FCV-530 and/or feedwater regulating bypass valve FW-1-FCV-1530, and subsequent SG swell during steam dump actuation. At that point, it could not be definitively determined which valve or whether both valves were leaking because the valves are in parallel.

Operators restored plant conditions to those which existed prior to the P-14 actuation and continued with plant startup without consulting with plant management.

On December 21, 1990, as a result of further investigation regarding feedwater system leakage, Operations determined that feedwater recirculation control valve FW-1-FCV-420 (SJ)(FCV) was leaking to the condenser. This valve was manually isolated.

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

None.

C. Dates and Approximate Times for Major Occurrences:

1. Dec. 8, 1990, 0025 hrs: MFP 1-2 is placed in service.

2. Dec. 8, 1990, 0031 hrs: Event\Discovery date. P-14 ESF signal occurred.

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3. Dec. 8, 1990, 0146 hrs: Four hour non-emergency report was made to the NRC in accordance with 10 CFR 50.72.

D. Other Systems or Secondary Functions Affected:

None.

E. Method of Discovery:

The event was apparent to control room personnel due to control room alarms and indications.

F. Operator Actions:

The operators stabilized the plant in Mode 2, verified that the SG levels were being maintained by the AFW system, and reestablished plant conditions to those existing prior to the P-14 signal.

G. Safety System Responses:

1. MFP 1-2 tripped.
2. The main turbine tripped.
3. The main feedwater isolation valves automatically closed.
4. The main feedwater regulating valves and regulating bypass valves automatically closed.

III. Cause of the Event

A. Immediate Cause:

The immediate cause for the SG high-high level signal was that SG 1-3 level exceeded the P-14 actuation setpoint due to overfill and subsequent swell during startup.

B. Root Cause:

The root cause of the event was leakage through feedwater regulating valve FW-1-FCV-530 and feedwater regulating bypass valve FW-1-FCV-1530. FW-1-FCV-530 was stroke tested during a Unit 1 forced outage which began on December 23, 1990, and it was determined that leakage through this valve was due to valve position controller drift. FW-1-FCV-1530 was disassembled and inspected during the same forced outage. Leakage through this valve was also due to valve position controller drift.

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C. Contributory Cause:

1. Backleakage through SG 1-3 inlet feedwater check valve FW-1-531 due to a slight misalignment of the check valve disc contributed to this event since the leakage through FW-1-531 in combination with leakage through FW-1-FCV-530 and FW-1-FCV-1530 initiated the level transient in SG 1-3.

2. On December 21, 1990, as a result of further investigation regarding feedwater system leakage, Operations determined that feedwater recirculation control valve FW-1-FCV-420 was leaking to the condenser. Leakage past the valve contributed to this event in that it provided a leak path to the condenser.

3. During recovery from the Unit 1 reactor trip on December 5, 1990, backleakage was experienced through SG 1-3 inlet feedwater check valve FW-1-531. Operations management believed that this leakage was within the capability of the operators control since similar leakage had been observed during previous startups from post-trip outages in February and June 1990. Plant management's failure to recognize the consequences of the combined leakage of the feedwater check and regulating valves contributed to this event.

4. Procedures for placing the main feedwater system in service (feeding forward to the steam generators) did not provide guidance for contingency actions to investigate feedwater regulating valve leakage. Plant management's failure to provide direction for either a timely repair of the feedwater valves or an adequate strategy for operators to enable them to startup the unit with leaking feedwater valves was a contributing cause.

IV. Analysis of the Event

A. Leaking Feedwater Regulating Valves

The high-high SG water level P-14 protective interlock is designed to protect the main turbine from water intrusion and subsequent damage caused by SG overfill. This feature is designed primarily for equipment protection, and functioned as designed. Leakage past the feedwater regulating valves is bounded by the analyzed event of a failed full-open feedwater regulating valve.

B. Leaking Feedwater Check Valve

Feedwater check valve FW-1-531 is a safety category I valve installed to protect the SG and RCS from excessive cooldown due to blowdown following a postulated line rupture in the safety category II portion of a feedwater

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line. The complete failure of FW-1-531 would constitute a main feedwater line failure, which is a previously analyzed FSAR Update accident.

The condition of the check valve has been evaluated for potential effect on feedwater line integrity and for potential effect on water supply to the SGs.

Valve integrity was investigated by physical inspection of the valve. The valve inspection showed no physical damage to the disc or seat that would impair disc movement or lead to catastrophic disc failure. The leakage was due to misalignment of the disc and the seat which resulted in lack of contact on the full circumference of the seat. At higher pressure differentials, the disc would elastically deflect and provide a seal, as was evident by successful surveillance test results. The inspection verified there was no impairment in integrity or in the ability to close in response to a differential pressure of greater than 500 psig.

The effect of FW check valve leakage on feedwater delivery to the SGs was investigated by evaluating the check valve leakage on the performance of the AFW system. Chapter 6 of the FSAR Update summarizes the flow requirements for the various design basis accidents. As shown in FSAR Update Table 6.5-2, a minimum flow of 440 gpm is required to be delivered to two SGs 10 minutes after a main feedwater line break. The main feedwater line break accident is the most limiting Chapter 15 accident with respect to AFW flow requirements. The feedwater line break accident was reviewed to evaluate the effect of a leak in the feedwater line check valve. The worst case scenario is considered to be a feedwater line break occurring in SG 2, one of the two steam generators which supplies steam for the turbine driven auxiliary feedwater (TDAFW) pump. The leaking feedwater line check valve is in the feedwater line to SG 3 which is the second source of steam to the TDAFW pump turbine. In this case, the feedwater line break would result in a rapid loss of steam pressure in the affected SG and the check valve leak would reduce auxiliary feedwater flow to the other steam generator (SG 3).

The availability of the steam sources for the TDAFW pump turbine were evaluated. SG 2 would be faulted due to the feedwater line break and would not be considered a viable steam source. Further, the delivery of auxiliary feedwater to SG 3 would be reduced due to backleakage through the feedwater check valve. SG 3 inventory and steam pressure may decay as a result of reduced auxiliary feedwater flow. The inventory and pressure decay would be slow, however, and SG 3 would provide steam for some period. Since the steam pressure response of the SG has not been quantified, a conservative approach was taken by assuming no steam would be available from SG 3. Thus, the TDAFW pump is conservatively assumed to be not operable.

This feedwater line break scenario was evaluated with the application of a single failure. The limiting single failure is bus H which takes motor driven auxiliary feedwater (MDAFW) pump 2 out of service. The bus H failure also removes power from the feedwater isolation valve on the SG 3 feedwater line. Without power, the feedwater isolation valve FW-1-FCV-440 would not close on the safety injection signal and would not isolate the check valve leak. With the leak not isolated, full AFW delivery to SG 3 could be impaired and, consequently, FSAR Update flow requirements may not be satisfied. This single failure results in MDAFW pump 3 feeding the feedwater line (SG 3) with the leaking check valve and feeding the unaffected SG 4. Due to the feedwater check valve leakage, the delivery of AFW to SG 3 could be reduced. This would result in SG 3 levels decreasing faster than SG 4 levels. This disparity would be evident to the control room operators. The operators would close the auxiliary feedwater regulating valves from the control room within 10 minutes to isolate auxiliary feedwater to the leaking feedwater line and to the damaged steam generator as required by procedure. Subsequently, flow from MDAFW pump 3 would be delivered to SG 4 only. The flow rate from the pump would not reach full design flow due to the increased hydraulic resistance in the auxiliary feedwater flow path to SG 4, since the pump is feeding only one line instead of two.

The condition of one steam generator being supplied from one MDAFW pump following a feedwater line break has been evaluated for DCPP by Westinghouse Electric Corporation (Reference letter PGE-89-526). The results of that evaluation show that the conclusions of the FSAR Update remain valid for this postulated condition. The following assumptions were made in the evaluation:

> The TDAFW pump turbine is disabled due to interruption of both steam sources following a feedwater line break.
>
> A single failure of the MDAFW pump which is not associated with the faulted steam generator occurs.
>
> Ten minutes after reactor trip, operator action isolates the feedwater line break and water is supplied to one intact SG. Estimated flow to the intact SG is 325 gpm.
>
> Thirty minutes after reactor trip, operator action increases the auxiliary feedwater delivery to 440 gpm. The additional auxiliary feedwater is fed to at least two SGs.

This evaluation was performed for two cases: with off-site power and without off-site power. The results were shown to be within FSAR Update limits by showing that no boiling would occur in the hot leg of the RCS and that the pressurizer would not fill.

The operators are trained to verify adequate auxiliary feedwater flow and SG levels following an accident. 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 instructs the operators to restore at least 460 gpm (indicated flow) of auxiliary feedwater flow to the SGs by performing local manual valve alignments as necessary to achieve the minimum flow requirements to two SGs. The operators could close the SG 3 feedwater isolation valve to isolate the back leakage through the check valve and resume auxiliary feedwater flow to SG 3 and/or open the MDAFW discharge line cross-tie to provide flow to SG 1 from MDAFW pump 3. These actions are consistent with the assumptions used in the Westinghouse evaluation.

The evaluation of the consequences of a leak in the feedwater line check valve, the results of the Westinghouse evaluation, and review of the DCPP emergency procedures show there were no adverse safety consequences resulting from the event or from the leaking check valve. Further, the P-14 actuation and the leaking feedwater valves resulted in no adverse safety consequences. Therefore, the health and safety of the public were not adversely affected by this event.

V. Corrective Actions

A. Immediate Corrective Actions:

None.

B. Corrective Actions to Prevent Recurrence:

1. Operating procedure OP L-3, "Secondary Plant Startup," was revised to include steps to check for main feedwater regulating and bypass valve leakage during startup, and to provide guidance for taking action to deal with any leakage identified.

2. Feedwater heater recirculation control valve FW-1-FCV-420 was manually isolated and corrective maintenance will be performed during the Unit 1 fourth refueling outage, scheduled to start February 3, 1991. Operating procedure OP L-3 was revised to require closing an additional manual isolation valve in the flow path of FW-1-FCV-420 when FW-1-FCV-420 is not in service.

3. The following corrective maintenance was performed during a Unit 1 forced outage which began on December 23, 1990:

a. The position controller for the feedwater regulating valve FW-1-FCV-530 was adjusted to correct for drift. However, the valve exhibited minor leakage during restart and, in accordance with revised Operating Procedure L-3, the valve was isolated until the Unit reached 12 percent power. During the next refueling outage, further valve corrective maintenance will be performed, including calibration of the position controller.

b. Feedwater regulating bypass valve FW-1-FCV-1530 was disassembled, the valve seat was lapped, and the valve plug was replaced. The valve was reassembled and given a successful leak check. The valve position controller was adjusted to correct for drift.

c. SG 1-3 inlet check valve FW-1-531 was disassembled and inspected, and a misalignment of the check valve disc was corrected.

4. A memorandum from the Plant Manager will be transmitted to department heads and other plant supervision regarding this event to emphasize that all equipment problems should be adequately evaluated in a timely manner and appropriate compensatory measures should be identified.

5. Following the P-14 actuation, operators continued with plant startup without consulting plant management. Although this is not a cause of the P-14 actuation, the Assistant Plant Manager, Operations, has reviewed with Operations supervisors the importance of obtaining plant management concurrence for recovery plans following ESF actuations and other significant plant transients.

VI. Additional Information

A. Failed Components:

None.

B. Previous LERs on Similar Events:

1. LER 2-90-007, "Unit 2 ESF Actuation, P-14 (High-High Steam Generator Level) Due To Unanticipated Steam Generator Swell At Low Power Levels." These P-14 actuations occurred during main turbine testing with the SG level controls in manual. Operators raised the SG levels high in the band in anticipation of shrink with main turbine intentional trip, but the SGs were fed at too high a rate and the warming water swelled to the P-14 actuation setpoint. An Operations Incident Summary was issued to sensitize operators during future similar evolutions. This corrective action would not have prevented the current event because the current event was caused by leaking feedwater valves.

2. LER 1-87-025, "High Steam Generator Water Level Main Turbine Trip and Feedwater Isolation During Startup Due To Lack Of Guidance For Operators On Proportional Integral Controllers." This P-14 actuation occurred due to a lack of guidance for

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operators when transferring from manual to automatic SG level control. An Operations Memorandum was issued at that time to provide operator guidance. This corrective action would not have prevented the current event because the current event was caused by leaking feedwater valves.

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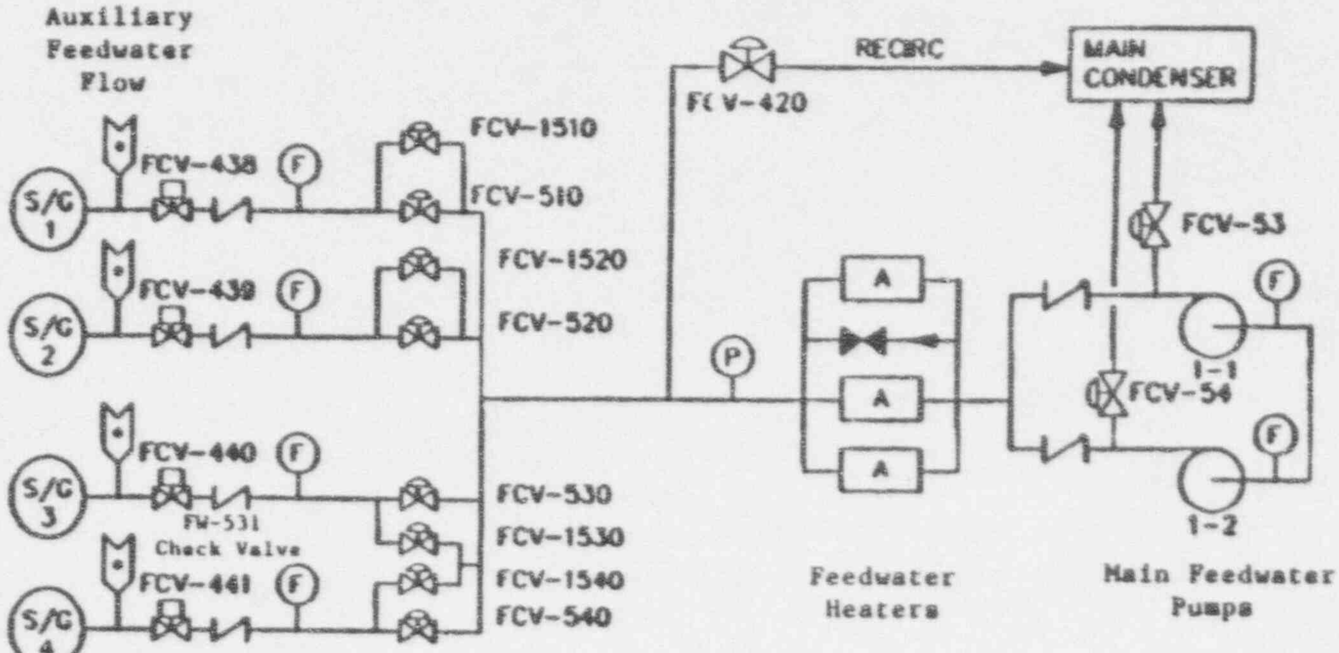
MAIN FEEDWATER SYSTEM
ONE LINE DIAGRAM



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