Florida Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302

February 14, 1997
3F0297-07

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555-0001

Subject: Licensee Event Report (LER) 96-024-01

Dear Sir:

Please find the enclosed supplement to Licensee Event Report (LER) 96-024-00 concerning an unanalyzed condition regarding a plant modification which could have prevented the fulfillment of a safety function involving emergency feedwater.

This report is submitted in accordance with 10 CFR 50.73 and includes a discussion of consequences, the results of a comprehensive root cause analysis, and identification of corrective actions.

Sincerely,

P.M. Beard, Jr.
Senior vice President
Nuclear Operations

PMB/TWC
Attachment

xc: Regional Administrator, Region II
Project Manager, NRR
Senior Resident Inspector

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NRC FORM 366 (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED OMB NO. 3150-0104

EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HOURS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
CRYSTAL RIVER UNIT 3 (CR-3)	05000302	1 OF 09

TITLE (4)
Plant Modification Creates Unanalyzed Condition Regarding Emergency Feedwater Availability

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)
									N/A	05000
10	11	96	96	— 024 —	01	02	14	97	N/A	05000

OPERATING MODE (9)	5
POWER LEVEL (10)	000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (CHECK ONE OR MORE OF THE FOLLOWING) (11)

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	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, NRC Form 366A)
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)	X	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)
T. W. Catchpole, Sr. Nuclear Licensing Engineer	352 563-4601

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

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

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)

On October 11, 1996, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE 5 (COLD SHUTDOWN). During a review of a diesel loading calculation, FPC engineers determined the calculation assumed the steam-driven Emergency Feedwater Pump (EFP) was running when the motor-driven EFP received an Engineered Safeguards (ES) automatic trip signal at Reactor Coolant System (RCS) pressure of 500 pounds per square inch gauge (psig) for Low Pressure Injection (LPI) actuation. A plant modification had removed the automatic start signal from the "A" side of the Emergency Feedwater Initiation and Control system to prevent runout and net positive suction head concerns with the steam-driven EFP during certain accident conditions when its flow control valves would fail open. This caused CR-3 to be in an unanalyzed condition with no emergency feedwater or LPI flow for the period of time between the 500 psig actuation signal and when RCS pressure is reduced below the LPI shutoff head (approximately 185 psig) when EFW is no longer required for residual heat removal. Subsequent analyses indicate the minimum time to depressurize the RCS using the steam generators, to at or below the LPI/EFP-1 trip setpoint, would provide sufficient time for operators to diagnose the event and take the necessary actions. Among the causes of the event were inadequate translation of assumptions from earlier designs, inadequate configuration management, and inconsistent 10CFR50.59 evaluations. Corrective actions include plant modifications, 10CFR50.59 program enhancements, additional training, corrections to design basis documents, and personnel accountability.

EVENT DESCRIPTION

On October 11, 1996, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE 5 (COLD SHUTDOWN). During a review of the Emergency Diesel Generator [EK,DG] (EDG) loading calculation, FPC engineers determined the calculation assumed the steam-driven Emergency Feedwater Pump [BA,P](EFP-2) was running when the motor-driven emergency feedwater pump, EFP-1, received an Engineered Safeguards (ES) automatic trip signal at Reactor Coolant System [AB](RCS) pressure of 500 pounds per square inch gauge (psig). The automatic trip feature was installed in June, 1990 in order to prevent electrical loads on the EDG's from being greater than the 30-minute engine rating during certain design basis accident scenarios which require Low Pressure Injection [BP](LPI) operation.

FPC engineers determined that the above assumption was invalidated by Modification Approval Record (MAR) 96-04-12-01, "ASV-204 EFIC Auto Open Removal," which was implemented in May, 1996 during Refueling Outage 10. The intent of the modification was to eliminate one of two automatic actuations of EFP-2 by removing the automatic opening of the redundant steam admission valve [SA,ISV](ASV-204) previously initiated from the "A" train of the Emergency Feedwater Initiation and Control [JB](EFIC) system in applicable transient conditions. The "B" train automatic actuation of EFP-2 remained active. This was determined to be necessary to address EFP-2 runout and net positive suction head (NPSH) concerns following a loss of the "B" DC power train [EJ,BTRY] concurrent with a Loss of Offsite Power (LOOP) and Loss of Coolant Accident (LOCA). The loss of DC power condition would result in failure to control the "B" train emergency feedwater flow control valves which would fail open. EFP-2 would remain available but require operator action to cross-tie EFP-2 flow to the EFP-1 flow path control valves which would have control power available during a loss of "B" DC Power train scenario.

The Emergency Feedwater System is required to assure an adequate emergency supply of feedwater to the Once-Through-Steam-Generators [AB,SG](OTSG) to ensure a heat sink to remove reactor decay heat until suitable conditions are attained to start the Decay Heat Removal System [BP](DH). The Decay Heat system functions as the LPI part of the Emergency Core Cooling System (ECCS) for LOCA's in the RCS. In the event of a small break LOCA, RCS pressure may not decrease to the point where the LPI pumps [BP,P] can inject water into the vessel [RPV]. Analyses exist indicating simultaneous operation of LPI and EFW is not required; however, no analysis existed at the time of this event was determined reportable to support operation without EFW between RCS pressures of 500 psig when the LPI pumps start, and the point at which RCS pressures are below the LPI shutoff head (approximately 185 psig). Therefore, with no credit taken for the manual operator action described above to cross-tie EFP-2 to EFP-1 flow path control valves, CR-3 operated in an unanalyzed condition. This condition existed between the time frame of startup from Refueling Outage 10 on May 17, 1996 and September 6, 1996 when the plant was taken to MODE 5 for a forced outage.

A Problem Report was initiated to describe the above conditions and was presented to the Shift Supervisor on Duty (SSOD) at 1750 hours on October 11, 1996. The SSOD then made a four-hour event notification at 1812 hours in accordance with 10CFR50.72(b)(2)(i) as an event found while the reactor was shut down that had it been found while the reactor was in operation, would have resulted in the plant being in an unanalyzed condition. This issue was assigned Event Number 31141.

This report is being submitted in accordance with 10CFR50.73(a)(2)(ii)(A) to describe an unanalyzed condition that could have compromised plant safety.

This event is the second reported issue related to the modification which removed the automatic open signal from ASV-204. LER 96-020 was provided as a voluntary report in October, 1996 to describe the impact of that modification on EDG loading.

BACKGROUND/SYSTEM DESCRIPTION

The Emergency Feedwater (EFW) system provides secondary coolant to the Once Through Steam Generators (OTSG) in the event the Main Feedwater System [SJ] is rendered inoperable and is unable to perform this function. The EFW system has two equipment trains (See Figure 1). Each train is capable of feeding both Steam Generators. The two trains take suction from a common line. The flow control valves associated with each pump operate on DC power. The valves are normally open, and require power to close. The valves remain open when the EFW system is in the standby mode. The "A" train pump is EFP-1 and is aligned to the "A" Emergency Diesel Generator during LOOP conditions. The "B" train pump is EFP-2. Motive Steam for the EFP-2 turbine is supplied from the Main Steam [SA](MS) header. The system includes two valves, ASV-5 and ASV-204, which open to admit steam to EFP-2 when the Emergency Feedwater system actuates. The valves are installed in parallel with one another. Only one of the valves must open in order to start the pump. Valve ASV-5 receives an OPEN command from an actuation of the "B" Emergency Feedwater Initiation and Control (EFIC) train. When ASV-204 was installed in 1985, the valve was powered from "B" Class 1E power sources and received its OPEN command from the "B" EFIC train. In 1987, FPC moved the ASV-204 power supply and OPEN command to the "A" side Class 1E sources so the valve opened on the "A" EFIC actuation train signal thus allowing use of EFP-2 for "A" EDG load reduction.

Since the "A" EDG did not have the capacity to support both EFP-1 and the Low Pressure Injection (LPI) pumps concurrently, a modification was installed in June, 1990 to trip EFP-1 and start the LPI pumps when RCS pressure dropped below the LPI setpoint of 500 psig. This scenario would occur during a postulated Loss of Offsite Power (LOOP) and a Small Break Loss of Coolant Accident (SBLOCA).

For Small Break LOCA events, the EFW system will be initiated by an ES signal. Under LOOP conditions, EFIC raises the level in the steam generators to the natural circulation setpoint automatically selected by EFIC. With EFIC in automatic, this process takes 30 minutes to 2 hours depending on OTSG pressures and initial levels.

For events involving loss of subcooling margin (LSCM), there is also an Emergency Core Cooling System (ECCS)/Loss of Subcooling Margin setpoint selected by pushbutton for LSCM and boiler condenser cooling after Reactor Coolant pumps [AB,P] are manually tripped.

EVENT EVALUATION

For the larger break sizes that result in a rapid depressurization of the RCS to 500 psig or lower, the break in itself is sufficient to prevent the RCS from repressurizing, and the energy absorption of the High Pressure Injection system [BQ](HPI) is sufficient to match core decay heat and residual stored energy in the metal. Since these break sizes are large enough to depressurize the RCS below the steam generator pressure, the OTSG's are a heat source to the RCS and EFW flow is not required for accident mitigation.

The Small Break LOCA analyses assume that EFW is always available, fills the OTSG's to the required level, and maintains that level. For smaller break sizes, operator action to steam the OTSG's is required to cool the RCS. It can be postulated, for a certain range of small breaks, that if EFW is lost during the cooldown due to the LPI/EFP-1 trip block, the RCS may repressurize resulting in inadequate HPI flow to cool the core. If the Borated Water Storage Tank [BP,TK](BWST) is emptied, requiring the HPI pumps to be in the "piggyback" mode, EFW flow could also be lost resulting in the same consequence. "Piggyback" describes the lineup in which LPI pumps take suction from the Reactor Building [NH] Sump in order to provide adequate NPSH to the HPI pumps. In these cases (SBLOCA, LOOP, and loss of "B" DC power), operator action is required to manually start EFP-2 to provide emergency feedwater flow to the OTSG's for secondary heat removal.

The minimum time to depressurize the RCS using the OTSG's, to at or below the LPI/EFP-1 trip block actuation setpoint, has been estimated to be approximately one hour using a conservative cooldown rate of 100 degrees Fahrenheit (F) per hour. For the break sizes of concern, it will require even a longer period of time to drain the BWST, approximately 2 to 2.5 hours. These time periods would have allowed sufficient time for operators to diagnose the event and take the necessary corrective actions. These actions were established in an Abnormal Procedure (AP-770) and included establishing a crosstie of EFP-2 to the EFP-1 flow control valves since these components would have power in a loss of the "B" DC bus scenario. EFP-2 could then be manually started from the main control room allowing EFW flow to maintain the required steam generator levels. Thus, the consequences of this event are acceptable from a safety analyses perspective and would not have had an impact on the safety of the general public.

CAUSE

The Root Cause investigation for this event involved extensive document research and interviews. There were several causes identified upon analysis of

inappropriate actions which fall into the following organizational and programmatic areas:

Failure to Adequately Translate Design Assumptions

The engineers who developed the EFP-1/LPI Interlock modification (MAR 88-05-24-01) installed in 1990, did not translate EFP-2 running assumptions needed when EFP-1 trips at 500 psig during LOOP conditions from source level documents into the MAR design input record. The source document used for MAR 88-05-24-01 does not discuss the difference between LPI actuation pressure and LPI pump shutoff head. However, it did state that FPC should verify that at the point in time when the motor driven EFW pump is shown to be tripped, plant cooling is assured through other means for all scenarios. Failure to identify the EFP-2 running assumptions and to identify within the above modification, the means for plant cooling in this scenario resulted in a less than adequate design basis document, FSAR, and NRC Safety Evaluation Report (NRC letter dated June 4, 1990) for license Amendment Number 130 issued in support of the modification.

Inadequate Configuration Management

A temporary change to the EFW/EFIC design basis document (DBD) was initiated in November, 1989, addressing the EFP-1/LPI Interlock modification at approximately the same time an effort was established to replace DBD's with "enhanced" design basis documents (EDBD's). The original issue of the EDBD for the EFW/EFIC systems involved B&W in its preparation and was issued in March, 1990. This original issue did not incorporate the November, 1989 temporary change and therefore did not reference the tripping of EFP-1 at 500 psig RCS pressure. This error contributed to the engineer responsible for the ASV-204 modification not being aware of key design input information regarding the EFW system.

Commitment to Formation of Project Teams

FPC management was not always sensitive to the need for formation of Project Teams prior to and during Refuel 10 for "small" projects. Existing guidance states, regardless of complexity or financial impact, the project team should consist of the design engineer, system engineer and project manager. While preparing the ASV-204 modification, the design engineer was aware of widespread dissatisfaction with the 1987 modification which aligned ASV-204 to the "A" train power supply and introduced cross-train dependency between the "A" EDG and EFP-2. Also, personnel considered the ASV-204 modification to be a minor project (consisting of lifting two leads). As a result, due to the small work scope, the common desire to eliminate cross-train dependency, and absence of a Project Team, the system engineer was not involved in the development of design inputs which contributed to the design engineer not being aware of certain known active failures which, had they been discussed, could have provided an opportunity to identify the effects of tripping EFP-1 at 500 psig RCS pressure under LOOP conditions.

Inconsistent Level of Quality in 10CFR50.59 Evaluations

FPC has not maintained a consistent level of quality in 10CFR50.59 evaluation development and reviews. As a result, design engineering personnel did not adequately respond to 10CFR50.59 questions regarding the reduced reliability of EFP-2 when the EFIC automatic start signal was removed from ASV-204 during Refuel 10. In addition, the Plant Review Committee (PRC) did not exercise sufficient questioning attitude in recommending approval of the modification based on their review of the 10CFR50.59 evaluation and supporting documentation.

Modification Package Review Expectations Not Adequately Conveyed

Modification package review expectations were not clearly established regarding reviews by Operations and System Engineering personnel prior to installation. This contributed to some confusion which resulted in inconsistent expectations between designers and reviewers. Due to outage work loads and rotating shifts, reviews of modification packages were not always reviewed by operations personnel who normally performed such reviews. In addition, the ASV-204 modification was reviewed by the backup EFIC system engineer and was not reviewed by the Diesel Generator, EFW, or primary EFIC system engineers.

IMMEDIATE CORRECTIVE ACTION

Due to the EFW/EDG issues, and other design-related issues, FPC management made a decision to keep the plant shut down in the current forced outage until these issues are adequately addressed. FPC notified the NRC of its decision to keep the plant shut down by FPC Letter 3F1096-22 dated October 28, 1996. This letter included a description of planned modifications to address this and other design margin issues. FPC has subsequently formed a Restart Panel patterned after the NRC 0350 process to manage the actions necessary to safely return CR-3 to power operation and ensure subsequent reliable operation. This decision was communicated to NRC by Letter 3F1296-05 dated December 13, 1996. The restart issues include design and system readiness reviews and other activities in place to determine the extent of conditions described in this report. For reference purposes, the following additional corrective actions are identified as applicable, with Restart Issue numbers found in Letter 3F1296-05.

ADDITIONAL CORRECTIVE ACTION

Appropriate emergency feedwater system modifications such as installation of cavitating venturis will be implemented to eliminate NPSH concerns and reduce operator burden prior to restart from the current forced outage. Appropriate license amendment request(s) will be submitted in reference to the operability requirements of EFP-2 and ASV-204. (Restart Issues D-5 and R-4)

Interim direction has been provided by nuclear engineering management stating that all design modifications will receive a 10CFR50.59 evaluation. Exceptions involving the use of 10CFR50.59 screening will require management authorization until 10CFR50.59 process changes have been made and a training and qualification program has been implemented. This direction also established that 10CFR50.59 evaluations will be "stand-alone" documents to eliminate any uncertainty with respect to Unreviewed Safety Questions (USQ). (Restart Issue OP-5)

A separate group has been established within the nuclear engineering organization, responsible for reviewing 10CFR50.59 evaluations for modifications. This group has been tasked with reviewing a sample of past modification 10CFR50.59 evaluations to determine the extent of condition and ensuring future 10CFR50.59 evaluations meet the "stand-alone" criteria established by nuclear engineering management. The group consists of experienced individuals knowledgeable of accident analyses and safety assessments. (Restart Issues D-15 and D-23)

Operating procedures, calculations and design basis documents will be updated to reflect the EFW system modifications implemented as a result of this event with emphasis on EFP-2 running assumptions.

Several accountability issues have been addressed with responsible personnel including senior management, the former Director of Nuclear Engineering and Manager, Nuclear Engineering Design, and the design and verification engineers who prepared the ASV-204 modification.

ACTION TO PREVENT RECURRENCE

1. The extent of condition of the omitted temporary change to the EFW/EFIC design basis document will be determined based on a review of currently installed temporary changes that were issued on individual system sections during the DBD enhancement process. These Temporary Changes will be checked against the EDBD to ensure they were indeed incorporated.

2. A System Ownership Team (SOT) will be developed for each system to include a Design Engineer, a System Engineer, and an Operations System Owner. The expectations for the SOT have been formally established, making them aware of their ownership of the assigned system.

3. Expectations for Project Team formation will be strengthened.

4. Expectations for review of modification packages will be established to address consistency of reviews by system engineers and plant operations representatives.

5. A Design Review Board (DRB) has been established consisting of interdisciplinary members from the design, systems, operations, maintenance,

training, and licensing organizations. The DRB has replaced the prior Design Engineering Review Board (DERB) whose membership was limited to engineering personnel. Generally, projects which have an effect on nuclear safety will be selected for DRB review.

6. Additional training regarding design basis accident analysis (FSAR Chapter 14) will be provided to selected engineering and operations personnel to upgrade the knowledge level of the CR-3 plant staff.

PREVIOUS SIMILAR EVENTS

This appears to be the first event involving an unanalyzed condition concerning emergency feedwater.

There have been two previous events involving the emergency feedwater system reported in accordance with 10CFR50.73(a)(2)(v) in which the condition was determined to have prevented the fulfillment of a safety function. LER 85-027 reported a condition wherein the steam-driven EFP was disabled per procedure and the motor-driven pump was disabled due to a spurious EFIC actuation while calibrating EFIC instrumentation. A second spurious actuation occurred resulting in no EFW response. LER 88-008 reported a condition wherein three out of four EFIC level transmitters were found sufficiently out of tolerance to have prevented the actuation of emergency feedwater to the "B" OTSG.

LER's 94-006, 95-015, and 95-016 reported setpoints for EFIC system instrumentation determined to be non-conservative relative to revised analyses using new setpoint methodology which resulted in questioning the systems ability to perform its intended safety function.

On October 10, 1996, FPC provided a voluntary LER to describe an unreviewed safety question (USQ) involving the EDG loading calculation that was developed in support of the plant modification which removed the automatic open signal from ASV-204.

ATTACHMENT

Figure 1 - Emergency Feedwater System (Simplified)

NRC FORM 366A (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104

EXPIRES 5/31/95

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
CRYSTAL RIVER UNIT 3 (CR-3)	05000302	96	024	01	09 OF 09

TEXT (If more space is required, Use additional NRC Form 366A's) (17)

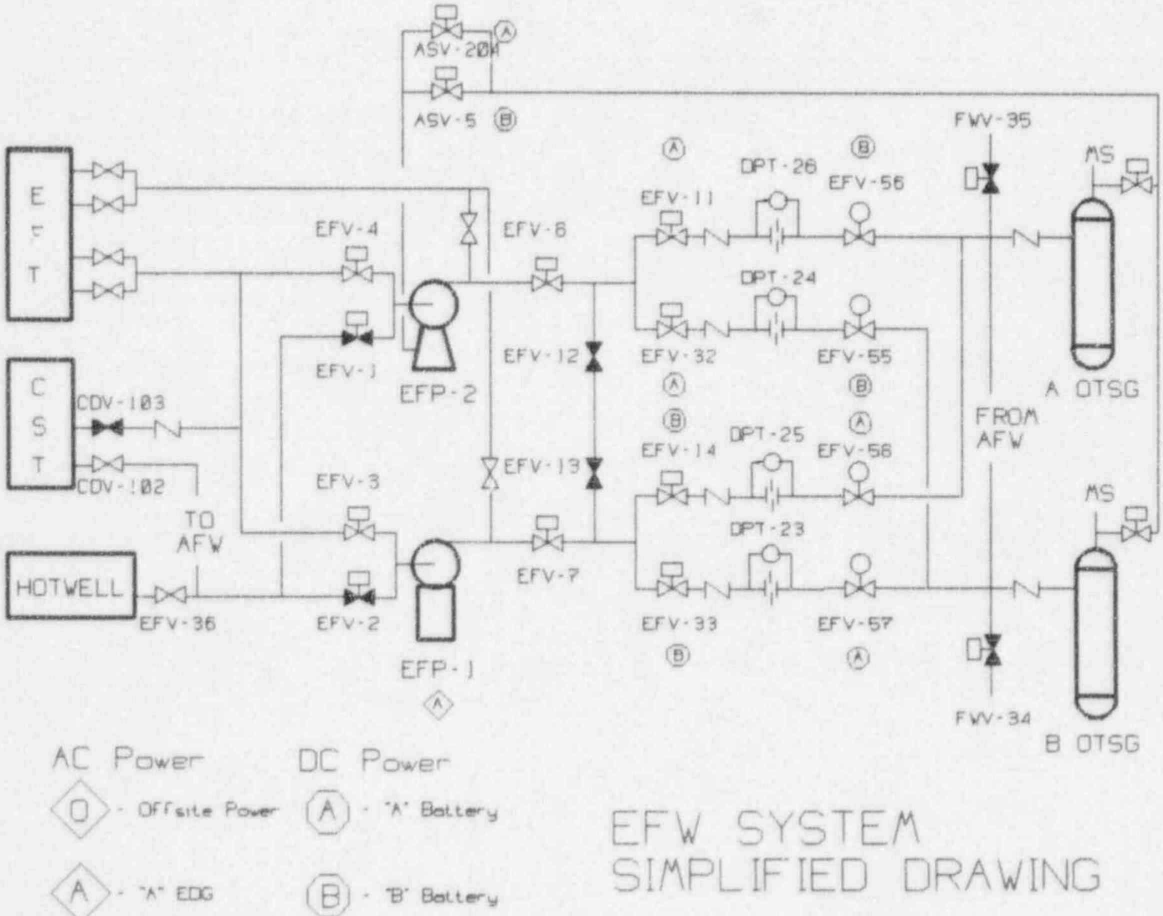

Figure 1