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

3F0391-10
March 8, 1991

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

Reference: FPC Letter to NRC dated 4/30/90
Licensee Event Report (LER) 90-005

FPC Letter to NRC dated 12/20/90
Licensee Event Report (LER) 90-005-01

Dear Sir:

Enclosed is Licensee Event Report (LER) 90-005-02 which is submitted in accordance with 10 CFR 50.73.

This supplement includes additional information relative to a concern that pH conditions may exceed design capability during a Small Break Loss of Cooling Accident (SBLOCA).

Sincerely,

Gary Boldt

G. L. Boldt
Vice President
Nuclear Production

WLR:mag

Enclosure

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

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NRC FORM 366 (6-89) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES 4/30/92

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
CRYSTAL RIVER UNIT 3	0 5 0 0 0 3 0 2	1 OF 0 8

TITLE (4) Calculation of RB Flood Level Shows Level Exceeds that of Safe Shutdown Equipment Due to Design Error Causing Operation Outside Design Basis

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)
0 3	2 9	9 0	9 0	— 0 0 5 —	0 2	0 3	0 8	9 1	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9): 6

POWER LEVEL (10): 0 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
W. K. BANDHAUER, NUCLEAR OPERATIONS SUPERINTENDENT	AREA CODE 9 0 4 7 9 5 - 6 4 8 6

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

On March 29, 1990 at approximately 1400, Florida Power Corporation (FPC) Crystal River Unit 3 (CR-3) was in MODE 5 (COLD SHUTDOWN) preparing for a refueling outage, when it was determined that CR-3 was operating outside the design basis due to a non-conservative Reactor Building (RB) maximum accident flood level calculation. This non-conformance was caused by an oversight in 1972 by the design engineer when a Net Positive Suction Head calculation for post-accident recirculation cooling was incorrectly used for the maximum flood level. The actual maximum RB flood level exceeds the level of some safe shutdown instrumentation and equipment.

Modifications and calculations performed after 1972 did not identify this problem because the design engineers assumed the original calculation was correct or were unaware of the original calculation. While evaluating the alternatives to this issue, FPC and Babcock & Wilcox Nuclear Service Company identified a concern that pH conditions may exceed design capability during a Small Break Loss Of Coolant Accident (SBLOCA).

FPC procedures have been revised to direct operators to limit the volume of water contributed by the Borated Water Storage Tank, and to close the Sodium Hydroxide (NaOH) outlet valve during a SBLOCA. To permanently resolve this issue, FPC will raise the affected instrumentation above the new flood level.

EVENT DESCRIPTION

On January 9, 1990, during an engineering review of a calculation performed for NRC Bulletin 79-01B, non-conservative assumptions were identified in the calculation of the Reactor Building (RB) [NH] flood level. This calculation review was performed as a part of the engineering configuration management verification and upgrade. The assumptions included using nominal or minimum tank volumes instead of maximum tank volumes permitted by Technical Specifications and assuming water from inside the primary shield wall does not reach the RB sump [NH,ACC]. Based on these concerns, the maximum RB flood level was re-calculated. The final results of the corrected calculations were received by Florida Power on March 28, 1990. Evaluation of the results on March 29, 1990, at approximately 1400, concluded the maximum level would exceed the level necessary to prevent submergence of essential safe shutdown instrumentation and equipment. This condition is considered to be outside the plant design basis.

While evaluating the alternative resolution strategies to the RB Flood Level issue, FPC and Babcock & Wilcox Nuclear Service Company (B&W) identified that the pH of the RB spray [BE] for a Small Break Loss Of Coolant Accident (SBLOCA) may not be enveloped by existing Loss Of Coolant Accident (LOCA) analysis. The system response for SBLOCA may create different drawdown rates on the Borated Water Storage Tank (BWST) [BP,TK] and Sodium Hydroxide (NaOH) Tank [BE,TK] than those assumed in the Large Break Loss Of Coolant Accident (LBLOCA) analysis as the worst case single active failure.

At the time of the verification that the plant was outside the design basis, March 29, 1990, CR-3 was in MODE 5 (COLD SHUTDOWN) in preparation for a refueling outage. No immediate actions were necessary.

At 1410 on March 29, 1990, a four-hour verbal report of this non-conformance was provided to the NRC Operations Center per 10CFR50.72(b)(2)(i) requirements. This written report is being provided per the requirements of 10CFR50.73(a)(2)(ii).

CAUSE

This non-conformance was caused by an oversight in 1972 by the architect design engineer performing the original RB flood level calculation. The original calculation used minimum or nominal Borated Water Storage Tank (BWST) [BP,TK], NaOH Tank [BE,TK], and Core Flood Tanks (CFT) [BP,TK] volumes and assumed that water within the primary shield area and other areas does not reach the RB sump area. These assumptions are appropriate for calculating the minimum water level available for Decay Heat Removal (DHR) and Building Spray (BSP) systems' pumps [BP,P][BE,P] Net Positive Suction Head (NPSH). The resultant minimum water level of 99.85 ft. plant datum was incorrectly assumed to be the maximum level above which critical instruments and equipment must be located.

Since the original 1972 calculation, other calculations and modifications have occurred which could have identified this problem. In 1981, during evaluation of NRC Bulletin 79-01B, the maximum RB flood level was reviewed and recalculated. These calculations were based on the faulty assumptions and incorrectly calculated a new flood level. In 1987, the primary shield wall was modified when drain holes were drilled in the wall to allow water to drain to the RB sump area. The utility design engineer contacted the Nuclear Steam Supply System vendor, the Architect Engineering firm, and a sister utility to determine why drains were not already installed, but no reason could be provided. The utility design engineer did not determine any impact on the original flood calculation since the impact to the original, or later calculations and assumptions, was not provided. In 1989, the flood level calculation was reviewed by a contract design engineer to determine the impact of the addition of equipment which would displace water and thus potentially raise the maximum water level. This calculation did not re-verify the previous level calculation, but simply evaluated the impact on the level. In conclusion, it appears these calculations and modifications did not identify the faulty assumptions because the design engineers assumed the original calculation was correct or were not provided the pertinent original design assumptions.

EVENT EVALUATION

FSAR Section 6.2.2.1 states:

> "In the event of a postulated LOCA [Loss of Coolant Accident], water will be pumped into the reactor building via the Reactor Building Spray System and Decay Heat Removal System as described in Sections 6.2 and 9.4, respectively. The reactor building will fill to an approximate elevation of 99.85 ft. prior to the initiation of the recirculation mode of the Emergency Core Cooling System (ECCS)."

This statement is incorrect. The essential safe shutdown instrumentation used during a LOCA is located approximately two to three inches above the 99.85 ft. elevation. The correct flood level elevation should be 101.7 ft.(101'8"), approximately 1.85 ft. above the incorrect flood level. As a result, the attached list of essential safe shutdown equipment may be subjected to an environment for which they are not qualified to perform their safety function.

The corrected flood level elevation of 101'8" is based on maximum BWST, NaOH, and CFT tank volumes and assumes the water level in the primary shield wall will equalize with the level in the RB sump area following a LOCA in the cold leg reactor coolant pump [AB,P] suction. Additionally, the entire contents of the Reactor Coolant System (RCS) [AB], less the Reactor Vessel [AB,RPV] volume, are assumed to contribute to the final maximum water level. These new assumptions are conservative because few accidents result in totally draining tanks and major portions of the RCS.

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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

Most of the instrumentation and equipment affected will perform their automatic safety function before the water level reaches the equipment. Automatic re-initiation of High Pressure Injection (HPI) [BQ] may not occur. Additionally, automatic actuation of Low Pressure Injection (LPI) [BP] may not occur. However, operators would be capable of manually initiating these safety systems. If the affected containment isolation valves [JM,ISV] have been opened, the associated containment penetration can still be automatically isolated with the isolation valve located outside the RB. However, because many RCS pressure instruments [AB,PIT] are affected, the operator may not have reliable pressure and subcooling margin information.

The standard analytical assumption has been that the pH of the sump water must be basic. To assure that the iodine did not re-evolve due to a LBLOCA, dose analysis in a plant of CR-3 vintage assumed a final RB sump water pH of 8.5 to 11.0 which is in agreement with NRC recommended limits. The initial LOCA analysis assumed simultaneous actuation of the HPI and LPI functions of the emergency core cooling systems (ECCS) [B] and the RB spray to evaluate the thermal-hydraulic performance of the reactor core [AB] and containment [NH]. The primary initial consequences of the LBLOCA are rapid depressurization of the reactor coolant system (RCS) [AB] and pressurization of the containment. A LBLOCA will actuate the HPI function at 1500 psig RCS pressure, the LPI function at 500 psig RCS pressure and the RB spray at a high RB pressure of 30 psig. These conditions are reached almost instantly during a LBLOCA.

The HPI, LPI and RB spray functions initially take suction from the BWST. The addition of NaOH to the RB spray is accomplished at CR-3 utilizing a passive, gravity feed connection to the RB spray pump suction path. The drawdown rates of borated water from the BWST and sodium hydroxide from the NaOH tank are functions of the hydraulic conditions for each tank and produce the desired pH ranges. The rapid RCS depressurization of a LBLOCA was utilized to establish the design requirements for the NaOH addition system to ensure the pH was met. Proper operation of the NaOH addition function during a LBLOCA was assured by considering the worst case single active failures expected to occur in the LPI and the RB spray systems. The HPI function is not assumed to play a part in mitigating a LBLOCA and has no significant, long term impact on the relative drawdown rates.

However, our SBLOCA analyses have shown that, depending upon the break size, a SBLOCA may actuate the HPI, but not actuate LPI or RB spray until some time after the break occurs. This scenario could create a difference in the relationship of the hydraulic heads on the BWST and NaOH tank. Consequently, the drawdown rates and relative amounts of the boric acid and NaOH supplied to the RB spray could be different when it is actuated. Thus, any LOCA scenario which does not produce simultaneous drawdown could produce a RB spray pH range which leads to a concern for the degradation of equipment in the RB.

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

CORRECTIVE ACTIONS

To permanently resolve the problem with the excessive flood level, Florida Power will relocate the RB instrumentation and equipment to an elevation higher than 102 ft. This will be achieved by the end of Refuel 8 (subject to material delivery).

In the interim, Florida Power will limit the volume of water contributed by the BWST. This has been accomplished by a procedural change in operator action. The operator will assure the RB flood level will not exceed the 99.85 ft. elevation. Following a LOCA, by procedure the operator will begin manual transfer of the ECCS pump suctions from the BWST to the RB sump when the RB flood level reaches the 97.6 ft. elevation (this level includes an allowance for instrumentation error and occurs later than 10 minutes into the event). In addition, an alarm will be received in the control room when the RB flood level reaches approximately 97.6 ft. The corresponding actual level will satisfy the ECCS pump NPSH, core cooling, shutdown and Ph requirements. The RB flood level will be less than 99.85 ft. when the switchover from the BWST to the RB sump is completed even under worst case large break LOCA flow rates. FPC has confirmed that this switchover can be accomplished in sufficient time. This action will assure that all equipment and instrumentation necessary to mitigate LOCA remain operable.

To resolve the potential pH problem FPC implemented procedural changes in the Emergency Operating Procedures to require the control room operators to close the NaOH tank outlet valve [BE,ISV] in response to symptoms indicative of a SBLOCA. The procedures advise the operators to verify, prior to closing the valves, that RCS pressure is remaining above 200 psig by monitoring existing RCS pressure instrumentation. This action will resolve the immediate concern and is acceptable for an interim corrective action since NaOH is only necessary to preclude the re-evolution of iodine during the recirculation phase for a LBLOCA. The procedural action will eliminate the possible high RB spray pH condition. The procedures have been revised and the licensed reactor operators have been trained on the procedure changes.

The potential to use inadequate assumptions and inputs in current calculations has been reduced by procedural enhancements and training. Procedures specify a format detailing design inputs and assumptions with reasons/bases clearly documented for future use. The potential for omitting the applicable design inputs in current MARs has been reduced by training, procedural enhancements and improved design basis documentation. These documents include enhanced design basis documents, analysis basis documents, and the Configuration Management Information System.

PREVIOUS SIMILAR EVENTS

This is the first report related to design error in the maximum RB flood elevation calculation. Two prior events were identified that also relate to RB equipment inoperability due to submergence. These previous events were concerned with locating equipment below the established RB flood elevation.

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SAFE SHUTDOWN EQUIPMENT AFFECTED

TAG NO.	EQUIPMENT DESCRIPTION
INSTRUMENTATION	
AH-536-TE [NH,TI]	Reactor Building temperature instrumentation, used for post accident monitoring
RC-1-LT1 [AB,LT] RC-1-LT3 "	Pressurizer level transmitters. This instrumentation is used by the operator and by the Integrated Control System to control makeup flow and pressurizer level.
RC-3A-PT3 [AB,PT] RC-3A-PT4 " RC-3B-PT3 "	Reactor Coolant System (RCS) pressure transmitters used by o Reactor Protection System for high pressure, low pressure and variable low pressure Reactor trips. o Engineered Safeguards System for High Pressure Injection and Low Pressure Injection automatic actuation. o Automatic Closure Interlock for overpressure protection of the Decay Heat Removal System. o Pressurizer spray, heaters and Pilot Operated Relief Valve for RCS pressure control. o Main Control Board indication for RCS pressure and Subcooling Margin.
RC-132-PT [AB,PT]	Low range RCS pressure transmitters used for pressure indication on the Main Control Board and for engineered safeguards actuation.
RC-158-PT [AB,PT] RC-159-PT "	Wide range RCS pressure transmitters located on the Main Control Board and the Remote Shutdown Panel.
RC-163A-LT1[AB,PT] RC-163B-LT1 " RC-164A-LT1 " RC-164B-LT1 " RC-163A-TE1[AB,TT] RC-163B-TE1 " RC-164A-TE1 " RC-164B-TE1 "	Reactor Coolant Inventory Tracking System level and temperature transmitters used for post-accident indication.

SAFE SHUTDOWN EQUIPMENT AFFECTED (CONT.)

TAG NO.	EQUIPMENT DESCRIPTION
SP-18-LT [JB,LT]	Emergency Feedwater Initiation and Control level transmitters used for initiation of Emergency Feedwater on low Steam Generator level.
SP-19-LT "	
SP-21-LT	
SP-22-LT	
SP-23-LT	
SP-24-LT	
SP-25-LT	
SP-27-LT	
SP-31-LT	
SP-32-LT	

VALVES

CAV-1 [KN,V]	RCS sampling valves and containment isolation valves.
CAV-3 "	
CAV-126 "	
MUV-40 [CB,V]	Letdown Cooler isolation valves and containment isolation valves.
MUV-41 "	
MUV-505 "	