Florida Power
CORPORATION

May 17, 1990
3F0590-06

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

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Reactor Building Flooding

Reference: Licensee Event Report No. 90-005, dated April 29, 1990

Dear Sir:

On March 29, 1990, Florida Power Corporation (FPC) determined that Crystal River Unit 3 (CR-3) was operating outside the plant design basis due to procedures which were based upon a non-conservative Reactor Building (RB) maximum accident flood level calculation. This problem is described in the reference LER (copy enclosed). The purpose of this letter is to describe the resolution which FPC will pursue.

The Borated Water Storage Tank (BWST) is the primary source of ECCS water to mitigate design basis accidents. During the initial phase of LOCA mitigation, the BWST provides the source of water for the High Pressure Injection (HPI), Low Pressure Injection (LPI) and Building Spray (BS) pumps to inject water into the reactor vessel and spray into the RB atmosphere. The water inventory from the reactor coolant system, the BWST, and the other tanks (Core Flood Tank, Makeup Tank, and Sodium Hydroxide Tank) fall to the RB floor, eventually reaching the RB sump located in the 95 ft floor elevation (plant datum). Flooding of this elevation will occur. In accordance with the current emergency operating procedures, when the BWST level reaches 2.5 ft, the operator manually transfers the LPI and BS pump suctions from the BWST to the RB sump, initiating the recirculation phase. This action would result in a RB flood

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level of approximately 102 ft elevation with maximum Technical Specification volumes in the various tanks at the beginning of the accident. A number of instruments necessary to mitigate the design basis accidents are installed on the walls below the 102 ft elevation, but above the 99.85 ft elevation (the maximum flood level described in FSAR Section 6.2.2.1). Thus, a design basis accident could cause these instruments to be submerged. It is, therefore, necessary to initiate operator action to limit the flood level to the FSAR value of 99.85 ft elevation.

To limit the flood level, there are three considerations which must be addressed. These are:

a. Water volumes and boron concentrations for core cooling and shutdown requirements must be satisfied,

b. Minimum required NPSH of the LPI and BS pumps must be satisfied before switchover to sump, and

c. Materials inside the RB must be compatible with the pH of the RB spray or sump fluid.

The licensing bases LOCA models consider the first item and limit the minimum volume of water and boron concentration in the BWST. The required levels for the LPI pumps and the BS pumps NPSH are 95.6 ft and 97.0 ft, respectively. Maintenance of the expected pH in the range of 7.2 to 11.0 by establishing a proper balance of sodium hydroxide and borated water will assure material compatibility. A limited duration excursion above a pH of 11.0 is verified to be acceptable in our EQ program documentation.

The approach which best satisfies these considerations and resolves the RB flooding issue is to limit the volume of water contributed by the BWST. This will be accomplished by a procedural change in operator action. The operator will ensure the RB flood level will not exceed the 99.85 ft elevation. Following a LOCA, the operator will begin manual transfer of the ECCS pump suctions from the BWST to the RB sump when he receives an alarm indicating that the RB flood level has reached approximately 97.6 ft elevation (this level includes an allowance for instrumentation error). 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 with the switchover from the BWST to the RB sump completed in 10 minutes even under worst case large break LOCA flow rates. FPC has confirmed that this switchover can be accomplished in this amount of time.

Prior to CR-3 restart from this refueling outage, FPC will perform the following actions:

1. A modification will be installed to add an alarm in the Main Control Room to indicate when the RB flood level has reached the point that operator action to initiate switchover from the BWST to the sump is to begin.

2. Operational procedures will be revised and issued. Operator training on the use of the RB flood level as the new switchover parameter will be completed.

If you have any questions, please contact Mr. Rolf C. Widell, Director, Nuclear Site Support at (904)563-4329.

Sincerely,

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

PMB/JWT

Enclosure

xc: Regional Administrator, Region II
Senior Resident Inspector

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

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.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): CRYSTAL RIVER UNIT 3

DOCKET NUMBER (2): 0 5 0 0 0 3 0 2

PAGE (3): 1 OF 0 6

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)
									N/A	0 5 0 0 0
0 3	2 9	9 0	9 0	0 0 5	0 0	0 4	3 0	9 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: L. W. MOFFATT, NUCLEAR SAFETY SUPERVISOR

TELEPHONE NUMBER: 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	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
N/A									
N/A									

SUPPLEMENTAL REPORT EXPECTED (14)

X YES (If yes, complete EXPECTED SUBMISSION DATE) | NO

EXPECTED SUBMISSION DATE (15): MONTH 1 1 | DAY 1 5 | YEAR 9 0

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

On March 29, 1990 at approximately 1400, Florida Power (FPC) determined that Crystal River Unit 3 (CR-3) was operating outside the plant design basis due to a non-conservative Reactor Building (RB) maximum accident flood level calculation. When re-calculated using conservative assumptions, the maximum RB flood level exceeds the level necessary to prevent submergence of safe shutdown instrumentation and equipment. At the time of this determination, CR-3 was in MODE 5 (COLD SHUTDOWN) preparing for a refueling outage. This non-conformance was caused by an oversight in 1972 by the design engineer when a calculation of the Net Positive Suction Head for post-accident recirculation cooling was incorrectly used for the maximum flood level. 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. FPC is evaluating possible solutions to resolve this design basis issue prior to restart from the refueling outage and is continuing to evaluate corrective actions to prevent recurrence.

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

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

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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.

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

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

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.

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 and Building Spray systems pump [BP,P][BE,P] Net Positive Suction Head. 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 feet 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 Reactor Coolant System.

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.

One impact of the new flood level is the information provided to the operator. Because many RCS pressure instruments [AB,PIT] are affected, the operator may not have reliable pressure and subcooling margin information.

CORRECTIVE ACTIONS

Florida Power is evaluating possible solutions to resolve this non-conformance prior to restart from the current refueling outage. These solutions include relocation of equipment necessary for accident mitigation, re-evaluation of the analytical methods which have established operator procedural steps, and reduction in tank volumes.

Florida Power is evaluating the modification and calculation controls for improvement to reduce the possibility of the types of errors which contributed to this non-conformance, in future modifications and calculations.

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.

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.
WD-303A-LT[NH,LT] WD-303B-LT " WD-304A-LT " WD-304B-LT "	Reactor Building Flood level transmitters.

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 "	