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

January 12, 1996
3F0196-10

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

Subject: Licensee Event Report (LER) 95-028-00

Dear Sir:

Please find the enclosed Licensee Event Report (LER) 95-028-00. This report is submitted by Florida Power Corporation in accordance with 10 CFR 50.73. A supplement to this report is expected to be issued by April 19, 1996.

Sincerely,

B J Hickle

B. J. Hickle, Director
Nuclear Plant Operations

TWC:ff

Attachment

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

IE22
1/1

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

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.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
CRYSTAL RIVER UNIT 3 (CR-3)	0 5 0 0 0 3 0 2	1 OF 0 9

TITLE (4): Operation Outside Design Basis Caused by Personnel Errors and Inadequate Documention of Borated Water Storage Tank Vacuum Breaker Capacity

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
									N/A	0 5 0 0 0
1 2	1 3	9 5	9 5	0 2 8	0 0	0 1	1 2	9 6	N/A	0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

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	3 5 2	5 6 3 - 4 6 0 1

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
B	B P	V A C	A 4 1 5	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 4	1 9	9 6

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

On December 13, 1995, Florida Power Corporation's Crystal River Unit 3 (CR-3) was in MODE ONE (POWER OPERATION) operating at 100% Reactor Power and generating 882 megawatts. During preparation for a test of Borated Water Storage Tank (BWST) vacuum relief valve DHV-70, the sensing line which connects the body of the vacuum breaker with the pilot assembly was found to be blocked. DHV-70 is one of two vacuum breakers required to provide vacuum relief to the BWST. Further investigation revealed the existing vacuum breakers did not individually have adequate relief capacity for their intended safety purpose. This was determined to be a condition outside the design basis and was reported as such on December 15, 1995. A screened rain-hood assembly is currently installed in place of DHV-70 to provide a vent path and assure OPERABILITY of the BWST. Corrective actions will include installation of replacement valves with adequate capacity, revisions to design basis documents to formalize the required relief capacity and strengthening the basis for BWST vacuum breaker redundancy. A formal analysis will determine the root cause of the corroded sensing line. Other actions will be identified after further evaluation of the issues involving change management and human performance which contributed to inadequate understanding of the design basis.

EVENT DESCRIPTION

On December 13, 1995, Florida Power Corporation's (FPC) Crystal River Unit 3 (CR-3) was in MODE ONE (POWER OPERATION), operating at 100% Reactor Power and generating 882 megawatts. During preparation for an ASME Section XI test of Borated Water Storage Tank [BP,TK] (BWST) vacuum relief valve [BP,VACB] DHV-70, the stainless steel sensing line which connects the body of the vacuum breaker with the pilot assembly was found to be totally obstructed with corrosion products. DHV-70 is one of two vacuum breakers required to provide vacuum relief to the BWST for protection during full Emergency Core Cooling System (ECCS)/Containment Spray [BE] drawdown. The system engineer was consulted and after an initial investigation, it was concluded the valve would not have worked properly with the sensing line clogged, although a test was not performed. When DHV-70 was removed from the system for testing, it was replaced with a screened rain-hood assembly to maintain an open vent path and prevent debris from entering the BWST. The system engineer conservatively elected to use the rain hood assembly because operators questioned the past practice of blanking-off the flanged connection while the removed valve was being tested. Clarification was sought to determine if one or both valves are required to protect the tank and prevent loss of Net Positive Suction Head (NPSH) to the Makeup & Purification (MU) pumps [CS,P] during maximum flow conditions. Subsequent investigation was completed on December 15, 1995 using the valve manufacturer's flow capacity information at different values of vacuum. This information indicated the two existing vacuum breakers (DHV-69 and 70) were not redundant, in that each considered alone, did not have adequate relief capacity for their intended purpose. A 1-Hour Notification was made to the Nuclear Regulatory Commission (NRC) using the Non-Emergency Event Notification system at 1041 hours on December 15, 1995. The notification was made in accordance with 10CFR72(b)(1)(ii)(B) as a suspected design basis issue and Event Number 29724 was assigned.

This report is being submitted in accordance with 10CFR50.73(a)(2)(ii)(B) as a condition outside the design basis of CR-3.

EVENT EVALUATION

The BWST and RB sump [BE,TK] provide borated water for the Decay Heat (DH) Removal pumps which function as the Low Pressure Injection [BP,P] pumps (LPI) during LOCA's, the Reactor Building Spray [BE,P] (BS) pumps, the Makeup Pumps [CB,P] which function as High Pressure Injection (HPI) pumps [BQ,P] during LOCA's, the Spent Fuel Cooling [DA,P] (SF) pumps, and the BWST recirculation pump. The BWST has four 8-inch penetrations at the top of the tank including the two vacuum breakers, a line to the Auxiliary Building Air Handling Exhaust Fans [UC,FAN] which continuously draw air from the top of the BWST, and an overflow line which has a large loop seal that affords tank overpressure protection. See Figure 1.

The basic function of DHV-69 and 70 is to equalize the pressure inside the BWST to provide protection against a collapse of the tank when vacuum is created during drawdown (see Figure 2). The valve setpoint is critical in maintaining adequate net positive suction head (NPSH) to the Makeup Pumps which are aligned to take suction from the BWST in their High Pressure Injection (HPI) mode during a large break loss-of-coolant-accident (LOCA). Analyses for Makeup/HPI Pump NPSH and Allowable Makeup Tank [CB,TK] (MUT) Overpressure assume a vacuum breaker setpoint for the BWST of 12 inches water vacuum for the most limiting case where the BWST is at its lowest level, the operating Makeup/HPI pump is Engineered Safeguards (ES) selected and Decay Heat/Low Pressure Injection and Reactor Building Spray pumps are being throttled for switchover to the RB sump. A vacuum in the BWST of greater than 12 inches water would reduce pressure on the BWST side of the tie-in point, thereby reducing NPSH available. As noted in the above Event Description, maintaining an open vent path in place of the removed vacuum breaker assures adequate vacuum relief capacity. With this arrangement, there is currently no impact on the health and safety of the public.

The BWST Vacuum Breaker valves are ISI Code Class 2 relief valves required to be tested in accordance with ASME OM-1 "Code for Operation and Maintenance of Nuclear Power Plants". DHV-69 and 70 were last tested on August 2, 1993 and August 6, 1993, respectively. Accurate valve pressure relief setpoints are necessary for the valves to fulfill their intended safety function. ASME OM-1 requires the valves to be repaired or replaced if the "as found" set pressure exceeds the set pressure by 3% or greater, (the upper allowable limit). In this case, the system engineer determined DHV-70 would not have passed the "as found" criteria.

Attachment 2 is provided as reference information to fully inform the NRC of activities which occurred subsequent to the event notification. The circumstances are not a direct part of the subject event but support the corrective actions identified.

CAUSE

The suspected inoperability of DHV-70 was due to blockage in the sensing line caused by accumulation of corrosion products. A more formal analysis will be developed by February 23, 1996 to determine the exact root cause.

The issue regarding whether or not DHV-69 and 70 were redundant, was caused by inadequate change management and human performance. Design basis documents were not specific regarding BWST Vacuum Breakers. Contributing causes included inadequate and outdated references. One example was a failure to update assumptions made in the original (1969) sizing calculation for the vacuum breakers. Another appears to be a reliance on "conservative estimates and engineering judgments" by the Architect-Engineer (A/E). In May, 1990, the A/E provided information regarding the effect the vacuum condition in the BWST had on the available NPSH for the Makeup/HPI pumps. The results were based on an evaluation of source documentation for Auxiliary Building Exhaust Fan pressure and the BWST vacuum breaker set pressure with a conclusion that sufficient NPSH margin for the Makeup/HPI pumps continued to exist.

IMMEDIATE CORRECTIVE ACTION

The BWST was determined to be fully OPERABLE with the screened rain-hood installed in place of DHV-70, thereby providing adequate relief capacity.

A work request was initiated to remove and inspect DHV-69 for a common failure mechanism (see Additional Corrective Actions).

ADDITIONAL CORRECTIVE ACTION

In order to drain the Fuel Transfer Canal, part of the normal Decay Heat flow is diverted to the BWST through the recirculation line, thus increasing the potential for a radioactive release through an open vent path in the BWST. Therefore, the following action items 1 through 5, necessary to replace DHV-69 and 70 will be completed prior to draining the Fuel Transfer Canal [DF] during the upcoming Refuel 10 Outage which is scheduled to begin February 29, 1996.

1. The relief capacity for the BWST Vacuum Breakers will be defined by verified, formal calculation.

2. FPC will ensure the relief capacity of the replacement vacuum breakers are adequate prior to installation.

3. The design basis document for the Decay Heat System will be revised to strengthen the information related to vacuum breakers and to update available references.

4. An adjustment will be made to the setpoint on the replacement vacuum breaker and the new valve will be installed prior to draining the Fuel Transfer Canal as noted in the Event Evaluation Section, above. The purpose of the adjustment is to establish added assurance that the valve will open and provide sufficient flow prior to reaching 12 inches of water column vacuum in the BWST. Prior to replacement of DHV-70, the screened rain-hood be installed in place of DHV-69.

5. A second vacuum breaker will be obtained as a replacement for DHV-69 and installed prior to draining of the Fuel Transfer Canal as noted above.

6. An evaluation will be conducted by April 7, 1996 of previous on-line maintenance and surveillance activities associated with DHV-69 and 70 and the previous practice of using a blank flange in place of the vacuum breaker removed for testing. A supplement to this report will be considered at that time to describe any past operability concerns with one of two old model vacuum breakers in service.

ACTION TO PREVENT RECURRENCE

1. A formal root cause for the blocked vacuum breaker sensing line will be documented by February 23, 1996. Based on the results, FPC will confirm whether or not other vacuum breakers installed at CR-3 are subject to the same failure mechanism.

2. A chronology of the BWST vacuum breaker design basis and change management will be developed by April 28, 1996. The information will be used to identify possible programmatic and/or human performance issues. Additional actions to prevent recurrence may be identified at that time. The chronology and recommendations/actions as appropriate, will be forwarded to design, systems, and procurement engineering for review as a "lessons learned".

PREVIOUS SIMILAR EVENTS

There have been no previous reportable events at CR-3 involving inadequate sizing of vacuum breakers. Based on a review of the Nuclear Plant Reliability Data System (NPRDS), CR-3 is the only plant that uses the AGCO Type 93 valve as a vacuum breaker. There have been no previous NPRDS reports made on these breakers by CR-3.

ATTACHMENT

Attachment 1 -Abbreviations, Definitions and Acronyms

Figure 1 - BWST Arrangements

Figure 2 - Type 93 Vacuum Relief Valve

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
CRYSTAL RIVER UNIT 3 (CR-3)	0 5 0 0 0 3 0 2	9 5	0 2 8	0 0	0 6 OF 0 9

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

ATTACHMENT 1 - ABBREVIATIONS, DEFINITIONS AND ACRONYMS

BS	Building Spray
CR-3	Crystal River Unit 3
ECCS	Emergency Core Cooling Systems
FPC	Florida Power Corporation
HPI	High Pressure Injection
LOCA	Loss of Coolant Accident
LPI	Low Pressure Injection
MODE ONE	POWER OPERATION (Greater Than 5 Percent Rated Thermal Power)
MU	Makeup and Purification System
NPSH	Net Positive Suction Head - a measure of the head (pressure) available to prevent cavitation.
Problem Report	A Problem Report documents a condition or event which impacts CR-3 and warrants evaluation, root cause analysis, or corrective actions beyond what it would receive if documented and processed by other methods.

NOTES: ITS defined terms appear capitalized in LER text {e.g. MODE ONE}

Defined terms/acronyms/abbreviations appear in parentheses when first used {e.g. Reactor Building (RB) }.

EIIS codes appear in square brackets {e.g. Makeup Tank [CB,TK] }

ATTACHMENT 2 - BACKGROUND AND TESTING ACTIVITIES RELATED TO REPLACEMENT FOR DHV-70

On December 18, 1995, while preparing to replace DHV-70 with a valve withdrawn from stock, a bench test was performed to establish the "as-found" data for the valve. The mechanical shop engineer noted the valve did not meet the requirement for the valve to fully open at 4.0 ounces per square inch vacuum which converts to 6.928 inches water column. The valve which failed the bench test had been purchased in 1994 using a safety related, commercial grade dedication procurement method. The dedication involved the use of source inspection as a basis for the dedication and the inspector witnessed the functional test, noting a set pressure of 5.9 inches water column. A Quality Material Problem Report (QMPR) was issued on January 5, 1996 with a disposition to return the valve to the vendor, Anderson, Greenwood & Company (AGCO). The return instructions require AGCO to perform an "as-found" lift test and inform FPC of the results along with recommended corrective action required, prior to proceeding with any repair. FPC intends to evaluate the results of "as-found" testing by AGCO of the returned Type 96A vacuum breaker along with any corrective action recommendations.

The above vacuum breaker is a Type 96A0606RS vacuum relief valve purchased in 1994 as a replacement for the existing Type 930608RC valve for DHV-69 and 70. The equivalency evaluation for the replacement indicates the decision was based on an attempt to standardize valve design presently used in another application and for ease of maintenance. Per discussion with the responsible procurement engineer, AGCO had also informed FPC the Type 93 valve was no longer being marketed as a vacuum relief device. The need for a spare replacement valve was related to several actions resulting from a review of the ISI classification for DHV-69 and 70 identified as part of a focused, configuration management effort. The review confirmed the valves should have been ISI Code Class 2 and that a Code Class 4 designation had been incorrectly assigned when ISI flags were originally added to CR-3 flow diagrams. In an attempt to justify the Code Class 4 designation, an evaluation was performed of alternate venting paths including the line to the Auxiliary Building Air Handling System and the BWST loop seal (see Figure 1). These paths were noted as not seismically designed and their integrity could not be assured. Problem Report EEPR 91-0014 was issued in July, 1991 to identify the need to replace several non safety-related parts installed based on the incorrect ISI designation and to change the drawings to identify the correct classification. One of the changes made, was to replace the carbon steel sensing line and fittings with stainless tube and fittings to provide better resistance to corrosion due to the salt air environment in which the valves are exposed. During performance of the work needed to change out the non safety-related parts, DHV-70 was found to be degraded and Problem Report 93-0194 was issued to determine if the operability of the BWST was affected. The problem report was determined to be not reportable on the basis a fully operable "redundant" DHV-69.

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)	0 5 0 0 0 3 0 2	9 5	0 2 8	0 0	0 8 OF 0 9

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

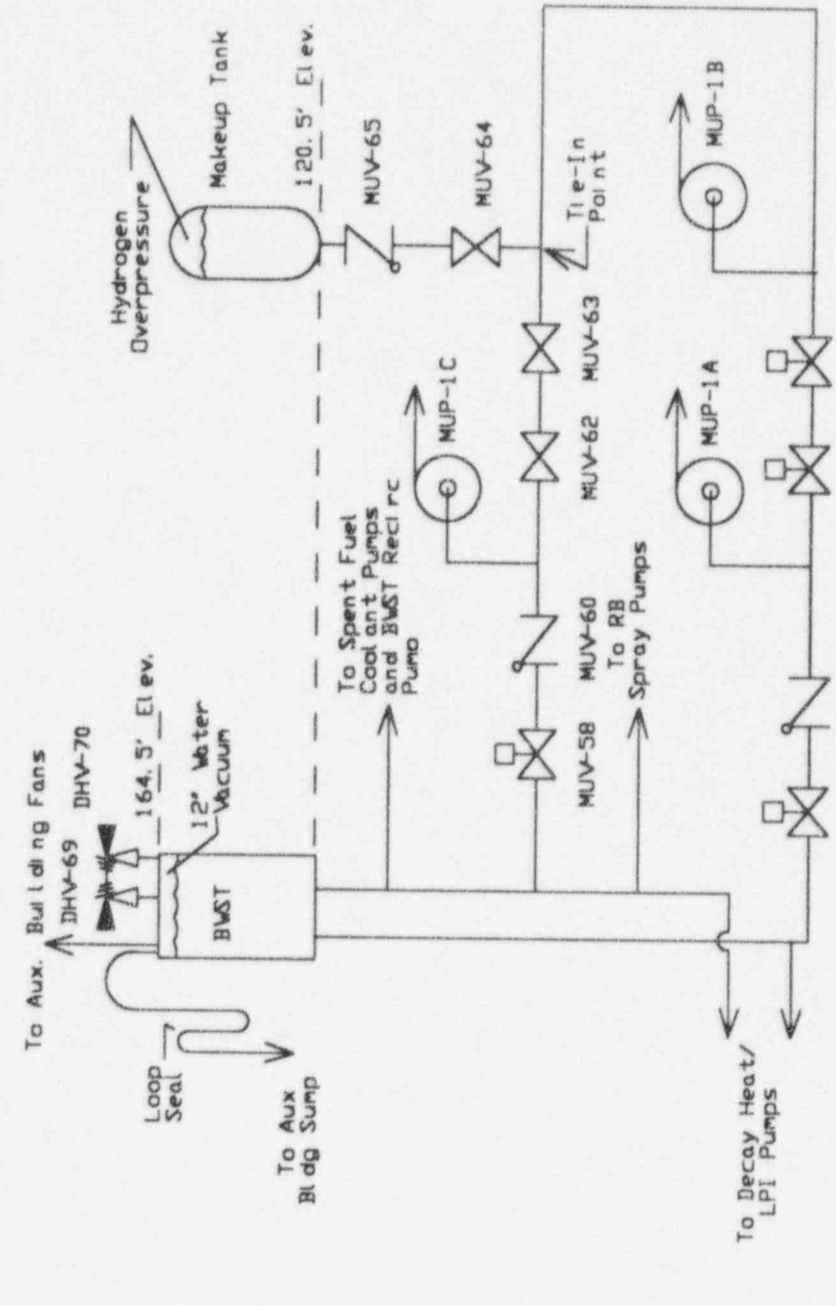
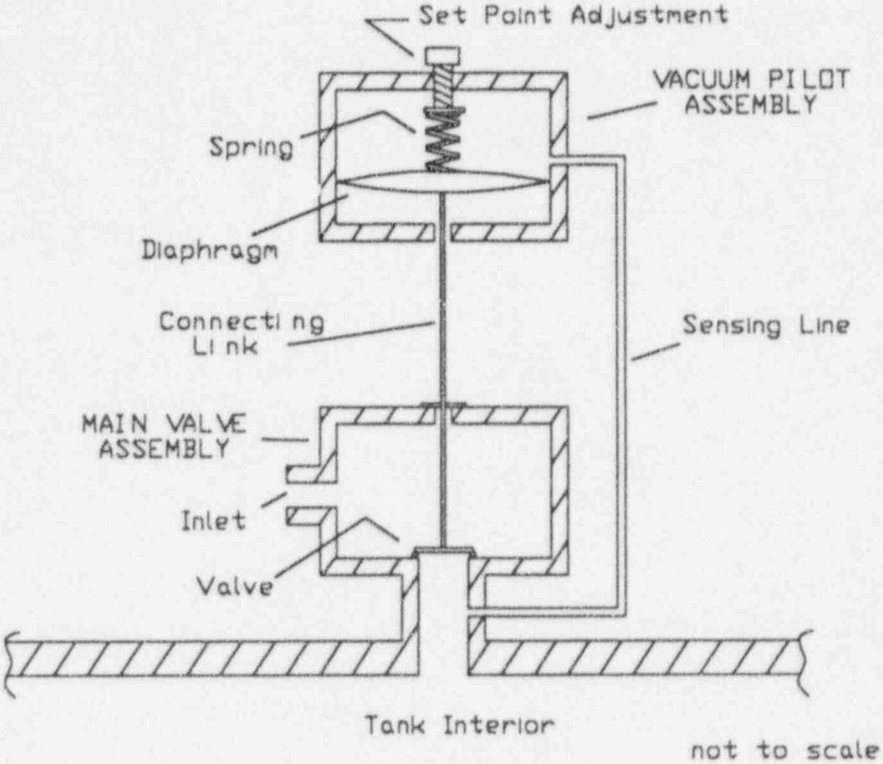


Figure 1

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)	0 5 0 0 0 3 0 2	9 5	0 2 8	0 0	0 9 OF 0 9

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



SIMPLIFIED VACUUM RELIEF VALVE

Figure 2